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

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MEMORANDUM FOR: Edward L. Jordan, Chairman
Committee to Review Generic Requirements

FROM: Eric S. Beckjord, Director
Office of Nuclear Regulatory Research

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

SUBJECT: DRAFT REGULATORY GUIDE ON STANDARD FORMAT AND CONTENT FOR LICENSE RENEWAL APPLICATIONS AND DRAFT STANDARD REVIEW PLAN FOR THE REVIEW OF LICENSE RENEWAL APPLICATIONS

Enclosed for review by the CRGR is a draft Regulatory Guide (Enclosure 1) which describes the standard format and content of technical information for applications to renew nuclear power plant operating licenses. The draft regulatory guide conforms to the standard format for regulatory guides and reflects a level of detail and character of content that are consistent with other regulatory guides. This draft has been formatted to track closely with the proposed License Renewal Rule (10 CFR 54) and to supplement and amplify the requirements conveyed by 10 CFR 54 so that licensees would meet these requirements in a manner acceptable to the NRC staff.

Also enclosed for review by CRGR is a draft Standard Review Plan for License Renewal (SRP-LR) (Enclosure 2) which has been prepared as guidance for staff use in performing safety reviews of applications for the renewal of power reactor licenses. The draft SRP-LR provides a framework for the staff relative to what information needs to be reviewed and provides acceptance criteria to assist the reviewers in evaluating the submitted information. The staff recognizes that the review procedures and acceptance criteria are rather general. However, the staff fully expects the draft SRP-LR to be a living document and that the review procedures and acceptance criteria in the draft SRP-LR will be periodically revised to reflect the staff experience gained from the review of the industry technical reports and the lead plant applications. The result of a staff review using this SRP-LR is to obtain assurance that license renewal will not lead to age-related degradation that would reduce the level of safety at an existing reactor facility below acceptable levels during the renewal term.

The development of the draft R.G. and the draft SRP-LR have been closely coordinated between the Offices of Nuclear Regulatory Research (RES) and Nuclear Reactor Regulation (NRR) to ensure that the two documents are complementary and consistent. Therefore, we request that the CRGR review of these two documents should occur simultaneously.

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For further information on the draft R.G., please contact Robert Bosnak, Deputy Director, Division of Engineering, RES, X23850. For information on the draft SRP-LR, please contact William Travers, Assistant Director for Special Projects, NRR, X21117.

The enclosed material is pre-decisional and is intended for internal use only.

ORIGINAL SIGNED BY

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Original signed by
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Enclosures:
1) Draft Regulatory Guide DG-1009
2) Draft Standard Review Plan for License Renewal

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U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REGULATORY RESEARCH

DRAFT REGULATORY GUIDE DG-1009

STANDARD FORMAT AND CONTENT OF TECHNICAL INFORMATION FOR APPLICATIONS TO RENEW NUCLEAR POWER PLANT OPERATING LICENSES

REVISION 5A

AUGUST 1990

Contact: J. Vora (301) 492-3854

Note: This draft Regulatory Guide is based upon the proposed license renewal rule, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," 10 CFR 54 (Federal Register, Vol. 55, No. 137, July 17, 1990). Future modifications to the proposed rule will be reflected in commensurate changes in the draft Regulatory Guide.

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STANDARD FORMAT AND CONTENT OF TECHNICAL INFORMATION FOR APPLICATIONS TO RENEW NUCLEAR POWER PLANT OPERATING LICENSES

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A. INTRODUCTION

The Nuclear Regulatory Commission (NRC) is proposing to supplement its regulations in Title 10 of the Code of Federal Regulations by adding Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."[a] Section 54.21 of the proposed rule specifies the technical information to be included as part of a License Renewal Application. This information is to be included in a supplement to the current, updated Final Safety Analysis Report (FSAR). The supplement is to be included in the application submitted by a nuclear power plant (NPP) licensee for a renewed operating license. The FSAR supplement will include an evaluation of the aging mechanisms that result in degradation of the plant's systems, structures, and components (SSCs) important to license renewal, as defined in 10 CFR 54.3(a). The FSAR supplement will also include a demonstration that the effects of such degradation will be effectively managed such that the current licensing basis, as defined in 10 CFR 54.3(a), for the NPP will be maintained throughout the renewal term. Each FSAR supplement is to contain the information required by 10 CFR 54.21.

Purpose

The purpose of this regulatory guide is to establish a uniform format and content acceptable to the NRC staff for structuring and presenting the technical information to be compiled by an applicant for a renewed NPP operating license and submitted by the applicant as part of an application for a renewed

(a) This draft regulatory guide is based upon the proposed License Renewal Rule, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," 10 CFR 54 (Federal Register, Vol. 55, No. 137, July 17, 1990). Future modifications to the proposed rule will be reflected in commensurate changes in the draft Regulatory Guide.

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license. This regulatory guide identifies the content of and provides technical criteria for the compiled technical information. Use of this format will help to ensure the completeness of the information compiled or provided, will assist the NRC staff and others in locating the information, and will aid in shortening the time needed for the process of reviewing the license renewal application.

Scope

This regulatory guide provides a standard format and content for the technical information required by 10 CFR 54 to be compiled or submitted in support of an application for a renewed operating license. Detailed technical information needs and a description of a standard format that is acceptable to the NRC staff are presented in Section C1 of this regulatory guide.

This regulatory guide also provides for meeting the technical information requirements of 10 CFR 54 including 1) content of technical information to be included in license renewal applications, 2) criteria for selection of SSCs important to license renewal and their constituent structures and components for which age-related degradation should be assessed and accounted for, 3) evaluation of design, operational, and environmental factors that contribute to age-related degradation, 4) identification of the aging mechanisms and specific sites involved in degradation processes, and 5) attributes of established effective programs and of acceptable actions taken or to be taken to understand and manage age-related degradation. Detailed guidance on understanding and managing aging that will be useful to a license renewal applicant in implementing these methods is contained in Appendix A to this regulatory guide.

Implementation of the guidance provided herein is expected to ensure that actions have been identified and have been taken or will be taken with respect to age-related degradation of those SSCs important to license renewal, such that there is reasonable assurance that the activities authorized by the renewed license can be conducted in accordance with the current licensing basis.

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Applicability

This regulatory guide applies to applications for renewal of operating licenses for commercial NPPs and to the constituent SSCs of these facilities that are designated important to license renewal as defined in 10 CFR 54.3(a).

Any information collection activities mentioned in this draft regulatory guide are contained as requirements in those sections of 10 CFR Part 54 that provide the regulatory basis for this guide. The proposed additions to 10 CFR 54 have been submitted to the Office of Management and Budget for clearance, as appropriate, under the Paperwork Reduction Act. Such clearance, if obtained, would also apply to any information collection activities mentioned in this guide.

B. DISCUSSION

The technical information developed by a license renewal applicant should allow the NRC staff to make the determination that the requirements of 10 CFR 54 are met. The format in which this information is presented should satisfy the requirements of 10 CFR 54 and should facilitate the NRC staff review of a license renewal application. Technical information submitted by an applicant should focus on aging mechanisms and the resultant age-related degradation that can lead, in the context of the renewed license term, to unacceptable deterioration of SSCs important to license renewal. The technical information content of a license renewal application should be sufficient to provide an NRC reviewer of the application with a sound understanding of the aging processes that contribute to degradation of SSCs important to license renewal and how these processes are or will be managed. The information needed to impart this understanding should address the integrated effects of materials, design, environment, stressors, and plant operating history on SSC degradation attributable to specific aging mechanisms. These effects are discussed in Appendix A to this regulatory guide.

Use of Standard Format

Conformance with the standard format described in Chapter C, Regulatory Position, is not required. License renewal applications with different formats will be acceptable to the staff if they provide an adequate basis for the findings requisite to the issuance of a renewed license. However, because it may be more difficult to locate needed information, the staff review time for such applications may be longer, and there is a greater likelihood that the staff may regard the application as incomplete.

Upon receipt of a license renewal application, the NRC staff will perform a preliminary review to determine if the application provides a reasonably complete presentation of the information needed for issuance of a license in accordance with 10 CFR 54.29. The purpose of this review will be to determine if the submittal is sufficient according to the provisions of 10 CFR 2.109(b). The standard format will be used by the staff as a guideline to identify the type of information needed unless there is good reason for not doing so. If the application does not provide a reasonably complete presentation of the necessary information, further review of the application will not be initiated and the provisions of 10 CFR §2.109(b) will not apply until a reasonably complete presentation is provided. The information provided in the application should be current with respect to the state of technology concerning age-related degradation in operating nuclear power plants and should take into account, as appropriate, recent changes in regulations and in industry codes and standards; results of recent developments in nuclear reactor safety; and experience in the construction and operation of nuclear power plants.

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C. REGULATORY POSITION

The methods described in this section are acceptable to the NRC staff for satisfying the requirements proposed in 10 CFR 54 and 10 CFR 2.109 pertaining to the technical information to be compiled or submitted in support of an application for an NPP operating license renewal. This information will be reviewed by the NRC staff to assess the effectiveness of an applicant's established or proposed programs for understanding and managing age-related degradation of SSCs important to license renewal during a renewed license term and to evaluate acceptability of the application for a renewed license.

1.0 FORMAT FOR TECHNICAL INFORMATION

An application for license renewal must meet the applicable provisions of 10 CFR 54. Provisions contained in 10 CFR 54.23 deal with environmental information to be submitted. Environmental issues are addressed in Regulatory Guide 4.2.

The license renewal application is to contain two distinct parts: a formal application and a supplement to the current FSAR. Regulatory Position 1.1 describes the basic information that should be included in the formal application. Regulatory Position 1.2 describes the information that should be included in the FSAR supplement, which will be an attachment to the formal application. The FSAR supplement will consist of a new chapter added to the current FSAR for the sole purpose of license renewal. This new FSAR chapter will contain the detailed technical information to be included as part of the application. As described in 10 CFR 54.17(e), the application may incorporate, by reference, information contained in previous submittals provided such references are clear and specific.

1.1 FORMAL APPLICATION

The formal application is to contain the following elements:

1. an introduction providing general information concerning the application. This should include:

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a) information specified in 10 CFR 50.33(a) through (e), (h), and (i) of Part 50,

b) earliest effective date and length of the renewal term,

c) a statement summarizing how and the extent to which the application meets the regulatory requirements for license renewal (10 CFR 54). Exceptions should be listed and justified.

d) description of scope and organization of remaining sections of the application,

e) information specified in 10 CFR 54.17(g), and

f) an acknowledgment that the commitments and requirements contained in the licensee's current Part 50 license not affected or superceded by the license renewal application will remain in effect when the Part 54 license is issued.

2. a characterization and summation of the licensee's findings

This section should provide the justification, in summary form, to support the conclusion that appropriate actions have been, or will be, taken to manage the effects of age-related degradation of the facility systems, structures, and components important to license renewal. Details supporting these findings are to be contained in the FSAR supplement.

3. an implementation plan that includes the following elements:

a) Summary of Commitments

List the commitments described in the license renewal application.

b) Description of Administrative Controls

Describe the administrative control program used by the licensee to establish and maintain the commitments described above. Such a program should ensure that changes to such commitments are evaluated for aging considerations prior to revision and conform to the requirements for an established effective program contained in 10 CFR 54.3(a).

c) Task and Schedule

Detail the commitments that will be completed following renewal of the operating license. These commitments may

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include system design changes, one-time replacements, program enhancements, and new programs.

A schedule should be established and provided for the specific actions committed to in this section, and this schedule should be consistent with the evaluations made in the license renewal application.

4. The formal application section should also include submittals of any Technical Specification and program changes and additions identified as necessary to manage age-related degradation.

5. A list of changes in calculations or analytical approaches resulting from licensing renewal activities.

1.2 FSAR SUPPLEMENT

The detailed technical information for a license renewal application is to be contained in a supplement to the current FSAR in accordance with 10 CFR 54.21. This supplement will consist of a new FSAR chapter that conforms to the administrative requirements for FSAR chapters. This new chapter should be cross-referenced, as necessary, to other chapters in the FSAR.

The supplemental FSAR chapter should contain sections that describe the licensee's methodology and results for satisfying each element of 10 CFR 54.21. The NRC staff will review the licensee's application on a systems basis according to review procedures set forth in a new Standard Review Plan chapter (SRP-LR) that deals with review of applications for renewed licenses. It is expected that the technical information compiled or submitted by a licensee in compliance with 10 CFR 54.21 will conform in format and content to the new SRP chapter. This expectation is based upon the requirements stated in 10 CFR 54.21 to assure that a facility's licensing basis will be maintained throughout the term of the renewed license and upon the definition of current licensing basis as stated in 10 CFR 54.3(a). Maintenance of the current licensing basis, as defined in 10 CFR 54.3(a), requires a licensee to comply, inter alia, with 10 CFR 50. This includes the requirement cited in 10 CFR 50.34(g) that NPP operating licenses docketed after May 17, 1982, include an evaluation of the facility against the SRP revision in effect six months prior to the docket date of the application.

Table I outlines the information that should be included in the FSAR supplement. This information is structured to conform to the process that will be followed by the NRC staff in reviewing applications for license renewal. As indicated in Table I, the FSAR supplement should contain three categories of information:

<u>Part A: General Information and Discussion</u>
Information of an introductory or general nature such as purpose, scope, definitions, organization and relationship to 10 CFR 54, conformance to regulatory guides, citations for referenced information, general technical information required by 10 CFR 54 and described in the following subsection 1.2.2, and a description of any deviations from the acceptance criteria contained in the SRP-LR.

<u>Part B: Systems</u>
Information specific to the principal systems and subsystems that comprise an NPP and that may contain structures or components important to license renewal.

<u>Part C: Generic Components</u>
Information related to structures and components important to license renewal for which age-related degradation may be generically addressed.

All of the items included in the lists of SSCs in Table I are not germane to all NPPs, and the lists may not include all SCCs important to license renewal for any particular NPP.

1.2.1 <u>Review Objectives</u>

The licensee's organization of the FSAR supplement should allow the NRC staff to reach the following conclusions:

1. Sufficient technical information has been submitted as part of the application to commence a review.

2. The licensee's screening methodologies and resulting list of SSCs important to license renewal and the lists of their constituent structures and components (SCs) are acceptable.

3. The licensee's methodologies for identifying established effective programs [10 CFR 54.3(a)] for SCs requiring evaluation of age-related degradation.

4. The licensee's established programs effectively manage age-related degradation of SCs important to license renewal.

5. The licensee actions to manage age-related degradation in SCs important to license renewal not currently subject to established effective programs are adequate.

1.2.2 Supporting Documentation

The FSAR supplement should provide the facility-specific technical information needed by the NRC staff to make the judgments cited above. The licensee should submit this information to meet the requirements stated in 10 CFR 54.21. Specifically, the FSAR supplement should contain:

1. a detailed description of the licensee's integrated plant assessment including the screening methodology for identifying SSCs important to license renewal and the methodology for determining if an established program is effective in managing age-related degradation.

2. information required by 10 CFR 54.21, specifically:

   a) a list of SSCs important to license renewal as defined in 10 CFR 54.3(a)

   b) a list of all SCs that are constituent elements of the SSCs listed in 2(a)

   c) a list of all SCs from 2(b) that require evaluation of age-related degradation

   d) justification for conclusions that any SCs listed in 2(b) but not listed in 2(c) do not contribute to the performance of a safety function of an SSC important to license renewal or that their failure would not prevent an SSC important to license renewal from performing its intended safety function

   e) a list of SCs for which age-related degradation is not significant with respect to the current licensing basis as defined in 10 CFR 54.3(a) through the renewed license period and documentation of the evaluations that support these findings

   f) a list of all SCs that are subject to an established effective program as defined in 10 CFR 54.3(a), the associated effective program(s) and the basis for continuing them through the renewed license period

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g) a description of, and the basis for, actions taken or to be taken to manage age-related degradation including, where appropriate, a description of revisions to replacement/refurbishment programs to demonstrate their adequacy

h) a list of all plant-specific exemptions granted pursuant to 10 CFR 50.12 and reliefs granted pursuant to 10 CFR 50.55(a)(3). This list must include an identification of those reliefs and exemptions granted on the basis of an assumed service life or period of operation bound by the original license term of the facility or otherwise related to SSCs subject to age-related degradation. Also, for reliefs and exemptions granted on the basis of an assumed service life or period of operation bounded by the original license term of the facility or otherwise related to SSCs subject to age-related degradation, justification for their continuation should be provided in either this section or the system-specific section below

i) a list and description of any proposed modifications to the facility or its administrative control procedures resulting from the integrated plant assessment [10 CFR 54.21(a)] or exemptions and reliefs described in 2(h). [10 CFR 54.21(b)]

3. information specific to the systems listed in Part B of Table I and the generic structures and components listed in Part C of Table I. Information related to components that can be grouped in terms of component type and expected age-related degradation may be cited once in generic form and that citation referenced for subsequent components that fit the appropriate grouping. For each system (Part B of Table I) or generic structure or component (Part C of Table I) applicable to the facility, the following information should be presented:

a) a SSC-specific list of constituent SCs that are important to license renewal. A reference to the lists provided in 2(a) and 2(b) will suffice, providing the lists in 2(a) and 2(b) clearly identify all SCs associated with that particular SSC.

b) identification of age-related degradation sites, site-specific aging mechanisms, and root causes, when practicable, for SCs listed in 2(c)

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c) for SCs important to license renewal, a summary discussion of the evaluation of key properties and parameters that may change with time and are affected by NPP operational and service conditions. The initial values at the start of operating life of these properties and parameters (such as fatigue cycle life, cable insulation dielectric strength fracture toughness, tensile strength, and pressure boundary wall thickness) as established by analyses or qualifications should be included, along with results of evaluations of past operating environments and service conditions to determine the rates of change experienced and residual values for these properties and parameters. This summary should also include a discussion of changes to analyses resulting from age-related degradation evaluations. These values should be used in trending and analyses to establish predicted, extended operating lives and to identify actions needed to maintain key properties and parameters within acceptable limits during the renewal term. (See Appendix A of this regulatory guide for further details.)

d) an identification of the structures and components from 3(a) that are subject to established effective programs including a technical justification of how these programs effectively manage age-related degradation. Reference may be made to the list described in 2(f), providing this list clearly identifies all components and structures associated with a particular system that are subject to established effective programs. Descriptions of established effective programs either should be provided as part of the license renewal FSAR or incorporated by reference from the most recent update of the facility FSAR or any other material referenced in the facility's docket (such as the FSAR, Quality Assurance Manual, Inservice Inspection and Testing Programs, and training programs). The program description should not be a general description of the overall program but should be specific and justify why the program is effective in managing age-related degradation as noted in 3(b) and 3(c) and described in Appendix A.

e) a description of actions taken or to be taken to manage age-related degradation in SSCs important to license renewal not currently subject to established effective programs and how programs resulting from these actions will be implemented and maintained effectively from an age-related degradation perspective. For structures and components applicable to the system under consideration and included in list 2(c) but not included in list 2(f), this description should contain proposed revisions to maintenance or other program elements, including administrative controls, that will be implemented and controlled throughout the renewal period to manage age-related degradation in these components. Alternatively, technical evaluations in the facility docket that provide adequate assurance that the SSCs will not degrade below acceptable levels of safety during the renewal term may be provided or referenced.

f) a summary of all current exemptions granted pursuant to 10 CFR 50.12 and reliefs granted pursuant to 10 CFR 50.55a(a)(3). For exemptions or reliefs that were granted based on an assumed service life or period of operation bounded by the original license term of the facility, a justification for continuing these exemptions and reliefs must be provided. A reference to the list provided in 2(h) will suffice, providing 2(h) clearly identifies all current exemptions and reliefs associated with each system. Justification for continuation can either be supplied in each system section or with the list in 2(h).

g) a description of provisions to be taken with respect to any proposed modifications to the facility or its administrative control procedures resulting from the integrated plant assessment [10 CFR 54.21(a)] or exemptions and reliefs described in 3(f). A reference to the list provided in 2(i) will suffice, providing 2(i) clearly identifies all proposed modifications associated with each system.

h) for existing and new programs identified as necessary for managing age-related degradation, a description of how these programs either are or will be implemented and controlled to ensure that their effectiveness is not degraded throughout the renewal term.

i) a description of the methods to be employed in obtaining and maintaining records of the documentation described in this section or to be generated in the course of performing activities described in this section. This should include identification of which records are to be kept, in what form, and over what period of time. Records that permit verification that all SSCs that are important to license renewal meet their specific performance requirements should be retained in an auditable and retrievable form for the renewal term plus whatever additional period is required in accordance with the licensing basis.

Additions or other changes to the Technical Specifications may be necessary to account for modifications in the plant design, age-related degradation, or

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limitations on plant operations during the renewal term. If such changes are deemed necessary, the license renewal application (1.1 Formal Application) should specifically request such changes and should provide the technical justification for the changes. Such changes should be limited to those necessary to address effects of age-related degradation.

Items 3(a) through 3(i) specify information that should be included in the FSAR supplement for each SSC important to license renewal. Additional information that should be submitted as part of a license renewal application or compiled and retained by the licensee is described in Table II. Some of the information items listed in Table II should be included in the FSAR supplement [information described in (1), (2), and (3) above]. The remainder of the information specified in Table II should be submitted as part of the application but independent of the FSAR supplement or retained by the licensee as appropriate.

2.0 TECHNICAL INFORMATION CONTENT

As required by 10 CFR 54.21, "The FSAR supplement (that presents the technical information requirement for license renewal) must include an evaluation of the aging mechanisms that are present and that result in the degradation of the plant's systems, structures, and components and a demonstration that the effects of such degradation will be effectively managed throughout the renewal term." To meet these requirements, such an evaluation must cover those SSCs that are important to license renewal and must be based upon principles of understanding and managing age-related degradation. The following subsections (2.1 and 2.2) provide guidance on selection criteria for SSCs important to license renewal and on programs and practices for understanding and managing aging that are acceptable to the NRC staff as the bases for the "evaluation" and "demonstration" required by 10 CFR 54.21 as noted above.

2.1 TYPES OF SSCs FOR WHICH AGING SHOULD BE CONSIDERED IN SUPPORT OF LICENSE RENEWAL

The process for identifying SSCs important to licensing renewal and their constituent SCs requiring evaluation of age-related degradation is illustrated

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in Figures 1A and 1B. These illustrations also identify some of the information to be collected by the licensee and either submitted as part of a License Renewal Application or retained in auditable, retrievable form for the term of the renewed operating license. Figure 1B defines approaches to information collection and evaluation for a licensee to demonstrate an acceptable level of understanding of age-related degradation in those SCs identified through the selection process illustrated in Figure 1A. Acceptable management of age-related degradation for important-to-license-renewal SCs requires SC-specific evaluations of the effectiveness of existing programs and practices for timely mitigation of age-related degradation. Figure 1B also delineates a process for identifying SC-specific deficiencies in existing programs for addressing age-related degradation and for specifying actions that will be pursued in addressing these deficiencies in support of a license renewal application.

Generic functional NPP SSCs that are composed, wholly or partially, of SSCs important to license renewal are listed in Table III. The information in Table III is included in this regulatory guide as a generic supplement to Table I. This information provides guidance to the licensee for compiling the plant-specific list of SSCs important to license renewal and is not intended to be all-inclusive. Appendix B provides a detailed system and component hierarchy for safety categories representative of a typical pressurized water reactor (PWR) and a typical boiling water reactor (BWR). The representative systems, structures, and components listed in Appendix B are referenced to corresponding Generic Functional NPP Elements (Table III), to corresponding sections of the Standard Review Plan (NUREG 0800), and to the Standard Technical Specifications. For NPPs to which these documents apply, they provide more detailed information on system functions, configurations, limitations, testing needs, habitability limits for personnel, and safe environmental limits for vital equipment.

2.2 INTEGRATED PLANT ASSESSMENT - PROGRAMS FOR ADDRESSING AGE-RELATED DEGRADATION[a]

Elements of the integrated plant assessment for the evaluation of age-related degradation should be based upon sound principles and practices for understanding and managing aging. SSC-specific understanding of aging mechanisms and the degradation sites at which they are operative is useful to evaluate the effectiveness of existing programs and replacement/refurbishment practices for managing aging or to develop acceptable new programs or practices. An established effective program should include, but is not limited to, inspection, surveillance, maintenance, trending, recordkeeping, replacement/refurbishment, and the assessment of operational life for the purpose of timely mitigation of the effects of age-related degradation. The prime criteria for such a program are that it be documented and that it ensures that SSCs important to license renewal will continue to perform adequately, ensuring that the current licensing basis will be maintained during the renewal period. In addition, an established effective program will 1) be clearly defined and documented in the FSAR, 2) be approved by onsite review committees, 3) be implemented by the facility operating procedures, 4) establish documented acceptance criteria against which the need for corrective action is evaluated to ensure that age-related degradation will not directly or indirectly prevent SSCs from performing their intended functions, and 5) ensure that corrective action is taken when applicable acceptance criteria are not met. Programs for understanding and managing aging should be implemented and maintained through a system of specific administrative controls that ensures continuing program effectiveness throughout the term of a renewed license.

Requirements contained in 10 CFR 54.21 provide ways for a licensee to demonstrate the adequacy of the plant program for addressing age-related degradation in structures and components important to license renewal. These are 1) by substantiating that established programs, i.e., ongoing programs that are currently in place, are effective, i.e., meet the criteria cited in the

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(a) For expanded discussions, see Appendix A to this regulatory guide and its associated bibliography.

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preceding paragraph [10 CFR 54.21(a)(3)]; 2) by taking actions or committing to actions to manage age-related degradation not adequately addressed by established programs [10 CFR 54.21(a)(4)]; or 3) by demonstrating that age-related degradation is not significant with respect to the current licensing basis [10 CFR 54.21(a)(4)]. Adequacy of the program for addressing age-related degradation must be demonstrated for each structure, component, or group of similar components important to license renewal using one of the aforementioned methods. Both existing programs and new actions taken or to be taken to manage age-related degradation are subject to the same effectiveness criteria. Criteria that relate to program structure and administration are cited in the preceding paragraph. Criteria that relate to established effective programs and to new actions taken or to be taken to manage aging are discussed in subsections 2.2.1 and 2.2.2, respectively. Subsection 2.2.3 relates to the exclusion of SCs not subject to significant age-related degradation during the renewed licensing term, and technical criteria for practices employed in understanding and managing aging are described in subsection 2.2.4 and 2.2.5, respectively.

2.2.1 Established Effective Programs

Established effective programs for managing age-related degradation are defined in 10 CFR 54.3(a). The structural and administrative criteria provided in that definition plus the technical criteria represented in Subsections 2.2.4 and 2.2.5 should be applied by a licensee in evaluating the current plant program for managing aging in structures and components important to license renewal. The methodology employed in performing these evaluations should be described, and results of the evaluations should be provided for all important-to-license-renewal structures and components that are subject to established programs. Two essential products of these evaluations will be identification of those important-to-license-renewal structures and components that are routinely replaced/refurbished at defined intervals and SCs for which established programs do not effectively address age-related degradation.

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2.2.2 Actions to be Taken to Manage Age-Related Degradation

It is expected that some important-to-license-renewal structures and components with potentially significant age-related degradation will not be subject to programs that address that degradation and that others will be subject to programs that are not fully effective. Such structures and components should be identified and the bases for actions that have been taken or will be taken to manage age-related degradation in these structures and components should be described by the licensee. These actions and judgements concerning their adequacy should be based upon the same criteria cited in Subsection 2.2.1 for established effective programs.

2.2.3 Structures and Components Not Subject to Significant Age-Related Degradation

For some structures and components important to license renewal, age-related degradation may not lead to significant reduction in the capacity of the structure or component to perform its safety functions, i.e., functions defined as important-to-license-renewal. If a licensee explicitly demonstrates that age-related degradation of an important-to-license-renewal structure or component will not compromise the current licensing basis during the license renewal term, that structure or component may be excluded from further consideration of age-related degradation.

2.2.4 Summary of Acceptable Programs and Practices for Understanding Aging

Programs to understand age-related degradation processes in SCs important to license renewal should be implemented on a plant-specific basis by qualified licensee staff using state-of-the-art knowledge of age-related degradation in NPPs. Efforts to understand aging mechanisms and degradation should be systematically structured, as illustrated in Figure 1B. For some SCs, e.g., reactor pressure vessel shells, it will be necessary to evaluate design, materials, and environmental and operational stressors and their interactions. Analysis of these factors will lead to identification of potential aging mechanisms, degradation sites, and root causes when practicable. This information is the basis for developing and implementing programs for monitoring and timely mitigation of age-related degradation. Not all important-to-license-renewal SCs require in-depth evaluation of the basic factors that contribute to age-related degradation. For some SCs, reference to empirical information such as operational records, manufacturers' information, test data, and ongoing regulatory requirements will be sufficient. To the extent practicable, equipment designers and manufacturers should be requested to identify aging mechanisms, rates, and any other pertinent information that they may possess. For other SCs, descriptions of surveillance or replacement programs will suffice.

The process of developing an understanding of age-related degradation involves integrating the relevant materials science concepts that describe degradation processes with plant-, SSC-, and SC-specific design and operational information to understand aging mechanisms, degradation sites and rates, and the consequences of degradation with respect to NPP safety. The individual and interactive influences of SC design, constituent materials, and both normal and abnormal stressors and environments establish the feasibility and determine the reaction rates for degradation mechanisms that can affect SSCs important to license renewal. An effective program to understand aging will selectively integrate a sound understanding of these basic principles with plant-specific design, operational experience, manufacturers' and design information, research and test data, applicable regulatory instruments and requirements, and qualified technical judgment to characterize age-related degradation processes that are operative in important-to-license-renewal SCs. These characterizations should be expressed in terms of specific degradation sites, site-specific aging mechanisms, root causes for degradation when practicable, and projected effects of degradation on SSC functions.

2.2.5 Summary of Acceptable Programs and Replacement/Refurbishment Practices for Managing Aging

Acceptable practices for managing aging in SCs for which no defined replacement/refurbishment programs exist may employ combined mechanistic and empirical approaches for understanding aging mechanisms and identifying degradation sites and root causes, where practicable. This information may form the basis for inspection, surveillance, condition monitoring, test

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methods and frequencies, and residual lifetime evaluations that will dictate timely and effective preventive and corrective maintenance methods and frequencies, and associated recordkeeping needs during the renewed license period.

Monitoring methods (e.g., inspection, surveillance, testing, condition monitoring) should reflect mechanistic and empirical assessments performed by licensee staff in their efforts to understand and mitigate age-related degradation. These methods should employ state-of-the-art non-destructive examination, e.g., ultrasonic testing, signature analysis, vibration analysis, dielectric performance measurements, and other measuring techniques, performed by qualified staff. Measurement results should be trended and analyzed with respect to implications for residual lifetime of SSCs important to license renewal.

Practices for SCs that are routinely replaced/refurbished at defined intervals should be evaluated for the adequacy of the replacement/refurbishment programs to ensure timely mitigation of age-related degradation during the renewed license period. The evaluation process should include reviews of the operational experience and, as appropriate, design and manufacturers' information, known aging mechanisms, and other available information.

The objective of aging management in support of license renewal should be to ensure that SSCs important to license renewal are subject to surveillance and maintenance that control, at intervals commensurate with expected component lifetimes, processes that could degrade their operability and reliability.

The maintenance program is the principal vehicle through which age-related degradation is managed. Operational and maintenance records and input from monitoring programs should be employed in the maintenance program for scoping and scheduling both preventive and corrective maintenance activities intended to manage age-related degradation. These activities should be carried out by experienced, qualified maintenance personnel and should lead to needed servicing, repair, refurbishment, or replacement with a frequency sufficient to maintain acceptable levels of SSC reliability. In evaluating the effectiveness of maintenance in managing aging, the maintenance and surveillance intervals should be considered along with 1) the probability of defect detection and diagnosis and 2) the probability of effective defect correction given defect detection and correct diagnosis.

A critical first step in implementing an aging management program is to define those surveillance, maintenance, or other mitigative program elements to be implemented to deal with the degradation processes revealed by programmatic efforts to understand aging (subsection 2.2.4, above). These program elements may include inspections, tests using a wide variety of non-destructive examination (NDE) and other methods, general surveillances, condition monitoring, diagnostic and root-cause analysis, preventive maintenance, corrective maintenance, predictive maintenance, and reliability-centered maintenance. The primary goal should be to develop effective aging management practices implemented through replacement, refurbishment, and repair programs that accurately reflect SC-specific residual lifetimes and the safety significance of anticipated, SC-specific degradation. The effectiveness of aging management programs should be evaluated on SC-specific bases using guidance such as that contained in relevant codes and standards, approved industry technical reports, and other sources acceptable to the NRC staff. The process for evaluating effectiveness should reveal those deficiencies that require correction through improved or new programs for managing aging. An accurate assessment of current program effectiveness, based upon the principles of understanding aging summarized in subsection 2.2.4 and in Appendix A, and implementation of program enhancements to address revealed deficiencies in aging management practices are prerequisites to operating license renewal in NPPs.

An important aspect of plant surveillance and maintenance programs is retention of essential data in complete, auditable, easily retrievable records. The records system and its contents should conform to good maintenance practices as well as the requirements of 10 CFR 50 Appendix B and the plant QA program to the extent that these documents apply to SSCs important to license renewal. A record of the documentation required by, or otherwise necessary to document compliance with the provisions of, 10 CFR 54 and a record of the administrative processes for controlling changes to such documents should be retained by the licensee in an auditable and retrievable form for the term of

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the renewed operating license. This should include a listing of and the justification for those structures, systems, and components important to license renewal and included in established effective programs as defined in 10 CFR 54.3(a) or subject to actions taken or to be taken. Records and related data should be employed to the extent practicable in trending degradation processes, thereby providing assurance of controlled, timely maintenance. Trends based upon data contained in the maintenance records should be used to monitor the effectiveness of aging management in selected SCs.

D. IMPLEMENTATION

This regulatory guide applies to all applications for renewal of operating licenses for commercial NPPs. Alternative methods of implementation may be proposed.

REGULATORY ANALYSIS

The regulatory analysis prepared in support of 10 CFR 54, NUREG 1362, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," provides the value/impact justification for this regulatory guide.

BACKFIT ANALYSIS

This regulatory guide presents for the first time NRC staff guidance on complying with a new rule, 10 CFR 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants." Accordingly, publication of this regulatory guide is not a backfit under 10 CFR 50.109, and no backfit analysis is necessary or has been prepared for this regulatory guide.

FIGURE 1.

Process for Selecting Systems, Structures, and Components Important to License Renewal and for Understanding and Managing Age-Related Degradation

INTRODUCTION TO FIGURES 1A AND 1B

Figures 1A and 1B constitute a flow chart that outlines a process, acceptable to the NRC staff for license renewal purposes, for selecting individual structures, and components (SCs) that are constituent elements of systems, structures, and components (SSCs) important to license renewal, and for developing and implementing programs for understanding and managing age-related degradation in these SCs during the renewed license term.

Figure 1A portrays the process for SSC and SC selection. Input to this process is defined by the four types of SSCs included in the definition of "important to license renewal" [10 CFR 54.3(a)]. Each of the input SSCs is subdivided into individual structures and components that are then screened [using the criteria contained in 10 CFR 54.21(a)(2)] to yield those SCs that require evaluation of age-related degradation (block 11, Figure 1A). This collection of SCs constitutes the input to the continuation of the process of evaluating are-related degradation as part of the Integrated Plant Assessment as shown in Figure 1B.

Two methods are described that provide guidance for determining the scope and depth of analysis necessary to define age-related degradation mechanisms and evaluate the adequacy of aging management programs. The first method may be applicable when evaluating SCs that are replaced or refurbished routinely at defined intervals. These SCs include items such as batteries, relays, and selected snubbers. The second method may be applicable when evaluating SCs that are not routinely replaced or refurbished. Such SCs typically were designed for and expected to be in place for the original 40 years of plant operation. These "long-lived" SCs include items such as the reactor coolant system, large-diameter piping, the reactor pressure vessel, steam generators, and cables.

SCs that are routinely replaced or refurbished may be evaluated based on a review of the adequacy of the replacement or refurbishment program(s) for timely mitigation of age-related degradation during the renewed license period. An acceptable approach would be to evaluate operational experience, replacement or refurbishment intervals, and, as appropriate, relevant design and manufacturers information, known aging mechanisms, and other available information. Based on this review and a conclusion that the SC will remain functional during the defined interval between replacement or refurbishment, an applicant may establish that the current program is adequate. When the ongoing replacement program is demonstrated to be adequate for timely mitigation of age-related degradation and if it is in the FSAR, approved by the onsite review committee, and implemented by the facility operating procedure, it is considered to be an "established effective program."

When the ongoing replacement or refurbishment program(s) are not adequate for timely mitigation of age-related degradation, the licensee should describe revisions to the replacement or refurbishment program and demonstrate its adequacy or perform detailed mechanistic analyses. The bases should be provided for actions taken or to be taken to manage age-related degradation during the renewed license period.

SCs that are not routinely replaced may be evaluated based on a detailed mechanistic analysis of age-related degradation mechanisms. When the analysis indicates that age-related degradation is not significant with respect to the current licensing basis throughout the renewed license term, the result of the analysis should be documented. For those SCs potentially susceptible to significant age-related degradation mechanisms, evaluations should be made to determine if they are subject to an established affective program. A list of SCs identified as being subject to an established effective program should be provided as well as descriptions of the programs and the basis for continuing them through the renewed license period. For those SCs that are not subject to an established effective program, the basis for actions taken or to be taken to manage age-related degradation during the renewed license period should be described and provided.

Throughout Figures 1A and 1B, individual activities and decision points have been referenced as appropriate to the specific parts and subparts of 10 CFR 54 that provide their justification. These identifications are denoted by references to the License Renewal Rule [54.XX(X)(X)] contained within appropriate nodes in the Figures.

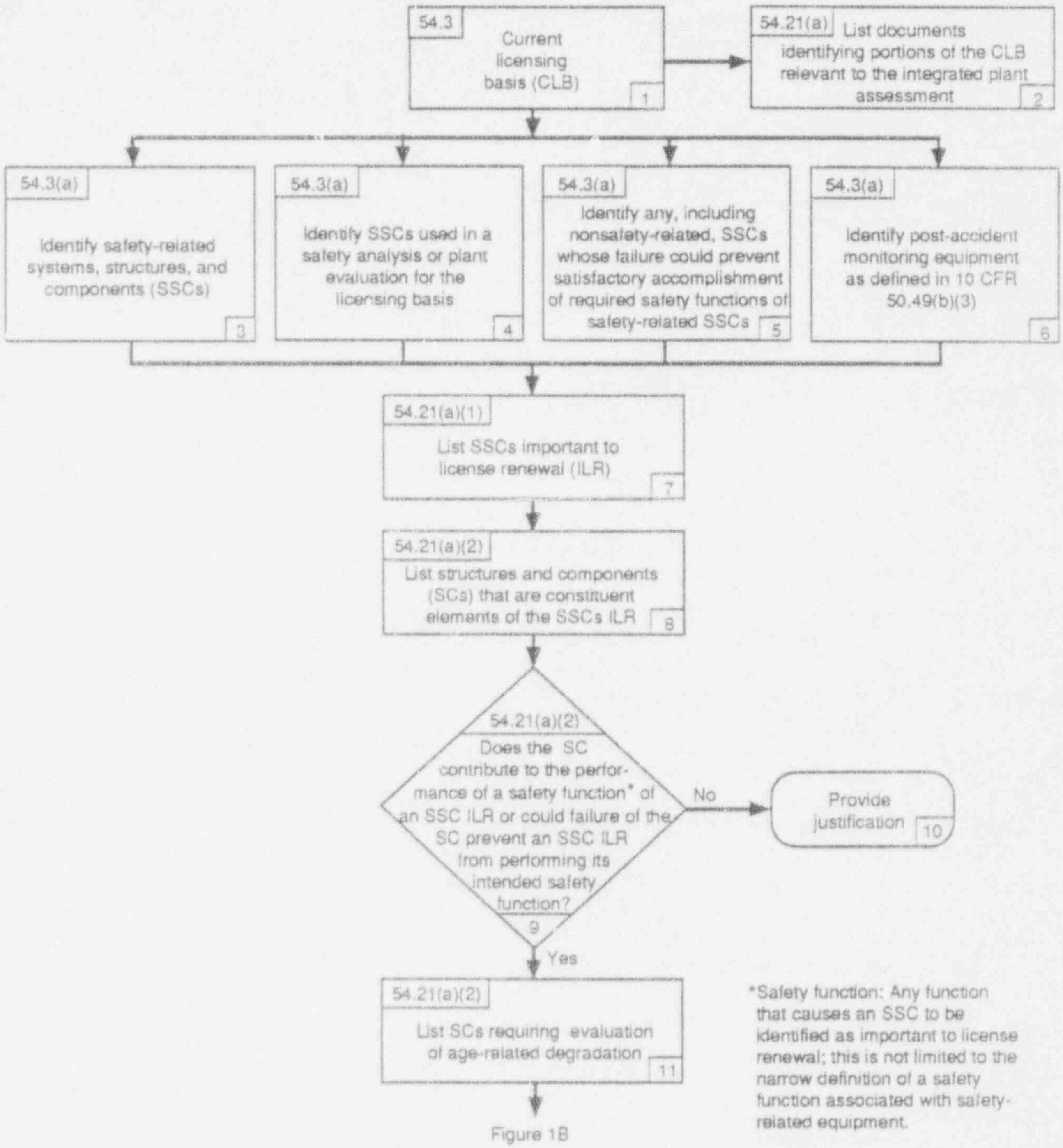

FIGURE 1A

Integrated Plant Assessment -- Identification of Important-To-License-Renewal SSCs and SCs Requiring Evaluation of Age-Related Degradation

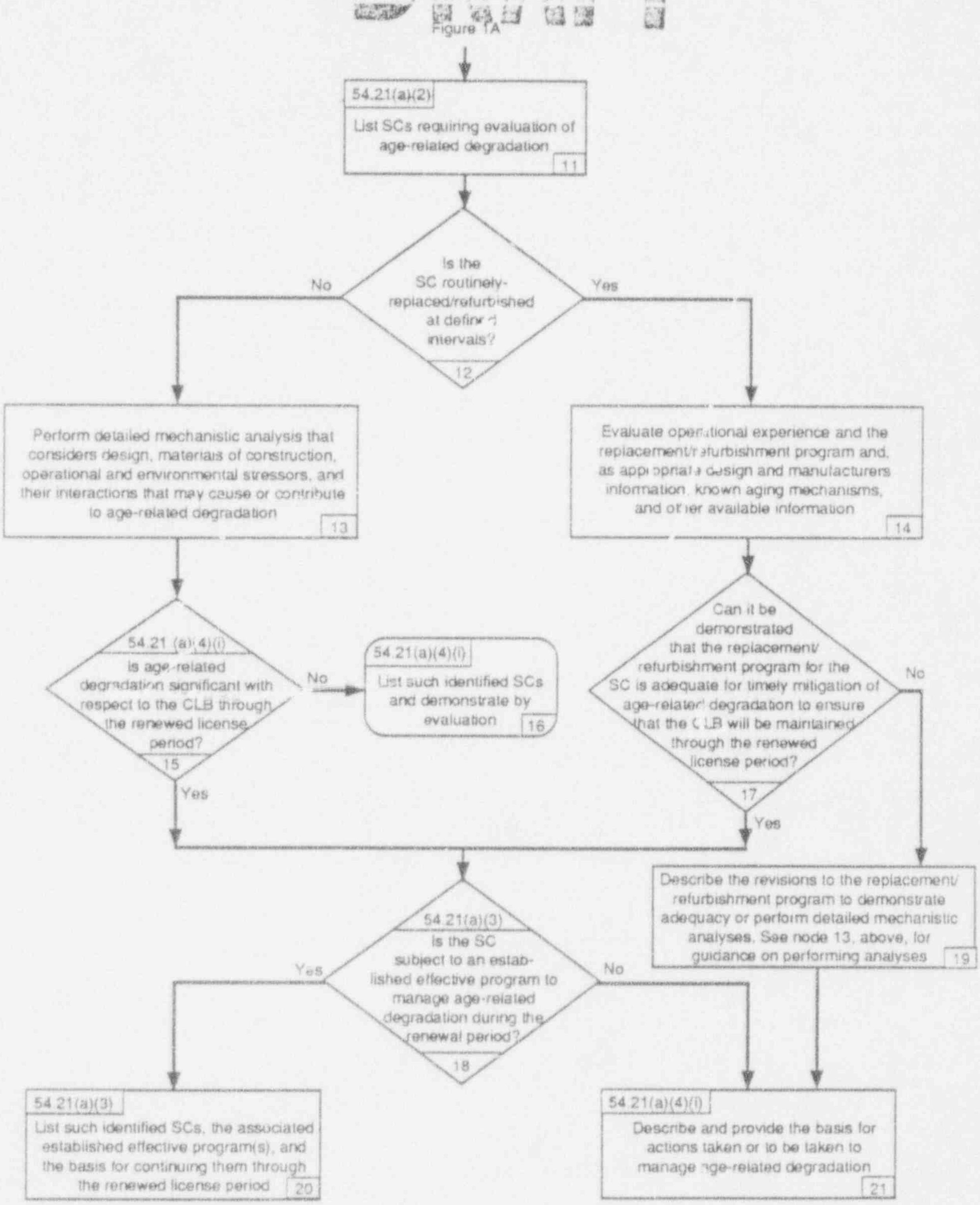

FIGURE 1B
Integrated Plant Assessment -- Evaluation of Age-Related Degradation

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Sources of Information and Instructions for Implementing Processes Described in Figures 1A and 1B.

1	Compile and maintain all documents describing the current licensing basis (CLB) in an auditable and retrievable form as per 10CFR54.21(a).
2	Submit a list of documents identifying portions of the CLB relevant to the integrated plant assessment (IPA) as per 10CFR54.21(a).
3, 4, 5, 6	These four categories are defined by 10CFR54.3(a) as being important to license renewal. Assignment of systems, structures, and components (SSCs) to the appropriate categories should be accomplished by starting with all plant SSCs and distributing these among the four categories by applying guidelines such as those contained in the sources referenced in the following four notes. SSCs that fit none of the four categories are not important to license renewal and require no further consideration for license renewal purposes.
3	Q List, Final Safety Analysis Report (FSAR), Quality Assurance Program and all other elements of the Current Licensing Basis as defined in 10CFR 54. SSCs used in a safety analysis or plant evaluation for the Licensing Basis include, but are not limited to, SSCs identified in the FSAR, the Technical Specifications, Plant Operations Manuals, Piping and Instrumentation Diagrams (P&IDs),Table III of this Regulatory Guide (Generic Functional SSCs Important to License Renewal), and evaluations submitted to show compliance with the Commission's regulations such as Anticipated Transients Without Scram (ATWS), Station Blackout, Fire Protection, Pressured Thermal Shock (PTS), and Environmental Qualification.
4	Current Licensing Basis.
5	FSAR, P&IDs, as-built current drawings, plant modification records, master equipment list, plant configuration control system data.
6	10CFR50.49 and associated regulatory guides, (e.g. R.G.1.89, R.G.1.97), FSAR.
7	Describe methodology for identifying SSCs important to license renewal, as defined in 10CFR54.3(a), and submit a list of the identified SSCs as part of the Integrated Plant Assessment (IPA).
8, 9, 10, 11	Describe methodology for identifying those structures and components (SCs) that are constituent elements of the SSCs important to license renewal and that require evaluation of age-related degradation.
11	Probabilistic risk assessment (PRA) techniques may also be used to supplement the deterministic approach shown in Figure 1A by adding additional SCs to the list of those requiring evaluation of age-related degradation.
12	Plant records and CLB.
13, 14	Input from the Nuclear Plant Aging Research (NPAR) Program; codes and standards; Non-Destructive Examination (NDE); industry studies and other programs related to effects of material variables, environment, and stressors on age-related degradation. Materials Characterization includes initial design, construction, installation plus changes introduced by modification, maintenance, or replacement. Stressors derive from both environmental and service conditions • Environmental Conditions include radiation, temperature, atmosphere, humidity, and chemical environment for all design basis events • Service Conditions include steady-state, cyclic, or other transient loadings imposed during normal operation, testing, or offnormal events. Mechanical and electrical loadings predominate.
15, 16	For those Structures and Components (SCs) for which it can be demonstrated that age-related degradation is not significant with respect to the CLB through the renewed license period, no further actions need be taken to manage their age-related degradation.
17, 18	Decisions to be based upon criteria developed by the licensee and subject to review by the NRC. Need for in-depth evaluation will reflect such considerations as safety significance, failure consequence, system and degradation process complexity, intensity or conservatism of current mitigation program, and other factors deemed important by the licensee and NRC for specific cases. Supporting information for these decisions will derive from various, plant-specific sources including operations/maintenance records, Technical Specifications, maintenance/surveillance procedures, Inservice Inspection Program (ASME Section XI), PTS analyses, preventive-predictive maintenance programs.
19, 20, 21	Programs for managing age-related degradation include inspection, testing, surveillance, condition monitoring, root cause analysis, predictive/preventive/corrective/ risk-based/reliability centered maintenance, record keeping and trending, other replacement/refurbishment/repair activities, residual life assessment, and responses to changes in operating and design parameters and environments. Relevant information can be found in operations/maintenance records, Technical Specifications, maintenance/surveillance procedures, Inservice Inspection Program (ASME Section XI), Pump and Valve Testing Programs, Environmental Qualification Programs, preventive-predictive maintenance program records, and the current body of regulatory requirements.
Note A:	Results of activities 2, 7, 8, 10, 11, 16, 19, 20, and 21 should be submitted with the license renewal application. These results plus results of activities 1, 3, 4, 5, 5, 13, 14, and 17 should be documented and retained by the licensee in auditable, retrievable form.

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TABLE I

Information To Be Included in the FSAR Supplement - General and SSC Specific

Part A: General Information and Discussion[a]
(Part A should include, but is not limited to, the following types of information)

1.0 Introduction

- 1.1 Purpose of the FSAR Supplement
- 1.2 Scope of the FSAR Supplement
- 1.3 Definition of Terms
- 1.4 Organization of the FSAR Supplement
- 1.5 Relationship of the FSAR Supplement to 10 CFR 54

2.0 Conformance to Applicable Regulatory Guides

3.0 Listing and Summary of Material Incorporated by Reference

4.0 Description of Integrated Plant Assessment (see Section C.1.2.2,1. of this Regulatory Guide)

5.0 Information Required by 10 CFR 54.21 (see Section C.1.2.2,2. of this Regulatory Guide)

6.0 Deviations From SRP-LR Acceptance Criteria

(a) An applicant for a renewed license will find it useful to consult the SRP-LR for descriptions of criteria and review procedures to be applied by the NRC to applications for license renewal.

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TABLE I (contd)

Part B: NPP Systems

(Numbers in brackets refer to corresponding chapters in the FSAR and in Regulatory Guide 1.70)

1.0 Nuclear Systems - the reactor core and those systems and subsystems that monitor and control the core's reactivity, remove heat from the core, and otherwise directly support the safe operation of the reactor.

1.1 (5,3) Reactor Pressure Vessel-(includes reactor core and internals)

1.2 (5,3) Reactor Coolant System-(includes piping, RCPs and Steam Generators)

1.3 (3) Reactor Control System

1.4 (3) Control Rod Drive System

1.5 (7,3) Reactor Protection System

1.6 (7,3) Nuclear Monitoring/Nuclear Instrumentation System

1.7 (5,6) Reactor Water Cleanup System (BWR)

1.8 (9,3,6) Standby Liquid Control System (BWR)

1.9 (3,5,6,9) Chemical and Volume Control System and Emergency Boration (PWR)

2.0 Engineered Safety Features - systems, other than containment systems, that are used to mitigate the effects of a reactor accident such as a LOCA.

2.1 (7,3) Engineered Safety Features Actuation System (PWR)

2.2 (3,5,6) Safety Injection Systems

2.2.1 Reactor Core Isolation Cooling (BWR)

2.2.2 High Pressure and intermediate pressure Safety Injection System (PWR)

2.2.3 Core Flood System (PWR)

2.2.4 RHR/Low Pressure Safety (Core) Injection (includes shutdown cooling function)

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TABLE I (contd)

2.2.5 Core Spray Systems (BWR)

2.2.6 High Pressure Coolant Injection (HPCI) System (BWR)

2.3 (3,6,10) Auxiliary Feedwater System (PWR)

2.4 (3,6) Automatic Depressurization System (BWR)

2.5 (7) Remote Shutdown System/Safe Shutdown Systems

3.0 Containment Systems - the containment (primary and secondary, as applicable) and those systems needed to prevent containment over-pressure, to prevent excessive leakage from the containment to the environment, and to provide a habitable atmosphere inside containment.

3.1 (3,6) Primary Containment Structure

3.2 (3,6) Secondary Containment

3.3 (6) Containment Heat Removal System

3.4 (6) Containment Isolation System

3.5 (6) Containment Purge System

3.6 (6) Standby Gas Treatment System (BWR)

3.7 (6) Containment Combustible Gas Control System

3.8 (6) Containment Spray System

3.9 (9) Containment Ventilation System

4.0 Electrical Systems - systems that supply electric power to the utility grid or other plant systems, or that are purely electric in nature.

4.1 (8) Main Power

4.1.1 Protective Relaying and Controls

4.2 (8) Plant AC Distribution System

4.2.1 Essential Power System

4.2.2 Nonessential Power System

4.2.3 HPCS Power System (BWR)

TABLE I (contd)

4.3 (8) Instrument and Control Power Systems

4.3.1 DC Power System

4.3.2 Instrument AC Power System

4.4 (8,9) Emergency Diesel Generators (EDG)

4.4.1 (8) EDG Instrumentation and Control Subsystem

4.4.2 (9) EDG Starting Subsystem

4.4.3 (9) EDG Cooling Subsystem

4.4.4 (9) EDG Fuel Oil Subsystem

4.4.5 (9) EDG Lubricating Oil Subsystem

4.5 (8,9) Plant Essential Lighting System

4.6 (7) Plant Computer

4.7 (8) Switchyard

4.7.1 DC Control Power System

4.8 (7) Information Systems Important to Safety

5.0 Process Auxiliary Systems - system and subsystems that support the plant systems directly involved in the process of safely producing electrical power.

5.1 (11) Offgas System (BWR)

5.2 (12) Radiation Monitoring System

5.3 (9) Component Cooling Water System

5.4 (9) Service Water System

5.5 (9) Ultimate Heat Sink

5.6 (9) Refueling System

5.7 (9) Spent Fuel Storage

5.8 (9) Compressed Air System

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TABLE I (contd)

6.0 Plant Auxiliary Systems - systems provided to support plant activities and personnel. They are typically nonsafety systems. Design of these systems varies greatly from plant-to-plant.

6.1 (9) Fire Protection System

6.2 (9) Communications

6.3 (6) Control Room Habitability System

6.4 (6) Auxiliary HVAC Systems

Part C: Generic Components

(These relate to various elements of the preceding subsection dealing with NPP Systems)

0.1 Generic Components and Structures Review Criteria

1.0 Mechanical

1.1 Piping

1.2 Valves

1.3 Pumps

1.4 Heat Exchangers

1.5 Tanks and Vessels

1.6 Equipment and Component Supports

2.0 Electrical

2.1 Cable and Wiring

2.2 Junctions

2.3 Electrical Penetrations

2.4 Relays, Circuit Breakers, and Switchgear

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TABLE I. (contd)

2.5 Transformers

2.6 Solenoid Operated Valves

2.7 Electric Motors

3.0 Instrumentation

3.1 Sensors

3.2 Electronic Components

3.3 Electronic Devices

4.0 Civil Structures

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TABLE II
Technical Information Needed for License Renewal (LR)

(Should demonstrate the current licensing basis for SSCs that are important to license renewal and that should be subject to established effective programs or subject to actions taken or to be taken to manage age-related degradation during the license renewal term.)

TECHNICAL INFORMATION TO BE GENERATED AND DOCUMENTED BY LICENSEE IN THE FORM OF AUDITABLE, RETRIEVABLE RECORDS	SUBMIT WITH LR APPLICATION? Y/N (Yes/No)
The principal vehicle for providing technical information in support of a license renewal application will be the FSAR supplement described in detail in Section C.1.2 of this regulatory guide. The FSAR supplement, which is to be submitted along with the Formal Application for License Renewal described in Section C.1.1, will contain or reference various compilations of technical information including, but not limited to, the following:	
1. The most recent update of the facility FSAR and any other manuals or program documents referenced in the FSAR, reports such as the Quality Assurance Manual, Emergency Response Plans, Inservice Inspection and Testing Programs, and training programs. These should be incorporated by reference in the License Renewal Application.	N
2. A list of all current exemptions granted pursuant to 10 CFR 50.12 and reliefs granted pursuant to 10 CFR 50.55(a)(3). For exemptions or reliefs that were granted based on an assumed service life or period of operation bounded by the original license term of the facility, a justification for continuing these exemptions and reliefs shall be provided.	Y
3. A description of any proposed modifications to the facility or its administrative control procedures resulting from the evaluation or analysis required by (2), above.	Y
4. A description of additions or other changes to the Technical Specifications as appropriate, including technical bases for these changes, that will be needed to account for the modifications to the plant design, age-related degradation, or limitations on plant operations during the renewal term. Technical Specification changes should not be contained in the FSAR supplement but should be contained and justified in the formal application.	Y

TABLE II. (contd)

TECHNICAL INFORMATION TO BE GENERATED AND DOCUMENTED BY LICENSEE IN THE FORM OF AUDITABLE, RETRIEVABLE RECORDS	SUBMIT WITH LR APPLICATION? Y/N (Yes/No)
5. A facility-specific list of SSCs that are important to license renewal as defined in 10 CFR 54.3(a) as required in 54.21(a)(1). Included with this list should be a description of the process used to identify SSCs important to license renewal (see Figure 1A for a schematic of such a process).	Y
6. A facility-specific list of structures and components (SCs) that are constituent elements of the SSCs important to license renewal, listed in (5), above. Included with this list should be a description of the process used to identify the SCs.	Y
7. Justification for conclusions that any selected SCs do not contribute to the performance of a safety function of an SSC important to license renewal or that their failure would not prevent an SSC important to license renewal from performing its intended safety function.	Y
8. A list of SCs requiring evaluation of age-related degradation as required in 54.21(a)(2).	Y
9. A list of SCs whose age-related degradation is not significant with respect to the CLB through the renewed license period and documentation of the evaluations that support these findings as required in 54.21(a)(4)(i).	Y
10. A list of the SCs subject to an established effective program, the associated established effective program(s), and the basis for continuing them through the renewed license period as required in 54.21(a)(3).	Y
11. A description of, and the basis for actions taken or to be taken to manage age-related degradation as required in 54.21(a)(4)(i) including changes in the refurbishment/replacement program to demonstration adequacy.	Y
12. For the SCs or similar SC groups cited in the facility-specific list specified in (6), above, identification of degradation sites, site-specific mechanisms, and when practicable, root causes.	Y

TABLE II. (contd)

TECHNICAL INFORMATION TO BE GENERATED AND DOCUMENTED BY LICENSEE IN THE FORM OF AUDITABLE, RETRIEVABLE RECORDS	SUBMIT WITH LR APPLICATION? Y/N (Yes/No)
13. For SCs important to license renewal, a summary discussion of the evaluation of key properties and parameters that may change with time and are affected by NPP operational and service conditions. The initial values at the start of operating life of these properties and parameters (such as fatigue cycle life, cable insulation dielectric strength, fracture toughness, tensile strength, and pressure boundary wall thickness) as established by analyses or qualifications should be included, along with results of evaluations of past operating environments and service conditions to determine the rates of change experienced and residual values for these properties and parameters. This summary should also include a discussion of changes to analyses resulting from age-related degradation evaluations. These values should be used in trending and analyses to establish predicted, extended operating lives and to identify actions needed to maintain key properties and parameters within acceptable limits during the renewal term. (See Appendix A of this regulatory guide for further details.)	Y
14. A description, including the technical bases, for all completed actions to incorporate the SCs listed in (6), above, into existing maintenance, surveillance, and inspection programs. These may include tests or inspections, maintenance and surveillance, and references to generic technical evaluations that provide assurance that the SSCs will not degrade below acceptable levels of safety during the renewal term.	Y
15. A specific description of maintenance or other program elements, including administrative controls, that will be implemented to provide for needed additional understanding and management of aging in SCs listed in (6), above.	Y
16. A description of the methods to be employed in maintaining records of the documentation described in this section or to be generated in the course of performing activities prescribed by this section. This	Y

TABLE II. (contd)

TECHNICAL INFORMATION TO BE GENERATED AND DOCUMENTED BY LICENSEE IN THE FORM OF AUDITABLE, RETRIEVABLE RECORDS	SUBMIT WITH LR APPLICATION? Y/N (Yes/No)
should include identification of which records are to be kept, in what form, and over what period of time. Records that permit verification that all SSCs that are important to license renewal meet their specific performance requirements should be retained in an auditable and retrievable form for the renewal term plus whatever additional period is required in accordance with the current licensing basis.	
17. A compilation of the facility's CLB. To assure auditability and retrievability, to the maximum extent possible, information comprising the CLB should be structured as or easily relatable to, the FSAR format.	N
18. A list of documents identifying portions of the CLB that are relevant to the integrated plant assessment.	Y

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TABLE III

Generic Functional NPP SSCs Important to License Renewal[a]

The following provides a generic basis for identifying SSCs important to license renewal for both PWR and BWR nuclear power plants:

a. All components which constitute the reactor coolant pressure boundary.

b. The reactor core and reactor vessel internals.

c. Systems or portions of systems that are required for 1) emergency core cooling, 2) postaccident containment heat removal, or 3) postaccident containment atmosphere cleanup (e.g., hydrogen removal system).

d. Systems or portions of systems that are required for 1) reactor shutdown, 2) residual heat removal, or 3) cooling the spent fuel storage pool.

e. Those portions of the steam systems of BWRs extending from the outermost containment isolation valve to the turbine stop valve, and connected piping of 2-1/2 inches or larger nominal pipe size to and including the first valve that is either normally closed or capable of automatic closure during all modes of normal reactor operation.

f. Those portions of the steam and feedwater systems of PWRs extending from and including the secondary side of steam generators to and including the outermost containment isolation valves, and connected piping of 2-1/2 inches or larger nominal pipe size to and including the first valve (including a safety or relief valve) that is either normally closed or capable of automatic closure during all modes of normal reactor operation.

g. Cooling water, component cooling, and auxiliary feedwater systems or portions of these systems, including the intake structures, that are required for 1) emergency core cooling, 2) postaccident containment heat removal, 3) postaccident containment atmosphere cleanup, 4) residual heat removal from the reactor, or 5) cooling the spent fuel storage pool.

h. Cooling water and seal water systems or portions of these systems that are required for functioning of reactor coolant system components important to safety, such as reactor coolant pumps.

(a) This table provides supplemental guidance for the development of plant-specific lists of SSCs important to license renewal. This guidance derives from Regulatory Guide 1.29, Seismic Design Classification.

TABLE III (contd)

i. Systems or portions of systems that are required to supply fuel for emergency equipment.

j. All electric and mechanical devices and circuitry between the process and the input terminals of the actuator systems involved in generating signals that initiate protective action.

k. Systems or portions of systems that are required for 1) monitoring of systems important to safety and 2) actuation of systems important to safety.

l. The spent fuel storage pool structure, including the fuel racks.

m. The reactivity control systems, e.g., control rods, control rod drives and boron injection system.

n. The control room, including its associated equipment and all equipment needed to maintain the control room within safe habitability limits for personnel and safe environmental limits for vital equipment.

o. Primary and secondary reactor containment.

p. Systems, other than radioactive waste management systems, not covered by items (a) through (o), above, that contain or may contain radioactive material and whose postulated failure would result in conservatively calculated potential offsite doses (using meteorology as recommended in Regulatory Guide 1.3, Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors) that are more than 0.5 rem to the whole body or its equivalent to any part of the body.

q. The Class 1E electric systems, including the auxiliary systems for the onsite electric power supplies, that provide the emergency electric power needed for functioning of plant features included in items (a) through (p), above.

r. Those portions of SSCs whose continued function is not required but whose failure could reduce the functioning of any plant feature included in items (a) through (q), above, to an unacceptable safety level or could result in incapacitating injury to occupants of the control room.

s. The first seismic restraint beyond the boundaries defined in items (a) through (r), above, and those portions of SSCs that form interfaces between Seismic Category I and non-Seismic Category I features.

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APPENDIX A

SUMMARY OF AGE-RELATED DEGRADATION PROCESSES AND THEIR MANAGEMENT IN OPERATING NUCLEAR POWER PLANTS

This appendix provides a discussion of the significant mechanisms that cause age-related degradation in NPPs and the principles involved in understanding and mitigating this degradation. Methods for selecting systems, structures, and components (SSCs) in which aging is a license renewal concern are also described. The information that follows is of a summary nature and is not intended to characterize in detail age-related degradation in NPPs.

As an NPP ages, various degradation mechanisms with potential for reducing SSC reliability are operative. Unmitigated, some of these processes could lead to reductions in safety levels below those defined in the NPP current licensing basis. Known aging mechanisms and criteria for understanding and mitigating them are described in the following sections. Many aging mechanisms and means for mitigating age-related degradation are addressed in ongoing regulatory programs. For NPP license renewal, however, some aspects of age-related degradation require additional attention. This regulatory guide, together with requirements stated in 10 CFR 54, provides the guidance needed to ensure that the technical information content of a license renewal application is adequate for the NRC staff to evaluate the effectiveness of the technical oversight and control applied to age-related degradation in SSCs that are important to license renewal. The guidance provided herein relates specifically to age-related degradation concerns that should be addressed by programs for understanding and managing aging during a renewed license term. Because these concerns center on aging mechanisms, many of which are operative over a number of years, oversight of these mechanisms must be in place before initiating a license renewal request to provide the auditable and retrievable documentation of SSC performance and maintenance needed to support a license renewal application.

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The following sections provide information that relates to 1) selecting SSCs important to license renewal, 2) understanding age-related degradation in SCs important-to-license-renewal SCs, and 3) managing aging in important-to-license-renewal SCs.

A.1 SELECTION OF SSCs IMPORTANT TO LICENSE RENEWAL

The process for selecting SSCs important to license renewal and for acquiring information that needs to be included in the license renewal application is outlined in Figures 1A and 1B. This process provides for selecting the SSCs for which age-related degradation should be addressed and for ensuring adequate understanding and management of age-related degradation in support of a license renewal application. As described in the Regulatory Position, products of this process represent a major part of the technical information to be compiled in support of, or included with, an application.

As required by 10 CFR 54.21, acceptable implementation of the process shown in Figures 1A and 1B should demonstrate that degradation of SSCs important to license renewal has been identified, evaluated, and accounted for in ensuring that the current licensing basis, as defined in 10 CFR 54.3(a), will be maintained throughout the license renewal term. Consistent with requirements for continued compliance with the current licensing basis, the selection process to be applied to SSCs with known safety functions emphasizes deterministically based evaluation of aging mechanisms and their effects. The license renewal applicant may also use probabilistic risk assessment (PRA) techniques as a supplement to the primarily deterministic methods to add additional components to the list of those designated as important to license renewal.

The process shown in Figures 1A and 1B utilizes the knowledge gained from engineering design information, tests, and operating experience. Also, data from in situ assessments, condition monitoring, maintenance and other records, and post-service examination and tests are recommended inputs to this process.

A.2 ELEMENTS OF AN EFFECTIVE PROGRAM TO ADDRESS AGING DEGRADATION

An effective program for addressing age-related degradation will provide for both understanding and managing the aging that occurs in NPPs. Aging

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mechanisms and their effects should be understood with sufficient accuracy and detail to provide the basis for developing and implementing aging management strategies that address, in a prioritized and timely fashion, actual or potential root causes of SSC failure.

A.2.1 Understanding Age-Related Degradation[a]

The aging mechanisms that occur in NPP SSCs should be understood if age-related degradation is to be effectively managed. The requisite understanding may be either empirical or mechanistic depending on the nature and potential consequences of a particular degradation mechanism. An understanding of age-related degradation requires a detailed awareness of SSC design, fabrication, installation, testing, inservice operation, and maintenance cycles. All of these elements in the life cycle of SSCs involve their interaction with stressors associated with service environments.

Age-related degradations of SSCs are time-dependent phenomena that depend upon the interactions of materials and environmental and operational stressors. Assessments of age-related degradation should consider the integrated effects of these interactions, and all SSCs that are important to license renewal should be evaluated in this context.

A.2.1.1 Materials

Most materials used in the fabrication of SSCs are subject to some level of age-related degradation. Whether this degradation can affect the operability or reliability of SSCs such that operation of the plant is reduced below acceptable safety levels is an important concern. It is important to understand how and at what rate the metallic, nonmetallic, and composite materials used in plant components degrade with time and how this degradation can be managed to ensure the operability or reliability of SSCs. This knowledge of material behavior is important in design and operations and in developing quality assurance, plant inspections, condition monitoring, and maintenance programs. As more is learned about the age-related behavior of materials and

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(a) For an expanded discussion, consult the annotated bibliography at the end of this appendix.

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how to use this knowledge in the design and operation of SSCs using these materials, confidence will grow in predictions of SSC lifetime behavior and plant operational safety.

A.2.1.2 Aging Stressors

Of the factors that can affect the age-related degradation of NPP SSCs, the stressors associated with environmental and service conditions are generally the most difficult to understand. Stressors due to service conditions assume various forms (e.g., mechanical, electrical) and can originate or are intensified during component fabrication, assembly, transportation, installation, operation, testing, and maintenance. Those who design, fabricate, operate, and maintain SCs should understand how stressors can degrade their operational capabilities.

A.2.1.2.1 Environmental Conditions. Environmental conditions under which SSCs are designed to function contribute individually and in concert with other stressors to age-related degradation. Environmental elements include ambient operating conditions (humidity and temperature within the plant or within a storage facility), chemicals that contact the material (pollutants, acids, lubricants, etc.), radiation, etc. Environmental effects can individually cause degradation or influence the rate at which degradation progresses or may act in combination with other factors (e.g., material type and condition, heat, and stress).

A.2.1.2.2 Service Conditions. Service conditions consist of steady-state, cyclic, or other transient loadings imposed on SSCs during normal operation, testing, or abnormal events. The principal loadings are mechanical in nature. Significant age-related degradation can also occur because of electrical loadings.

1. Mechanical loads are generally associated with physical movements, pressure differentials, and dimensional changes. The operation of SSCs either during normal operation, including testing or under accident conditions, usually induces time-dependent mechanical stresses. These stresses are caused by dynamic loads, internal or external pressure changes, impact, vibration loads, temperature

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changes, component test loads, and seismically induced motions. The operational motions of active SSCs (e.g., valve operation and pump rotations) produce time-dependent distortions and inertial stresses as well as wear. The effects of these loads in degrading SSCs are generally understood; but degradation rates are usually only estimates obtained from the analysis of inservice monitoring data, inspection reports, and maintenance information. Proper maintenance can mitigate much or all of the degradation caused by mechanical loads. Internal and external pressure loads approaching operational or accident design limits also can produce high stresses that can cause distortions and, after sufficient cycles, can result in strain hardening and fatigue damage to SSC materials. If these stresses are combined with vibration and thermal stresses, measurable degradation can occur in a period that is short relative to the anticipated operational life of the SSCs. Seismic events or similar but more localized events, e.g., water hammer, can inflict immediate damage to SSCs at any point during their operational life. Even though the SSCs may not fail during the impact, their functional capability may be degraded such that the operational life is shortened. The extent of the damage to SSCs resulting from external sources must be understood to anticipate any associated reduction in lifetime. Vibrational loads can cause fatigue damage. Methods of analyzing vibrational fatigue damage are available; however, the results often include large uncertainties. These uncertainties are associated with material fatigue properties and the distribution and magnitude of the induced dynamic stresses. Vibrational stresses may be induced by plant operational modes, during transportation if a component is not properly isolated, and by ground or seismic vibrations. The source of vibrational loads that develop during the operational life of SSCs, the distribution of the associated stresses, and the endurance limits of the materials must be known for lifetime prediction. Thermal stresses develop in SSCs because of temperature-gradient-induced differences in thermal expansion and the fact that different materials expand at different rates when heated. Differential expansions may be resisted internally or by interference with adjacent component surfaces. This resistance results in time-dependent thermal stresses that can cause age-related degradation, either separately or when combined with the effects of other stressors. Typical of such degradation are the thermal fatigue cracks that have appeared in high temperature coolant water piping and nozzles and embrittlement of insulating materials.

2. Electrical stresses are induced in the insulating materials used in the fabrication of electrical and electromechanical parts and components. Both passive SSCs (cables, connectors, electrical penetrations, transformers, terminal boards, etc.) and active SSCs (motors, circuit-breakers, relays, voltage and current activated devices, etc.) experience voltage gradients during normal operation and testing. Of primary concern are the higher levels of electrical stresses that are generated during switching operations and during accident and post-accident situations. The nature of electrical voltage loads varies depending upon the design and functional application of the device. Voltage gradients can be very high and may be imposed by d.c., a.c., or nonperiodic, fast or slow transients. The most severe voltage gradients are experienced when a device is subjected to various combinations of these voltages superimposed at the same time. The magnitude and duration of voltage- and current-related stresses in plant electrical SCs should be accurately assessed during normal operating conditions, test sequences, and accident and post-accident situations.

A.2.1.3 Aging Mechanisms

Stressors and environments act in concert on SSC constituent materials to cause age-related degradation. Many mechanisms potentially can contribute to degradation processes. Extensive analytical and experimental efforts by both government and industry have identified numerous aging mechanisms that are operative in nuclear power plants. These mechanisms vary widely in terms of their potential effects. Some mechanisms affect numerous types of SSCs over wide variations in environment and stressor level; others are limited in their effects to specific components or materials over narrow ranges of conditions. Aging mechanisms of concern in NPPs include the following.

1. Corrosion

Corrosion is a common form of degradation in NPPs, resulting in wall thinning in steam and condensate systems, pitting in service water systems, and transport of activated corrosion products. Many localized corrosion processes are operative in NPPs, e.g., crevice corrosion, pitting corrosion, galvanic corrosion, various types of stress-enhanced or irradiation-enhanced corrosion, microbiologically influenced corrosion, etc. These processes can result in local wall thinning that may lead to failure.

Oxidation to produce a surface oxide scale takes place in metals by direct reaction with an oxidizing atmosphere. If the scale is nonporous and completely covers the surface, the reaction rate will decrease as the oxide thickens because the transport of reactive species through the scale becomes rate controlling. Factors such as electrical potential, concentration gradients, or preferential migration paths through the film may control the overall corrosion rate. The breakdown of surface scales, typically through mechanical or chemical processes, often leads to a loss in protective quality of the scale.

Pitting is a localized form of corrosion that results in small craters or holes in the metal. Pitting is potentially one of the most insidious forms of corrosion because it can lead to component failure by perforation while producing only a small loss of metal. Because of their small size and because the pits are often covered with corrosion products, they can be difficult to detect. Pitting occurs when one area of a metal surface becomes anodic with respect to the rest of the surface or when highly localized changes in the environment in contact with the surface cause accelerated attack. Causes of pitting include local inhomogeneities on or beneath the metal surface, local loss of passivity, mechanical or chemical rupture of the protective oxide surface film, galvanic corrosion from a relatively distant cathode, and the formation

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of a metal ion or oxygen concentration cell under a solid deposit (crevice corrosion). The rate of penetration into the metal by pitting may be 10 to 100 times greater than for general corrosion. The most common causes of pitting in steels are surface deposits that set up local concentration cells and dissolved halides that produce local anodes by rupture of the protective surface scale. With corrosion resistant alloys, such as stainless steels, the most common cause of pitting is the highly localized destruction of passivity through contact with a halide-containing environment.

Uniform attack is normally characterized by a chemical or electrochemical reaction which proceeds uniformly over the entire exposed surface or over a large area. The metal becomes thinner and eventually fails. Wall thinning of steam generator tubes has occurred because of uniform attack by acid phosphate residues concentrated in low flow areas. Uniform attack of carbon or low alloy steel by concentrated boric acid has also been observed.

Intergranular attack is preferential dissolution of the grain boundary regions of a metal with only slight or negligible attack of the grain matrix. This preferential attack can be enhanced by segregation of specific elements or impurities, by enrichment of one of the alloying elements in the grain boundaries, or by the depletion of an element that imparts corrosion resistance to the grain boundary areas. Susceptibility to intergranular attack usually develops during thermal processing such as welding or heat treatments. The susceptibility to intergranular attack can often be corrected by redistributing alloying elements more uniformly through solution heat treatment, by modifying the alloy to increase resistance to segregation, or by using a completely different alloy.

Stress corrosion cracking (SCC) is an aging mechanism that occurs in engineering materials by the combined and synergistic interaction of a chemically aggressive environment, a susceptible material, and a tensile stress or radiation field. The material fails by slow, environmentally-induced crack growth that occurs with little or no attendant macroscopic

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plastic deformation. Although a tensile stress is not necessary for irradiation-assisted SCC, it can aggravate the phenomenon. The stresses required to cause SCC are usually below the yield strength and are tensile in nature. These stresses can be either applied or residual and may result from the fabrication process or inservice loading of the component or structure. Common sources of stress include thermal processing and stress risers created during surface finishing, fabrication, or assembly. The length of time required to produce SCC decreases for increasing stress level. The minimum stress at which cracking will occur depends on the temperature, the composition and microstructure of the alloy, and the environment. SCC may initiate at pre-existing mechanical cracks or other surface discontinuities such as pits produced by chemical attack.

Microbiologically influenced corrosion (MIC) occurs when biological organisms affect corrosion processes on metals by directly influencing the anodic and cathodic reactions, by affecting the protective surface scales on metals, by producing corrosive substances, or by creating solid deposits. These organisms include microscopic forms such as bacteria and macroscopic types such as algae and barnacles. Microscopic and macroscopic organisms have been observed to live and reproduce under broad ranges of pressure, temperature, humidity, and pH; thus, biological organisms may influence corrosion in a variety of environments. MIC effects on carbon steel may result in random pitting, general corrosion, or severe hydraulic effects due to formation of tubercles and massive corrosion product deposits. MIC attack on stainless steel is characterized by pitting, most commonly at weldments.

Saline water attack has resulted in the degradation of reinforced concrete structures. The degradation mechanism involves water seepage into the concrete thereby providing a high chloride environment to the reinforcing bars. The reinforcing bars corrode resulting in expansion, which leads to and cracking and spalling of the concrete. This aging mechanism is of particular concern for Category I structures, or parts thereof, that cannot be routined inspected or examined because of submergence in water or physical inaccessibility due to intense radioactivity.

2. Erosion

Erosion caused by high velocity steam, water, or two-phase mixtures (which may include silt or other particulates) has contributed to failures of NPP equipment. Degradation processes of importance include cavitation and particulate wear. Erosion caused by cavitation involves the creation of a two-phase gas-liquid zone in the vicinity of high-speed, rotating parts (e.g., pump impellers) or in components in which steep pressure gradients occur (such as throttling valves and orifices).

Erosion-corrosion is an accelerated form of corrosion caused by the relative motion of a corrosive fluid with respect to a metal component. The corrosion process is accelerated because of erosive destruction of the protective oxide film resulting in chemical attack or dissolution of the underlying metal. The carbon steel secondary piping systems in NPPs are susceptible to erosion-corrosion. The damage morphology is usually characterized by grooves, waves, and valleys oriented in a consistent direction. Highest erosion rates tend to occur in regions where the metal is in contact with wet steam. Alloy additions to carbon steel can reduce or eliminate erosion-corrosion in most cases. Chromium is the most effective alloying element for improving resistance. Other elements such as copper and molybdenum also have a beneficial effect.

3. Embrittlement

Embrittlement, of metals and polymers used as electrically insulating barriers, because of structural or chemical changes induced by radiation, elevated temperature, or atmospheric contaminants can lead to fragility and failure under dynamic loading. Metallic components are most susceptible to embrittlement from neutron radiation; thus, components in proximity to the reactor core are most affected. Embrittlement with loss in toughness for critical components such as pressure vessels and supports represents the most significant contribution of radiation to

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aging. Organic and electronic materials are particularly susceptible to radiation damage from gamma rays. Thermal embrittlement is associated with chemical or metallurgical changes and results from such processes as thermal aging leading to reduced toughness of ferrous alloys, high temperature sensitization to intergranular stress corrosion cracking in austenitic stainless steels, and oxidation or cross linking of polymers with a resultant loss in toughness and dielectric strength. Hydrogen absorption by metallic alloys can also lead to loss of toughness and brittle fracture.

<u>Neutron irradiation</u> of metal components can result in a significant increase in yield strength with accompanying decreases in ductility and fracture toughness. Irradiation embrittlement is primarily caused by irradiation-induced precipitation of fine-scale copper precipitates and formation of radiation-induced point defect clusters. These mechanisms produce barriers to dislocation movement, thereby causing an increase in the yield stress of the steel, a shift in the ductile-to-brittle transition temperature, and a decrease in fracture energy. The major variables controlling irradiation embrittlement in reactor steels are the copper and nickel content of the steel and the neutron fluence. Other factors that contribute include irradiation temperature, neutron spectrum and flux, phosphorus content, thermomechanical history, and concentrations of other impurities and minor alloying elements.

<u>Thermal embrittlement</u> can occur in cast austenitic-ferritic (duplex) stainless steel piping. The embrittlement is associated with the formation of precipitates in the ferritic phase, leading to cleavage of the ferrite or separation of the ferrite/austenite phase boundaries. The degree of aging is related to the volume fraction of ferrite in the material. In addition, the precipitation and growth of phase-boundary carbides or nitrides can lead to brittle fracture. In general, low carbon grades of cast stainless steel are the most resistant, and molybdenum-containing high carbon grades are the most susceptible to thermal embrittlement.

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Hydrogen damage is an environmentally assisted degradation process that usually results from the combined action of hydrogen and residual or applied tensile stresses. Hydrogen damage occurs in several ways such as hydrogen embrittlement, blistering, and cracking from hydride formation. Hydrogen embrittlement is usually associated with loss of tensile ductility in carbon steels and high strength alloys and is a function of the stress level and time. Steel can be embrittled by only a few parts per million hydrogen, which can originate from the fabrication process or inservice corrosion reactions. A similar effect may occur in austenitic stainless steels, but required hydrogen levels are many times the levels for carbon steels. Above about 400°F, steels are not affected by hydrogen embrittlement.

4. Mechanical Degradation Mechanisms

Fatigue is a common degradation process that occurs in rotating or reciprocating equipment or under other service conditions that place periodic or cyclic loads on SSCs. Fatigue damage results in progressive, localized structural change in materials subjected to fluctuating stresses and strains. Associated failures may occur at either high or low cycles in response to various kinds of loads, e.g., mechanical or vibrational loads, thermal cycles, pressure cycles, etc. The process of fatigue consists of three stages: 1) initial fatigue damage leading to crack initiation, 2) crack propagation, and 3) sudden fracture of the remaining ligament. Fatigue cracks initiate and propagate in regions of stress concentration that intensify strain, e.g., structural defects. The fatigue life of any SSC is the number of stress or strain cycles required to cause failure. This number is a function of several variables such as stress level, stress state, cyclic waveform, fatigue environment, and the metallurgical condition of the material. Stress cycles can be generated by the direct application of mechanical loads, differential thermal expansion of mechanically constrained components, or temperature fluctuations. Although the loading conditions are different, the resultant fatigue is considered to be additive. Fatigue cracks form at the point of maximum local stress and minimum local strength. The

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local stress pattern is governed by the geometry of the SSC, including local features such as surface and metallurgical imperfections that concentrate stress, and by the type and amplitude of the loading. Surface imperfections such as scratches, mars, burrs, and other fabrication flaws are locations where fatigue cracks may start. Inclusions, hard precipitates, and crystal discontinuities such as grain boundaries are examples of microscopic stress concentrators. Pitting corrosion, SCC, and other effects of a hostile environment may also be important. For example, many fatigue failures originate in fretted areas. In many large, structural components, the existence of a crack does not necessarily imply imminent failure of the component. Significant structural life may remain before the crack grows to a size at which failure occurs. The growth of a fatigue crack under cyclic loading is principally controlled by the maximum load and the ratio of maximum to minimum load.

Wear is a general concern for rotating or other sliding surfaces where tolerances can affect performance. Lubricant loss or degradation, e.g., because of contaminants or chemical breakdown, can greatly accelerate wear. Fretting is a wear phenomenon that occurs between tight-fitting surfaces that are subjected to a cyclic, relative motion of extremely small amplitude. Fretting is frequently accompanied by corrosion. Common sites for fretting are in joints that are bolted, keyed, pinned, press fit, or riveted; in oscillating bearings, couplings, spindles, and seals; in press fits on shafts; and in universal joints. Under fretting conditions, fatigue cracks may be initiated at stresses well below the endurance limit of nonfretted specimens. The initiation of fatigue cracks depends mainly on surface residual stresses superimposed on applied cyclic stresses.

Shrinkage or creep can occur in most materials and are common phenomena in plastics and in metals at high temperatures. Polymers and composites used as electrical insulators, supports, and protective coatings may exhibit dimensional changes caused by exposure to high temperatures, moisture, mechanical stresses, or radiation. These effects can lead to

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deterioration in insulating and structural properties. Shrinkage of concrete in NPPs is due mainly to long-term dehydration. Dimensional changes in concrete as it ages do not degrade the properties of concrete; however, when these dimensional changes cause interference (e.g., with other components in prestressed reinforced concrete structures), degradation can occur. Shrinkage is the main contributor to the loss of prestressing forces in prestressed concrete containments.

A.2.1.4 Degradation Resulting From Operational Environment

The operational environment of an NPP has age-related degradation implications over the plant operating history that should be properly accounted for. Some SSCs were initially designed or qualified for a finite lifetime (usually 40 years or less) with an associated design margin or safety factor that may, in practice, change during service. In effect, the original design or qualification provided initial values and minimum acceptable values for key design properties and parameters such as minimum values of wall thickness, fatigue cyclic life, dielectric strength, fracture toughness, tensile strength, etc. These properties and parameters may change with time as SSCs are subjected to loadings and environmental stressors from design basis events and also from events not included in the original design. In the license renewal process, each licensee should return to the initial design or qualification analyses as supplied by the original equipment manufacturer (including all modifications and revisions thereto), evaluate the past service experience to determine residual values, and determine actual rates of change for key design properties and parameters. Actual rates of change together with minimum acceptable values of key properties and parameters will be useful in establishing an acceptable extended operating license period. An example of an event that may not have been included in the original design but should be considered is leakage of hot primary cooling water into low temperature piping. While such leakage would have been evaluated as an isolated event at the time of occurrence, other related aspects of the plant operation should be evaluated to ensure that a fatigue effect does not go unevaluated. Each event with aging consequences should be evaluated and reconciled with the original design or original qualification to both ensure that the design conditions were not exceeded and that transients did not contribute to limiting the expected lifetime of the affected SSC. Normal operating, testing, and environmental stressors including those due to electrical, mechanical, and thermal loadings also contribute to age-related degradation that should be evaluated prior to extended life. Original equipment designers and manufacturers should be consulted for identification of aging mechanisms specific to particular SSCs.

A.2.1.5 Degradation Sites

Most SSCs are not uniformly susceptible to degradation. Certain sites exhibit more deterioration than others; and for many SSCs, degradation is limited to only a specific location. Factors that affect vulnerability to degradation include localized chemical or metallurgical variations, geometry with respect to fluid flow or chemical potential gradients, proximity to mechanically or chemically incompatible materials, and localized high stresses. Examples of site-specific degradation include 1) localized erosion/corrosion in ferritic steel piping because of local high fluid velocities, 2) enhanced intergranular stress corrosion cracking in heat-affected zones near welds in austenitic stainless steels, 3) excessive hinge pin wear in check valves subject to flutter, 4) rapid degradation of pump impeller blades when cavitation occurs, 5) wear or galling of sliding contacts, 6) crevice corrosion, and 7) fatigue cracking in regions experiencing tensile stresses. An understanding of age-related degradation requires a knowledge of which sites degrade by what mechanisms and at what rates. This information is fundamental to selection of effective monitoring methods and where, how, and with what frequency monitoring should be implemented to reliably trend and mitigate degradation.

A.2.2 Managing Aging Degradation[a]

When the interactive effects of materials, designs, and stressors due to environmental and service conditions are understood, the root causes of age-related degradation can be identified and programs to ensure that SSCs will adequately perform their intended functions can be implemented. Inspections and surveillance to monitor degradation in important-to-license-renewal SSCs

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(a) For expanded discussions, consult the bibliography at the end of this appendix.

should be regularly performed. Selectively applied condition monitoring and trending can also be useful in this respect. Effective management of aging will permit timely repair, replacement, or servicing through preventive or corrective maintenance.

Effective maintenance programs require understanding of what to maintain, when to maintain, and how to maintain plant SSCs. Depending upon their intended function, these programs take various forms (e.g., inspections, surveillance, tests, condition monitoring, trending, recordkeeping, predictive maintenance, preventive maintenance, corrective maintenance, and reliability centered maintenance). The mix of elements that comprise an overall maintenance program should reflect both the technical nature and the potential consequences of the age-related degradation processes that the program is intended to mitigate.

From an aging management perspective, the key steps in determining when to maintain and how to maintain specific SSCs are:

1. Identify monitorable indicators that can be trended to show aging effects on the performance or reliability of SSCs important to license renewal.

2. Develop and implement methods for monitoring the indicators identified in (1), above.

3. Retain information acquired by monitoring programs in auditable, retrievable form.

4. Trend performance measures and functional indicators for each SSC under observation and analyze the impact of rate of change; retain information in auditable, retrievable records.

5. Determine minimum acceptable functional capability at the end of service life for normal operation and for accident mitigation.

6. Develop criteria for effective surveillance, maintenance, refurbishment, and replacement programs.

7. Interpret, analyze, and make decisions for maintenance or replacement.

Both predictive and preventive maintenance programs are needed to manage aging. The aging management program will provide useful input for making decisions for the full spectrum of maintenance-related activities including quality assurance and quality control, engineering support, and plant modifications.

A.2.2.1 Root-Cause Determination

In order to avoid recurrences of excessive degradation, it is necessary to understand the basic underlying causes of observed deterioration, i.e., root cause. Root cause is defined as the most basic reason or collection of reasons for the degradation that if corrected, will prevent future similar deterioration. Root causes may be associated with intrinsic SSC characteristics, such as composition, metallurgical structure, or design features or may reflect situational factors, i.e., departures from design envelopes, extremes in environmental factors or stressors, operational variables, or combinations of these and other factors. An analytical program should exist to evaluate instances of unexpected or excessive degradation in terms of their root causes. Root-cause analysis relies upon the availability of accurate, sufficiently detailed, retrievable records to provide the facts needed to evaluate the potential engineering, procedural, operational, and environmental contributors to the observed degradation. Given this information, knowledgeable staff can generally track causes and effects to successively more basic levels until the root causes are revealed. When the root causes are understood, methods for preventing recurrence of similar degradation will generally become evident.

A.2.2.2 Monitoring Aging Degradation

Monitoring and trending of age-related degradation are the bases for predictive maintenance. The overall goal of the predictive maintenance program is to provide information concerning degradation rates and residual lifetimes that can be used to predict and prevent failures. Tools used in doing this include nondestructive examination (NDE), condition-monitoring, residual life assessment, and information analysis and trending. Trends and defined action levels provide guidance needed by the preventive maintenance program to schedule services with a frequency that will avoid failure of SSCs important to license renewal. Monitoring and trending of the effects of age-related degradation provide opportunities for identifying and eliminating sources of unnecessary degradation through root-cause analysis and corrective action. Approaches to monitoring degradation include the following.

A.2.2.2.1 Nondestructive Examination. Various nondestructive techniques are employed as part of in-service inspection and testing programs to detect and characterize flaws or other evidence of degradation that may be failure precursors. Commonly used methods include visual inspection, dye-penetrant and magnetic particle treatments, radiography, eddy current testing, ultrasonic testing, electrical signature analysis, and acoustic emission monitoring. Each of these methods has its advantages and limitations. The limitations derive mainly from the fact that NDE techniques were developed primarily as quality control tools for detecting manufacturing flaws. New or improved NDE methods are continuously being developed. Techniques that will provide the quantitative characterizations of flaws required for fracture mechanics analysis and that will allow on-line monitoring of deterioration in mechanical properties during long-term inservice exposure are expected to be available in the future.

A.2.2.2.2 Condition Monitoring. For some SSCs that are important to license renewal, integrated monitoring programs that might involve a combination of sensors and evaluation methods to ensure reliability may be in order. Condition monitoring should be employed when justified in terms of the consequences of potential failures.

A.2.2.2.3 Surveillance, Testing, and Inspection Programs. Detailed and comprehensive requirements for monitoring degradation in SSCs are conveyed by various regulatory instruments including the Inservice Inspection (ISI) requirements in Section XI of the ASME Boiler and Pressure Vessel Code and the surveillance testing requirements stated in the plant technical specifications. These oversight programs can provide useful indications of age-related degradation. These programs are supplemented by nonmandatory surveillance, inspections, and tests that reflect good engineering practices.

A.2.2.2.4 Residual Life Assessment. For monitored trends in age-related degradation to have meaning in terms of service, replacement, or refurbishment frequency, it is necessary to correlate the level of monitored parameters with expected SSC residual lifetimes. These correlations are difficult to establish at best; and, as a consequence, the technology for assessing residual life is not well developed. Methods employed include surveillance specimen testing, monitoring of operational parameters, evaluation of SSCs that have been in service, and mechanistic and empirical modeling to provide bases for predictions. Improvements in the technology, accruing from more sophisticated and reliable models, better archiving, development of miniature specimen testing and reconstituted specimen testing techniques, and in situ monitoring of the effects of aging-related degradation, are expected to greatly increase the scope and confidence of future residual lifetime assessments.

In summary, degradation monitoring methods, e.g., inspection, surveillance, testing, condition monitoring, should reflect mechanistic and empirical assessments performed by qualified staff in their efforts to understand and mitigate age-related degradation. These methods should employ state-of-the-art NDE, e.g., ultrasonic testing, signature analysis, vibration analysis, dielectric performance measurements, and other measuring techniques performed by qualified staff. Measurement results should be documented, trended, and analyzed with respect to implications for residual SSC lifetime and for frequency and nature of preventive and corrective maintenance.

A.2.2.3 Mitigating Aging

Timely mitigation of age-related degradation through regular service, repair, refurbishment, or replacement of SSCs is the prime function of the maintenance program. Some or all of the monitoring activities discussed in the preceding sections are generally included under the auspices of the Maintenance Department. For present purposes, mitigation of aging is construed as the collection of activities that relate directly to physical maintenance of important-to-license-renewal SSCs.

Maintenance activities range from simple, straightforward tasks to complex activities that require extensive coordination, training, and technical expertise. The level of oversight and resources devoted to these activities should reflect their complexity and importance to plant safety and reliability. A maintenance program has many important elements. Those considered

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here as being particularly relevant to age-related degradation include preventive maintenance, corrective maintenance, reliability centered maintenance, and recordkeeping and trending. Most of these elements have clear interfaces and interdependencies with the monitoring activities discussed in the preceding sections. In addition, the scope and nature of the various maintenance elements should reflect the as-built plant specifications; manufacturer's recommendations; operating experience--both internal and external; relevant recommendations and information from the NRC, the nuclear power industry, and its vendors; and general good engineering practices.

A.2.2.3.1 Preventive Maintenance. Preventive maintenance includes the planned and scheduled actions performed to prevent equipment failure. Preventive maintenance relies heavily upon information generated by monitoring programs to define necessary activities and the frequency at which they should be performed. In addition to input from monitoring programs, preventive maintenance action should be based on equipment histories, other plant performance experience, vendor recommendations (to support life extension programs as well as the current licensing basis), and good engineering practice. Preventive maintenance conducted to support license renewal should be so identified and should be comprehensive in nature. Planned actions and schedules should be documented, and departures from these plans should be justified on technical grounds and subject to management review and approval. Clear, comprehensive procedures are vital for preventive maintenance and other oversight and maintenance activities.

A.2.2.3.2 Corrective Maintenance. Corrective maintenance is performed to restore failed or malfunctioning equipment to service. For some types of equipment (e.g., items lacking severe failure consequences), a corrective rather than preventive approach is preferred. Malfunctions that represent significant challenges to plant safety or reliability should be prevented. A major responsibility of the maintenance organization is to be cognizant of the significance of potential malfunctions and to ensure that severe consequence events are averted by adequate preventive maintenance. As with other maintenance activities, corrective maintenance priorities should be based on the relative importance of the equipment and on plant safety and reliability

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objectives. Added functions of corrective maintenance are to determine root causes of malfunctions and carry out appropriate corrective action to prevent recurrences.

A.2.2.3.3 Reliability-Centered Maintenance. The traditional approach to defining maintenance program objectives and priorities is based upon engineering judgment supported by vendor and industry data, maintenance and operating histories, and regulatory requirements and guidance. These will continue to be essential considerations in structuring a maintenance program. They are likely to be supplemented by new approaches that quantitatively correlate priority with safety significance and reliability as key factors in prioritizing maintenance activities. Reliability-centered maintenance uses formalized decision logic to prioritize preventive maintenance activities and to limit maintenance and oversight to those SSCs having low safety or economic consequences of failure. The general product of applying reliability-centered prioritization of preventive maintenance will be:

- a list of SSCs whose failure or loss of function could have significant safety consequences. These SSCs require scheduled preventive maintenance that may be further prioritized based upon risk, operating experience, and expert opinion.
- a list of SSCs whose failure or loss of function would not be self-evident. These SSCs should also be subject to scheduled oversight and/or maintenance.
- all other SSCs. Failure or loss of function for these will have economic consequences only. Preventive maintenance is at the discretion of the plant owner and, presumably, would require justification on economic grounds.

Results of risk-based analyses can be used to prioritize reliability-centered maintenance activities. These methods employ quantitative failure mode and effect analyses, e.g., PRA, to quantitatively identify SSCs, in the context of their service and systems environments, whose malfunction could jeopardize plant safety. In this way, SSCs can be ranked in terms of safety significance, and oversight and maintenance efforts can be commensurately focused upon the most risk-significant equipment.

A.2.2.4 Recordkeeping and Trending

Recordkeeping and trending are essential elements of both monitoring and maintenance programs. The sole product of monitoring programs is information. In order to be useful, this information must be translated into effective maintenance practices. This requires that 1) the information obtained by monitoring activities be recorded in adequate, unambiguous detail in a form that allows ready retrievability and 2) the information be reliably relatable to specific maintenance practices that effectively address the age-related degradation that is actually occurring. Records that meet these requirements are needed to prioritize maintenance resources and to correlate actual operating environments and stressors with design assumptions and computed lifetimes so that SSC lifetimes and maintenance intervals can be realistically anticipated.

Maintenance records serve to establish performance histories for the SSCs that comprise the plant. This information and its continuous feedback are useful in specifying what, how, and when equipment should be maintained; what information should be collected; and how it should be recorded. Maintenance histories and equipment performance trends should be documented and kept current. Requirements for records retention and retrieval should be established to meet the needs of other elements of programs to understand and manage age-related degradation. These requirements should be consistent with quality assurance program requirements related to records.

Recordkeeping can be supplemented or requirements offset by conservative maintenance practices based upon equipment history or conservative condition assessments for selected SSCs; however, detailed, usable, and retrievable records of such practices and condition assessments and supplementary raw data should be maintained. This is a task that, in principle, can be simplified greatly by modern computer technology, which has enhanced the technical and economic feasibility of maintaining high quality records. Trending of information obtained by monitoring activities may be a straightforward process that leads directly to maintenance recommendations. More often, however, trending intended to lead to improved oversight and control requires considerable initial development of the basic trending program and qualification of the

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measures to be trended if results are to be meaningful. Records of component failure data can be trended and monitored to assess maintenance program effectiveness. Process indicators, such as post-maintenance test results, surveillance test results, ratio of preventive to corrective maintenance, maintenance backlog, and rework frequency, should also be trended to provide indications of overall maintenance effectiveness and areas requiring improvement.

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APPENDIX B - REPRESENTATIVE SYSTEMS, STRUCTURES, AND COMPONENTS POTENTIALLY IMPORTANT TO LICENSE RENEWAL[a]

I. PRESSURIZED WATER REACTORS

	Standard Review Plan, NUREG-0800	Generic[b] Functional Criteria	Standard Technical Specifications
A. RELIED UPON FOR PRESSURE BOUNDARY INTEGRITY, SHUTDOWN AND ACCIDENT MITIGATION			
1. Reactor Coolant Pressure Boundary			
Reactor Vessel	5.3.3	a	3/4.2.1, 4.9.1, 4.10, 3/4.9.10
Steam Generator	5.4.1, 5.4.2.2	a	3/4.4.5, 3/4.4.6, 3/4.4.10
Reactor Coolant Pump	5.4.1	a	3/4.4.1, 3/4.4.10
Piping	5.4.3	a	3/4.4.1.1, 3/4.4.10,
Pressurizer	5.4.10	a	3/4.4.3, 3/4.4.10, 3/4.4.10
Instrumentation	7.1	a	3/4.3
Valves	5.4.12	a	3/4.4.2, 3/4.4.4
2. Power Operated Relief Valves, Block Valves, and Interconnected Piping			
Pressurizer PORV	5.4.13	a	3/4.4.4, 3/4.4.9.3, 3/4.4.10
Pressurizer Block Valves	5.4.13	a	3/4.4.4, 3/4.4.10
Pressurizer Piping	5.4.3	a	3/4.4.10
Safety Valves			3/4.4.2, 3/4.4.10
3. Reactor Protection System		j	
Detector	7.2	j	3/4.3.1, 3/4.2, 2.2.1
Signal Comparator	7.2	j	3/4.3.1, 3/4.2, 2.2.1
Logic Circuit	7.2	j	3/4.3.1, 3/4.2, 2.2.1
Master Relay	7.2	j	3/4.3.1, 3/4.2, 2.2.1
Slave Relay	7.2	j	3/4.3.1, 3/4.2, 2.2.1
Connecting Wire/Cable	7.2	j	3/4.3.1, 3/4.2, 2.2.1
4. Engineered Safety Features Actuation System		k	
Detector	7.3	k	3/4.3.2, 2.2.1
Signal Comparator	7.3	k	3/4.3.2, 2.2.1
Logic Circuit	7.3	k	3/4.3.2, 2.2.1
Master Relay	7.3	k	3/4.3.2, 2.2.1
Slave Relay	7.3	k	3/4.3.2, 2.2.1
Connecting Wire/Cable	7.3	k	3/4.3.2, 2.2.1
5. Control Room and Auxiliary Shutdown	7.4	n	
Cable	7.4	n	3/4.3.3.5, 2.2.1
Instrumentation	7.4	n	3/4.3.3.5, 2.2.1
6. Nuclear Instrumentation		j,k,	
Source Range Detectors	7.2	j,k	3/4.3.1
Intermediate Range Detectors	7.2	j,k	3/4.3.1
Power Range Detectors	7.2	j,k	3/4.3.1
Connecting Cable		j,k	3/4.3.1

(a) References to NUREG-0800 and the Standard Technical Specifications are directly relevant only to NPPs that were reviewed against NUREG-0800. For older NPPs, these references should be viewed as illustrative only; and the licensee should consult the plant-specific CLB, which includes the current FSAR, for comparable sources of information.

(b) Table III of this regulatory guide.

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	Standard Review Plan NUREG-0800	Generic[b] Functional Criteria	Standard Technical Specifications
A. RELIED UPON FOR PRESSURE BOUNDARY INTEGRITY, SHUTDOWN AND ACCIDENT MITIGATION (contd)			
7. Non-nuclear Instrumentation		j,k	
Temperature, RCS	7.2	j,k	3/4.3.1
Pressure, RCS	7.2, 7.3	j,k	3/4.3.2, 3/4.3.1
Pressurizer Level	7.2	j,k	3/4.3.1
Flow, RCS	7.2	j,k	3/4.3.1
Reactor Vessel Level Instrumentation	7.5	--	3/4.3.3.6
Sub-cooling	7.5	j,k	3/4.3.3.6
Pressurizer Pressure	7.2, 7.3, 7.4,	j,k	3/4.3.1, 3/4.3.3, 3/4.3.2, 3/4.3.3
Steam Generator Level	7.2, 7.4, 7.5	j,k	3/4.3.1, 3/4.3.3.6
Impulse Pressure	7.7	j,k	
Steam Flow	7.7, 7.2	--	3/4.2
Feedwater Flow	7.7, 7.2	--	3/4.3.3.5 (AF)
Steam Pressure	7.7, 7.2	j,k	3/4.3.3.5
Feedwater Pressure		--	
8. In-Core Instrumentation		j,k	
Flux Detector	7.2	--	3/4.3.3.2
Thermocouple	7.5	--	3/4.3.3.6
Drive Assembly	7.2	--	3/4.3.3.2
Transfer Device	7.2	--	3/4.3.3.2
Connecting Tubing	7.2	--	
Drive Cable		--	3/4.3.3.2
Readout/Control Equipment		--	3/4.3.3.2
Gas Purge System		--	
Leak Detection System		--	
9. Seismic Category I Piping, Raceways, Cables, Hangers, Structures		a,c	
Piping	5.2.4, 5.2.3	a,c	3/4.4.10, 3/4.1.2, 3/4.4.10
Raceways	5.2.4	c	
Cables		c	
Hangers	5.2.4	c	3/4.7.9
Structures		c	
10. Auxiliary Feedwater System	7.4/10.49	g	
Pumps		g	3/4.4.10, 3/4.7.1.2
Motor		g	3/4.7.1.2
Turbine		g	3/4.7.1.2
Valves		g	3/4.4.10, 3/4.7.1.2
Piping		g	3/4.4.10, 3/4.7.1.2
Pipe Supports		g	3/4.7.9
Pipe Restraints		g	3/4.7.9
Condensate Storage Tank		g (not always)	3/4.7.1.3, 3/4.4.10
Automatic Steam Generator Overfill Protection		m	3/4.3.1
Control Air			
11. Emergency Diesel Generators	8.3.1	q	
Diesel Engine		q	3/4.8.1
Alternator		q	3/4.8.1
Starting Air Compressor	9.5.6	q	3/4.8.1
Aftercooler		q	
Air Dryer		q	

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	Standard Review Plan NUREG-0800	Generic[b] Functional Criteria	Standard Technical Specifications
A. RELIED UPON FOR PRESSURE BOUNDARY INTEGRITY, SHUTDOWN AND ACCIDENT MITIGATION (contd)			
11. Emergency Diesel Generators (contd)	8.3.1	q	
Air Receiver		q	
Filters		q	
Valves		q	
Piping		q	
Pipe Supports		q	
Pipe Restraints		q	3/4.7.9
Intake Air Filter	9.5.8	q	3/4.8.1
Silencers		q	
Intercoolers		q	
Ducting		q	
Turbocharger		q	
Exhaust Air Silencer	9.5.8	q	3/4.8.1
Fuel Oil Storage Tank	9.5.4	q	3/4.8.1
Day Tank		q	3/4.8.1
Transfer Pumps	9.5.4	q	3/4.8.1
Filters		q	
Strainers		q	
Piping		q	
Valves		q	
Injector Pumps		q	3/4.8.1
Drain Tank		q	3/4.8.1
Drain Tank Pump		q	3/4.8.1
Intercooler Heat Exchanger		q	3/4.8.1
Jacket Water Heat Exchangers		q	3/4.8.1
Jacket Water Pumps		q	3/4.8.1
Jacket Water Auxiliary Pump		q	3/4.8.1
Lube Oil Cooler		q	3/4.8.1
Valves		q	3/4.8.1
Jacket Water Heaters		q	3/4.8.1
Expansion Tank		q	3/4.8.1
Piping		q	3/4.8.1
Instrumentation		q	3/4.8.1
Lube Oil Pumps	9.5.7	q	3/4.8.1
Auxiliary Lube Oil Pump		q	3/4.8.1
Motor		q	3/4.8.1
Electric Heater		q	3/4.8.1
Filter		q	3/4.8.1
Strainers		q	3/4.8.1
Valves		q	3/4.8.1
Heat Exchan_ rs		q	3/4.8.1
Auxiliary Tank		q	3/4.8.1
Rocker Lube Oil Pump		q	3/4.8.1
Pre-Lube Pump		q	3/4.8.1
Motor		q	3/4.8.1
Reservoir		q	3/4.8.1
Gas Ejector		q	3/4.8.1
Separator		q	3/4.8.1
Sump		q	3/4.8.1
Tubing		q	3/4.8.1
Instrumentation		q	3/4.8.1
12. Station Batteries and Vital Power Supplies)	8.3.2	q	
Battery		q	3/4.8.2
Battery Charger		q	3/4.8.2
Cable		q	3/4.8.2
Breakers		q	

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	Standard Review Plan NUREG-0800	Generic(b) Functional Criteria	Standard Technical Specifications
A. RELIED UPON FOR PRESSURE BOUNDARY INTEGRITY, SHUTDOWN AND ACCIDENT MITIGATION (contd)			
13. Electrical Distribution, Safety Related	8.3.1	q	3/4.8.3
All Components with Safety Function		q	
14. Containment Building		o	
Containment Lines	3.8.1-3	o	3/4.6.1, 3/4.6.1.7
Shield Building		o	3/4.6.1
Primary Shield Wall		o	
Missile Shield		o	
Refueling Cavity		o	
Recirculation Sump		o	
Base Mat		o	3/4.6.1
Relief Valves			3/4.6.7
Tendons		o	3/4.6.1.7
Isolation Valve		o	3/4.6.4, 3/4.6.1.2, 3/4.9.9
Air Locks		o	3/4.6.1.3, 3/4.9.4
15. Containment Isolation System	6.2.4	o	3/4.6.1.4
Cable		o	
Instrumentation		o	
16. Containment Spray System	6.5.2	c	
Containment Spray Pumps		c	3/4.6.2.1, 3/4.4.10
Spray Additive Tank		c	3/4.6.2.2, 3/4.4.10
Piping		c	3/4.4.10
Nozzles		c	3/4.6.2.1
Instrumentation		c	
Valves		c	3/4.4.10
17. Containment Air Cooling System	6.2.2	c	
Fans		c	3/4.6.2.3
Motors		c	
Coolers		c	3/4.6.2.3
Roughing Filters		c	3/4.6.1.9
HEPA Filters		c	3/4.6.1.9
Dampers		c	
Ductwork		c	
Instrumentation		c	
Moisture Separator		c	
Relief Devices		c	
Charcoal Filters		c	3/4.6.4, 3/4/6.1.9
18. Component Cooling Water System	9.2.1	g	
Pumps		g	3/4.7.3, 3/4.4.10
Heat Exchangers		g	3/4/7.3, 3/4.4.10
Surge Tanks		g	3/4.4.10
Valves		g	3/4.4.10, 3/4.7.3
Piping		g	3/4.4.10, 3/4.7.3
Instrumentation		g	
19. Service Water System, Safety Related	9.2.1	g	
Pumps		g	3/4.7.4, 3/4.4.10
Strainers		g	
Piping		g	3/4.4.10, 3/4.7.4
Valves		g	3/4.4.10, 3/4.7.4
Instrumentation		g	
Cooling Towers		g	3/4.7.5

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	Standard Review Plan NUREG-0800	Generic[b] Functional Criteria	Standard Technical Specifications
A. RELIED UPON FOR PRESSURE BOUNDARY INTEGRITY, SHUTDOWN AND ACCIDENT MITIGATION (contd)			
20. Emergency Core Cooling System	6.3	c	
Accumulators		c	3/4.5.1, 3/4.4.10
Boron Injection Tank		c,m	3/4.5.4.1, 3/4.4.10
Refueling Water Storage Tank		c	3/4.5.5, 3/4.4.10
Intermediate Head Injection System		c	3/4.5.2, 3/4.4.10
Low Head Injection System		c	3/4.5.2, 3/4.4.10
High Head Injection System		c	3/4.1.2.2, 3/4.1.2.4, 3/4.5.2, 3/4.1.2.1, 3/4.4.10
Containment Recirculation Sump		c	3/4.5.2
Valves		c	3/4.5.2, 3/4.4.10
Piping		c	3/4.5.2, 3/4.4.10
21. Residual Heat Removal System	5.4.7	d	
Pumps		d	3/4.5.2, 3/4.4.10, 3/4.9.8
Heat Exchangers		d	3/4.5.2, 3/4.4.10, 3/4.9.8
Valves		d	3/4.5.2, 3/4.4.10, 3/4.9.8
Piping		d	3/4.5.2, 3/4.4.10, 3/4.9.8
Instrumentation		d	3/4.3.3.5, 3/4.9.8
22. Chemical and Volume Control System	9.3.4	d,m	
Regenerative Heat Exchanger		--	3/4.4.10
Letdown Heat Exchanger		--	3/4.4.10
Ion Exchangers		--	3/4.4.10
Volume Control Tank		--	3/4.1.2, 3/4.4.10
Primary Water Storage Tank		--	3/4.1.2, 3/4.4.10
Boric Acid Tanks		d,m	3/4.1.2, 3/4.4.10
Boric Acid Batch Tank		d,m	3/4.1.2, 3/4.4.10
Boric Acid Transfer Pumps		d,m	3/4.1.2, 3/4.4.10
Filter		d,m	
Blender		d,m	
Excess Letdown Heat Exchanger		--	3/4.4.10
Valves		d,m	3/4.1.2, 3/4.4.10
Piping		d,m	3/4.1.2, 3/4.4.10
Instrumentation		--	
Positive Displacement Pump		--	3/4.4.10
23. Combustible Gas Control	6.2.5	c	
Post-Accident Hydrogen Venting System		c	3/4.6.5.3
Post-Accident Hydrogen Sampling System		c	3/4.6.5.1
Post-Accident Hydrogen Mixing System		c	3/4.6.5.4
Internal Hydrogen Recombiners		c	3/4.6.5.2
External Hydrogen Recombiners		c	3/4.6.5.2
24. HVAC, Control Room and ESF		c	
Purge and Exhaust System	6.4	c	3/4.7.7
Reactor Containment Fan Cooler System		c	3/4.6.2.3

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	Standard Review Plan NUREG-0800	Generic[(b)] Functional Criteria	Standard Technical Specifications
A. RELIED UPON FOR PRESSURE BOUNDARY INTEGRITY, SHUTDOWN AND ACCIDENT MITIGATION (contd)			
24. HVAC, Control Room and ES[e] (contd)			
Containment Activated Charcoal Filter Units System		c o	3/4.6.1.9, 3/4.6.4
Reactor Cavity and Excore Instrumentation Ventilation System		--	
Control Rod Drive Mechanism Ventilation System		--	
Manipulator Crane Ventilation System		--	3/4.9.12
Pressure Vacuum and Relief System		--	3/4.6.7
Control Room Ventilation System		n	3/4.7.7
25. Instrument Air System	9.3.1	r	
Compressors		r	
After Cooler		r	
Receiver		r	
Dryer/Filter Train		r	
Accumulators		r	
Instrumentation		r	
26. Fuel Pool Structure and Cooling System	9.1.3	d	
Pumps		d	3/4.9.12, 3/4.4.10
Heat Exchanger		d	3/4.9.12, 3/4.4.10
Purification Pumps		--	3/4.9.12, 3/4.4.10
Demineralizer		--	3/4.4.10
Piping		d	3/4.9.12, 3/4.4.10
Strainers, Filters		--	3/4.9.12
Valves		d	3/4.9.12, 3/4.4.10
27. Fire Protection	9.5.1	r	
Pumps		--	3/4.7.11.1, 3/4.7.11.2
Valves		--	3/4.7.11.1, 3/4.7.11.2
Piping		--	3/4.7.11.1, 3/4.7.11.2
Tanks		--	3/4.7.11.1, 3/4.7.11.2
Instrumentation		--	3/4.3.3.8
Halon		--	3/4.7.11.4
CO_2		--	3/4.7.11.3.3
28. Untimate Heat Sink	9.2.5	--	3/4.7.5
NA			
B. FAILURE CAN AFFECT FUNCTIONING OF CATEGORY A SSC			
1. Condensate/Feedwater System, Including Reheat	10.4.7, 10.3.6	f,r	
Main Condenser	10.4.1	--	3/4.4.10
Condensate Pumps		--	3/4.4.10
Demineralizers		--	3/4.4.10
LP Feedwater Heaters		--	3/4.4.10
Piping		--	3/4.4.10
Valves		--	3/4.4.10, 3/4.7.1
Main Feed Pumps		--	3/4.4.10
HP Feedwater Heaters		--	3/4.4.10
Startup Feedwater System		--	3/4.4.10
Heater Drain System		--	3/4.4.10
Condensate Storage and Transfer System		--	3/4.4.10, 3/4.7.1.3

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	Standard Review Plan NUREG-0800	Generic[(b)] Functional Criteria	Standard Technical Specifications
B. FAILURE CAN AFFECT FUNCTIONING OF CATEGORY A SSC (contd)			
2. Turbine, Main Generator, Controls	10.2	f,r,s	
HP Turbine		--	
LP Turbines		--	
Valves		--	
Piping		--	
Gland Steam Condenser		--	
Condenser Exhausters		--	
Regulators		--	
Piping		--	
Valves		--	
Oil Pumps		--	
Oil Reservoir		--	
Oil Coolers		--	
Turning Gear		--	
Ejector		--	
Moisture Separator Reheater		--	
Main Generator		--	
Excitation System		--	
Instrumentation		--	3/4.3.4
3. Main Steam System	10.3	a,f,r,s	
Steam Generator			3/4.7.2, 3/4.4.5, 3/4.4.10
Piping			3/4.4.10
Valves			3/4.4.10, 3/4.7.1
4. Reactor Control System			
Control Rod Drive Mechanism	3.9.4	d,m	3/4.1.3, 3/4.3.3
Logic Cabinet			
Power Cabinet			
Instrumentation			
5. Condenser Cooling System	10.4.5	f,r	
Circulating Water Pumps		--	
Valves		--	
Piping		--	
Condenser		--	
Cooling Towers		--	
6. Instrument/Service Air	9.3.1	r	
Compressors		--	
After Coolers		--	
Air Receivers		--	
Dryer/Filter Train		--	
Instrumentation		--	
C. OTHER SSCs IMPORTANT TO LICENSE RENEWAL			
1. Reactor Post-Accident Monitoring System	7.5	j,k	3/4.3.3.6
Instrumentation			
2. Safety Parameter Display System		k	
Computer	7	--	
Instrumentation		--	

	Standard Review Plan NUREG-0800	Generic[b] Functional Criteria	Standard Technical Specifications
C. OTHER SSCs IMPORTANT TO LICENSE RENEWAL (contd)			
3. Waste Systems: Liquid, Gas, Solid	11.4, 11.2 11.3	p	
Liquid Subsystems			
Solid Subsystems			
Gaseous Subsystems			
4. Fuel Handling Systems		l,p	3/4.9
New Fuel Storage Area			
Spent Fuel Storage Pool			3/4.4.10, 3/4.9.11
Fuel Storage Building Crane			
Spent Fuel Bridge Crane			
New Fuel Elevator			
New Fuel Handling Tool		--	
Spent Fuel Handling Tool		--	
Refueling Cavity			
Transfer Canal			3/4.4.10
Polar Crane			3/4.9.7
Manipulator Crane			3/4.9.6
Red Cluster Assembly Change Fixture		--	
Reactor Vessel Head Lifting Device		--	
Reactor Internals Lifting Device		--	
Stud Tensioner		--	
Refueling Tools		--	
Conveyer Car Assembly			
Drive Frame Assembly			
Lifting Mechanism		--	
Valve			3/4.3.3.1, 3/4.9.2
Instrumentation			
Controls			
5. Radiation and Environmental Monitoring		k	
Containment Air			
Particulate Detector	11.3	--	3/4.3.3.1
Containment Noble Gas Monitor	11.3	--	3/4.3.3.1
Containment Purge Exhaust Monitor	11.3	--	3/4.3.3.1
Auxiliary Building Ventilation System Monitor	11.3	--	3/4.3.3.1
Plant Vent Stack Monitor	11.3	--	3/4.3.3.1
Control Room Air Intake Monitor	11.3, 9.4.1	--	3/4.3.3.1
Condenser Air Ejection Gas Monitor	11.3	--	3/4.3.3.1
Steam Generator Blowdown Liquid Monitor	11.2	--	
Component Cooling Water System Monitor		--	
Service Water Effluent Discharge Monitor		--	
Waste Disposal System Liquid Effluent Monitor		--	
Gas Decay Tank Effluent Gas Monitor		--	
6. Communications Equipment	9.5.2	k	3/4.9.5
Telephone System		--	
Radio System		--	
Page System		--	

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	Standard Review Plan NUREG-0800	Generic[b] Functional Criteria	Standard Technical Specifications
C. OTHER SSCs IMPORTANT TO LICENSE RENEWAL (contd)			
7. Intrusion Detection		--	
Motion Detection System		--	
Sound Monitoring System		--	
Television System		--	
RF Field System		--	
E-Field System		--	
8. Access Control		k	
Door Control System		--	
Badging/ID System		--	
9. Guard Response Support			
Weapons Systems		--	
Communications Systems		--	
10. Alarm Station Operation		j	
Instrumentation		--	
11. Area Radiation Monitors		j,k	3/4.3.3.1
Area Radiation Monitoring System		--	
12. Radiation Survey Instruments			
Radiation Monitoring Systems		--	
13. Personnel Monitoring Devices			
Radiation Detectors	12.3/4	--	3/4.6
14. Personnel Protection Barriers			
Machinery		--	
Structural		--	

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II. BOILING WATER REACTORS

	Standard Review Plan NUREG-0800	Generic Functional Criteria	Standard Technical Specifications
A. RELIED UPON FOR PRESSURE BOUNDARY INTEGRITY, SHUTDOWN AND ACCIDENT MITIGATION			
1. Reactor Coolant Pressure Boundary		a	
Reactor Vessel	5.3.1	a	2.0, 3/4, 4.3
MSIVs	5.3.3, 5.2.3	a	3/4, 4.7
Core Spray Isolation Valves	5.3.3, 5.2.3	a	3/4, 3.2, 4.3
Core Injector Isolation Valves	5.3.3, 5.2.3	a	3/4, 3.2, 4.3
Recirculation Loops	5.3.3, 5.2.3	a	3/4, 3.2, 4.3
CRDM(s)	4.5.1	a	3/4, 3.2, 4.3
Feedwater Isolation Valves	5.3.3, 5.2.3	a	3/4, 3.2, 4.3
Head Spray Isolation Valves	5.3.3, 5.2.3	a	3/4, 3.2, 4.3
2. Reactor Protection System		j,d	
MG Sets	7.2	j,d	3/4, 3.1
Detectors (LPRM, APRM, etc.)	7.2	j,d	3/4, 3.1
Divisions Channels	7.2	j,d	3/4, 3.1
Analog Comparator Units (ACU)	7.2	j,d	3/4, 3.1
A.D. Converters	7.2	j,d	3/4, 3.1
Optical Isolators	7.2	j,d	3/4, 3.1
Logic Circuits	7.2	j,d	3/4, 3.1
Solenoid Control Logic	3.9.4, 7.2	j,d	3/4, 3.1
Scram Air Operated Pilot Valves	3.9.4, 7.2	j,d	3/4, 3.1
Back Up Solenoid Scram Values	3.9.4, 7.2	j,d	3/4, 3.1
Scram Discharge Volume Pilot Valves	3.9.4, 7.2	j,d	3/4, 3.1
3. Control Rod Drive System		--	
Suction Filters	4.5	h	3/4, 1.3, 1.4
Pumps	4.5	h	3/4, 1.3, 1.4
Isolation Valves	4.5.1	a	3/4, 1.3, 1.4
HCUs	3.9.4	d	3/4, 1.3, 1.4
Accumulators	3.9.4	d	3/4, 1.3, 1.4
Scram Discharge Volume	3.9.4	a,d	3/4, 1.3, 1.4
Control Rod	4.6	d	3/4, 1.3, 1.4
4. Standby Liquid Control System		m	
Storage Tank	9.3.5	m	3/4, 1.5
Pumps	9.3.5	m	3/4, 1.5
Squib Valves	9.3.5	m	3/4, 1.5
Neutron Absorption System	9.3.5	m	3/4, 1.5
5. Control Room and Auxiliary Shutdown		d	
Remote S/D Panel	7.4	d	3/4, 3.7
6. Neutron Monitoring System		j,k	
Source Range Monitor	7.1	j,k	3/4, 3.7
Intermediate Range Monitor	7.1	j,k	3/4, 3.7
LPRM/APRM	7.1	j,k	3/4, 3.7

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	Standard Review Plan NUREG-0800	Generic[(b)] Functional Criteria	Standard Technical Specifications
A. RELIED UPON FOR PRESSURE BOUNDARY INTEGRITY, SHUTDOWN AND ACCIDENT MITIGATION (contd)			
7. Seismic Category 1 Piping, Raceways, Cables, Hangers, and Structures to Support Dynamic Loads		a	
Reactor Vessel, System	3.0, 3.10	a,b	3/4, 4.6
Recirculation System	3.0, 3.9, 6, 3.10	a	3/4, 4.6
Main Steam System	3.0, 3.9	a,c	3/4, 4.7
Condensate and Feedwater System	3.0, 3.7	a,g	3/4, 4.4
Automatic Reactor Vessel Overfill Protection	3.0, 3.7	m	3/4.3.1
Reactor Core Isolation Cooling System	5.46	a,g	3/4, 7.3
Reactor Water Cleanup System	5.4.8	a,p	3/4, 4.4
8. Primary Containment		--	
Reactor Building Foundation	3.2.1	o,s	3/4, 6.1, 6.2, 6.3, 6.4, 6.5
Drywell	3.2.1	o	3/4, 6.1, 6.2, 6.3, 6.4, 6.5
Drywell Access Penetrations	3.2.1	o,s	3/4, 6.1, 6.2, 6.3, 6.4, 6.5
Drywell Electrical Penetrations	3.0	o,s	3/4, 6.1, 6.2, 6.3, 6.4, 6.5
Drywell Pipe Penetrations	3.0	o,s	3/4, 6.1, 6.2, 6.3, 6.4, 6.5
Horizontal Vents and Weir Wall	3.0	o	3/4, 6.1, 6.2, 6.3, 6.4, 6.5
Containment	3.0	o	3/4, 6.1, 6.2, 6.3, 6.4, 6.5
Fuel Transfer Tube	9.1	o	3/4, 6.1, 6.2, 6.3, 6.4, 6.5
Suppression Pool	9.0, 3.0	o,g	3/4, 6.1, 6.2, 6.3, 6.4, 6.5
Containment Upper Pool	9.1	o	3/4, 6.1, 6.2, 6.3, 6.4, 6.5
Primary Containment HVAC System	9.4	o	3/4, 6.1, 6.2, 6.3, 6.4, 6.5
Primary Containment Auxiliary System	9.4	o	3/4, 6.1, 6.2, 6.3, 6.4, 6.5
Containment Spray	9.4, 6, 5.2	o,g	3/4, 6.1, 6.2, 6.3, 6.4, 6.5
9. Containment Air Cooling		c	
Drywell Recirculation System	6.2.2	c	3/4, 6.7
Drywell Purge Ventilation System	6.2.2	c	3/4, 6.7
Containment Normal Ventilation System	6.2.2	c	3/4, 6.7
Containment High Flow Purge System	6.2.2	c	3/4, 6.7
Containment Recirculation System	6.2.2	c	3/4, 6.7
10. Hydrogen Control System		c	
Containment Combustible Gas Control System	6.2.5	c	3/4, 6.7
Distributed Igniter System	6.2.5	c	3/4, 6.7
Containment Atmospheric Monitoring System	6.2.5	c	3/4, 6.7

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	Standard Review Plan NUREG-0800	Generic Functional Criteria (b)	Standard Technical Specifications
A. RELIED UPON FOR PRESSURE BOUNDARY INTEGRITY, SHUTDOWN AND ACCIDENT MITIGATION (contd)			
11. Station Batteries and Vital Power Supplies		q	
4.16 KV Switchgear	8.3.1	q	3/4, 8.1, 8.2, 8.3, 8.4
Division 1&2 Diesel Generators	9.5	q	3/4, 8.1, 8.2, 8.3, 8.4
Division 3 Diesel Generators	9.5	q	3/4, 8.1, 8.2, 8.3, 8.4
480 V Switchgear	8.3.1	q	3/4, 8.1, 8.2, 8.3, 8.4
Essential AC Power Supplies	8.3.1	q	3/4, 8.1, 8.2, 8.3, 8.4
Batteries 125, 250, VDC	8.3.2	q	3/4, 8.1, 8.2, 8.3, 8.4
Battery Chargers	8.3.2	q	3/4, 8.1, 8.2, 8.3, 8.4
Nuclear System Protection Separate Divisional Power Supplies	8.1	k	3/4, 8.1, 8.2, 8.3, 8.4
12. EDG (Including Air Storage, Fuel Storage and Transmission and Cooling)		i	
Cooling Water System	9.5.5	c	3/4, 5.1
Lube Oil System	9.5.7	c	3/4, 5.1
Air Compressors	9.5.6	c,i	3/4, 5.1
Air Storage Tanks	9.5.6	c,i	3/4, 5.1
Diesel Engine	9.5.8		3/4, 8.1-8.4
Generator	9.5.8.1		3/4, 8.1-8.4
Intake & Exhaust	9.5.8		3/4, 8.1-8.4
Fuel Oil System	9.5.4		3/4, 8.1-8.4
Instrumentation & Control	9.5		3/4, 8.1-8.4
13. Electrical Distribution - Safety Related		q	
Divisional Power 1,2,3,4,5	8.1	q	3/4, 8.1-8.4
Static Bypass Switch	8.1	q	3/4, 8.1-8.4
Inverter	8.1	q	3/4, 8.1-8.4
14. Reactor Core Isolation Cooling System (Including Isolation Condenser)		a,c,e,g	
Steam Isolation Valves	5.4.6	a,c,e	3/4, 7.3
Steam Flow Elements	5.4.6	a,c,e	3/4, 7.3
Turbine Trip Throttle Valve	5.4.6	a,c,e	3/4, 7.3
Turbine Governor Valve	5.4.6	a,c	3/4, 7.3
Turbine	5.4.6	a,c	3/4, 7.3
Turbine Oil System	5.4.6	c	3/4, 7.3
Gland Seal Elements	5.4.6	g,c	3/4, 7.3
Exhaust Piping	5.4.6	g,c	3/4, 7.3
Suction Strainer	5.4.6	g,c	3/4, 7.3
Suction Valves	5.4.6	g,c	3/4, 7.3
Water Log Pump	5.4.6	g,c	3/4, 7.3

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	Standard Review Plan NUREG-0800	Generic(b) Functional Criteria	Standard Technical Specifications
A. RELIED UPON FOR PRESSURE BOUNDARY INTEGRITY, SHUTDOWN AND ACCIDENT MITIGATION (contd)			
14. Reactor Core Isolation Cooling System (Including Isolation Condenser) (contd)		a,c,e,g	
RCIC Pump	5.4.6	g,c	3/4, 7.3
Auxiliary Equipment Cooling	5.4.6	g,c	3/4, 7.3
Minimum Flow Bypass Line	5.4.6	g,c	3/4, 7.3
Test Recirculator Line	5.4.6	g,c	3/4, 7.3
Testable Check Valve	5.4.6	g,c	3/4, 7.3
Flow Controller	5.4.6	g,c	3/4, 7.3
15. High Pressure Cooling Injection System		g,c	
Suction Path	6.3	g,c	3/4, 5.1
HPCS(I) Pump	6.3	g,c	3/4, 5.1
Discharge Path	6.3	g,c	3/4, 5.1
HPCS(I) Water Leg Pump	6.3	g,c	3/4, 5.1
Leak Detection System	6.3	g,c	3/4, 5.1
Valve Interlock	6.3	g,c	3/4, 5.1
16. Automatic Depressurization System		a	
Safety/Relief Valves	6.0	a	3/4, 5.1
Air Supply	6.0	--	3/4, 5.1
Vacuum Breakers	6.0	--	3/4, 5.1
17. Core Spray System or Low Pressure Injection System		--	
Suction Path	5.4.7	c	3/4, 5.1
LPCS Pump	5.4.7	c	3/4, 5.1
Discharge Path	5.4.7	c	3/4, 5.1
LPCS Water Leg Pump	5.4.7	c	3/4, 5.1
LPCI System	5.4.7	c	3/4, 5.1
18. Reactor Circulation System		a,h	
Recirculation Loop Suction Piping	5.2.3	a,h	3/4, 4.1
Suction Isolation Valve	5.2.3	a,h	3/4, 4.1
Recirculation Pumps	5.4	a,h	3/4, 4.1
Recirculation Pump Shaft Seals	5.4	a,h	3/4, 4.1
Recirculation Pump Discharge Piping	5.4	a,h	3/4, 4.1
Flow Control Valve	5.4	a,h	3/4, 4.1
Discharge Isolation Valve	5.4	a,h	3/4, 4.1
Reactor Water Sample Connection	5.4	a,h	3/4, 4.1
Discharge Manifold and Risers	4.5.2	a,h	3/4, 4.1
Jet Pumps	4.5.2	a,h	3/4, 4.1
Recirculation Pump Motors	5.4	a,h	3/4, 4.1
Low Frequency Motor Generator Sets	5.4	h	3/4, 4.1

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	Standard Review Plan NUREG-0800	Generic[(b)] Functional Criteria	Standard Technical Specifications
A. RELIED UPON FOR PRESSURE BOUNDARY INTEGRITY, SHUTDOWN AND ACCIDENT MITIGATION (contd)			
19. Residual Heat Removal System (Including Drywell Spray)		c	
Suction Strainers	5.4.7	c,d,g	3/4, 4.9, 4.1
RHR Water Leg Pump	5.4.7	c,d,g	3/4, 4.9, 4.1
RHR Pumps	5.4.7	c,d,g	3/4, 4.9, 4.1
RHR Heat Exchangers	5.4.7	c,d,g	3/4, 4.9, 4.1
Motor Operated Valves	5.4.7	c,d,g	3/4, 4.9, 4.1
Testable Check Valves	5.4.7	c,d,g	3/4, 4.9, 4.1
Containment Spray Spargers	5.4.7	c,d,g	3/4, 4.9, 4.1
Air Operated Control Valves	5.4.7	d	3/4, 4.9, 4.1
Electro Pneumatic Controllers	5.4.7	d	3/4, 4.9, 4.1
20. RHR/Shutdown Service Water System		--	
Ultimate Heat Sink Basin and Towers	9.2.1, 9.2.5	g	3/4, 5.1, 7.1
Standby Service Water Pumps	9.2.1	g	3/4, 5.1, 7.1
Heat Exchangers	9.2.5	g	3/4, 5.1, 7.1
21. Emergency Equipment Cooling		c	
Heat Exchangers	9.2.1	c	3/4, 7.1
Closed Cooling Water System	9.2.1	c	3/4, 7.1
22. HVAC-Control Room and ESF		n,r	
Supply Air Handling Units	6.4	n,r	3/4, 7.2
Recirculation Fans	6.4	n,r	3/4, 7.2
Makeup Air Cleaning Units	6.4	n,r	3/4, 7.2
23. Instrument Air--Important to Safety		n,r	
Service and Instrument Air Compressors	9.3.1	n,r	N/A
Air Receiver	9.3.1	n,r	N/A
Refrigeration Air Dryers and After Filters	9.3.1	n,r	N/A
Dessicant Air Dryers	9.3.1	n,r	N/A
Booster Instrument Air Compressors	9.3.1	n,r	N/A
24. Fuel Pool Structure and Cooling System		d,l	
Skimmers Weifs and Scuppers	9.1	d,l	3/4, 9.1-9.12
FPCC Drain Tank	9.1	d,l	3/4, 9.1-9.12
FPCC Pumps	9.1.2	d,l	3/4, 9.1-9.12
FPCC Heat Exchangers	9.1.2	d,l	3/4, 9.1-9.12
Filter/Demineralizers	9.1.2	d,l	3/4, 9.1-9.12
Diffusers	9.1.2	d,l	3/4, 9.1-9.12
Spent Fuel Pool	9.1.3	d,l	3/4, 9.1-9.12
Transfer Pool	9.1.3	d,l	3/4, 9.1-9.12

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	Standard Review Plan NUREG-0800	Generic[b] Functional Criteria	Standard Technical Specifications
A. RELIED UPON FOR PRESSURE BOUNDARY INTEGRITY, SHUTDOWN AND ACCIDENT MITIGATION (contd)			
25. Fire Protection (Including Suppression)		r	
Fresh Water Supplies	9.5.1	r	3/4, APPENDIX "R"
Fire Water Supplies	9.5.1	r	3/4, APPENDIX "R"
Fire Jockey Pumps	9.5.1	r	3/4, APPENDIX "R"
Fire Main	9.5.1	r	3/4, APPENDIX "R"
Manual Hose Station	9.5.1	r	3/4, APPENDIX "R"
Preaction Type Sprinkler System	9.5.1	r	3/4, APPENDIX "R"
Deluge Type Sprinkler System	9.5.1	r	3/4, APPENDIX "R"
Wet Pipe Type System	9.5.1	r	3/4, APPENDIX "R"
High Pressure CO_2	9.5.1	r	3/4, APPENDIX "R"
Low Pressure CO_2	9.5.1	r	3/4, APPENDIX "R"
Halon System	9.5.1	r	3/4, APPENDIX "R"
Heat Detection Systems	9.5.1	r	3/4, APPENDIX "R"
Smoke Detectors	9.5.1	r	3/4, APPENDIX "R"
Flame Detectors	9.5.1	r	3/4, APPENDIX "R"
26. Ultimate Heat Sink		r	
Circulating Water Pumps	9.2.5	r	3/4, 7.1
Cooling Towers	9.2.5	r	3/4, 7.1
Main Condenser	9.2.5	r	3/4, 7.1
Major Valves	9.2.5	r	3/4, 7.1
Major Piping	9.2.5	r	3/4, 7.1
B. FAILURE CAN AFFECT FUNCTIONING OF CATEGORY A SSC			
1. Condensate/Feedwater System Including Reheat	10.4.7	r	4.4
2. Turbine-Generator and Controls	10.2	r,s	3/4, 3.8
3. Main Steam System	10.3	r,s	3/4, 4.7
4. Reactor Control System	7.1	d,m	3/4, 4.1
5. Condenser Cooling System (Circulation Water System)	10.4.5	r	N/A
6. Instrument Air/Service Air, Not S.R.	9.3.1	r	N/A
7. Switchyard	8.2	r	3/4, 8.1-8.4
C. OTHER SSCs IMPORTANT TO LICENSE RENEWAL			
1. Reactor Post-Accident Monitoring System	9.3.2	j,k	3/4, 3.1-3.9
Instrumentation	9.3.2	k	3/4, 3.1-3.9
2. Safety Parameter Display System		k	3/4, 3.1-3.9
Computer	7.1	--	3/4, 3.1-3.9
Instrumentation	7.1	--	3/4, 3.1-3.9
3. Waste Systems: Liquid, Gas, Solid	11.0	p	3/4, 11.1-11.4
Liquid Subsystems	11.0	--	3/4, 11.1-11.4
Solid Subsystems	11.0	--	3/4, 11.1-11.4
Gaseous Subsystems	11.0	--	3/4, 11.1-11.4

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	Standard Review Plan NUREG-0800	Generic[b] Functional Criteria	Standard Technical Specifications
C. OTHER SSCs IMPORTANT TO LICENSE RENEWAL (contd)			
4. Fuel Handling Systems	9.0	l,p	3/4, 9.1-9.12
New Fuel Storage Area	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
Spent Fuel Storage Pool	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
Fuel Storage Building Crane	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
Spent Fuel Bridge Crane	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
New Fuel Elevator	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
New Fuel Handling Tool	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
Spent Fuel Handling Tool	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
Refueling Cavity	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
Transfer Canal	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
Polar Crane	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
Manipulator Crane	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
Rod Cluster Assembly Change	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
Fixture	9.1.3	--	3/4, 9.1-9.12
Reactor Vessel Head Lifting	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
Device	9.1.3	--	3/4, 9.1-9.12
Reactor Internals Lifting	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
Device	9.1.3	--	3/4, 9.1-9.12
Stud Tensioner	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
Refueling Tools	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
Conveyer Car Assembly	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
Drive Frame Assembly	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
Lifting Mechanism	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
Valve	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
Instrumentation	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
Controls	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
5. Radiation and Environmental Monitoring		k	
Containment Air	12.1-12.5		3/4, 3.7
Particulate Detector	12.1-12.5	--	3/4, 3.7
Containment Noble Gas Monitor	12.1-12.5	--	3/4, 3.7
Containment Purge Exhaust Monitor	12.1-12.5	--	3/4, 3.7
Auxiliary Building Ventilation System Monitor	12.1-12.5	--	3/4, 3.7
Plant Vent Stack Monitor	12.1-12.5	--	3/4, 3.7
Control Room Air Intake Monitor	12.1-12.5	--	3/4, 3.7
Condenser Air Ejection Gas Monitor	12.1-12.5	--	3/4, 3.7

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	Standard Review Plan NUREG-0800	Generic(a) Functional Criteria	Standard Technical Specifications
C. OTHER SSCs IMPORTANT TO LICENSE RENEWAL (contd)			
5. Radiation and Environmental Monitoring (contd)		k	
Steam Generator Blowdown Liquid Monitor	12.1-12.5	--	3/4, 3.7
Component Cooling Water System Monitor	12.1-12.5	--	3/4, 3.7
Service Water Effluent Discharge Monitor	12.1-12.5	--	3/4, 3.7
Waste Disposal System Liquid Effluent Monitor	12.1-12.5	--	3/4, 3.7
Gas Decay Tank Effluent Gas Monitor	12.1-12.5	--	3/4, 3.7
6. Communications Equipment	9.5.2	k	N/A
Telephone System	9.5.2	--	N/A
Radio System	9.5.2	--	N/A
Page System	9.5.2	--	N/A
7. Intrusion Detection	13.6	--	N/A
Motion Detection System	13.6	--	N/A
Sound Monitoring System	13.6	--	N/A
Television System	13.6	--	N/A
RF Field System	13.6	--	N/A
E-Field System	13.6	--	N/A
8. Access Control	13.6	k	N/A
Door Control System	13.6	--	N/A
Badging/ID System	13.6	--	N/A
9. Guard Response Support			
Weapons Systems	13.6	--	N/A
Communications Systems	13.6	--	N/A
10. Alarm Station Operation		j	
Instrumentation	13.6	--	N/A
11. Area Radiation Monitors		j,k	
Area Radiation Monitoring System	12.0	--	3/4, 3.7
12. Radiation Survey Instruments			
Radiation Monitoring Systems	12.0	--	3/4, 3.7
13. Personnel Monitoring Devices			
Radiation Detectors	12.0	--	3/4, 3.7
14. Personnel Protection Barriers			
Machinery	13.6	--	N/A
Structural	13.6	--	N/A

DRAFT NUREG 1299

STANDARD REVIEW PLAN

License Renewal

U.S. Nuclear Regulatory
Commission

Office of Nuclear Reactor Regulation

August 1990

Note: This draft SRP-LR is based upon the proposed license renewal rule, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," 10 CFR 54 (Federal Register, Vol. 55, No. 137, July 17, 1990). Future modifications to the proposed rule will be reflected in commensurate changes in the draft SRP-LR.

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STANDARD REVIEW PLAN FOR LICENSE RENEWAL (SRP-LR)

PART A: GENERAL INFORMATION AND DISCUSSION

I. INTRODUCTION

A. BACKGROUND

The Nuclear Regulatory Commission (NRC) regulations contained in Title 10 of the Code of Federal Regulations (10 CFR) have been supplemented by the addition of Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants." The requirements stated in 10 CFR Part 54 are based on the following two important principles:

1. Except for age-related degradation, the current licensing basis for each operating nuclear power plant provides and maintains an acceptable level of safety for operation during any renewal period. This principle is founded on the Commission's initial finding of adequate protection for the initial design and construction of a plant, as well as in the Commission's continuing oversight and regulatory actions for these plants.

2. A plant's current licensing basis must be maintained during the renewal period, in part through a program to manage age-related degradation of systems, structures, and components (SSCs) that are important to license renewal. This principle is a necessary complement to the first principle. The Commission has already made a generic finding for all nuclear power plants that the reasonable assurance findings for issuance of an operating license continue to be true at the time of the renewal application and accordingly need not be made at the time of license renewal. Therefore, the focus of 10 CFR Part 54 is on age-related concerns requiring license renewal applicants to take the necessary actions to provide assurance that age-related degradation will be effectively managed so that the plant will continue to meet an acceptable level of safety during the renewal term.

Given the above principles, the Standard Review Plan for License Renewal (SRP-LR) is based on the staff position that reasonable assurance must be provided to demonstrate that license renewal will not lead to age-related degradation that would reduce the level of safety at an operating nuclear power plant below the level established by the current licensing basis as defined in 10 CFR 54.3(a).

B. PURPOSE AND SCOPE OF SRP-LR

The SRP-LR has been prepared as guidance to staff reviewers in the Office of Nuclear Reactor Regulation for performing safety reviews of applications to renew operating licenses in accordance with the requirements set forth in 10 CFR Part 54. The SRP-LR parallels Regulatory Guide DG-1009, "Standard Format and Content of Technical Information for Applications to Renew Nuclear Power Plant Operating Licenses," which has been prepared to provide guidance to license renewal applicants on how to structure and present the technical information to be compiled, including the information to be submitted, as part of an application for license renewal.

Regulatory Guide DG-1009 provides guidelines for the following: (1) specific format and content of technical information to be included in license renewal applications; (2) criteria for selecting SSCs important to license renewal and structures and components (SCs) requiring evaluation of age-related degradation; (3) design, operational, and environmental factors that contribute to age-related degradation; (4) aging mechanisms and degradation sites; and (5) attributes of established, effective programs for understanding and managing age-related degradation.

The primary purpose of the SRP-LR is to ensure the quality and uniformity of staff reviews and to ensure that these reviews are focused on the license renewal concerns described in 10 CFR Part 54. It is also the intent of the SRP-LR to make information about regulatory matters widely available and to improve communication between the NRC, interested members of the public, and the nuclear power industry and increase understanding of the review process. Specifically, it provides guidance to the staff regarding items that should be reviewed and provides acceptance criteria to help the reviewer evaluate the information submitted as part of the license renewal application as specified in 10 CFR 54.17, 54.19, and 54.21. Guidance in the SRP-LR represents approaches that are acceptable to the staff, but licensees are not required to conform with this guidance. If a licensee proposes new or different approaches, the staff is likely to require more time and effort to complete the review. The specific technical information that must be submitted as part of a license renewal application is described in Regulatory Guide DG-1009 and in Section II, "Requirements of the License Renewal Rule," which follows. Review criteria for environmental concerns to satisfy 10 CFR 54.23 will be addressed as part of revisions to 10 CFR Part 51.

The staff review of an application for license renewal is not intended to be a review of the current licensing basis. Therefore, guidance provided in the SRP-LR differs from that provided in NUREG-0800, "Standard Review Plan for the

Review of Safety Analysis Reports for Nuclear Power Plants." The emphasis of the SRP-LR is to provide guidance on how to evaluate those programs and processes that the license renewal applicant utilizes or will utilize in managing age-related degradation of selected SSCs important to license renewal so as to maintain the plant's licensing basis throughout the renewal term requested in the application.

The manner in which the staff applies the SRP-LR can vary from plant to plant and within a single plant for different areas. In some cases, the staff may be able to complete some portions of the review on a generic basis; in other cases, the staff may need to review plant-specific features. The staff need not review every step for every license renewal application in detail, but may select and emphasize particular aspects of each SRP-LR section as appropriate for the application.

The SRP-LR is part of a continuing regulatory standards development activity that documents current methods of review and provides the basis for orderly modifications to the review process. It will be revised periodically, as needed, to clarify content, correct errors, and incorporate modifications approved by the Director of the Office of Nuclear Reactor Regulation. This version of the SRP-LR is a draft document which serves as a frame on which to produce a more detailed document. The SRP-LR is a living document to be revised based upon experience gained during the review of the lead plant applications and industry technical reports. Comments and suggestions for improvement will be considered and should be sent to the Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC, 20555. Notice of errors or omissions should also be sent to the same address.

C. ORGANIZATION OF SRP-LR

The SRP-LR is divided into three parts. Part A provides the general context for the purpose and scope of the SRP-LR. In addition, Section II of Part A briefly describes the integrated plant assessment as required in 10 CFR Part 54, which is a main focus of staff review of the license renewal application. Section III assigns review responsibilities and describes the criteria for making the preliminary determination of whether or not a license renewal application is sufficient before conducting the detailed technical review. Appendix A.1 to Part A provides review guidance for the generic requirements in the first two steps of the integrated plant assessment as required in 10 CFR 54.21(a). These two steps involve identifying SSCs important to license renewal and selecting specific SCs from the initial list of SSCs that need to be evaluated for age-related degradation. Appendix A.1 is organized into subsections similar to those described below for Parts B and C.

Part B is organized on a plant system basis. The first section (B.0.1) covers generic license renewal review guidance for systems in general. The remaining sections (B.1.1 through B.6.4) provide additional guidance to the staff for particular systems within the plant that may need to be evaluated to determine whether or not the licensee has established an effective program and what actions might be needed to manage age-related degradation during the renewal term.

The generic and system-specific sections in Part B focus on providing guidance to the staff for reviewing systems important to license renewal to determine if, in fact, the applicant has established an effective program to manage aging of the SCs in the system. If no such program exists, the reviewer needs to evaluate whether or not the actions the applicant has taken, or intends to take, would ensure that the degradation of the SCs important to license renewal due to aging will not impair a system's own safety function and that age-related degradation of SCs within the boundaries of the system will not prevent the intended safety function of any other SSC important to license renewal. The systems in Part B refer to the appropriate component section in Part C for guidance for review of types or classes of some generic components and structures.

Each section in Part B is organized in the same manner as NUREG-0800. The major subsections consist of the following:

- o Review Responsibilities: This subsection assigns primary and secondary staff responsibilities for the review of a license renewal application.

- o Areas of Review: This subsection describes the scope of review by the staff having primary review responsibility. It contains a description of the system and basic degradation mechanisims that must be evaluated as part of the particular system being reviewed.

- o Acceptance Criteria: This subsection identifies the NRC requirements that are applicable and the technical bases for determining the acceptability of the programs within the scope of the area of review of the SRP-LR section. The technical bases consist of specific criteria such as those given in 10 CFR Part 54, regulatory guides, and industry codes and standards.

  - o Review Procedures: This subsection discusses how the review is performed. It consists generally of a step-by-step procedure that the reviewer utilizes to conclude with reasonable assurance that the applicable acceptance criteria have been met. It also contains a discussion of the information needed or the review expected from other supporting staff to permit the staff having primary review responsibility to complete its review.

  - o Findings: This subsection presents the type of conclusions that are drawn from the review. The conclusions are included in the staff's safety evaluation report (SER), which documents the results of the review.

- o Implementation: This subsection states that unless the licensee proposes an alternative method for complying with specified portions of the SRP-LR, the staff will conduct its evaluation according to the methods described in SRP-LR.

- o Genera Information: This subsection contains supplemental information that may assist the staff during the review process. It may contain information on recent technological advances in the assessment of age-related degradation and in surveillance, monitoring, and inspection techniques. It may also address results obtained from the Nuclear Plant Aging Research Program (NPAR) that may be useful in understanding aging mechanisms and ways for managing aging for a particular system during the period of license renewal. The general information section serves solely as background material and should not be used as review criteria or review procedures.

- o References: This subsection lists the references considered useful in the review process.

Part C deals with generic classes of SCs that are constituent elements of the systems addressed in Part B. The organization of Part C parallels the structure of Part B.

The approach taken in structuring the detailed review process in the SRP-LR, first by system and then by components and structures common to various systems, arises out of a need for an integrated systems analysis of age-related degradation. Figure A-1 is a flow chart showing how a systems-level review provides assurance that the evaluation findings and their implementation will be consistent with the requirements of 10 CFR Part 54.

II. REQUIREMENTS OF THE LICENSE RENEWAL RULE

A key element of 10 CFR Part 54 is the requirement for license renewal applicants to perform and submit an integrated plant assessment which demonstrates that age-related degradation of the facility's SSCs has been identified, evaluated, and accounted for as needed to ensure that the facility's licensing basis will be maintained throughout the term of the renewed license.

The technical information required by 10 CFR Part 54 will be documented in the supplement to the final safety analysis report (FSAR) that is submitted as part of the license renewal application. The FSAR supplement will include the results and technical bases for the integrated plant assessment that is illustrated in Figure A-2. The four major steps of the integrated plant assessment are as follows:

1. Identify the SSCs important to license renewal using the definition in 10 CFR 54.3. The SCs that are constituent elements of the SSCs important to license renewal included in the initial list are also identified.

2. Select the SCs requiring evaluation of age-related degradation by identifying those SCs that contribute to the safety functions of an SSC important to license renewal and those SCs whose failure can interfere with the performance of a safety function of an SSC important to license renewal. The term "safety function" in 10 CFR Part 54 refers to any function that causes an SSC to be identified as important to license renewal. This definition is not limited to the narrow definition of a safety function associated with safety-related equipment, but includes certain other functions, such as those associated with non-safety-related SCs and post-accident monitoring equipment.

3. Determine which SCs identified in step 2 above are subject to established, effective programs for managing age-related degradation as defined in 10 CFR 54.3a. This determination may be made either through a detailed mechanistic analysis or through evaluation of operational experience and other data pertaining to aging.

4. For those SCs that are not part of an established, effective program for managing aging, the applicant for license renewal must determine if aging is significant to the plant's current licensing basis, and if not, then such finding must be demonstrated by evaluation. If aging is found to be significant, then the applicant must describe and provide the basis for actions taken or to be taken to manage aging.

As indicated in the second step of the integrated plant assessment, 10 CFR Part 54 requires the applicant for license renewal to list those SCs important to license renewal that should be further evaluated for aging, and to focus the evaluation or whether or not the applicant has established an effective program for these SCs.

However, a systems approach is essential for reviewing the licensee's integrated plant assessment. The system-specific environmental and operating conditions may contribute to aging mechanisms and the degradation process; hence, any established program for a structure or component has also to be evaluated in terms of its effect on the whole system.

Appendix A.1 provides more detailed standard review criteria for the process of identifying SCs that need to be evaluated for age-related degradation.

III. SUFFICIENCY OF LICENSE RENEWAL APPLICATION

Title 10 CFR 2.109 allows the licensee's current license to remain in effect during the time that the staff is reviewing the license renewal application, provided the licensee files a sufficient application at least 3 years before the current license is scheduled to expire. Although the staff expects to complete its review of a renewal application in a timely manner before the existing license expires, in some instances it may take more time to fully evaluate the technical adequacy of the application and make a final determination on the application. The staff has to perform an initial review of the application in order to determine if the application is sufficient to commence the detailed review. Therefore, one of the first actions to be taken on an application for license renewal is to determine if it is "sufficient" so that the timely renewal provision of 10 CFR 2.109 applies.

A. REVIEW RESPONSIBILITIES

The License Renewal Project Directorate (LRPD) Office of Nuclear Reactor Regulation, has primary responsibility for determining the sufficiency of an application for license renewal.

B. REVIEW PROCEDURES

Regulatory Guide DG-1009 provides guidance to the licensee on the content of the technical information that should be submitted as part of the license renewal application. The general checklist that the project manager should use as guidance in determining sufficiency is shown in Figure A-3. This checklist is consistent with the guidelines in Regulatory Guide DG-1009.

APPLICATION RECEIVED

APPLICATION SUFFICIENT TO COMMENCE DETAILED REVIEW

REVIEW OF METHODOLOGY

REVIEW FROM SYSTEMS PERSPECTIVE

REVIEW FROM COMPONENT AND STRUCTURE PERSPECTIVE

INTEGRATION INTO COMPOSITE SAFETY EVALUATION REPORT

Figure A-2. Overall perspective of the review process

NOTES:

1. LRPD is the primary reviewer listed in all sections of the SRP-LR. However, the staff anticipates that significant review responsibility will be assigned to the secondary reviewer(s) with LRPD performing the coordinating functions. All reviewers will follow the guidance contained in the SRP-LR.

2. The review of methodology includes the following:

   a. The methodology for identifying systems, structures, or components (SSCs) important to license renewal (ILR) and requiring further evaluation of age-related degradation is reviewed.

   b. The methodology for determining the effectiveness of established programs to monitor and manage age-related degradation is reviewed.

3. The review from the systems perspective includes the following:

   a. All appropriate systems are identified.

   b. All structures and components (SCs) ILR within the system boundary are identified.

   c. Appropriate environmental and operating conditions ILR are identified.

4. The review from the component perspective includes the following:

   a. Established effective programs (EEPs) are identified and justified, or

   b. New programs to identify and manage age-related degradation are identified and justified, or

   c. Justification that age-related degradation is not significant during the renewal term is submitted.

   d. Justification on continuing reliefs or exemptions previously granted which may be affected by age-related degradation concerns is submitted.

Figure A-2. Integrated plant assessment for license renewal

CHECKLIST FOR REVIEWING SUFFICIENCY OF APPLICATION FOR LICENSE RENEWAL

The reviewer should use this checklist to verify that the basic requirements of 10 CFR Part 54 have been addressed in the license renewal application. A check in the "No" column without sufficient explanation may be justification for rejecting the application.

	Yes	No
1. The application for renewed license is filed in accordance with: [54.17(a)]		
A. 10 CFR Part 2, Subpart A	____	____
B. 10 CFR 50.4	____	____
C. 10 CFR 50.30	____	____
2. Applicant is eligible to apply for a license [54.17(c)]	____	____
3. Application is within specified time frame [54.17(d)] [2.109(a)(b)]	____	____
4. References to information contained in previous applications are clear and specific [54.17(f)]	____	____
5. Restricted data agreement is present [54.17(g)]	____	____
6. Information specified in 50.33(a) through (e), (h), and (i) is provided or referenced [54.19]	____	____
A. Name of applicant	____	____
B. Address of applicant	____	____
C. Business description	____	____
D. Citizenship or corporation	____	____
1. Where incorporated	____	____

Figure A-3 Checklist for Review of a License Renewal Application for Sufficiency

2. Board of directors and principal officers ______ ______

3. Ownership details ______ ______

E. Class of license ______ ______

F. Construction/alteration dates ______ ______

G. Regulatory agencies with jurisdiction ______ ______

7. A statement is provided that summarizes how and the extent to which the application meets the regulatory requirements for license renewal [10 CFR Part 54] ______ ______

8. Implementation plan contains: ______ ______

A. Summary of commitments ______ ______

B. Description of administrative controls ______ ______

C. Task and schedule (detail of commitments that will be completed following the renewal of the operating license) ______ ______

9. FSAR supplement includes an evaluation of aging mechanisms and a demonstration that the effects of degradation will be effectively managed throughout the renewal term [54.21]. The FSAR supplement includes:

A. Integrated plant assessment [54.21(a)] ______ ______

B. Methodology to identify all SSCs important to license renewal [54.21(a)(1)] ______ ______

C. List of all SSC's important to license renewal [54.21(a)(1)] ______ ______

D. List of structures and components (SCs) that are constituent elements of SSCs important to license renewal (optional) [54.21(a)(2)] ______ ______

Figure A-3 (continued)

E. Methodology, including selection criteria, to identify those SCs from 9.D that contribute to the performance of the safety function of an SSC important to license renewal or whose failure could prevent an SSC from performing its intended safety function [54.21(a)(2)] ____ ____

F. List of SCs identified in 9.E [54.21(a)(2)]. ____ ____

G. Methodology to identify those SCs from 9.F that are subject to an established effective program as defined in 54.3(a) [54.21(a)(3)] ____ ____

H. List of those SCs identified in 9.G [54.21(a)(3)] ____ ____

I. Established effective programs for the SCs in 9.H [54.21(a)(3)] ____ ____

J. Information of the following type for SCs on the list from 9.F and not on the list from 9.H [54.21(a)(4)]: ____ ____

1. Describe and provide basis for action taken to manage age-related degradation; or ____ ____

2. Describe and provide basis for action to be taken to manage age-related degradation; or ____ ____

3. Demonstrate by evaluation that the age-related degradation is not significant with respect to the current licensing basis ____ ____

10. List of all plant-specific exemptions granted pursuant to 10 CFR 50.12 and reliefs granted pursuant to 10 CFR 50.55(a)(3) is provided [54.21(b)] ____ ____

11. Justification is provided for continuing those exemptions and reliefs granted on the basis of an assumed service life or period of operation bound by the original license term or otherwise related to SSCs subject to age-related degradation [54.21(b)] ____ ____

Figure A-3 (continued)

12. Description is provided of proposed plant modifications to the facility or its administrative control procedures resulting from analysis or evaluations of Section 54.21 [54.21(c)] ____ ____

13. Description is provided of additions or other changes to the Technical Specifications, when applicable, including technical bases for these changes that will account for the modifications to the plant design, age-related degradation, or limitations on plant operations during the renewal term ____ ____

14. List of documents is provided identifying portions of CLB that are relevant to the integrated plant assessment and a brief description of the administrative controls for and location of these documents is provided [10 CFR 54.37] ____ ____

15. Environmental report complies with the requirements of subpart A of 10 CFR Part 51 ____ ____

Figure A-3 (continued)

APPENDIX A.1

SELECTION OF STRUCTURES AND COMPONENTS FOR EVALUATION OF AGE-RELATED DEGRADATION

This appendix covers the portion of the integrated plant assessment related to selection of structures and components (SCs) for detailed evaluation of age-related degradation. The objective of the integrated plant assessment is to demonstrate that age-related degradation of the facility's systems, structures, and components (SSCs) has been identified, evaluated, and accounted for as needed to ensure that the facility's licensing basis will be maintained throughout the term of the renewed license. The evaluation of aging mechanisms, degradation sites, and root causes is an important part of this process, but not every SSC in the facilityis evaluated. Rather, a subset of SCs is selected from the set of all SSCs using a process described in the rule, and a detailed evaluation of aging is performed only for these SCs. This selection process, which is described in the first two steps of the integrated plant assessment, 10 CFR 54.21(a)(1) and 54.21(a)(2), is illustrated in Figure A-2 from Part A of SRP-LR. Review guidance to the staff is provided below.

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - All technical review branches

I. AREAS OF REVIEW

The licensee should submit several lists of systems, structures, and components as part of the license application: a list of SSCs important to license renewal, a list of SCs that are constituent elements of the SSCs important to license renewal, and a list of the SCs requiring further evaluation of age-related degradation.

The next section provides guidance to the staff in reviewing the methodologies for preparaing the necessary lists.

II. ACCEPTANCE CRITERIA

The acceptability of the licensee's selection of SSCs important to license renewal and of SCs requiring evaluation of age-related degradation will be based on criteria in the following three categories:

A. SCs IMPORTANT TO LICENSE RENEWAL

The licensee has submitted a list of SSCs important to license renewal and has described the methodology for identifying these SSCs. If the licensee has used the methodology described in Regulatory Guide DG-1009 or a method that was previously approved and documented in an NRC safety evaluation report (SER), the standard methodology can be referenced with enough supporting information to demonstrate that the methodology has been correctly applied and its limitations understood. If a new methodclogy has been used, the licensee should provide specific detail concerning the selection of SSCs.

B. SCs THAT ARE CONSTITUENT ELEMENTS OF THE SSCs IMPORTANT TO LICENSE RENEWAL

The licensee has submitted a list of SCs that are constituent elements of the SSCs important to license renewal and has described the methodology used for preparing this list. The description of the methodology should include the means of identifying system boundaries from which the list of SCs is derived.

C. SCs REQUIRING FURTHER EVALUATION OF AGE-RELATED DEGRADATION

The licensee has submitted a list of SCs from Item II.B above that have one or both of the following characteristics:

- The structure or component contributes to the performance of a safety function of an SSC important to license renewal or

- The failure of the structure or component could prevent an SSC important to license renewal from performing its intended safety function.

The licensee also has described the methodology, including selection criteria, used to assess the relevance of each SC to safety functions of SSCs important to license renewal.

The term "safety function" in 10 CFR Part 54 refers to any function that causes an SSC to be identified as important to license renewal. This definition is not limited to the narrow definition of a safety function associated with safety-related equipment, but includes certain other functions, such as those associated with non-safety-related equipment and post accident monitoring equipment.

III. REVIEW PROCEDURES

The reviewer should adher to the following review procedures to assess the adequacy of the licensee's methodology for identifying SSCs important to license renewal and SCs to be evaluated in detail and to provide reasonable assurance that the methodology has been appropriately applied. NRC will focus its review differently for licensees referencing methodologies that have already been approved by the NRC than for licensees proposing new methodologies.

If the NRC has already issued an SER on the proposed approach, the principal responsibility of the primary reviewer is to confirm that the implementation of the methodology falls within the bounds and conditions specified in the SER. This may involve auditing selected aspects of the analysis to ensure that it is carried out in an appropriate manner and carefully reviewing any deviations from the standard methodology. If the licensee proposes a new methodology, the NRC staff must review all aspects of the methodology more thoroughly. The NRC reviewer must be able to conclude that the method will, with reasonable assurance, be comprehensive and meet the intent of the rule. In either case, the reviewer should select and emphasize areas in which there are major deviations from standard procedures, as appropriate for a particular case. Specific guidance follows for reviewing the licensee's program for selecting SSCs and SCs based on the acceptance criteria in Item II "Acceptance Criteria" above.

A. SSCs IMPORTANT TO LICENSE RENEWAL

The reviewer should confirm that the licensee's application includes deterministic methodology and criteria for identifying SSCs important to license renewal in the following four categories, as described in 10 CFR 54.3(a):

(i) Safety-related SSCs

(ii) All SSCs used in a safety analysis or plant evaluation for the licensing basis

(iii) Any, including non-safety-related, SSCs whose failure could keep safety-related equipment from satisfactorily performing required safety functions

(iv) Post accident monitoring equipment as defined in 10 CFR 50.49(b)(3)

The rule does not prescribe how to identify these SSCs, and criteria and methodologies may vary among plants or among different systems within the same plant. The reviewer should examine the criteria and methodology provided by the licensee, paying particular attention to deviations from previously approved approaches or to new approaches developed by the licensee. The reviewer should confirm that probabilistic techniques are used only to supplement the list obtained from deterministic considerations. The reviewer also should confirm that no SSCs are eliminated at this stage on the basis of aging considerations.

The methodology and the list of SSCs should be reviewed at a level that provides reasonable confidence that all SSCs important to license renewal have been properly identified. The reviewer should examine the licensee's final safety analysis report (FSAR) and other licensing-basis documents to the extent necessary. The review should assume that all safety-related systems as well as non-safety-related systems, such as overfill protection systems and ATWS mitigation systems have been identified as important to license renewal. For methodologies that have not been previously approved by NRC, the reviewer may use the generic list of SSCs from Appendix B to Regulatory Guide DG-1009 as a starting point for assessing the completeness of the licensee's list. Although this generic list is likely to contain some SSCs that are not relevant to every plant, it may be used to identify areas in which additional justification or analysis is warranted.

B. SCs THAT ARE CONSTITUENT ELEMENTS OF THE SSCs IMPORTANT TO LICENSE RENEWAL

The reviewer should confirm that the licensee has described the methodology for converting SSCs important to license renewal into SCs that are their constituent elements. This is an implicit requirement in 10 CFR 54.21(a)(2) and is shown as a separate step in Figure A-2. In this step, any structure or component that was initially identified as important to license renewal is automatically included as a constituent SC. In addition, all SCs from any of the systems identified as important to license renewal are added to the second list.

The focus of this review is to examine the methodology for defining system boundaries to ensure SCs are not eliminated inappropriately. The system boundaries should be broadly defined, reserving the question of safety function for the next step. The reviewer could refer to Table III in Regulatory Guide DG-1009 for guidance concerning appropriate system boundaries when evaluating methodologies that have not been approved by the NRC.

The reviewer may wish to use the licensee's FSAR and other licensing-basis documents to check the implementation of the licensee's methodology for identifying constituent SCs for selected systems.

C. SCs REQUIRING EVALUATION OF AGE-RELATED DEGRADATION

The reviewer should confirm that the licensee has submitted the methodology and selection criteria for identifying the SCs requiring aging evaluation from the list of SCs in Item III.B above. The objective of this review is to develop reasonable assurance that the list of SCs is comprehensive since these are the only SCs that will be analyzed in detail with respect to age-related degradation. The licensees may have to review previously completed analysis of aging issues and identify questions concerning the analysis (including its assumptions. The reviewer should ensure that the licensee has not eliminated SCs on the basis of consideration about aging at this step. Guidance for the review of the detailed aging evaluations of SCs identified at this step can be found in SRP-LR Parts B and C.

The reviewer should confirm that the licensee has identified SCs that contribute to the performance of the safety function of an SSC important to license renewal or whose failure could prevent an SSC important to license renewal from performing its intended safety function. As noted in SRP-LR Part A, the term "safety function" in 10 CFR Part 54 refers to any function that causes an SSC to be identified as important to license renewal.

It is possible that some constituent SCs in a system may not contribute to the performance of the system or to the performance of other SSCs important to license renewal. It is also possible that these same SCs do not have failure modes that could prevent SSCs important to license renewal from performing their intended safety functions. Although SCs of this nature need not be included in the list of SCs requiring further evaluation regarding the reviewer should confirm that the licensee has provided the justification for eliminating them from further consideration. The reviewer should check the implementation of the licensee's procedure for some subset of the SCs that were eliminated to ensure that the methodology is properly applied.

The reviewer should confirm that deterministic analyses form the basis for the principal conclusions about safety functions. However, insights from probabilistic assessments may be used to the extent that they supplement the list of SCs for further evaluation.

IV. FINDINGS

The reviewer should determine and verify the licensee has provided sufficient information and the review supports the conclusion to be included in the staff's SER that: (1) the licensee's methodology for identifying SSCs important to license renewal and for identifying SCs requiring aging evaluation is acceptable and (2) the licensee's implementation of this methodology is acceptable.

V. IMPLEMENTATION

Except in those cases in which the licensee proposes an acceptable alternative method for complying with specified portions of SRP-LR, the staff plans to use the methods described herein during its review.

VI. GENERAL INFORMATION

VII. REFERENCES

PART B: PLANT SYSTEMS

Part B addresses the review of various systems requiring evaluation of age-related degradation. Key plant systems that are likely to be identified as important to license renewal have been included in Part B. Plant-specific analysis could id... fy additional systems as important to license renewal that are not treated se arately in Part B, such as main steam, turbine generator, and turbine bypass systems. Part C provides guidance for reviewing the constituent structures and components of all systems important to license renewal.

B.0.1 SYSTEM REVIEW CRITERIA

This first section, B.0.1, provides information and criteria which are applicable to all systems important to license renewal. Standard acceptance criteria, review procedures, findings, and implementation applicable to each system are given below. The remaining sections contain the information which is applicable to selected specific systems. Sections of individual systems include system descriptions, requirements, and information unique to a particular system.

The review of systems not addressed in an individual system section should follow the generic guidance in this section (B.0.1).

I. AREAS OF REVIEW

A. See the section on the specific system for a description of the subject system, its function, and its boundaries.

B. The SRP-LR addresses age-related degradation of the subject system that must be understood and controlled with sufficient certainty to permit the staff to consider issuing an operating license for the requested renewal period while maintaining the current licensing basis. The licensee's staff has conducted an integrated plant assessment (IPA) to identify potential age-related degradation of systems, structures, and components and to evaluate the adequacy of the licensee's programs to identify and mitigate age-related degradation for the renewal term. The FSAR supplement for license renewal provides a list of systems, structures, and components identified as important to license renewal and a list of structures and components requiring evaluation of age-related degradation. The reviewer's safety evaluation for the subject system will be contained in the corresponding section of the safety evaluation report (SER) for license renewal.

The review of issues contained in this SRP-LR is not intended to be a review of the existing licensing basis. However, the actual licensing basis for an individual plant is contained, in part, in the FSAR specific to that facility. The NRC staff documented its review of the FSAR in the safety evaluation report (SER) it prepared to accompany the original operating license.

The areas of aging concern should be reviewed in accordance with site-specific conditions and experience as documented in the IPA.

C. See the section on the specific system for a discussion of the typical age-related degradation associated with the subject system.

II. ACCEPTANCE CRITERIA

Acceptance and performance criteria for structures and components within the boundaries of the subject system are typically contained, in part, in such sources as technical specifications; ACI, AISC, ASME, and IEEE codes and standards; root-cause analysis; failure-mode analysis; equipment performance history; branch technical positions; approved topical and other industry reports; vendor criteria; and regulatory guides. For specific components, the vendor recommendations for extending their life through the renewal period could be critical in such areas as (1) applicability of current maintenance practices, (2) applicability of the current technical manual, and (3) design limitations for the specific component which may require replacement of selected parts. The licensees may have to review previously completed analysis if aging issues identify questions concerning the analysis (including its assumptions). The acceptability of the licensee's proposed program for identifying age-related degradation, monitoring aging degradation, and mitigating the effects of age-related degradation in the subject system will be based on the following criteria:

A. The licensee has performed and documented an IPA which demonstrates that degradation related to the aging of the subject system has been identified, evaluated, and accounted for as necessary to ensure that the current licensing basis will be maintained throughout the term of the renewed license. The review focuses on the following items:

1. The licensee has listed all structures and components within the boundary of this system that contribute to the performance of a safety function of an SSC important to license renewal or whose failure could prevent an SSC important to license renewal from performing its intended safety function (refer to Appendix A.1, Part A of SRP-LR).

2. The licensee has listed the structures and components within the boundary of this system from II A.1 above that are subject to an established effective program. This program must continue to ensure the capability of the structures and components to either perform their safety functions during the renewal term or not to impair the safety functions of other SSCs. In accordance with the requirements of 10 CFR 54.3(a), an established effective program shall include as appropriate,

but is not limited to, inspection, surveillance, maintenance, trending, recordkeeping, replacement, refurbishment, and assessment of operational life for timely mitigation of the effects of age-related degradation. An established effective program must satisfy the following three criteria:

a. The program is documented in the FSAR, approved by the onsite review committee, and implemented by the facility operating procedures.

b. The program ensures that all SSC safety functions affected by age-related degradation of the subject system are properly reviewed by adhering to the program procedures.

c. The program establishes acceptance criteria against which the need for corrective action is to be evaluated and requires that timely corrective action be taken when these criteria are not met.

Programs and practices acceptable to the staff are discussed in Regulatory Guide DG-1009, "Standard Format and Content of Technical Information for Applications to Renew Nuclear Power Plant Operating Licenses." Such programs and practices contain the following important elements: (1) use of state-of-the-art knowledge of age-related degradation in nuclear power plants; (2) integration of relevant materials science concepts, which describe degradation processes, with plant-specific design and operational information; and (3) use of state-of-the-art monitoring methods that reflect the mechanistic and empirical assessments performed by the licensee to understand age-related degradation and mitigate its effects.

Some existing programs will have to be modified in order to be classified as established effective programs for the renewal period. For example, for selected electrical components the licensee may claim the equipment qualification (EQ) program required by 10 CFR 50.49 is an established effective program. But for a subset of these components, either extensive additional testing is required or a reanalysis (with appropriate justification documented or selected verification tests, as appropriate) must be performed in order for the EQ program to be applied to the renewal period.

3. For those structures or components within the system's boundary identified as requiring evaluation of age-related degradation but which are not included in an established effective program, the licensee has described and provided the bases for actions taken, or to be taken, to manage the age-related degradation or has demonstrated, by evaluation, that the age-related degradation is not significant with respect to the current licensing basis. This action will include one of the following:

a. Discuss specific aging-management actions, including inspection, maintenance, surveillance testing, condition monitoring, replacement, refurbishment, recordkeeping, and any adjustments made to the operating environment of the SSCs, as appropriate; or

b. Demonstrate that age-related degradation is not significant and that the subject system will continue to meet the current licensing basis without additional action during the term of the renewed license.

B. The licensee has identified plant-specific exemptions granted pursuant to 10 CFR 50.12, "Specific Exemptions," and reliefs granted pursuant to 10 CFR 50.55a(a)(3), "Codes and Standards." The licensee should justify continuing those exemptions and reliefs that were granted on the basis of an assumed service life or period of operation bounded by the original license term of the facility, or otherwise related to SSCs subject to age-related degradation.

C. Additional criteria are discussed in sections on specific systems as applicable.

III. REVIEW PROCEDURES

Upon request from the primary reviewer (LRPD), the coordinating review branches will provide material for the areas of review identified in Item I above. The primary reviewer obtains and uses such information as required to ensure that this review procedure is complete.

These procedures should be followed for reviewing specific systems to determine whether or not: (1) the structures and components within the boundaries of the subject system requiring evaluation of age-related degradation have been identified, (2) the potential aging mechanisms have been identified by the licensee for specific components and structures within the boundaries of the subject system (typical examples are provided in Item I.C of each SRP-LR Part B chapters), (3) the established or new programs for managing age-related degradation are adequate, (4) exemptions and reliefs based upon assumed service life will continue to be appropriate during the renewal terms, and (5) proposed modifications to the administrative procedures are adequate to manage age-related degradation.

The reviewer should perform the following steps to evaluate the licensee's program for license renewal based on the acceptance criteria given in Item II above.

A. The reviewer should confirm that an IPA has been documented and submitted which demonstrates that age-related degradation related to the subject system has been identified and evaluated in conformance with 10 CFR 54.21(a). Typical degradation mechanisms of concern for a specific system are discussed in Item I.C of each chapter of SRP-LR Part B. However, the actual mechanisms of concern for a particular facility should be addressed in its IPA. The methodology for identifying structures and components within the boundaries of the subject system that require evaluation of age-related degradation, should be reviewed as described in SRP-LR Part A to ensure it has been adequately applied to this system.

B. The reviewer should verify the licensee has presented information that demonstrates acceptable performance from an aging perspective for each structure and component in an established effective program. The reviewer should confirm that the licensee identifies the method for evaluating age-related degradation and the adequacy of the aging-management program for each structure and component. For structures or components identified as being routinely replaced or refurbished at defined intervals, the reviewer should ensure that the licensee demonstrates ongoing programs are adequate for timely mitigation of age-related degradation. The support for this determination could focus on review of joperational experience, replacement or refurbishment intervals, and, as appropriate, design and manufacturer information, known aging mechanisms, and other relevant information. For structures and components not routinely replaced or refurbished, the reviewer should ensure that the licensee's support for the conclusion that the structure or component is subject to an established effective program includes a detailed mechanistic analysis of age-related degradation mechanisms. The reviewer should confirm that:

1. The established program is documented in the FSAR, approved by the onsite review committee, and implemented by the facility operating procedures.

2. All SSC safety functions affected by age-related degradation are reviewed.

3. The program establishes acceptance criteria against which the need for corrective action is to be reviewed and requires that timely corrective action be taken when these criteria are not met. Replacement, refurbishment, and inspection schedules that may be necessary to manage age-related degradation are implemented by ensuring that the plant program defines inspection methods used, inspection frequency, replacement and refurbishment frequency, and meets current licensing-basis requirements.
The reviewer should ensure that the acceptance criteria are based on an industry standard or technically acceptable report and that the action taken or to be taken is timely and will restore the component or structure to an acceptable performance condition in accordance with the facility's current licensing basis.

C. For structures and components in the system that are not subject to established effective programs, the reviewer should verify one of the following:

1. Current programs have been or will be revised to provide for timely mitigation of age-related degradation for this structure or component, or a new program will be developed specifically for this structure or component. The reviewer should confirm that the licensee's evaluation of the adequacy of the aging-management program includes detailed mechanistic analyses for all structures and components not routinely replaced or refurbished. These analyses may also be required for structures and components that are routinely replaced or refurbished if analysis of operational experience is not sufficient to demonstrate adequacy of the replacement or refurbishment program to provide for timely mitigation of age-related degradation.

2. Evaluation is provided to demonstrate that age-related degradation is not significant with respect to the current licensing basis for this structure or component and to justify why the structure or component is not required to be part of an aging-management program.

D. Exemptions and reliefs granted on the basis of assumed service life have been reviewed to determine if they will continue to be valid for the term of the license renewal.

E. Additional review procedures are discussed in the sections on specific systems, as applicable.

IV. FINDINGS

The reviewer should determine and verify that sufficient information has been provided and the review supports the following conclusions to be included in the staff's SER regarding license renewal.

A. The licensee's analysis of the subject system acceptably identifies the structures and components requiring evaluation of age-related degradation. The generic components and structures reviewed from SRP-LR Part C which are applicable to the system under review are included in this finding.

B. The licensee demonstrated compliance with the requirements of 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," and demonstrated through the IPA that age-related degradation had been identified, evaluated, and accounted for as necessary to ensure that the licensee's current licensing basis will be maintained throughout the term of the renewed license.

C. The licensee is implementing an effective program for license renewal which uses existing programs and any necessary new procedures and methods to identify the plant-specific age-related degradation mechanisms, to manage age-related degradation, to ensure that the activities authorized by the renewed license can be conducted in accordance with the current licensing basis over the renewed term.

D. The licensee provided a list of exemptions related to the subject system granted pursuant to 10 CFR 50.12, "Specific Exemptions," and reliefs granted pursuant to 10 CFR 50.55a(a)(3), "Codes and Standards." The justifications for continuing the exemptions and reliefs are acceptable for the renewal term.

E. The licensee adequately addressed and identified any proposed modifications to the facility or its administrative control and plant procedures.

V. IMPLEMENTATION

Except in those cases in which the licensee proposes an acceptable alternative method for complying with specified portions of the SRP-LR, the staff plans to use the methods described herein during its review.

VI. GENERAL INFORMATION

Addressed in specific system sections.

VII. REFERENCES

Addressed in specific system sections.

B.1.1 REACTOR PRESSURE VESSEL

REVIEW RESPONSIBILITIES

Primary - LRPD Secondary - EMEB/EMCB

I. AREAS OF REVIEW

A. This section addresses the reactor pressure vessel (RPV).

1. Description

The RPV is cylindrical with a welded hemispherical bottom and a removable flanged hemispherical upper head. The RPV is typically constructed of a manganese molybdenum alloy; all surfaces in contact with reactor coolant are clad with stainless steel or nickel-chromium-iron. The RPV contains the core, core support structures, control rods, internal, components and other associated items.

The RPV support structure is designed to permit thermal growth and to simultaneously restrain vertical, lateral, and rotational movement resulting from seismic and pipe break events. For PWRs, head penetrations include those for control rod drive mechanisms (CRDMs) adapters, head vent, and, for those plants so equipped, upper head injection adapters. The bottom contains penetrations for the incore nuclear instrumentation. Typically, inlet and outlet nozzles are located in a horizontal plane below the head flange, but above the top of the fuel assemblies. However, there are variations in some earlier RPV designs where the inlet and outlet nozzles are located near the top of the fuel assemblies.

For BWRs, head penetrations include those for the vent and head spray and a spare penetration. The bottom penetrations include those for the bottom drain, CRDM housings, incore nuclear instrumentation, and other incore instrumentation. The various inlet and outlet nozzles are located at separate elevations on the cylindrical portion of the RPV.

The RPV is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the specific facility.

2. System Function

The RPV provides a barrier to the release of fission products, forms part of the primary coolant boundary, and supports and aligns the reactor core, control rod drive assemblies, and other RPV internals/components.
The main functions of the reactor internal components are to

provide orientation and support for the reactor core and to guide and protect the reactor control rod drive assemblies. They also provide a passageway, support, and protection for any in-vessel instrumentation and direct water flows as necessary.

3. System Boundaries

The RPV boundary extends to the end of each penetration nozzle, housing, or adapter.

B. See Section I, "Areas of Review," of SRP-LR B.0.1 for Item I.B.

C. A variety of age-related degradation mechanisms can affect the safe, continued operation of the RPV and internal components. The principal RPV degradation mechanisms and degradation sites are given in the following paragraphs.

In PWRs, the beltline region is subject to irradiation embrittlement, which is the primary RPV aging concern, and to thermal and pressure-induced fatigue. Inlet and outlet nozzles are susceptible to thermal and mechanical fatigue and irradiation embrittlement. Instrumentation and CRDM housing nozzles are susceptible to thermal and mechanical fatigue. Flange closure studs are subject to mechanical fatigue and stress corrosion cracking. Head flanges and vessel flanges are subject to corrosion, erosion, and mechanical wear. Core support pads are subject to mechanical wear and irradiation embrittlement. The RPV head dome is susceptible to corrosion damage and the head penetrations may experience fatigue.

In BWRs, the feedwater nozzles and safe-end welds are subject to high-cycle thermal fatigue and mechanical fatigue. Recirculation inlet and outlet nozzles and dissimilar metal welds are exposed to stress corrosion cracking (SCC) and thermal and mechanical fatigue. SCC has been found in welds to CRDM stub tubes and RPV interval attachments. The beltline region is subject to irradiation embrittlement and thermal fatigue. Closure studs are susceptible to mechanical fatigue and corrosion.

The potential degradation mechanisms for RPV internal components for both BWRs and PWRs are neutron embrittlement, stress corrosion cracking, mechanical wear, low- and high-cycle fatigue, and stress relaxation.

The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LP B.0.1 for Items II.A and II.B.

C. Licensees shall verify using plant-specific fatigue analyses for the reactor vessel and all safety-related vessel intervals, that the ASME Section III cumulative usage factor allowable of 1.0 will not be exceeded during the projected lifetime of the plant.

D. All of the current programs to monitor and/or mitigate age-related degradation of the RPV shall remain in effect throughout the license renewal term.

Tables B.1.1-1 and B.1.1-2 (Ref. 1) summarize the aging concerns and the current NRC requirements for addressing these concerns for PWRs and BWRs respectively.

Also listed in Tables B.1.1-1 and B.1.1-2 are various alternative methods, proposed in Reference 1, that may assist in managing or mitigating the aging concerns.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LP B.0.1 for Items III.A through III.D.

E. Using the criteria of Item II.C, the reviewer should confirm that the overall RPV surveillance and monitoring program submitted by the licensee is adequate to manage aging effects. The program should present reviews, current status, evaluations, and engineering analysis to provide reasonable assurance that the licensee understands and is managing aging for this important component.

Elements that should be included in the licensee's RPV aging management program, as appropriate, are the following:

1. Microhardness and tensile testing of notched RPV material specimens

2. Reconstituted and miniature test specimens, or accelerated testing for plants that do not have adequate surveillance material for the license renewal period for evaluating irradiation embrittlement

3. Reviews of nickel, copper, and phosphorus contents of RPV materials

4. Reliable detection, sizing, and characterization of RPV defects and crack growth, using ASME Code methods or additional improved techniques

5. Fatigue crack growth curves, including the effects of the specific reactor environment

6. Acoustic emission monitoring to monitor crack growth for on-line fatigue monitoring

7. Evaluation of high-cycle fatigue

8. Monitoring and evaluation of corrosion

9. Evaluation of hydrogen water chemistry (BWRs)

10. Examination, procedures, and schedules for interior attachment welds

11. Inspections and evaluations of pressure-retaining welds

F. An acceptable conservative approach to satisfy the staff's fatigue concerns would be to verify that the licensee has accomplished the following:

a. List the original design basis calculated cumulative usage factor for all reactor vessel and internal components. These calculations should have been based on the estimated number of plant transients and cycles for a plant life of 40 years.

b. Provide the additional number of transients and cycles to be used as the design basis for the extended life of the plant, e.g., if the projected life is an additional 20 years for a total of 60 years, the original design basis transients from a. above should be increased by 50%. For all components, calculated the cumulative usage factors for this additional increment of time.

c. List any known cycles due to unanticipated plant transients which were not considered as design basis events in a or b above. For all components, calculate cumulative usage factors for these additional transients.

d. Add the cumulative usage factors calculated from a, b and c above to arrive at the total end of life fatigue usage factors for all components.

e. If the rate of actual plant cycles indicate that the design basis cycles will be exceeded at the end of life of the plant, the procedures in a, b and c above should be adjusted to account for these additional cycles.

f. The above analyses should be in accordance with ASME Section III, Subsections NB-3222.4(e) and NB-3228.5 or NG-3222.4(e) and NG-3228.3. If the total number of stress cycles is estimated to be greater than 10^6, the licensee should provide the basis for appropriate design fatigue curves.

g. In the above analyses, the licensee should include an evaluation of environmental effects on the fatigue crack initiation to the extent needed to show that the analyses are conservative.

h. All of the above evaluations should be based on elastic analyses. The use of elastic-plastic or fully plastic approaches as a means to remove conservatism in fatigue analyses may be acceptable if a detailed description of the analysis techniques and the basis on which these techniques have been qualified are submitted to the staff for review and evaluation prior tc using such procedures.

i. Each licensee should list any plant-specific history of failure due to fatigue in any reactor vessel or vessel interval component.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR Section B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

A. Aging Degradation

The aging concerns for the RPV are very similar for both BWRs and PWRs, differing primarily in magnitude. The major areas of concern for the PWR RPVs are (Refs. 1 and 2):

- o Irradiation embrittlement in the beltline region
- o Thermal and mechanical fatigue at the inlet and outlet nozzles (penetrations)
- o Thermal and mechanical fatigue at the CRDM penetrations and incore instrument penetrations

The major areas of concern for the BWR RPVs are (Refs. 1 and 2):

- o High-cycle thermal fatigue and mechanical fatigue at the feedwatei inlet nozzles (penetrations)
- o Thermal and mechanical fatigue and intergranualar stress corrosion cracking (IGSCC) at the other nozzles (recirculation loop inlet and outlet nozzles and various emergency core cooling system penetrations)
- o Thermal and mechanical fatigue and potential for IGSCC at the CRDM and incore instrumentation penetrations and internal component attachments
- o Irradiation embrittlement in the beltline region and in the reactor internals components

o Thermal and mechanical fatigue of the closure studs

The important degradation sites, mechanisms, and consequences are summarized in Tables B.1.1-3 and B.1.1-4 for PWR and BWR RPVs respectively. The various age-related degradation mechanisms acting on the beltline region, penetrations, and internal components are discussed below (Refs. 1 and 2)

1. Beltline Region

The primary concern for PWR RPVs is to ensure against catastrophic failure caused by irradiation embrittlement combined with thermal and mechanical stresses of the ferritic steel adjacent to the reactor core. Irradiation embrittlement is not as severe for BWR RPVs as it is with PWR RPVs; the greater amount of water between the reactor core and the vessel wall generally reduces the expected neutron fluence accumulated during the initial 40 years of operation to about 5×10^{18}/cm2 (E-1 MeV) for BWR RPVs, compared with 4×10^{19}/cm2 (E-1 MeV) typically expected for PWR RPVs. Thus, although irradiation embrittlement should not be ignored, it is of lesser concern in BWR RPVs than in PWR RPVs. BWR RPVs are more difficult to inspect because of jet pump placement and have often been exempted from certain inspection requirements.

Irradiation reduces the energy required to fracture the steel and increases the ductile-to-brittle transition temperature. The amount of embrittlement caused by a given amount of radiation exposure (neutron fluence) depends primarily on the irradiation temperature and the composition of the steel. Lower irradiation temperatures increase the rate of embrittlement as a function of neutron fluence. The presence of some elements, notably copper and nickel, also increase the rate of irradiation embrittlement.

Restrictions are imposed on operational limits to account for the effects of irradiation embrittlement. If these restrictions become, or are predicted to become, too severe from an operations standpoint, one alternative is to thermally anneal the RPV beltline region so as to remove a significant portion of the embrittlement. ASTM Standard Guide E509-86 (Ref. 3) covers the general procedures to be considered for conducting an inservice thermal anneal of the RPV and for demonstrating the effectiveness of the procedure. Thermal and mechanical transients, which arise both from normal operational cycles and abnormal events, can also produce fatigue damage in the beltline region. Cycling of the stresses can initiate crack growth and cause ductile fracture and leaking at the welds. This is discussed in more detail in Item V1.A.2, "Penetrations".

2. Penetrations

Stresses are concentrated at weld regions, especially where dissimilar metals are joined and where geometrical discontinuities occur. The stress distributions at these positions vary in response to temperature and pressure changes during normal and abnormal events. Cycling of these stresses results in fatigue damage of the stressed regions. It can also cause growth of existing cracks and accelerate stress corrosion cracking of susceptible materials. At the inlet and outlet nozzles of PWRs, irradiation embrittlement may contribute to degradation induced by thermal and mechanical cycling.

The primary concern is that an abnormal event might lead to overpressurization, possibly combined with thermal shock, leading to ductile fracture and leaking at one of the welds. Periodic inservice inspection of the welds and fatigue-damage modeling, based on thermal and mechanical transient histories, are used to ensure that the penetration welds could sustain such an abnormal event.

Procedures to predict crack initiation, addressed in ASME Code, Section III, (Ref. 4), are based on the classical stress-strain/life approach, with no provision for accounting for the sequence of loading events. In contrast, the procedures for predicting crack growth are based on the damage-tolerant fracture-mechanics approach, as defined in ASME Code, Section XI (Ref. 5). Additional research is being conducted, under the NRC Pressure Vessel Research Program, to improve the life-prediction procedures. NUREG/CR-4731 (Ref. 1) discusses a modified approach for fatigue-damage modeling for residual life assessment. This modified approach bases crack initiation on the local strain history, including the sequence of the stress-strain cycles, and bases crack propagation on integration of a realistic crack-growth relationship (local-strain approach). It is anticipated that this approach will supplant those currently incorporated in ASME Code, Sections III and XI.

3. Internals

One of the potential failures of the RPV internal components, (i.e., cooling water jetting), has led to the degradation of fuel rod cladding and the disbursement of fuel into the coolant in certain reactors (Ref. 1). Failure of the RPV internal components could also relocate fuel away from the control rods or prevent the control rods from inserting properly and lead to an operational transient without scram.

The key RPV internal components susceptible to aging degradation are the lower core plate, the baffle-former assembly, the upper support column bolts, the control rod guide tube sheaths and support pins, the thermal shield bolts, the core barrel bolts, the incore instrument nozzles, and the flux thimble tubes.

The potential degradation mechanisms are neutron embrittlement, stress corrosion cracking, mechanical wear, low- and high-cycle fatigue, and stress relaxation. The low-cycle fatigue is caused by the loads resulting from changes in power levels, vessel inlet and outlet temperature differences, and coolant pressures and flow rates. The high-cycle fatigue is a result of the flow-induced vibrations. The control rod guide tube sheath and support pins and flux thimble tubes are locations that may experience significant mechanical wear.

B. Managing Aging Degradation

1. PWR RPVs

Inservice inspection (ISI) is required by ASME Code, Section XI. Inspection intervals, during which certain welds must be examined, are 10 years long. All shell, head, shell-to-flange, head-to-flange, and repair welds must be subjected to a 100 percent volumetric examination during the first inspection interval. The staff anticipates implementing a required 100 percent volumetric examination for all successive ISI. Successive inspection intervals require fewer beltline region, head, and repair weld examinations. The nozzle-to-vessel welds must all be subjected to a volumetric examination during all inspection intervals. Twenty-five percent of the partial-penetration nozzle welds (CRDM and instrumentation) are required to have a visual, external surface examination during each inspection interval, leading to total coverage of all nozzles. All nozzle-to-safe-end butt welds with dissimilar metals (i.e., ferritic steel nozzle to stainless steel or Inconel) must be subjected to volumetric and surface examinations during each inspection interval. Any integrally welded attachments are required to have surface or volumetric inspections of welds during each inspection interval.

The inspection plan is very complete and results in close monitoring of potential fatigue crack formation and growth. Additional monitoring and recording of transients are usually done in accordance with the plant technical specifications. Irradiation embrittlement of RPV beltline materials is normally monitored by testing specimens that have been irradiated in surveillance capsules located near the vessel wall. ASTM Standard Practice E185-82 (Ref. 6) provides guidelines for designing and conducting a minimum surveillance program. This practice covers selection and characterization of materials, type and number of specimens, specification and monitoring of irradiation conditions, conduct of the test program, and reporting of results. ASTM Standard Practice E636-83

(Ref. 7) provides guidance on recommended supplemental test methods and procedures to be used in conjunction with those required by Standard Practice E185-82. These supplemental test methods permit acquisition of additional information on irradiation-induced changes in fracture toughness and strength properties of RPV steels.

It is recommended in the referenced standards that several surveillance capsules be installed before to reactor startup, each containing Charpy V-notch and tensile specimens that have been cut from base material, welds, and the heat-affected zone near the welds. At specified intervals over the design life of the pressure vessel, capsules should be removed and the enclosed specimens tested to determine the changes in the ductile-to-brittle transition temperature and the upper-shelf energy.

2. BWR RPVs.

The methods and requirements of ASME Code Section XI, BWR RPVs are the same as those for PWR vessels However, many older BWRs have very limited accessibility for external ISI of the vessel. Typically, 75 to 90 percent of the vessel weld lengths are exempted because of inaccessibility. The only alternative is ISI methods of examination from the inside surface. This is of particular importance at the beltline welds. Minor and major repairs were made to shell plates during construction, but some of these cannot be examined because of limited accessibility. This is also true for some of the pipe-to-nozzle welds that were not configured to facilitate ISI. This was changed in later reactors after the requirements of ASME Code, Section XI, were published.

Surveillance for irradiation embrittlement is dictated for BWRs as well as PWRs. Therefore, monitoring of actual changes in Charpy V-notch and tensile properties with regard to accumulated fluence for the most critical vessel materials is under way.

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "Residual Life Assessment of Major Light Water Reactor Components - Overview," Volume in another section 1, NUREG/CR-4731 (EGG-2469, Vol. 1) June 1987.

2. U.S. Nuclear Regulatory Commission, "A Review of Information Useful for Managing Aging in Nuclear Power Plants," NUREG/CR-5562 (PNL-7323), June 1990.

3. American Society for Testing and Materials, ASTM Standard Guide E509-86, "In-Service Annealing of Light-Water Cooled Nuclear Reactor Vessels," Annual Book of ASTM Standards, Vol. 12.02, September 1987.

4. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Division 1, New York, 1983.

5. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," New York, 1983.

6. American Society for Testing and Materials, ASTM Standard Practice E185-82, "Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," Annual Book of ASTM Standards, Vol. 12.02, September 1987.

7. American Society for Testing and Materials, ASTM Standard Practice E636-83, "Supplemental Surveillance Tests for Nuclear Power Reactor Vessels," Annual Book of ASTM Standards, Vol. 12.02, September 1987.

TABLE B.1.1-1 Summary of PWR Pressure Vessel Aging Concerns

Understanding Aging (materials stressors, and environmental interactions)		Managing Aging		Mitigation
		Inservice Inspections, Surveillance, and Monitoring		
Sites	Aging Concerns	NRC Requirements	Recommendations	
Beltline Region	Irradiation embrittlement - Chemical composition of vessel materials (Cu, Ni, P) - Drop in upper shelf energy - Shift in reference nil-ductility-transition-temperature Environmental fatigue (thermal and mechanical)	Surveillance program to asses irradiation damage, i.e., shift in RT_{NDT}* and drop in USE* (10 CFR 50 Appendix H, Regulatory Guide 1.99, Revision 2) PTS screening criteria (10 CFR 50.61) Damage evaluation (10 CFR 50, Appendix G) Volumetric examination of all welds during each inspection interval (10 CFR 50.55a, IWB-2500) Flaw detection and evaluation (10 CFR 50.55a, IWB-3000) Leakage and hydrostatic pressure tests (10 CFR 50.55a, IWA-5000)	Include fracture toughness and tensile test specimens in surveillance programs Develop use of reconstituted and miniature specimens Accelerated irradiation of reconstituted specimens Revise Regulatory Guide 1.99, Revision 2 to account for phosphorus with low copper Use state-of-the-art NDE techniques for improved reliability of defect detection, sizing, and characterization Use fatigue crack growth curves (ASME Section XI, Appendix A) Develop acoustic emission monitoring to detect crack growth (Nonmandatory appendix is being developed by ASME Section XI)	Neutron flux reduction Inservice annealing (ASTM E 509-86) Determine affects of annealing and re-embrittlement rate

* RT_{NDT} = Reference Temperature, Nil Ductility Transition
USE = Upper Shelf Energy

TABLE B.1.1-1 Continued

Understanding Aging (materials stressors, and environmental interactions)		Managing Aging		Mitigation
		Inservice Inspections, Surveillance, and Monitoring		
Sites	Aging Concerns	NRC Requirements	Recommendations	
Outlet/Inlet Nozzles	Environmental fatigue Irradiation embrittlement - Function of nozzle elevation - Potential impact of Regulatory Guide 1.99, Revision 2	Volumetric examination of all nozzle-to-vessel welds and nozzle inside radius section during each inspection interval (IWB-2500) Volumetric and surface examination of all dissimilar metal welds during each inspection interval (IWB-2500)	Use on-line fatigue monitoring (monitoring of pipe wall temperatures and coolant flow, temperature, pressure) Evaluate irradiation embrittlement damage	
Instrumentation Nozzles/CRDM Housing Nozzles	Environmental fatigue	Visual examination of external weld surface of 25% of nozzles during system hydrostatic test (IWB-2500)		
Flange Closure Studs	Environmental fatigue Boric acid corrosion (if leakage occurs)	Volumetric and surface examination of all studs and threads in flange stud holes during each inspection interval (IWB-2500)		

TABLE B.1.1-2 Summary of BWR Pressure Vessel Aging Concerns

Understanding Aging (materials stressors, and environmental interactions)		Managing Aging		Mitigation
		Inservice Inspections, Surveillance, and Monitoring		
Sites	Aging Concerns	NRC Requirements	Recommendations	
Feedwater Nozzles and Safe End Welds	High-cycle thermal fatigue caused by feedwater leakage Environmental fatigue (thermal and mechanical)	Volumetric examination of all nozzle-to-vessel welds and nozzle inside radius sections during each inspection interval (IWB-2500)	Use on-line fatigue monitoring (monitoring of pipe wall temperature, and pressure) Develop criteria for assessing high-cycle fatigue damage	Revise design and operating procedures and remove feedwater nozzle cladding to prevent fatigue cracking
Recirculation/Inlet Outlet Nozzles and Dissimilar Metal Welds	IGSCC crack initiated in HAZ may propagate in base metal Environmental fatigue	Volumetric and surface examination of all dissimilar metal welds during each inspection (IWB-2500)	Develop on-line corrosion monitoring Evaluate long-term effects of hydrogen water chemistry	Implement hydrogen water chemistry to mitigate corrosion fatigue
Welds - Control rod drive stub tubes - Interior attachments	IGSCC crack initiated in HAZ may propagate in base metal by corrosion and/or environmental fatigue	Visual examination of all accessible interior attachment welds during each inspection interval (IWB-2500)	Develop remote inspection technique for interior attachment welds	
Beltline Region	Irradiation embrittlement - Chemical composition of materials (Cu, Ni, P) - Drop in upper shelf energy (USE) - Shift in reference nil-ductility-transition-temperature (RT_{NDT}) - Welds are more susceptible than base metal - Flux is lower than that in PWR vessel Environmental fatigue	Surveillance program to assess shift in RT_{NDT} and drop in USE (10 CFR 50 Appendix H, Regulatory Guide 1.99, Revision 2) Damage evaluation (10 CFR 50 Appendix G) Volumetric examination of all shell welds during each inspection interval (10 CFR 50.55a, IWB-2500) Flaw detection and evaluation of all shell welds during each inspection interval (10 CFR 50.55a, IWB-300, Regulatory Guide 1.150, Revision 1) Leakage and hydrostatic pressure tests (10 CFR 50.55a, IWA-5000, IWB-5000)	Revise Regulatory Guide 1.99, Revision 2 to account for phosphorous when copper content is low Use state-of-the art inspection techniques for improved reliability of defect detection, sizing, and characterization Develop robotics system for remote probe inspection positioning and scanning Include fracture toughness and tensile test specimens in surveillance program Develop use of reconstituted and miniature specimens and accelerated irradiation of reconstituted specimens	Inservice annealing (ASTM E 509-86) Determine effects of annealing and reembrittlement rates Implement neutron flux reduction program

TABLE B.1.1-2 Continued

Understanding Aging (materials stressors, and environmental interactions)		Managing Aging		Mitigation
		Inservice Inspections, Surveillance, and Monitoring		
Sites	Aging Concerns	NRC Requirements	Recommendations	
			Use fatigue crack growth curves (ASME Section XI, Appendix A) Develop acoustic emission monitoring to detect crack growth (nonmandatory appendix is being developed by ASME Section XI)	
Closure Studs	Fatigue, fretting	Volumetric and surface examination of all studs and threads in flange stud holes during each inspection interval (IWB-2500)		

TABLE B.1.1-3 Summary of Degradation Processes for PWR Reactor Pressure Vessels

Rank of Degradation Site*	Degradation Site	Stressors	Degradation Mechanisms	Potential Failure Modes	ISI Surveillance Methods
1	Beltline Region	Neutron irradiation, mechanical and thermal stresses	Irradiation embrittlement (degree is dependent on individual vessel materials and flux spectrum history)	Ductile high-energy tearing leading to leakage (net section over-load)	100% volumetric during first inspection; one weld for subsequent inspection
			Stress corrosion cracking	Brittle fracture (i.e., pressurized thermal shock)	Surveillance program for assessing irradiation damage is required by law
				Ductile low-energy tearing (low upper-shelf toughness)	
			Environmental fatigue (thermal and pressure induced fatigue)	Ductile overload leading to a leak; possible brittle fracture if PTS occurs	
2	Outlet/Inlet nozzles	Mechanical and thermal stresses	Fatigue crack initiation and propagation	Ductile overload leading to a leak; possible brittle fracture if pressurized thermal shock occurs with some irradiation embrittlement	All nozzle welds inspected volumetrically at each interva
3	Instrumentation nozzles (penetrations) and control rod drive mechanism housing nozzles	Mechanical and thermal stresses	Fatigue crack initiation and propagation	Ductile overload leading to a leak	Visual, inspection of external surface; 25% of nozzles inspected at first interval; remaining 75% spread out over next three intervals
4	Flange closure studs	Mechanical and thermal stresses	Fatigue crack initiation and and propagation (possibly corrosion assisted)	Ductile overload failure (can be replaced)	Volumetric and surface inspection of all studs and threads in flange stud holes at each interval

* Rank of Degradation Site: 1 is highest in ranking priority

TABLE B.1.1-4 Summary of Degradation Processes for BWR Reactor Pressure Vessels

Rank of Degradation Sites*	Degradation Site	Stressors	Degradation Mechanisms	Potential Failure Modes	ISI Surveillance Methods
1	Nozzles (including instrument and CRD penetrations) plus safe end welds	Mechanical and thermal stresses	Fatigue crack initiation and and propagation, IGSCC	Ductile overload leading to a leak	All large nozzle welds inspected volumetrically at interval; visual, external surface inspections of small nozzles/penetrations
2	Closure studs, flange bushings, stud holes	Mechanical and thermal stresses	Fatigue crack initiation and propagation, fretting, corrosion	Ductile overload failure (can be replaced)	Volumetric and surface inspections of all studs, threads in flange stud holes and bushings
3	Beltline Region	Irradiation embrittlement	Neutron irradiation (extent depends on vessel materials)	Ductile high-energy tearing leading to a leak (not a serious problem)	100% volumetric inspection; surveillance as required by federal law
4	External attachments	Mechanical and thermal stresses	Fatigue	Ductile overload failure	Volumetric and surface inspections

* Rank of Degradation Site: 1 is highest in ranking priority

B.1.2 REACTOR COOLANT SYSTEM

REVIEW RESPONSIBILITIES

˜.imary - LRPD
Secondary - SRXB/EMEB/EMCB

I. AREAS OF REVIEW

A. This section addresses the reactor coolant system (RCS).

1. Description

a. The PWR RCS consists of two, three, or four heat transfer loops connected in parallel to the reactor pressure vessel (RPV). Components that are associated with the loops include steam generators, primary coolant pumps (PCPs), a pressurizer, interconnecting piping and fittings, and safety relief and isolation valves. The BWR RCS consists of RPV recirculation pumps, jet pumps, interconnecting piping and fittings, and isolation valves. The BWR RCS is often referred to as the reactor recirculation system (RRS).

b. Steam generators (SGs) are typically vertical shell and U-tube/straight tube heat exchangers constructed of carbon steel. Metal surfaces in contact with reactor coolant are made from or clad with an appropriate corrosion-resistant material.

c. The pressurizer (Pzr) is a carbon steel vessel connected to the RCS by surge and spray lines. All surfaces in contact with reactor coolant are made from or clad with an appropriate corrosion-resistant material.

The RCS is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the facility.

2. System Function

a. In PWRs the RCS functions to transfer the heat produced in the reactor to steam generators, where steam is produced and transported to the turbine generator. In BWRs the function of the RCS is to provide a forced flow of coolant through the reactor core and to control the core power level. The RCS also functions as a barrier against fission product release to the environment.

3. System Boundaries

The RCS boundary includes each component of the system that is subject to full RCS operation pressure. The boundary extends to and includes the second isolation valve in each line. The major components of a PWR are the steam generators (tube side), the reactor coolant pumps, the pressurizer, and connecting piping and valves. The major components of a BWR are the recirculation pumps and connecting piping and valves; jet pumps, which are an integral part of the recirculation system, are covered in Section B.1.1, "Reactor Pressure Vessel."

The instrumentation and control boundaries of the RCS include equipment that is considered significant to the RCS's safety function. Most of these components are addressed as generic I&C equipment and include but are not limited to control switches, relays, controllers, sensors, transmitters, recorders, computational modules, and circuit breakers. For BWRs the reactor recirculation flow control is covered in Section B.1.3, "Reactor Control System."

B. See Section I, "Areas of Review," of SRP-LR B.0.1 for Item I.B.

C. A variety of age-related degradation mechanisms can affect the safe, continued operation of the RCS, including the Pzr and SGs. Typical examples of age-related degradation associated with the RCS are listed below (Refs. 1-4). The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

- Oxidation
- Pitting
- Crevice corrosion
- Intergranular attack (IGA)
- Stress corrosion cracking (SCC)
- Intergranular stress corrosion cracking (IGSCC)
- Microbiologically influenced corrosion (MIC)
- Uniform corrosion (wastage)
- Erosion/corrosion
- Thermal embrittlement
- Hydrogen embrittlement
- Fatigue crack initiation and propagation
- Thermal fatigue
- Corrosion fatigue
- Denting
- Fretting

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1 for Items II.A and II.B.

C. Licensees should verify using plant-specific fatigue analyses for all ASME Class 1 components in the RCS that the ASME Section III cumulative usage factor allowable of 1.0 will not be exceeded during the projected lifetime of the plant.

D. The provision of ASME Code, Section XI (Ref. 5) for inservice inspection and inservice testing (ISI/IST) of RCS components are implemented through the license renewal period.

For those components important to license renewal that are not included in the facility's ISI/IST program (i.e., because they are small), a 10 percent minimum sample shall be inspected to ensure design adequacy is maintained throughout the license renewal period.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for Items III.A through III.D.

E. The RCS components not specifically addressed in this chapter may be addressed by the generic aging topic reviews in Part C of the SRP-LR. The reviewer should ensure that structures and components included as part of the generic SRP-LR topics are adequately reviewed for this system. The following sections of SRP-LR Part C should be used for the review C.1, "Mechanical" (all); C.2, "Electrical" (all); and C.3, "Instruments" (all).

The various control systems associated with the RCS (i.e., SG level control, pressurizer pressure control, recirculation flow control, etc.) are addressed in SRP-LR B.1.3, "Reactor Control System." Reviews of these systems may require other staff input.

The reviewer should verify:

F. 1. That the ISI/IST program will continue throughout the license renewal period.

2. That wall-thinning mechanisms such as erosion are specifically addressed for components that are located in lines with high-velocity fluids.

3. That on a one-time basis the licensee inspects a 10 percent sample of components that are important to license renewal and that are not part of the ASME Code, Section XI ISI/IST program. This testing shall be performed in accordance with Section XI or its equivalent.

G. An acceptable conservative approach to satisfy the staff's fatigue concerns would be to verify that the licensee has accomplished the following:

a. List the original design basis calculated cumulative usage factors for all components. These calculations should have been based on the estimated number of plant transients and cycles for a plant life of 40 years.

b. Provide the additional number of transients and cycles to be used as the design basis for the extended life of the plant, e.g., if the projected life is an additional 20 years for a total life of 60 years, the original design basis transients from a. above should be increased by 50 percent. For all components, calculate the cumulative usage factors for this additional increment of tiem.

c. List any known cycles due to unanticipated plant transients which were not considered as design basis events in a or b above. For all components, calculate cumulative usage factors for these additional transients.

d. Add the cumulative usage factors calculated from a, b and c above to arrive at the total end of life fatigue usage factors for all components.

e. If the rate of actual plant cycles indicate that the design basis cycles will be exceeded at the end of life of the plant, the procedures in a, b and c above should be adjusted to account for these cycles.

f. The above analysis should be in accordance with ASME Section III, Subsections NB-3222.4 (e)(5) and NB-3228.5. If the total number of stress cycles is estimated to be greater than 10^6, the licensee should provide the basis for appropriate design fatigue curves.

g. In the above analyses, the licensee should include an evaluation of environmental effects on fatigue crack initiation to the extent needed to show that the analyses are conservative.

i. All of the above evaluations should be based on elastic analyses. The use of elastic-plastic or fully plastic approaches as a means to remove conservatism in fatigue analyses may be acceptable if a detailed description of the analyses techniques and the basis on which these techniques have been qualified are submitted to the staff for review and evaluation prior to using such procedures.

j. Each licensee should list any plant-specific history of failure due to fatigue in any RCS component.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "Residual Life Assessment of Major Light Water Reactor Components - Overview," Volume 1, NUREG/CR-4731, June 1987.

2. U.S. Nuclear Regulatory Commission, "Residual Life Assessment of Major Light Water Reactor Components - Overview," Volume 2, NUREG/CR-4731, October 1989.

3. U.S. Nuclear Regulatory Commission, "Life Assessment Procedures for Major LWR Components," Volume 4, NUREG/CR-5314, November 1989.

4. U.S. Nuclear Regulatory Commission, "A Review of Information Useful for Managing Aging in Nuclear Power Plants," NUREG/CR-5562 (PNL-7323), 1990.

5. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1986.

B.1.3 REACTOR CONTROL SYSTEM

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SICB

I. AREA OF REVIEW

A. This section addresses the reactor control system.

1. Description

The RCSs consists of the instrumentation and control elements that sense plant conditions and are used for normal operations. The RCS are not relied on to perform safety functions following anticipated operational occurrences or accidents but are relied on to control plant processes having a significant impact on safety.

The RCS is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the facility.

2. System Function

The RCS controls the reactor during startup, power operation, and shutdown via the reactivity control systems; and controls and maintains reactor coolant pressure, temperature, flow, and inventory; controls secondary system pressure and flow controls; and controls the environmental control systems for safety-related instruments and instrument sensing lines.

3. System Boundaries

The RCS includes the power sources, sensors, transmitters, initiation circuits, logic matrices, bypasses, permissive relays, interlocks, racks, cables, panels, control boards, and actuation and actuated devices that are required to control the reactor during startup, power operation, and shutdown via the reactivity control systems; to control and maintain reactor coolant pressure, temperature, flow, and inventory; to control secondary system pressure and flow controls; and to control the environmental control systems for safety-related instruments and instrument sensing lines. The review of the controls, permissive relays, inhibits, and interlocks for the withdrawal, the insertion, normal and scram, and the selection and sequencing of control rods is also addressed in this section.

The objectives of this review are to confirm that the RCS satisfies the acceptance criteria and guidelines for age-related degradation applicable to the control system and that this system will perform its intended function during plant conditions for which it is required.

Typical reactor control systems are (other reactor control systems may also be used):

a. Reactivity control systems
b. Reactor coolant pressure control systems
c. Reactor coolant temperature control systems
d. Reactor coolant flow control systems
e. Condensate and feedwater control systems
f. Environmental control systems for safety-related instruments and instrument sensing lines

Typical secondary control systems are (other secondary control systems may also be used):

a. Secondary system pressure control systems
b. Secondary system flow control systems
c. Environmental control systems for safety-related instruments and instrument sensing lines

The RCS system is isolated from the engineered safety features actuation system (ESFAS) and the engineered safety features (ESF) and essential auxiliary supporting (EAS) systems, the reactor trip system (RTS), and the normal and emergency ac and dc power systems by circuit breakers, isolation amplifiers, isolation transformers, actuation logic, fuses, and other approved isolation devices.

B. See Section I, "Area of Review," of SRP-LR B.0.1 for Item I.B.

1. Typical examples of age-related degradation associated with the RCS are the following:

   a. Age-related degradation due to setpoint drift

   b. Age-related degradation due to functional testing cycles and trips

   c. Age-related degradation due to improper maintenance repair

   d. Age-related degradation of sensors, connectors, cables and wiring, circuit breakers, relays and electronic components, etc.

   The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

2. Electrical systems and instrumentation and control systems and their components are subject to age-related degradation. A wide variety of age-related degradation mechanisms can affect the ability of these systems and components to operate reliably. Several of these systems and components are listed below. The age-related degradation mechanisms for these systems and components are discussed in Part C, of the SRP-LR.

   a. Electrical Systems and Components

      - Cables and wiring
      - Junctions
      - Penetrations
      - Relays and switchgear
      - Transformers
      - Solenoid-operated valves
      - Electrical motors

   b. Instrumentation and control systems and components

      - Sensors
      - Electronic Components
      - Electronic Devices

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 Items III.A. through III.D.

E. Components of the RCS not specifically addressed in this section may be addressed in the generic aging topic reviews in Part C of the SRP-LR. The following sections of SRP-LR Part C are applicable to the RCS and should be used for the review: C.2.1, "Cable and Wiring"; C.2.2, "Junctions"; C.2.3, "Electrical Penetrations"; C.2.4, "Relays, Circuit Breakers, and Switchgear"; C.2.5, "Transformers"; C.2.6, "Solenoid-Operated Valves"; C.2.7, "Electrical Motors"; and all of C.3 "Instruments." This may require other staff input.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

Reference 1 indicates that, in general, the current testing programs are adequate for the intended purpose of verifying RCS operability and performance.

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "Nuclear Plant Aging Research of Reactor Protection Systems," NUREG/CR-4740 (TI88 007920), January 1988.

B.1.4 CONTROL ROD DRIVE SYSTEM

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SICB/EMEB

I. AREAS OF REVIEW

A. This section addresses the control rod drive system which

1. Description

The PWR control rod drive mechanisms (CRDMs) are located at the top of the reactor pressure vessel. Each CRDM is linked to its assembly by a detachable coupling. An assembly can be withdrawn or inserted by its CRDM at speeds consistent with the reactivity changes required for reactor operation or can be held at a desired location. The coupled assemblies and CRDM drive rods can also be released to drop into the core by gravity for maximum negative reactivity insertion (scram).

In BWRS, the CRDMs are located on the bottom of the reactor pressure vessel. The CRDMs used for positioning the control rods are mechanically latched, hydraulically actuated devices that rely on hydraulic fluid pressure differential for rod insertion. The CRDM provides a mechanical latch to hold the control rod in position until the hydraulic system is actuated. The hydraulic control units (HCUs) provide pressurized water, on command, to the CRDM. The HCU is a system of valves that actuate in various sequences to provide normal or scram movement of the control rod.

The CRDS is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the facility.

2. System Function

The CRDS allows manual positioning of control rods within the reactor core. In PWRs the CRDS also allows automatic positioning of control rods. This provides one means for regulating the reactivity in the core for startup and shutdown, as well as the maintenance of a programmed average temperature during power operations.

The CRDS is also designed to respond to scram signals, from the reactor protection system, by rapidly inserting withdrawn control rods. The CRDS is designed to prevent control rods from withdrawing as a result of a single malfunction.

3. System Boundaries

a. PWR

The boundaries of the PWR CRDS extend to the internal latching assemblies, rod drive assemblies, and magnetic coil stacks. The CRDM nozzle (or pressure housing), which contains the CRDM, is discussed in SRP-LR B.1.1, Reactor Pressure Vessel."

b. BWR

The typical CRDS for a BWR consists of four major components:

1. Control rod drive mechanisms
2. Hydraulic control units
3. Hydraulic system
4. Control rod drive supports

The hydraulic system supplies pressurized water to the HCUs. The hydraulic supply system consists of the following components:

1. Drive water pumps
2. Filters/strainers
3. Flow control station
4. Pressure control station
5. Scram discharge volume

The control rod drive supports are horizontal beams installed directly below the bottom of the reactor vessel between the rows of control rod drive housings. These supports ensure that control rod movement, following a control rod drive housing failure, is limited to prevent an inadvertent reactivity addition.

The following is a general description of the system interfaces involved with the typical BWR CRDS.

1. The hydraulic water typically comes from the condensate storage system and then is discharged into the reactor vessel during normal operation. In the event of a scram, some of the water is discharged into the scram discharge volume and then, in turn, is drained to a liquid radwaste system.

2. The reactor manual control system inputs (insert or withdrawal) signals to the HCU directional control valves to actuate normal rod movement. The reactor protection system inputs scram signals to open the scram inlet and outlet valves, which will rapidly insert the rod.

3. The CRDMs are bolted to the reactor pressure vessel and, therefore, interface with the primary pressure boundary.

B. See Section I, "Areas of Review," of SRP-LR B.0.1 for Item I.B.

C. Typical examples of age-related degradation associated with CRDMs are provided in Tables B.1.4.-1 and B.1.4.-2 (Ref. 1). The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP LR B.C.1 Items A., and B.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 Items III. A through III. D.

E. CRDS components not specifically addressed in this section may be addressed by the generic aging topic reviews in Part C of the SRP-LR. The reviewer should ensure that structures and components included as part of the generic SRP-LR topics are adequately reviewed for this system. The following sections of SRP-LR Part C should be used for the review: C.1.1, "Piping"; C.1.2, "Valves"; C.2.1, "Cable and Wiring"; C.2.4, "Relays, Switchgear"; C.2.6, "Solenoid-Operated Valves" (BWR only); C.2.7, "Electrical Motors" (PWR only); C.3.1, "Sensors"; C.3.2, "Electronic Components"; and C.3.3, "Electronic Devices." This may require other staff input.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1

VI. GENERAL INFORMATION

The technology of CRDM replacement is well established; thus, degradation of CRDM components is generally not a limiting factor in license renewal. However, periodic rotation of PWR CRDMs to different locations could allow more even wear and extend their life.

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "A Review of Information Useful for Managing Aging in Nuclear Power Plants," NUREG/CR-5562 (PNL-7323), June 1990.

Table B.1.4.-1 Summary of degradation process center for BWR control rod mechanisms

Degradation site	Stressors	Degradation mechanisms	Potential failure modes	
Pressure housing stub tube	Corrosive water, thermal stress, residual stress	Intergraded stress corrosion cracking	Cracking leading to leak	
Latching mechanism (collect assembly and index tube)	Thermal transients, corrosive water, rubbing, impacting	wear, IGSCC, fatigue	Binding, stuck rods	
Piston seal C-spring	Preloads, corrosive water	IGSCC	Stuck rods	
Hydraulic Control system	Thermal stress, corrosive water, debris, improper maintenance, overpressure misalignment	Valve diaphragm embrittlement and cracking	Stuck rods, unintentional rod movement	
Piston seals	Temperature, corrosive water	Embrittlement wear	Stuck rods	
Latch assembly	Loose parts, impacting	Fretting, wear, spalling	Binding, stuck rods	None
Coil stack	Moisture, temperature, radiation	Insulation breakdown, electrical shorting	Dropped rods, stuck rods	None
Drive rod	Rubbing, impacting	wear, low-cycle fatigue	Uncoupling of control assembly	None
External components	Boric acid (if leak is present)	Boric acid corrosion	Leaks	None

B.1.5 REACTOR PROTECTION SYSTEM

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SICB

I. AREAS OF REVIEW

A. This section addresses the Reactor Protection System (RPS):

1. Description

The RPS consists of the instrumentation and control elements that sense plant conditions and activate equipment to mitigate the consequences of abnormal conditions.

The RPS includes the power sources, sensors, transmitters, initiation circuits, logic matrices, bypasses, interlocks, racks, cables, panels, control boards, air systems, fluid systems, and actuation and actuated devices that are required to initiate a reactor shutdown and actuate the emergency equipment.

The RPS is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the utility.

2. System Function

The first function of the RPS is to ensure that specified plant safety limits are not exceeded by automatically de-energizing the power sources to the control rod drive mechanisms, which allow the control rods to insert into the core. This function is accomplished by the reactor trip system (RTS) and is addressed in this section of the SRP-LR. The second function of the RPS is to actuate emergency equipment in the event of a loss of primary or secondary system coolant inventory. This function is accomplished by the engineered safety features actuation system (ESFAS), which is addressed in SRP-LR B.2.1. The review of the controls, permissive relays, inhibits, rod blocks, and interlocks for the withdrawal, insertion, sequencing and selection of control rods is addressed in SRP-LR B.1.3. (RC) system.

3. System Boundaries

The RTS, including its sensors, generally shares many of its components with the ESFAS and the associated engineered safety features and essential auxiliary supporting systems. All shared components should be identified as such, however, they will be reviewed under the applicable specific system sections of this

SRP-LR. In regard to the remaining system boundaries, the RTS is isolated from the reactor control systems, and the associated normal and emergency ac and dc power systems by circuit breakers, isolation amplifiers, isolation transformers, actuation logic, fuses, or other approved isolation devices.

The objectives of this review are to confirm that the RTS satisfies the acceptance criteria and guidelines for age related degradation applicable to the protection system and that this system will perform its intended function during the plant conditions for which it is required.

Actuation of the RTS to scram the control rods is initiated by the following typical parameters. Other parameters may also be used.

- o High reactor power
- o Nuclear overpower based on reactor coolant flow and axial
- o imbalance
- o Loss of power to reactor coolant pumps
- o High T-hot
- o High reactor coolant system pressure
- o Low reactor coolant system pressure
- o High reactor building pressure
- o Anticipatory loss of main feedwater
- o Turbine trip
- o Overtemperature delta temperature
- o Overpower delta temperature
- o Low steam generator level
- o Low feedwater flow
- o Safety injection signal actuation
- o Turbine stop valve closure
- o Turbine control valve fast closure
- o Reactor vessel low water level
- o Main steamline isolation
- o Scram discharge volume high water level
- o Main steamline high radiation
- o Main condenser low Vvacuum
- o Mode switch in shutdown
- o Manual operator action

B. See Section I, "Area of Review," of SRP-LR B.0.1 for Item I.B.

Typical examples of age-related degradation associated with the RTS are the following:

The areas of aging for a facility should be reviewed to evaluate the acceptablity of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

1. Age-related degradation due to setpoint drift

2. Age-related degradation due to functional testing cycles and trips

3. Age-related degradation due to improper maintenance or repair

4. Age-related degradation of sensors, connectors, cables and wiring, circuit breakers, relays, electronic components, etc.

Electrical systems and instrumentation and control systems and their components are subjected to age-related degradation. A wide variety of age related degradation mechanisms can affect the ability of these systems and components to operate reliably. Several of these systems and components are listed below. The age-related degradation mechanisms for these systems and components are discussed in Part C of the SRP-LR.

1. Electrical Systems and Components:

   a. Cables and wiring
   b. Junctions
   c. Penetrations
   d. Relays and switchgear
   e. Transformers
   f. Solenoid operated valves
   g. Electrical motors

2. Instrument and Controls Systems and Components

   a. Sensors
   b. Electronic Components
   c. Electronic Devices

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of B.0.1 Items A and B.

III. REVIEW PROCEDURES

See Section III, "Review Procedures", of SRP-LR B.0.1 Items III.A through III.D.

E. Components of the RPS not specifically addressed in this section may be addressed in the generic aging topic reviews in Part C of the SRP-LR. The following sections of SRP-LR Part C are applicable to the RPS and should be used for the review: C.2.1, "Cable and Wiring"; C.2.2, "Junctions"; C.2.3, "Electrical Penetrations"; C.2.4, "Relays, Circuit Breakers, and Switchgear"; C.2.5, "Transformers"; C.2.6, "Solenoid Operated Valves"; C.2.7, "Electrical Motors"; and all of C.3.0 "Instruments". This may require other staff input.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

Regerence 1 indicates that, in general, the current testing programs are adequate for the intended purpose of verifying RPS operability and performance.

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "Nuclear Plant Aging Research of Reactor Protection Systems," NUREG/CR-4740 (TI88 007920), January 1988.

B.1.6 NEUTRON MONITORING SYSTEM

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SICB

I. AREAS OF REVIEW

A. This section addresses the neutron monitoring system.

1. Description

The nuclear instrumentation system provides the detectors and electronic circuitry needed to monitor the leakage neutron flux (which is proportional to reactor power) from the reactor under all conditions from shutdown to full power or overpower excursions.

In BWRs, incore detectors are used for core monitoring and all automatic safety protection. Some BWRs have, in addition, an excore neutron monitoring instrumentation system for post accident monitoring. The incore power level signal provides indication, control, and protective functions in the reactor control system and the reactor protection system. Any excore power level signal is for indication only.

In PWRs, excore detectors are used for core monitoring and all automatic safety functions. The excore nuclear instrumentation system provides the detectors and electronic circuitry needed to monitor the leakage neutron flux (which is proportional to reactor power) from the reactor under all conditions from shutdown to full power or overpower excursions. The power level signal thus developed is used to provide indication, control, and protective functions in the reactor control system and the reactor protection system.

The neutron monitoring system is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the facility.

2. System Function

Indication and recording of the various ranges are provided in the control room for the BWR incore and PWR excore instrumentation, including the status of various nuclear instrumentation trip and permissive bistable trip devices. BWR excore instrumentation, if provided, is also indicated and recorded in the control room.

3. System Boundaries

The BWR incore nuclear instrumentation is a component of the reactor protection system. Only the power range detectors have stable mountings. The source and intermediate detectors can be inserted or withdrawn from the core region depending on the core power level. Alarms, trips, and protective actions are generated in the instrumentation and transferred to the reactor protection system. Any BWR excore nuclear instrumentation is provided for indication only.

The PWR excore nuclear instrumentation is a component of the reactor protection system. Alarms, trips, and protective actions are generated in the instrumentation and transferred to the reactor protection system.

Power is typically provided from the reactor protection system 120-V ac or a 120-V ac vital instrument bus.

B. See Section I, "Areas of Review," of SRP-LR B.0.1 for item I.B.

C. Typical examples of age-related degradation associated with neutron monitoring instrumentation are provided below. The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

All detectors inside the containment are qualified to a specific life and replaced before that lifetime ends. Potential problems include cable aging, detector burnup, and connector aging. The neutron monitoring instrumentation is subject to random electronic component failure and system drift. Most age-related degradation is identified during routine testing and calibration programs.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1 for Items A. and B.

C. In addition to the above items, include the following:

1. Section 4.4 of IEEE Standard 279-19711) (Ref. 1) requires, for those plants issued a construction permit after 1971, type test data (or reasonable engineering extrapolation of type test data) that verify that this instrumentation meets, ON A CONTINUING BASIS [emphasis added], the system performance requirements. Older plants may also be committed to IEEE Standard 279-1971.

2. Regulatory Guide 1.97 (Ref. 2), which was backfitted on all plants as part of NUREG-0737, Supplement (Ref. 3), requires neutron flux instrumentation that is installed and maintained

in accordance with Category 1 criteria. Some licensee' installed, or are installing, additional excore neutron monitoring instrumentation in response to this requirement. Other licensees (both BWR and PWR) upgraded, or are upgrading, their existing neutron flux monitoring instrumentation in response to this requirement. Some PWR licensees had instrumentation that met this criteria.

3. These criteria should be part of an established ongoing licensee program to ensure the availability of the neutron flux instrumentation. The licensee may also propose a one-time or new periodic inspection of components of the neutron flux instrumentation. For example, the licensee may institute an in-containment cable check and evaluation (Ref. 4). In addition, the licensee's program should contain provisions for adding new testing and evaluation criteria to monitor newly detected aging degradation (Ref. 5).

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LRB.0.1 for Items III.A through III.D.

E. Components of the neutron monitoring system not specifically addressed in this section may be addressed in the generic aging topic reviews in Part C of the SRP-LR. The following sections of Part C are applicable to the neutron monitoring system and should be used for the review: C.2.1 "Cable and Wiring"; C.2.2, "Junctions"; C.2.3, Electrical Penetrations"; and all of C.3. "Instruments". This may require other staff input.

F. The plant license renewal review should show the existence of a well-planned and implemented surveillance and maintenance program. This program should include the periodic replacement of the neutron flux detectors. The review should also establish that the licensee has shown the acceptability of or replaced the in-containment cables associated with the neutron flux monitoring instrumentation.

The licensee's program for circumventing age-related degradation of the neutron monitoring system should include a continuing program that assesses the acceptability of the in-containment cables associated with this system.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

The electronic circuitry, including amplifiers and power supplies, is located outside the containment. The primary contributors to failure of this equipment are overheating and electrical transients. Electrolytic capacitors, fuses, indicators, transformers, and semiconductors are susceptible to aging degradation that is accelerated by these stresses. Aging deterioration can be detected by periodic equipment surveillance, output analysis, and component (mostly capacitor) parameter measurements. A comprehensive maintenance program will address inspection, cleanliness, testing, predictive maintenance, and corrective maintenance.

A. The detectors are part of the environmental qualification program. A detector is replaced periodically so that it is always at the point in its qualified life when it can perform during and following an accident. Cables run through the containment and in auxiliary building areas are addressed as part of SRP-LR C.2.1 of this SRP-LR. The electronic circuitry is addressed as part of SRP-LR C.1.5. That review will include the power supplies and the nuclear instrument racks.

VII. REFERENCES

1. Institute of Electrical and Electronics Engineers, IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations."

2. Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident."

3. U. S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements," NUREG-0737, Supplement 1.

4. U.S. Nuclear Regulatory Commission, "Inspection, Surveillance, and Monitoring of Electrical Equipment Inside Containment of Nuclear Power Plants - With Applications of Electrical Cables," Vol. 1, NUREG/CR-4257, August 1985.

5. U.S. Nuclear Regulatory Commission, "Nuclear Plant-Aging Research on Reactor Protection Systems," NUREG/CR-4740, January 1988.

B.1.7 REACTOR WATER CLEANUP SYSTEM (BWR)

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SPLB/SRXB/EMCB

I. AREA OF REVIEW

A. This section addresses the reactor water cleanup system (RWCS).

1. Description

The RWCS consists of a pumping system that takes suction on both recirculation loop suction lines and the reactor vessel bottom head, pumps the water through heat exchanger and ion exchange facilities, and pumps the water back to the reactor vessel via the feedwater piping or recirculation loop discharge line.

The RWCS is described in the most recent revision of the fina safety analysis report (FSAR) or updated safety analysis report (USAR) for the facility.

2. System Function

The RWCS maintains water effluent conductivity typically less than 0.3 "micro- and undissolved solids at less than 0.01 par per million (PPM); and chlorides and sulphates less than 15 parts per billion (PPB). The RWCS is normally operated continuously during all phases of reactor plant operation, startup, shutdown, and refueling.

3. System Boundaries

The boundaries of the RWCS extend from the reactor pressure vessel and recirculation loop piping back to the point where the coolant enters the feedwater piping. All pumps, pipes, valves, heat exchangers, ion exchangers, and instrumentation within the boundary of the RWCS.

B. See Section I, "Area of Review," of SRP-LR B.1.0 for Item I.B.

C. No specific aging concerns are associated with the RWCS other than those generic aging concerns discussed in Part C of the SRP-L for the various components in the RWCS.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.1.0.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.1.0.

E. RWCS components not specifically addressed this section may be addressed by the generic aging topic reviews in Part C of the SRP-LR. Specifically, the reviewer should review all of the generic license renewal topics except for C.2.5, "Transformers," and C.4.0, "Civil Structure." This may require other staff input.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.1.0.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.1.0.

VI. GENERAL INFORMATION

VII. REFERENCES

B.1.8 STANDBY LIQUID CONTROL SYSTEM (BWR)

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SICB/SRXB/EMCB

I. AREAS OF REVIEW

A. This section addresses the standby liquid control system (SLCS).

1. Description

The SLCS is a high-pressure pumping system that injects a neutron absorber solution (sodium pentaborate) into the core. The highly concentrated boron solution is maintained in solution by trace heating. A batch mix tank supplies the boron solution to a storage tank, which in turn supplies two 100 percent capacity pumping trains. The system also has an accumulator at the pump discharge to absorb pressure swings and relief valves to prevent piping damage due to overpressurization. A miniflow line provides a flow path from the test tank through the pumps returning to the test tank.

Each pumping train consists of a piston-type positive displacement pump with an associated explosive discharge valve and motor-operated suction valve. The explosive discharge valves of both trains discharge into a common line that enters the drywell and has a facility-specific path for injecting the sodium pentaborate solution into the reactor pressure vessel.

2. System Function

The SLCS provides an alternate method independent of the control rods for inserting negative reactivity into the core to render the reactor subcritical. If the control rods fail to insert on demand anticipated transient without scram (ATWS), the SLCS is initiated manually or automatically to inject the boron solution to shut down the reactor. Sufficient solution is injected to ensure adequate margin in order to maintain the reactor subcritical at cold shutdown conditions.

3. System Boundaries

The SLCS includes the boron storage tank, two independent pumping trains, a system test tank, and a delivery path to the reactor vessel. The injection point varies at different facilities from an injection sparger in the bottom head to an injection nozzle into the high-pressure core spray line. All pumps, tanks, accumulators, valves, piping, interconnections, and cross-connections are included in the SLCS, but these individual components will be reviewed during the generic reviews.

The electrical boundaries include electrical power for heaters and for the pumps.

The instrumentation and control (I&C) boundaries of the SLCS include equipment that is required to perform plant-specific functions that are considered significant to the SLCS's safety function. Most of these components are addressed as generic I&C equipment and include, but are not limited to, control switches, relays, controllers, sensors, transmitters, recorders, computational modules, and circuit breakers.

B. See Section I, "Area of Review," of SRP-LR B.0.1 for Item I.B.

C. A review of licensee event reports showed that four types of SLCS failures have affected the operability of the SLCS. These failures are listed below and are issues that should be reviewed the license renewal process.

1. System Relief Valve Set Point Drift

   The set point for SLCS relief valves drifted to a relief point that was the technical specification limit. The relief point ranged from as low as 600 psig to as high as 1540 psig. In at least one case, crystallized sodium pentaborate was found on valve surfaces; the reasons for the other occurrences remain unexplained.

2. Heat Trace Failures

   Two cases of failures of heat tracing occurred. These cases were both corrected by modification. Heat trace failures are detected by routine temperature monitoring and surveillances.

3. Test Loop Throttle Valve Failure - Disc Separation

   Two cases of a failure of the test loop throttle valve occurred in which the valve disc/plug separated from the valve stem. In each case the valve was repaired by replacing the stem and disc/plug. This type of failure appears to be due to a cyclic stress mechanism and thus is an aging concern.

4. Accumulator Precharge Loss

   Three cases of loss of accumulator precharge occurred. In two cases, a leak at the nitrogen charging valve stem was the cause; in the third case, a ruptured bladder was the cause. These type of failures are common to accumulators and are covered in SRP-LR C.1.5, "Tanks and Vessels."

   Typical examples of age-related degradation associated with the SLCS are provided above. The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for Items III.A. through III.D.

E. SLCS components not specifically addressed in this section may be addressed by the generic aging topic reviews in Part C of the SRP-LR. The reviewer should ensure that structures and components included as part of the generic SRP-LR topics are adequately reviewed for this system. The following sections of SRP-LR Part C should be issued for the review: C.1.1, "Piping"; C.1.2, "Valves"; C.1.3, "Pumps"; C.1.5, "Tanks and Vessels"; C.1.6, "Equipment and Component Supports"; all of C.2.0, "Electrical" except C.2.5, "Transformers"; all of C.3.0, "Instrument". This may require other staff input.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

The failure events discussed in Item I.A.3.C above were identified by a review of licensee event reports for the last 10 years. Age-related failures were screened by component. Pacific Northwest Laboratory is conducting research on the SLCS aging effects. The results of that research will be included as a section in NUREG/CR-5562, "A Review of Information Useful for Managing Aging in Nuclear Power Plants" (Ref. 1).

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "A Review of Information Useful for Managing Aging in Nuclear Power Plants," NUREG/CR-5562 (PNL-7323), 1990.

B.1.9 CHEMICAL AND VOLUME CONTROL SYSTEM AND EMERGENCY BORATION SYSTEM (PWR)

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SRXB/EMCB

I. AREAS OF REVIEW

A. This section addresses the chemical and volume control system (CVCS) and the emergency boration system.

1. Description

The CVCS or the emergency location system consist of those components and controls that provide the normal makeup and purification function for the reactor coolant system. This system also includes all control and components providing the emergency or rapid boration capability.

The CVCS or the emergency boration system are described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for a specific facility.

2. System Function

The portions of the CVCS that serve the emergency core cooling system, the reactor coolant pump shaft seals and the emergency boration system are designed to seismic Category I. These systems provide the following functions:

a. Adjusts the reactor coolant system (RCS) boric acid concentration

b. Maintain the proper water inventory in the reactor coolant system

c. Provide seal water flow to the reactor coolant pump shaft seals

d. Maintain the proper concentration of corrosion-inhibiting chemicals in the reactor coolant

e. Purify the reactor coolant by removing impurities to maintain RCS ionic inpurities and fission products within operating limits

f. Provide borated water for emergency core cooling

g. Process reactor coolant for reuse of boric acid and reactor makeup water

h. Degass the reactor coolant system

i. Provide a means of emergency boration of the RCS

j. Provide a hydrostatic pressure capability for those systems that use a positive displacement pump

k. Provide dissolved hydrogen to control and seavenge oxygen generated by radioloysis of water in the reactor core

3. System Boundaries

Extraction from and input to the reactor coolant system form two major boundaries. Extraction flow, normally termed "letdown", generally taps off an intermediate section of cold-leg RCS piping through two series isolation valves and a letdown delay pipe to a regenerative heat exchanger. Input or charging occurs with one to three charging pumps which supply preheated water to a choice of RCS loops.

Other system boundaries may be established at tie-in points for such systems as the reactor makeup system, seal injection system, and excess letdown system (Westinghouse only) and at the check valves leading back to the boric acid tanks in the emergency boration system (called rapid boration system at some plants).

B. See Section I, "ARea of Review," of SRP-LR B.0.1 for Item I.B.

C. No system-specific areas of aging concern have been identified at this time other than those associated with pumps, piping, heat exchangers, and tanks and those associated with the instrumentation.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for Items III.A through III.D.

E. CVCS components not specifically addressed in this section may be addressed by the generic aging topic reviews in Part C of the SRP-LR. The reviewer should ensure that structures and components included as part of the generic SRP-LR topics are adequately reviewed for this system. All the sections of SRP-LR Part C (Sections C.1.0, "Mechanical"; C.2.0, "Electrical"; and C.3.0, "Instrument" should be used for the review.

In addition, the reviewer should coordinate the review of the CVCS and the emergency boration system with that of the reactor control system (SRP-LR B.1.3).

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "A Review of Information Useful for Managing Aging in Nuclear Plants," NUREG/CR-5562, (PNL-7323), June 1990.

B.2.1 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (PWR)

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SICB

I. AREAS OF REVIEW

A. This section addresses the engineered safety features actuation system (ESFAS).

1. Description

The ESFAS portion of the reactor protection system (RPS) consists of the instrumentation and control elements that sense plant conditions and actuate emergency equipment in the event of a reactor transient or accident, including a loss of primary or secondary system coolant inventory. Engineered safety features actuation is necessary for providing emergency core cooling, for maintaining reactor building integrity, and for mitigating the consequences of a reactor transient or accident.

The ESFAS is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the facility.

2. System Function

The ESFAS actuates engineered safety features equipment in the event of a reactor transient or accident, including a loss of primary or secondary system coolant inventory, to provide emergency core cooling, to maintain reactor building integrity, and to mitigate the consequences of the reactor transient or accident.

3. System Boundaries

The ESFAS includes both the automatic and manual initiation of the engineered safety features (ESF) systems and the essential auxiliary supporting (EAS) systems. The ESFAS includes the power sources, sensors, transmitters, initiation circuits, logic matrices, bypasses, interlocks, racks, cables, panels, and control boards, and actuation and activated devices that are required to actuate and control the ESF and EAS equipment. This section of the SRP-LR also includes the review of control systems that regulate the operation of ESF components following their initiation by the protection system. The review of instrumentation and control elements for EAS systems are addressed in the SRP-LR sections applicable to each specific system.

The objectives of this review are to confirm that the ESFAS satisfies the acceptance criteria and guidelines for age-related degradation applicable to the ESFAS and ESF systems. The review also confirms that the controls for ESF systems satisfy the acceptance criteria and guidelines applicable to ESF systems, including their performance requirements.

Actuation of the ESFAS and subsequent initiation of ESF systems are initiated by the following typical parameters. Other parameters may also be used.

a. Reactor coolant system pressure low/pressurizer pressure low
b. Reactor building pressure high/containment pressure high
c. Steam generator pressure low
d. Steam generator differential pressure high
e. Main steamline flow high
f. Manual operator action

Typical ESF systems are (other ESF systems may also be used):

a. Containment and reactor vessel isolation systems
b. Emergency core cooling systems
c. Containment heat removal and depressurization systems
d. Auxiliary/emergency feedwater systems
e. Containment air purification and cleanup systems
f. Containment combustible gas control systems

The ESFAS is isolated from the ESF systems it actuates, the reactor control systems, the reactor trip system excluding sensors that may be shared with the ESFAS, and the associated normal and emergency ac and dc power systems by circuit breakers, isolation amplifiers, isolation transformers, actuation logic, fuses, and other approved isolation devices.

B. See Section I, "Area of Review," of Section B.0.1 for Item I.B.

C. Measures must be taken to monitor systems, components, and interfaces to detect degradation and if necessary, to restore integrity through maintenance, repair, or replacement (Ref. 1).

1. Typical examples of age-related degradation associated with the ESFAS are the following:

   a. Age-related degradation due to setpoint drift

   b. Age-related degradation due to functional testing cycles and trips

c. Age-related degradation due to improper maintenance or repair

d. Age-related degradation of sensors, connectors, cables and wiring, circuit breakers, relays, electronic components. etc.

The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

2. Electrical systems and instrumentation and control systems and their components are subject to age-related degradation. A wide variety of age-related mechanisms can affect the ability of these systems and components to operate reliably. Several of these systems and components are listed below. The age-related degradation mechanisms are discussed in Part C of the SRP-LR.

a. Electrical Systems and Components

- o Cables and wiring
- o Junctions
- o Penetrations
- o Relays and switchgear
- o Transformers
- o Solenoid-operated valves
- o Electrical motors

b. Instrumentation and Control Systems and Components

- o Sensors
- o Electronic components
- o Electronic devices

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for Items A. through D.

E. Components of the ESFAS not specifically addressed in this section may be addressed in the generic aging topic reviews in Part C of the SRP-LR. The following sections of Part C are applicable to the ESFAS and should be used for the review: C.1.6, "Supports"; C.2.1 "Cable and Wiring"; C.2.2, "Junctions"; C.2.3, "Electrical Penetrations"; C.2.4, "Relays, Circuit Breakers, and Switchgear"; C.2.5, "Transformers"; C.2.6, "Solenoid-operated Valves"; C.2.7, "Electrical Motors"; and all of C.3.0 "Instruments." This may require other staff input.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

Reference 2 indicates that, in general, the current testing programs are adequate for the intended purpose of verifying RPS operability and performance.

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "An Aging Failure Survey of Light Water Reactor Safety Systems and Components," Vol. 2, NUREG/CR 47-47 (EGG-2473) July 1988.

2. U.S. Nuclear Regulatory Commission, "Nuclear Plant Aging Research of Reactor Protection Systems," NUREG/CR-4740 (TI88 007920), January 1988.

B.2.2.1 REACTOR CORE ISOLATION COOLING SYSTEM (BWR)

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SRXB

I. AREA OF REVIEW

A. This section addresses the reactor core isolation cooling (RCIC) system.

1. Description

The typical BWR RCIC system provides a limited source of makeup water and cooling water to the reactor vessel during shutdown conditions when the main feedwater is unavailable and the reactor is pressurized with steam. The RCIC system is capable of startup independent of auxiliary ac power, plant service air, and external cooling water systems.

The RCIC system is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the facility.

2. System Function

The RCIC system is classified as an engineered safety feature (ESF) in many BWR plants and as a safe shutdown system in some plants. At least one plant classifies the RCIC system as not being safety related. Early model BWRs (typically BWR/2 and some BWR/3 plants) may have an isolation condenser system or emergency condenser system instead of a RCIC system.

3. System Boundaries

The main components of the RCIC system include the steam-turbine and pump unit and associated instruments, controls, piping, and valves.

RCIC system boundaries may differ from plant to plant and are described in each licensee's FSAR. In general, boundaries exist at the reactor vessel main steamline inboard of the steam isolation valve. At this interconnection, steam from the reactor vessel is supplied to the RCIC turbine stop valve during standby conditions and to the turbine during RCIC system operation. An additional reactor vessel boundary exists at the head spray nozzle where makeup water is supplied to the vessel during RCIC system operation. Other boundaries exist at the condensate storage tank and suppression pool where makeup water is supplied and where steam exhaust is directed to the suppression pool. For plants that use a steam-condensing mode of residual heat removal (RHR), an RCIC/RHR system boundary exists in the RCIC pump line where makeup water can be drawn from the RHR heat exchanger. All of the associated piping, valves, lube oil cooler systems, water leg pumps, and gland seal compressors should be considered a portion of the RCIC system.

B. See Section I, "Area of Review," of SRP-LR B.0.1 for Item I.B.

C. Aging-related failures have been experienced in virtually all of the main components of the RCIC system (Ref. 1) including the following:

- Valves and operators
- Instrumentation
- Electrical components
- Piping and supports
- Turbine and pump

These observed failures are the result of a number of age-related degradation mechanisms, including corrosion, wear, temperature effects, etc.

The above failures are typical examples of age-related degradation associated with the RCIC system. The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1 for Items A. and B.

C. Specific areas of concern that should be addressed by the licensee for monitoring aging in the RCIC system include the following:

1. An inspection and surveillance program that addresses the condition and on-demand response of RCIC system components using information presented in Table B.2.2.1-1 (Ref. 1) to guide the licensee in targeting components that are susceptible to age-related failures.

Table B.2.2.1-2, (Ref. 2) provides aging processes typical for the RCIC system.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for items A. through D.

E. Components of the RCIC system not specifically addressed in this section may be addressed in the generic aging topic reviews in Part C of the SRP-LR. The following sections of Part C are applicable to the RCIC system and should be used for the review: C.1.1, "Piping"; C.1.2, "Valves"; C.1.3, "Pumps"; C.1.6, "Supports"; C.2.1, "Cable and Wiring"; and all of C.3.0, "Instruments." This may require other staff input.

F. The reviewer should ensure that the licensee's aging management program specifically addresses a one-time inspection and surveillance program to assess RCIC system components susceptible to age-related degradation as indicated in Tables B.2.2.1-1 and B.2.2.1-2 and to inspect components exempt from ASME Code, Section XI requirements.

The reviewer should ensure that the licensee's aging management program establishes appropriate increases to future inspection frequencies for marginal conditions identified.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

In a light-water-reactor (LWR) aging survey, time-dependent failure contributions of safety system components were analyzed using data compiled from the Nuclear Plant Reliability Data System (NPRDS) data base (Ref. 1). The failure data are identified by system and grouped in five failure categories, one of which includes age-related failures. Table 2.2.1-1 illustrates that for the RCIC system, approximately 30 percent of the observed failures could be associated with age-related degradation. RCIC system components contributing to failures within the system are listed in Table B.2.2.1-1, which identifies the component failure fractions by failure category and ordered by aging fractions. Of the failures associated with aging recorded, valve failures had the highest occurrence with approximately 50 percent of these failures attributed to aging. Although valve operators and instrumentation switches also had a relatively high number of failures, a lower percentage of these failures were attributed to aging. Supports, instrumentation controllers, transmitters, and recorders, and turbines had failures attributed to aging in each case of less than 35 percent.

The RCIC system is important to plant safety. However, aging mechanisms associated with the degradation of RCIC system performance are not well defined. Aging mechanisms in other LWR systems that include components within the RCIC system can provide some guidance. In general, the turbine is a component in which aging mechanisms have been identified.

Although the turbines, themselves, have been shown to be relatively reliable and rugged, the turbine auxiliary systems (e.g., governor controls, trip and throttle valves) have contributed significantly to operational failures in this system (Ref. 2).

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "An Aging Failure Survey of Light Water Reactor Safety Systems and Components," NUREG/CR-4747 (EGG-2473), July 1987.

2. U.S. Nuclear Regulatory Commission, "A Review of Information Useful for Managing Aging in Nuclear Power Plants," NUREG/CR-5562, 1990.

Table B 2.2.1-1 Reactor core isolation cooling component failure category fractions

Component	Total	Design	Aging	Testing	Human	Other
Motor	7	0.143	0.571	0.143	----	0.143
Pipe	2	----	0.500	----	----	0.500
Instrumentation: Computation Module	10	0.100	0.500	0.100	----	0.300
Valve	199	0.080	0.492	0.075	0.020	0.332
Instrumentation Recorder	26	0.115	0.346	----	0.077	0.462
Support	36	0.056	0.278	0.028	----	0.639
Valve Operator	144	0.090	0.229	0.111	0.007	0.562
Instrumentation: Transmitter	44	0.045	0.227	0.023	0.023	0.682
Circuit Breaker	34	0.147	0.206	0.059	----	0.588
Pump	5	0.200	0.200	0.200	----	0.400
Instrumentation: Controller	27	0.111	0.185	0.074	----	0.630
Mechanical Function Unit	17	0.118	0.176	0.059	----	0.647
Turbine	31	0.129	0.161	0.129	0.032	0.548
Relay	10	0.200	0.100	0.300	0.100	0.300
Instrumentation: Switch	176	0.085	0.097	0.108	0.017	0.693
Generator/Invertor/ Alternator	5	----	----	----	----	1.000
Electrical Conductor	1	----	----	----	----	1.000
Total	774					

Table B.2.2.1-2 Summary of aging process for the high pressure injection system

Major component	Stressors	Degradation mechanisms	Potential failure modes	Inservice inspection methods
Nozzles and thermal sleeves	System operating transients, thermal cycling, vibration, water-hammer	Fatigue crack initiation and propagation	Leaks through wall, loose parts	Visual inspection, volumetric inspection
Valves	System operating transients, maintenance, testing	Wear, foreign material, mechanical leakage, faults	Leakage, failure to operate, blockage, command faults	Visual and operational tests
Air-operated valves	System operating transients, contaminated air supply	Sticking, blockage, fouling water, oil	Failure to operate	Visual and operational tests
Instrumentation and controls	Electrical transients, thermal cycles, maintenance	Corrosion, loose connections failure (catastrophic)	Open, shorts, failure to operate	Testing
Pumps	System operating transients, thermal cycles	Wear, vibration, fatigue	Seal leaks, failure to start, failure to run	Testing, visual inspection
Pipe Supports	Vibration, water hammer	Fatigue, loosening of abrasive water	Breaking loose	Visual inspection
Piping	Vibration, water-hammer, thermal cycles	Thermal fatigue abrasive water	Through the wall leaking or cracks	Visual inspection, volumetric inspections

B.2.2.2 HIGH-PRESSURE AND INTERMEDIATE-PRESSURE INJECTION SYSTEM (PWR)

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SRXB

I. AREA OF REVIEW

A. This section addresses the high-pressure injection system (HPIS) and the intermediate pressure injection system.

1. Description

The HPIS and SIS are emergency core cooling systems (ECCSs). Plant-specific designs are subject to considerable variation with respect to the equipment used and its interface with normal operation of the plant at some Westinghouse and Combustion Engineering plants, the normal makeup (charging) pumps provide high-pressure injection capability during small-break loss-of-coolant accidents (LOCAs) which do not rapidly depressurize the reactor at these plants, additional intermediate head safety injection pumps (SIS) are used as part of the HPIS to provide coolant injection. At Babcock and Wilcox (B&W) plants, HPIS pumps provide reactor coolant injection for the full range of LOCAs up to that which depressurizes the reactor sufficiently for the low pressure injection system to operate (typically the residual heat removal system, which is discussed in Section B.2.2.4 of the SRP-LR). The B&W HPIS pumps also provide reactor makeup water during normal operation.

In most designs, the HPIS includes two fully independent 100 percent capacity trains, which pump borated water from a storage tank to the reactor pressure vessel (RPV) cold-leg piping under small-break LOCA conditions. Some HPIS equipment is also used to provide reactor makeup water during normal operation, which is not an ECCS function. In those plants where the intermediate-pressure system (SIS) is used, typically two independent 100 percent capacity trains are provided which pump borated water from a storage tank to the RPV cold-leg piping under small to intermediate-break LOCA conditions. The SIS also can be manually switched over to provide hot-leg injection during long term LOCA events to mitigate boric acid precipitation in the core, which could hinder core cooling.

The HPIS and SIS are described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the facility.

2. System Function

The HPIS and SIS are designed to provide sufficient coolant injection to ensure cooling of the reactor core in the event of a small to intermediate-break in the reactor coolant pressure boundary. They are also designed to provide emergency boration which ensures adequate shutdown margin to mitigate such events as secondary steamline breaks, steam generator tube ruptures, and control rod drive mechanism housing ruptures (rod ejection accident).

3. System Boundaries

The boundaries of the HPIS and SIS extend from the refueling/borated water storage tank through the injection pump to the cold leg(s) supplying water to the reactor. The boundaries of the SIS also include connections to the hot legs. All valves, piping, and interconnections are included.

The normal charging and suction lines to the injection pumps are only included up to the block valves nearest to the pumps.

B. See Section I, "Area of Review," of SRP-LR B.0.1 for Item I.B.

C. Table B.2.2.2-1 provides typical examples of age-related degradation mechanisms and failure modes for the various components of the HPIS (Refs. 1 and 2). Similar age-related degradation mechanisms and failures have been experienced in the SIS.

The areas of aging concern for a facility sl uld be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for Items IV.A. through III.D.

E. HPIS and SIS components not specifically addressed in this section may be addressed by the generic aging topic reviews in Part C of the SRP-LR. The reviewer should ensure that structures and components included as part of the generic SRP-LR topics are adequately reviewed for this system. The following sections in SRP-LR Part C should be used for the review: C.1.1, "Piping"; C.1.2, "Valves"; C.1.3, "Pumps"; C.1.6, "Equipment and Component Supports"; all of C.2.0, "Electrical"; and all of C.3.0, "Instruments." This may require other staff input.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

The HPIS and SIS are designed with redundant channels, which interact with many of the other reactor systems making item vulnerable to common-mode failures. However, for failures associated with the HPIS and SIS, maintenance error is the leading cause of system failure. The effect of aging on these systems is primarily through aging of the components. The HPIS components with the highest frequency of failure were valves, followed by instrumentation and control (I&C) components, pumps, and pipes. The pipe failures usually were leaks through cracks. For example, one of the piping concerns involves cracks in nozzles and thermal sleeves that had occurred in B&W and Westinghouse plants. In this case, the cracks were attributed to fatigue due to thermal cycling. Boric acid leaks are potentially serious because of the corrosive action of boric acid on carbon steel. An example might be corrosion of bolts, which could lead to leaks in the pressure boundary. Flow blockages may also occur if boric acid crystals are allowed to form as a result of loss of temperature or boration concentration control.

Because of redundancy, only about 0.7 percent of the HPIS and SIS failures caused total loss of system function, and 21.3% of the component failures were age related. Plant records reveal that many small leaks and problems with I&C are required before major failures develop. Thus, the use of plant records is one method for identifying incipient failures (Ref. 3).

Personnel at the plants usually follow the manufacturer's recommendations for maintenance of major components. About 28 percent of abnormal occurrences at PWRs is due to faulty maintenance and surveillance testing. Current inspection methods include visual inspection, volumetric inspection, and operational tests (Ref. 3).

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "Nuclear Plant-Aging Research on High Pressure Injection Systems," NUREG/CR-4967 (EGG-2514), November 1987.

2. L.C. Meyer, "Nuclear Plant Aging Research on the High Pressure Injection System," Proceedings of the Topical Meeting on Nuclear Power Plant Life Extension, July 31-August 3, 1988, Snowbird, Utah Vol. 1.

3. U.S. Nuclear Regulatory Commission, "A Review of Information Useful for Managing Aging in Nuclear Power Plants," NUREG/CR-5562 June 1990.

Table B.2.2.2-1 Summary of aging process for the high pressure injection system

Major component	Stressors	Degradation mechanisms	Potential failure modes	Inservice inspection methods
Nozzles and thermal sleeves	System operating transients, thermal cycling, vibration, water-hammer	Fatigue crack initiation and propagation	Leaks through wall, loose parts	Visual inspection, volumetric inspection
alves	System operating transients, main-tenance, testing	Wear, foreign material, mecha ical leakage, fault	Leakage, failure to operate, blockage, command faults	Visual and operational tests
Air-operated valves	System operating transients, contam-inated air supply	Sticking, blockage, fouling water, oil	Failure to operate	Visual and operational tests
Instrumentation and controls	Electrical tran-sients, thermal cycles, maintenance vibration	Corrosion, loose connections failure (catastrophic)	Open, shorts, failure to operate	Testing
Pumps	System operating transients, thermal cycles	Wear, vibration, fatigue	Seal leaks, failure to start, fail to run	Testing, visual inspection
Pipe supports	Vibration, water-hammer	Fatigue, loosening of abrasive water	Breaking loose	Visual inspection
Piping	Vibration, water-hammer, thermal cycles	Thermal fatigue abrasive water	Through the wall leaking, or cracks	Visual inspection, volumetric inspection

B.2.2.3 CORE FLOOD SYSTEM (PWR)

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SRXB

I. AREAS OF REVIEW

A. This section addresses the core flood system (CFS).

1. Description

The CFS in addition to a high pressure injection system and a low pressure injection system is a part of the emergency core cooling system (ECCS) of a PWR.

Typically, the core flood tanks contain borated water and are pressurized to about 600 psig. Nitrogen is used to provide the charging pressure. The outlet of each tank is connected to a check valve that directs flow out of the tank. In series with the check valve is a motor-operated isolation valve. Under normal operating conditions, the motor-operated valves (MOVs) are open and reactor coolant pressure against the check valve outlets keeps the check valves closed. In the event of a large loss-of-cool-and-accident (LOCA), reactor vessel pressure will decrease. When the pressure has decreased below the charging pressure in the tanks, the check valves will open, injecting cooling water into the vessel. The tanks contain enough borated water to cover the reactor core. In the cold shutdown condition when reactor vessel pressure is not high enough to keep the check valves closed, the MOVs are kept closed.

The CFS is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the facility.

2. System Function

The CFS system is a passive system that requires no external signal or power source to operate. It is designed to inject cooling water rapidly into the reactor vessel when vessel pressure falls below a predetermined level. It also provides sufficient borated water to cover the reactor core in the event of a large LOCA.

3. System Boundaries

The CFS for a PWR extends from the connections for the supply of fill water through the core flood tanks to the points of water injection into the reactor core. All valves, piping, and interconnections are included. Also included is the gas system used to pressurize the core flood tanks and an interface with the radioactive liquid waste system for draining the tanks.

B. See Section I, "Area of Review," of SPR-LR B.0.1 for Item I.B.

C. Typical examples of age-related degradation associated ith the CFS are the following:

1. Corrosion stressors - electrical, mechanical, and thermal

2. Environmental factors - chemical, atmospheric, and underground

3. Corrosion/fouling phenomena - uniform and pitting corrosion

4. Intergranular stress corrosion cracking - alloy selection and treatment

The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of Section B.0.1 for Items III A. through III D.

E. CFS components not specifically addressed in this section may be addressed by the generic aging topic reviews in Part C of the SRP-LR. The reviewer should ensure that structures and components included as part of the generic SRP-LR topics are adequately reviewed for this system. The following sections of SRP-LR Part C should be used in the review: C.1.1, "Piping"; C.1.2, "Valves"; C.1.5, "Tanks and Vessels"; C.1.6, "Equipment and Component Supports"; and C.3.0, "Instruments." This may require other staff input.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

VII. REFERENCES

B.2.2.4 RESIDUAL HEAT REMOVAL SYSTEM

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SRXB

I. AREAS OF REVIEW

A. This section addresses the residual heat removal (RHR) system.

1. Description

The typical BWR RHR system performs one or more of the following functions, depending on plant design:

Restores and maintains desired water level in the reactor vessel following a loss-of-coolant-accident (LOCA); condenses steam and reduces airborne activity in the containment following a LOCA; removes heat from the suppression pool; removes decay heat from the core following a reactor shutdown; condenses reactor steam and returns the condensate back to the reactor vessel via the reactor core isolation cooling (RCIC) System; provides fuel pool cooling if capacity beyond the normal system is required; and floods the containment if required for long term post LOCA recovery operations.

For a PWR plant, the main function of the RHR system is to remove decay heat from the reactor during plant shutdown. Some PWR plants use the RHR pumps for LPI service.

The specific description of the RHR system is contained in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the facility.

2. System Function

The RHR system is used to bring the reactor to safe shutdown condition and to mitigate the consequences of an accident. The RHR pumps restore and maintain reactor coolant inventory following a large-break LOCA. The system removes heat from the suppression pool (BWR) or the reactor coolant (PWR) in long term cooldown mode following a LOCA. The RHR system can also provide cooling to the spent fuel pool if cooling capacity is needed beyond the normal system capacity.

3. System Boundaries

Typically the RHR system interfaces with reactor systems, containment systems including suppression pool, the service water system, and the instrument air system.

The normal water supply of the RHR system is the suppression pool for BWR and the refueling water storage tank for PWR. Some BWR facilities may have RHR cross-ties to the recirculation system for shutdown cooling or the fuel pool and cleanup for supplemental fuel pool cooling.

B. See Section I, "Area of Review," of SRP-LR B.0.1. item I.B.

C. There are a variety of age related mechanisms that can affect the ability of the RHR to continue to operate safely and efficiently. Most of these concerns are generic in nature and are addressed by the generic topics of Part C of the SRP-LR. In addition to the generic aging concerns spray nozzles and spargers are subject to erosion/corrosion and radiation embrittlement. Table B.2.2.4-1, Summary of Aging Processes for the High Pressure Injection System (Ref. 1), provides aging processes typical for the RHR.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1.

III. REVIEW PROCEDURES

See Section II, "Review Procedures," of SRP-LR B.0.1 for items III.A through III.D.

E. RHR system components not specifically addressed in this section may be addressed by the generic aging topic reviews in Part C of the SRP-LR. The reviewer should ensure that structures and components included as part of the generic SRP-LR topics are adequately reviewed. The sections of SRP-LR Part C applicable to the RHR system are: C.1.0, "Mechanical" (all); C.2.0 "Electrical" (except C.2.5, "Transformers); and C.3.0 "Instrument" (all). This may require other staff input.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "A Review of Information Useful for Managing Aging in Nuclear Power Plants," NUREG/CR-5562 (PNL-7323), June 1990.

Table B.2.2.4-1 Summary of aging process for the high pressure injection system

Major component	Stressors	Degradation Mechanisms	Potential Failure modes	ISI methods
Nozzles and thermal sleeves	System operating transients, thermal cycling, vibration, water hammer	Fatigue crack initiation and propagation	Leaks through wall, loose parts	Visual inspec-, tion volumetric inspection
Valves	System operation transients, maintenance, and testing	Wear, foreign material, mechanical leakage, faults	Leakage, fail to operate, blockage, command faults	Visual and operation tests
Air-operated valves	Systems operating transients, contaminated air supply	Sticking, blockage, fouling water, and oil	Fail to operate	Visual and operational tests
I&C	Electrical transients, thermal cycles, maintenance vibration	Corrosion, loose connections failure (catastrophic)	Open, shorts, fail to operate	Testing
Pumps	Systems operating transients, thermal cycles	Wear, vibration, fatigue	Seal leaks, fail to start, fail to run	Testing, visual inspection
Pipe Supports	Vibration, water hammer	Fatigue, loosening of abrasive water	Breaking loose	Visual inspection
Piping	Vibration, water hammer, thermal cycles	Thermal fatigue abrasive water	Through the wall leaking, or cracks	Visual inspection, volumetric inspections

B.2.2.5 CORE SPRAY SYSTEMS (BWR)

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SRXB

I. AREAS OF REVIEW

A. This section addresses the high pressure and low pressure core spray (HPCS and LPCS) systems.

1. Description

Both the HPCS and LPCS systems are classified as emergency core cooling systems (ECCSs).

Early BWRs have a two-loop LPCS system with a single-loop steam turbine-driven high pressure coolant injection (HPCI) system (the HPCI system is addressed in Section B.2.2.6). The more recent BWR-5 and BWR-6 plants have both the HPCS and LPCS systems, each containing a single loop.

The HPCS and LPCS systems are described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the facility.

2. System Function

The HPCS system is designed to provide spray cooling to the reactor core and to maintain reactor pressure vessel inventory following small-pipe breaks, which do not rapidly depressurize the reactor pressure vessel (RPV). It is a single-loop system taking suction from either the condensate storage tank or the suppression pool and discharging water above the core directly on the fuel bundles.

The LPCS system is designed to provide spray cooling to the reactor core and to assist other emergency core cooling systems mitigate the consequences of loss-of-coolant accidents for which the RPV is depressurized. The LPCS system is either a single- or two-loop system taking suction from the suppression pool and discharging water through a core spray sparger ring.

3. System Boundaries

The HPCS and LPCS systems include components from the suppression pool through the spray nozzles and include all pumps, piping, valves, instrumentation, controls and logic. The keep-full pumps, test lines, and all associated components are included as part of the core spray systems. The core spray systems also interface with the following systems:

a. Primary Containment

The suppression pool, which is part of the primary containment system, is used for normal suction and as a return for the minimum-flow and full-flow test lines. The primary containment is addressed in SRP-LR B.3.1.

b. Standby Auxiliary AC Power System

The diesel generators provide emergency backup power for the LPCS and HPCS systems. The emergency diesel generators are addressed in SRP-LR B.4.4.

c. Condensate Transfer and Storage System

This system is used for flushing the HPCS and LPCS systems.

d. DC Power System

The 125 Vdc system provides logic power and control and instrument power. The 24 Vdc system provides analog instrument power to the logic. These systems are addressed in SRP-LR B.4.3.1.

B. See Section I, "Area of Review," of SRP-LR B.0.1 for Item I.B.

C. A variety of age-related mechanisms can affect the ability of the HPCS and LPCS systems to continue to operate safely and efficiently. Most of these concerns are generic in nature and are addressed by the generic topic reviews in Part C of the SRP-LR. In addition to the generic aging concerns, spray nozzles and spargers are subject to erosion/corrosion and radiation embrittlement. Table B.2.2.5-1 (Ref. 1) provides aging processes typical for the HPCS and LPCS systems.

The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for Items III.A through III.D.

E. The HPCS and LPCS components not specifically addressed in this section may be addressed by the generic aging topic reviews in Part C of the SRP-LR. The reviewer should ensure that structures and components included as part of the generic SRP-LR topics are adequately reviewed for this system. The following sections of Part C should be used for the review: C.1.1, "Piping"; C.1.2, "Valves"; C.1.3, "Pumps"; C.1.6, "Equipment and Component Supports"; C.2.1, "Cable and Wiring"; C.2.3, "Electrical Penetrations"; C.2.4, "Relays, Circuit Breakers, and Switchgear"; C.2.7, "Electrical Motors"; and all of C.3.0, "Instruments". This may require other staff input.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "A Review of Information Useful for Managing Aging in Nuclear Power Plants," NUREG/CR-5562 (PNL-7323), June 1990.

Table B.2.2.5-1 Summary of aging process for the high-pressure injection system

Major component	Stressors	Degradation mechanisms	Potential failure modes	Inservice inspection methods
Nozzles and thermal sleeves	System operating transients, thermal cycling, vibration, water-hammer	Fatigue crack initiation and propagation	Leaks through wall, loose parts	Visual inspection, volumetric inspection
Valves	System operating transients, maintenance, testing	Wear, foreign material, mechanical leakage, faults	Leakage, failure to operate, blockage, command faults	Visual and operational tests
Air-operated valves	System operating transients, contaminated air supply	Sticking, blockage, fouling water, oil	Failure to operate	Visual and operational tests
Instrumentation	Electrical transients, thermal cycles, maintenance vibration	Corrosion, loose connections failure (catastrophic)	Open, shorts, failure to operatet	Testing
Pumps	System operating transients, thermal cycles	Wear, vibration, fatigue	Seal leaks, failure to start, fail to run	Testing, visual inspection
Pipe Supports	Vibration, water-hammer	Fatigue, loosening of abrasive water	Breaking loose	Visual inspection
Piping	Vibration, water-hammer, thermal cycles	Thermal fatigue, abrasive water	Through the wall leaking, cracks	Visual inspection, volumetric inspections

B.2.2.6 HIGH-PRESSURE COOLANT INJECTION SYSTEM (BWR)

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SRXB

I. AREAS OF REVIEW

A. This section addresses the high pressure coolant injection (HPCI) system.

1. Description

The HPCI system typically consists of a turbine-driven pump and a booster pump with associated piping, valves, and instrumentation. The turbine-driven pump is used to supply clean makeup water to the reactor vessel. The HPCI system is normally in a standby condition and will automatically start on a low reactor water level or high drywell pressure initiation signal. Normal suction for the HPCI system is the condensate storage tank with automatic switchover to the suppression pool.

The HPCI system is described in the most recent revision fo the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the facility.

2. System Function

The purpose of the HPCI system is to provide cooling in the reactor core under loss of-coolant-accident conditions that do not result in rapid depressurization of the reactor vessel. The HPCI system allows for complete plant shutdown while maintaining sufficient reactor water inventory until the reactor is depressurized to a point where low-pressure cooling systems can be placed into operation. The HPCI system is capable of operation independent of auxiliary ac power, plant service air, or external cooling water systems.

3. System Boundaries

The HPCI system is bounded by interconnections with other plant systems. The HPCI turbine-driven pump system supplies clean demineralized makeup water to the reactor vessel normally via the feedwater system. The HPCI water supply typically comes from the condensate storage tank, as an alternate source of makeup water is available from the suppression pool. The HPCI turbine is driven by steam from the main steam system. The HPCI steam supply line taps into a main steamline upstream of the main steam isolation valves. The steam exhausted from the HPCI turbine is discharged into the suppression pool.

The HPCI system interfaces with plant electrical and reactor instrumentation systems. The electrical system supplies ac and dc power to operate associated valves, turbine control, indicating lights, and system relay logic. The instrumentation system provides automatic initiation signals from drywell pressure and reactor vessel level instrumentation. Condensate storage tank (CST) level and suppression pool (SP) level provide actuation signals for automatic switchover of HPCI suction from the CST to the SP.

B. See Section I, "Area of Review," of SRP-LR B.0.1 for Item I.B.

C. Typical age-related degradation mechanisms that could affect the operation and safety of the HPCI system are the following (Ref. 1):

1. Fatigue crack initiation and propagation have been found in nozzles and thermal sleeves.

2. Wear, foreign material, and faults have resulted in valve leaks and operability problems.

3. Wear, vibration, and fatigue have resulted in pump reliability problems.

4. Vibration, waterhammer, and thermal cycles have affected the system piping and pipe supports.

5. Errosion/corrosion of the steamlines of the turbine-driven pump have resulted in wall thinning.

The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for Items A. through D.

E. The HPCI components not specifically addressed in this section may be addressed by the generic aging topic reviews in Part C of the SRP-LR. The reviewer should ensure that structures and components included as part of the generic SRP-LR topics are adequately reviewed for this system. The following sections of Part C should be used for the review: C.1.1, "Piping"; C.1.2, "Valves"; C.1.3, "Pumps"; C.1.6, "Equipment and Component Supports"; all of C.2.0, "Electrical"; and all of C.3.0 "Instrument." This may require additional staff input.

F. The reviewer should ensure that the licensee's IPA addressees the cumulative effects of applicable aging prosesses identified in Table B.2.2.6-1.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "A Review of Information Useful for Managing Aging in Nuclear Power Plants," NUREG/CR-5562 (PNL-7323), June 1990.

Table B.2.2.6-1 Summary of aging process for the high pressure injection system

Major component	Stressors	Degradation mechanisms	Potential failure modes	Inservice Inspection methods
Nozzles and thermal sleeves	System operating transients, thermal cycling, vibration, waterhammer	Fatigue crack initiation and propagation	Leaks through wall, loose parts	Visual inspection, volumetric inspection
Valves	System operating transients, maintenance, testing	Wear, foreign material, mechanical leakage, faults	Leakage, failure to operate, blockage, command faults	Visual and operational tests
Air-operated valves	System operating transients, contaminated air supply	Sticking, blockage, fouling water, oil	Failure to operate	Visual and operational tests
Instrument and controls	Electrical transients, thermal cycles, maintenance vibration	Corrosion, loose connections failure (catastrophic)	Open, shorts, failure to operate	Testing
Pumps	System operating transients, thermal cycles	Wear, vibration, fatigue	Seal leaks, failure to start, fail to run	Testing, visual inspection
Pipe supports	Vibration, waterhammer	Fatigue, loosening of abrasive water	Breaking loose	Visual inspection
Piping	Vibration, waterhammer, thermal cycles	Thermal fatigue, abrasive water	Through the wall leaking, cracks	Visual inspection, volumetric inspections

B.2.3 AUXILIARY FEEDWATER SYSTEM (PWR)

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SPLB

I. AREA OF REVIEW

A. This section addresses the auxiliary feedwater system (AFWS) of pressurized-water reactors.

1. Description

A typical AFW consists of redundant auxiliary feedwater (AFW) trains, with a 50 percent capacity motor-driven pump in each train feeding directly to the steam generators and a 100 percent capacity steam turbine-driven pump able to supply either of the redundant trains. There are variations in this AFWS arrangement; however, the factors that affect the aging process are independent of the type of system design.

The AFWS is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the facility.

2. System Function

The AFWS supplies feedwater to the steam generators to allow secondary-side heat removal from the primary system when main feedwater is unavailable. The system is capable of functioning for extended periods either to hold the plant at hot standby or to cool the plant down to temperature and pressure levels at which the low-pressure residual heat removal system can operate.

3. System Boundaries

The boundaries of the AFWS extend from the condensate storage tank and the backup seismic Category I water supply to the connections with the steam generators, which are made either through a connection to the main feedwater piping or through separate auxiliary feedwater piping connected directly to the steam generators. Pumps, valves, piping, interconnections, and cross-connections are included in the AWFS.

B. See Section I, "Area of Review," of SRP-LR B.0.1 for Item I.B.

C. A variety of age-related mechanisms can affect the ability of the AFWS to continue to operate safely and efficiently. Typical examples of the areas of degradation in the AFWS are the following (Refs. 1-4):

1. The combined failures of motor and air operators for valves have been found to result in approximately the same level of degradation of the AFWS as the turbine drives alone. Pump failures and check valve failures are also significant contributors to system degradation.

2. The single, largest source of historical AFWS degradation is the turbine drive for AFW pumps. It should be noted that the turbine proper has been a relatively reliable and rugged piece of equipment. However, the turbine auxiliaries, including the governor control and trip throttle valve have contributed substantially to the overall turbine problems.

3. Instrumentation and control (I&C)-related failures dominated the group of failures that were detected during demand conditions (as opposed to failures detected as the result of periodic monitoring or routine observations made by operators or other personnel). Many of the potential failure sources that were found to not be detectable by the current monitoring practices were related to the I&C portion of the system.

The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1 for Items II. A. and II. B.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for Items III. A. through III. D.

E. AFWS components not specifically addressed in this section may be addressed by the generic aging topic reviews in Part C of the SRP-LR. The reviewer should ensure that structures and components included as part of the generic SRP-LR topics are adequately reviewed for the AFWS. The following sections should be used for the review: C.1.1, "Piping"; C.1.2, "Valves"; C.1.3, "Pumps"; C.1.6 "Equipment and Component Supports"; all of Section C.2.0, "Electrical," except C.2.5, "Transformers," and all of Section C.3.0, "Instrument". This may require other staff support.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

Historical failure data indicate that the turbine drive is the most significant source of AFWS degradation. The turbine drive is the single, largest source of system degradation associated with on-demand failures. The turbine drive itself has been fairly reliable; however, a number of problems have developed with the governor and controls for the turbine-driven pumps.

The other significant types of AFWS components, including pumps, check valves, and air and motor operators, have been, or are being, reviewed in detail as part of the Nuclear Plant Aging Research (NPAR) Program. In light of the significance of the turbine to historical AFWS degradation, as well as the fact that the turbines used on AFW pumps are similar to those used on some safety-related pumps in boiling water reactor plants, turbine drives in general, and more specifically the turbine controls, will be reviewed further as part of on-going programs.

The fraction of AFWS degradation that has historically been found during demand events, as well as the number and types of failure and degradation sources that were found to not be detectable by the monitoring methods in place at the reference plant for NPAR studies, indicates the need for improvements in certain aspects of the current monitoring practices. Although there are no guidelines to establish what is an acceptable fraction of failures detected during demand, the rate indicated by the failure data review (about 18 percent of all system degradation was detected during demand conditions) appears excessive. This is particularly the case for certain component types and parts (e.g., the pump driver group and the turbine-driven pump instrumentation and controls I&C and governor controls).

During the reference-plant review, the NRC staff also found that the ability of some components to function as required under design-basis or off-normal conditions is not verified periodically. This was found to be the case particularly where multiple component interaction is involved. Decidedly adverse effects could result from routine testing of some of these currently non-tested areas (such as checking the ability of the AFW pumps to successfully negotiate the suction transfer from the condensate storage tank to emergency service water). Other areas could be checked fairly easily with little additional effort and no adverse consequences (such as verification of pump capability by monitoring additional parameters during the full stroking of discharge check valves).

Another observation made was that some components, or certain parts or aspects of components, appear to be tested in excess of what failure history indicates to be appropriate. On the other hand, as can be gathered from the comments above, other aspects of certain parts of the AFWS are either never tested or receive less than thorough testing. It appears that enhanced testing requirements are needed in order to reduce excessive testing while at the same time ensuring that performance is thoroughly verified periodically.

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "A Review of Information Useful for Managing Aging in Nuclear Power Plants," NUREG/CR-5562 (PNL-7323), June 1990.

2. U.S. Nuclear Regulatory Commission, "Aging and Service Wear of Auxiliary Feedwater Pumps for PWR Nuclear Power Plants, "Volume 1, "Operating Experience and Failure Identification," NUREG/CR-4597 (ORNL-6282/V1), July 1986.

3. U.S. Nuclear Regulatory Commission, "Aging and Service Wear of Auxiliary Feedwater Pumps for PWR Nuclear Plants," Vol. 2, "Aging Assessments and Monitoring Method Evaluations," NUREG/CR-4597 (T188 012499), June 1989.

4. U.S. Nuclear Regulatory Commission, "Auxiliary Feedwater System Aging Study," Vol 1 "Operating Experience and Current Monitoring Practices," NUREG/CR-5404 (ORNL-6566/V1), Draft, 1989.

B.2.4 AUTOMATIC DEPRESSURIZATION SYSTEM (BWR)

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SPLB

I. AREAS OF REVIEW

A. This section addresses the automatic depressurization system (ADS).

1. Description

The ADS is an emergency core cooling system (ECCS). The system consists of pneumatically operated pressure relief valves designed to relieve reactor coolant system pressure automatically or manually during small or intermediate loss-of-coolant accidents.

The ADS is described in the most recent update of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the facility.

2. System Function

The ADS utilizes certain main steam safety relief valves (SRVs) to depressurize the reactor vessel so that the low-pressure ECCSs can inject sufficient coolant into the vessel to cool the core.

3. System Boundaries

The ADS includes all the piping, accumulators, and pneumatic and shuttle valves from the SRVs to the isolation valve of the compressed air system. The SRVs are considered part of the main steam system, but the actuating devices for the ADS are part of the ADS. The dc power system is only used to supply power. All of the ADS logic and indicating lights are part of the ADS. Annunciators are excluded.
There are some differences in the ADS for the different BWR product lines; however, the system functions are the same. The number of SRVs varies with plant size, the logic is slightly different, and manual initiation is not available in the older plants.

B. See Section I, "Area of Review," of SRP-LR B.0.1 for Item I.B.

C. The primary aging concern identified for the ADS is foreign material in the pneumatic system. A large fraction of BWRs have experienced problems with the pneumatic valves because of foreign material in the system. This material has been attributed to age-related scaling, dirty air supply, and other causes.

The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SPR-LR B.0.1 for Items III. A. through III. D.

E. The ADS components not specifically addressed in this section may be addressed by the generic aging reviews in Part C of the SRP-LR. The reviewer should ensure that structures and components included as part of the generic SRP-LR topics are adequately reviewed for this system. The following sections in Part C should be used for the review: C.1.1, "Piping"; C.1.2, "Valves"; C.2.1, "Cable and Wiring"; C.2.3, "Electrical Penetrations"; C.2.4, "Relays, Circuit Breakers, and Switchgear"; C.2.6, "Solenoid-Operationed Valves"; and all of C.3.0, "Instruments." In addition, since most of the problems with the ADS are associated with the air supply, the reviewer should review Section B.5.8, "Compressed Air System," for those portions of the ADS exposed to the air supply. This may require additional staff support.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

VII. REFERENCES

B.2.5 REMOTE SHUTDOWN SYSTEM/SAFE SHUTDOWN SYSTEMS

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SICB

I. AREA OF REVIEW

A. This section addresses the remote shutdown/safe shutdown systems:

1. Description

These systems are used to achieve and maintain a safe shutdown of the plant.

The remote shutdown/safe shutdown systems specific description is contained in the most recent update of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the specific facility.

SRP-LR B.4.6 addresses "Information Systems Important to Safety" for information about which controls are used for the control of systems required for remote shutdown/safe shutdown.

2. System Function

The functions of the Remote Shutdown/Safe Shutdown Systems are to achieve and maintain safe shutdown of the plant. Some engineered safety features (ESF) systems are used to both mitigate accidents and to achieve and maintain a safe shutdown. The review of the systems in this section is limited to those system features that are unique to safe shutdown, and does not include those directly related to accident mitigation (those features unique to accident mitigation are addressed in SRP-LR, B.2.1.).

3. System Boundaries

The remote shutdown/safe shutdown systems controls are isolated from the systems components which they control and the normal and emergency ac and dc power systems by circuit breakers, isolation amplifiers, isolation transformers, actuation logic, fuses, or other approved isolation devices.

The objectives of this review are to confirm that the remote shutdown/safe shutdown systems and controls satisfy the requirements of the acceptance criteria and guidelines for age-related degradation affecting those systems.

This review covers sensor couplings, sensors, initiating circuitry, logic bypasses, interlocks, redundancy features, and actuated devices of those systems which provide the necessary instrumentation and control functions to achieve safe shutdown.

Typical systems and features required for remote shutdown/safe shutdown are:

Remote Shutdown Panel
Auxiliary Feedwater System
Residual Heat Removal System
Chemical and Volume Control System (Boration Control)
Reactor Protection System
Neutron Monitoring System
Mode Switch

B. See Section I, "Area of Review," of SRP-LR B.0.1 for Item I.B.

Typical examples of age-related degradation associated with Remote Shutdown/Safe Shutdown are provided in this section. The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

1. Age-related degradation due to setpoint drift.

2. Age-related degradation due to functional testing cycles and trips.

3. Age-related degradation due to improper maintenance/repair.

4. Age-related degradation of sensors, connectors, cables and wires, circuit breakers, relays and electronic components, etc.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 Items III. A through III. D.

E. Components of the remote shutdown/safe shutdown system not specifically addressed in this section may be addressed in the generic aging topic reviews in Part C of the SRP-LR. Specifically the following sections of SRP-LR Part C are applicable to the remote shutdown/safe shutdown system and should be reviewed: C.2.1 "Cable and Wiring," C.2.2 "Junctions," C.2.3 "Electrical Penetrations," C.2.4 "Relays, Circuit Breakers, and Switchgear," C.2.5 "Transformers," C.2.6 "Solenoid Operated Valves," C.2.7 "Electrical Motors," and all of C.3.0 "Instrument." This may require other staff input.

IV. EVALUATION FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

NUREG/CR-4740, Nuclear Plant-Aging Research of Reactor Protection Systems (Ref. 1) indicates that, in general, the current testing programs are adequate for the intended purpose of verifying RPS operability and performance. This, in conjunction with the redundancy built into the RPS systems, makes the RPS and similar systems, i.e., the remote shutdown/safe shutdown systems not susceptible to many of the age-related degradation concerns.

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "Nuclear Plant Aging Research of Reactor Protection Systems," NUREG/CR-4740, (TI88 007920), January 1988.

B.3.1 PRIMARY CONTAINMENT STRUCTURE

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - ESGB

I. AREAS OF REVIEW

A. This section addresses the primary containment structure (PCS).

1. Description

Primary containments are either of the free-standing ASME steel vessel type, which accommodate both structural and leak-tightness requirements, or of the concrete containment type, which use a reinforced or prestressed concrete structure with a carbon steel liner for leak-tightness.

PWR containments enclose the steam generators, the reactor coolant pumps, the pressurizer, and the reactor pressure vessel; therefore, PWR containment structures tend to be quite large, providing room for the expansion of steam resulting from a loss-of-coolant accident (LOCA). Some PWR containments include an ice condenser which provides a large passive heat sink for initial LOCA heat loads. The use of an ice condenser reduces peak post-LOCA containment pressure,thereby allowing the use of a smaller containment volume. The ice condenser containment is addressed further in Section B.3.3, "Containment Heat Removal System," of the SRP-LR.

BWR containments are based on a pressure-suppression concept whereby the LOCA fluid is channeled to a large water-filled pool (suppression pool) where the steam is condensed, thereby reducing peak containment pressure. As with the ice condenser containment, the pressure-suppression design allows for a smaller containment volume. The suppression pool is also used to condense the steam released by the safety relief valves (SRVs) during actuation. BWRs experience a number of plant transients that result in actuation of one or more of the SRVs.

BWR containments have evolved through three distinctly different designs. The first BWR design, referred to as the "Mark I containment," is made up of a drywell in the general shape of a inverted light bulb and a toroidal shaped suppression chamber (pool) located below and encircling the drywell. Circular vent pipes with expansion joints connect the drywell and the pressure-suppression chamber. The drywell head closure is made with a double tongue-and-groove seal, which permits periodic leak checks without pressurizing the entire vessel.

The second BWR containment design, referred to as the "Mark II containment," includes a cylindrical/conical drywell with a steel closure head. A cylindrical suppression pool is located directly beneath the drywell. Vertical vent pipes connect the drywell and suppression chamber.

The BWR Mark III design is the third and most recent BWR containment design. The Mark III design includes a cylindrical drywell with steel closure head. The suppression pool is contained within concrete walls in an annulus below and encircling the reactor vessel and to the outside of the drywell. Horizontal vents with a weir wall provide the flow path from the drywell to the suppression pool.

All primary containment designs include numerous penetrations through the containment boundary for piping, electrical, and instrument sensing lines. Additional penetrations include personnel access hatches in the drywell and suppression chamber, equipment hatch(s) in the drywell, and the drywell closure head.

The PCS is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the facility.

2. System Function

The primary containment provides the main barrier to the release of fission products to the environment in the event of core damage. As such, the primary containment, including penetrations, is designed to accommodate, without exceeding the design leakage rate, the calculated temperature and pressure conditions resulting from any LOCA.

The primary containment design also includes a sump (PWR) or a suppression pool (BWR) which provides the source of water for operation of the emergency core cooling system during LOCA conditions after depletion of the water or transfer from the borated water/condensate storage tank.

3. System Boundaries

The primary containment includes the containment structure (and liner) and all penetrations. Isolation valves and isolation logic are addressed in SRP-LR B.3.4, "Containment Isolation System," of the SRP-LR. BWR containments include the drywell, suppression pool, and connecting vent pipes.

B. See Section I, "Area of Review," of SRP-LR B.0.1 for Item I.B.

C. Aging concerns include the potential for loss of structural integrity and leak-tightness of the boundary. One concern for concrete is the loss of bound water and associated degradation of the shielding properties that can be caused by nuclear heat. Corrosion of the reinforcing steel can cause cracking and spalling of the concrete, which degrades the structural integrity of the containment or its shield. The steel vessel or steel liner and suppression pool (BWR) and vent pipes (BWR) are subject to corrosion from the internal corrosive environment. The expansion joints on the vent pipes are susceptible to fatigue from thermal cyclic loading. The occasional actuation of the SRVs adds to the thermal and cyclic loads placed on the containment and suppression pool for BWRs, (Refs. 1-3) furnish additional information on containment aging.

A unique aging concern for prestressed concrete containment structures is loss of tendon preload as a result of corrosion of the tendons, creep or other aging phenomena.

Another concern is the degradation of the protective coatings used on the containment liner, the drywell, and the suppression pool walls.

Considerable information relative to the aging concerns of containment structures can be found in References 1 through 6. The following discussion summarizes many of those concerns and identifies recommendations for managing the age-related degradation experienced by containment structures.

1. Aging Concerns and Mechanisms

   In addition to the potential for degradation of the containment liner or structure, there is a potential for degradation of seals at penetrations for piping, access doors, hatches, and ventilation openings. These sites represent pathways for escape of radioactivity irrespective of the containment liner or vessel.

   a. Concrete Containments

      Reference 2 discusses aging and degradation of concrete structures. In general, the strength of concrete tends to increase with age. Steel that is enclosed in concrete (rebar plates and tendons) is well protected from corrosion; however, small cracks, porous areas, or voids that allow the penetration of moisture and air through the concrete provide conditions conducive to the corrosion of the steel. The iron oxide corrosion products are less dense than steel; thus, formation of the oxides produces tensile stresses in the concrete that tend to expand the area that is accessible to corrosion.

      A steel liner forms the inner wall of the containment and serves as the primary barrier against leakage of radioactive materials to the external environment. Corrosion or cracking of this steel barrier could allow radioactive gases or liquids to escape the containment system.

Table B.3.1-1 summarizes the degradation concerns for concrete containments. In reinforced concrete containments, parts of the concrete may be placed in tension; as a result, cracks may open and expose the reinforced steel to corrosive conditions. The same stress conditions may exist in basemats; here, however, moisture and chemicals in the soil may compound the seriousness of the situation. In contrast, prestressed concrete vessels are intended to be maintained in compression, which tends to prevent the formation (or opening) of cracks. However, experience has shown that prestressing is reduced over time. Significant loss of prestressing may occur and lead to degradation of the containment. Corrosion of the steel tendons or anchorage assemblies (caused by the ingress of moisture, breakdown of the grease protection material, or microbial action) must be prevented to ensure that the concrete remains in compression. Partial failures of anchorage assemblies and individual strands of the tendons have occurred as a result of improper chemistry or heat treatment of the steel.

b. Steel Containments

Tables B.3.1-2 and B.3.1-3 (excerpted from Reference 1) summarize the degradation processes of concern for steel containments. The major concern is corrosion; the most significant experience to date has been the corrosion of the exterior surface of a BWR drywell near the sand pocket.

Corrosion in the BWR steam pressure-suppression system presents special problems; in addition to the potential for microbial growth (and increased corrosion) in the suppression pool, stagnation of the pool water increases the possibility of corrosion at any site where the protective coating is damaged or has deteriorated. In addition, stainless steel bellows in the piping that connects the drywell to the suppression chamber are susceptible to fatigue and stress corrosion cracking.

2. Managing Aging Degradation

Reference 3 discusses various elements of a program to manage the age-related degradation of PWR concrete containments. The potential degradation mechanisms along with the actions proposed in Reference 3 are summarized in Table B.3.1-4.

These recognized problems are expected to continue through-out the life of the plant, including the license renewal term. The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1 for Items II.A and II.B.

C. The results of the inspection and reviews required in III below shall comply with the requirements of References 7-9, as appropriate.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for Items III.A through III.D.

E. PCS components not specifically addressed in this section may be addressed by the generic aging topic reviews in Part C of the SRP-LR. The reviewer should ensure that structures and components included as part of the generic SRP-LR topics are adequately reviewed for this system. The following sections of Part C should be used for the review: C.1.1, "Piping"; C.2.3, "Electrical Penetrations"; and C.4.0, "Civil Structures." This may require other staff input.

F. The reviewer should confirm that the licensee has committed to implement the inservice inspection requirements of Reference 7 during the license renewal period. For prestressed concrete containment structures, the licensee's inservice inspections should also address the provisions of References 8 or 9.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

VII REFERENCES

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2. U.S. Nuclear Regulatory Commission, "Concrete Component Aging and Its Significance Relative to Life Extension of Nuclear Power Plants." NUREG/CR-4652 (ORNL/TM-10059), September 1986.

3. W.B. Dodson, P.F. McHale, P. Beament, and M. Lapides, "Life Extension Considerations for Pressurized Water Reactor Containment Structures," Proceedings of the Topical Meeting on Nuclear Power Plant Life Extension, July 31-August 3, 1988, Snowbird, Utah, Vol. 1, pp. 68-75.

4. R.F. Sammataro, "Preservice and Inservice Requirements for Containments Structures - A Status Report," Third Workshop on Containment Integrity, May 21-23, 1986, Washington, D.C.

5. R.F. Sammataro, "Codes and Standards for Nondestructive Examination of Concrete Containments," American Society of Mechanical Engineers 1989 Summer Pressure Vessels and Piping Division Conference, July 23-27, 1989, Honolulu, Hawaii.

6. D. Naus, M. Marchbanks, and G. Arndt. "Evaluation of Aged Concrete Structures for Continued Service in Nuclear Power Plants," Proceedings of the Topical Meeting on Nuclear Power Plant Life Extension, July 31 - August 1988, Snowbird, Utah, Vol. 1, pp. 57-67.

7. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section XI.

8. U.S. Nuclear Regualtory Commission, "Inservice Inspection of Upgrouted Tendons in Prestressed Concrete Containment Structures," Regulatory Guide 1.35.

9. U.S. Nuclear Regulatory Commission, "Inservice Inspection of Prestressed Concrete Containment Structures With Grouted Tendons," Regulatory Guide 1.90.

TABLE B.3.1-1 Summary of concrete containment degradation processes

Degradation site	Stressor	Degradation mechanisms	Potential failure modes
Reinforcing bars	Corrosive environment stray currents	Corrosion, fatigue	Loss of structural integrity
Post-tensioning system anchors *	Trapped water, steady-state stress	Hydrogen embrittlement, corrosion	Loss of stress
Posttensioning tendon wire or strand *	Moisture, trapped water, microorganisms, steady-state stress	Pitting, microbiologically be induced corrosion, relaxation	Loss of stress
Steel liner dome and wall	Moisture, acidic environment, stress	Corrosion	Liner-concrete interaction, leakage of radioactive gases
Steel liner over base slab	Moisture, acidic environment, stress	Corrosion	Leakage of radioactive material
Suppression pool steel liner below water line **	Cyclic thermal and mechanical loads, corrosive internal environment, microorganisms	Corrosion due to differential aeration, fatigue, microbiologically influenced corrosion	Leakage of radioactive material
Drywell steel liner, suppression pool steel above water line **	Moisture, corrosive internal environment, cyclic thermal and pressure loads	Corrosion, fatigue	Leakage of radioactive gases
Concrete	Aggressive environment, internal chemical reactions	Cracking, spalling	Loss of integrity, corrosion of reinforcing steel
	Nuclear heat **	Loss of bound water **	Degradation of shielding properties **

* Prestressed concrete containments only.
** Boiling-water reactors only.

Table B.3.1-2 Summary of PWR steel containment degradation processes

Degradation site	Stressor	Degradation mechanisms	Potential failure modes
Shell welds and base metal	Stresses, vibration, cyclic loading, aggressive environment	Corrosion	Loss of structural integrity, leakage of radioactive gases
Interface between shell and concrete slab at base of shell	Stresses, vibration, cyclic loading, aggressive environment	Corrosion	Loss of structural integrity, leakage of radioactive gases
Discontinuities in the shell such as hatches and penetrations	Stresses, vibration, cyclic loading, aggressive environment	Corrosion	Leakage of radioactive gases
Steel bottom of shell embedded in concrete	Aggressive environment	Corrosion	Leakage of radioactive gases
Base slab concrete	Aggressive environment, internal chemical reactions	Cracks, spalling	Corrosion of reinforcing steel, corrosion of steel bottom of containment shell
Exterior surface of drywell base near sand pocket	Moisture, microorganisms, degraded fill material corrosion	Aqueous corrosion, crevice corrosion, microbiologically influenced	Leakage of radioactive gases
Exterior surface of drywell	Degraded fill material, moisture	Crevice corrosion, aqueous corrosion	Leakage of radioactive gases
Embedded shell region	Cyclic thermal loading, corrosive internal environment	Thermal fatigue, crevice corrosion and pitting	Loss of structural integrity
High energy pipe line penetrations, hatches, vent lines	Cyclic thermal loading, pressure testing, corrosive internal environment	Thermal and mechanical fatigue, environmentally assisted fatigue	Leakage of radioactive gases

Table B.3.1-2 (continued)

Degradation site	Stressor	Degradation mechanisms	Potential failure modes
Stainless steel bellows	Corrosive internal environment, cylic thermal loading, pressure testing	Intergrnular stress corrosion cracking at heat-affected zone, intergranular stress corrosion cracking, fatigue	Leakage of radioactive gases
Submerged portion of suppression pool	Corrosive internal environment, safety relief valve discharge tests, pressure testing, microorganisms	Differential aeration, mechanical fatigue, pitting, microbilogically influenced corrosion	Leakage of radioactive gases
Transition region from cylindrical to sperical portion of drywell shell at the core	Cyclic thermal loading, pressure testing, corrosive dry- internal environment, neutron radiation	Thermal and mechanical fatigue, environmentally assisted fatigue, irradiation embrittlement	Leakage of radioactive gases
Dissimilar metal welds	Corrosive internal environment, cyclic thermal loading, pressure testing	Galvanic corrosion, fatigue	Leakage of radioactive gases

TABLE B.3.1-3 Summary of BWR (Mark I) Steel Containment Degradation Processes

Degradation Site	Stressor	Degradation Mechanisms	Potential Failure Modes
Exterior surface of drywell base near sand pocket	Moisture, microorganisms, degraded fill material	Aqueous corrosion, crevice corrosion, microbial influenced corrosion	Leakage of radioactive gases
Exterior surface of drywell	Degraded fill material, moisture	Crevice corrosion, aqueous corrosion	Leakage of radioactive gases
Embedded shell region	Cyclic thermal loading, corrosive internal environment	Thermal fatigue, crevice corrosion and pitting	Loss of structural integrity
High energy pipe line penetrations, hatches, vent lines	Cyclic thermal loading, pressure testing, corrosive internal environment	Thermal and mechanical fatigue, environmentally assisted fatigue	Leakage of radioactive gases
Stainless steel bellows	Corrosive internal environment, cylic thermal loading, pressure testing	IGSCC at heat affected zone, IGSCC, fatigue	Leakage of radioactive gases
Submerged portion of suppression pool	Corrosive internal environment, safety relief valve discharge tests, pressure testing, microorganisms	Differential aeration, mechanical fatigue, pitting, microbial influenced corrosion	Leakage of radioactive gases
Transition region from cylindrical to spherical portion of drywell, drywell shell at the core	Cyclic thermal loading, pressure testing, corrosive internal environment, neutron radiation	Thermal and mechanical fatigue, environmentally assisted fatigue, irradiation embrittlement	Leakage of radioactive gases
Dissimilar metal welds	Corrosive internal environment, cyclic thermal loading, pressure testing	Galvanic corrosion, fatigue	Leakage of radioactive gases

Table B.3.1-4 Proposed elements of a program for extended life of PWR concrete containments

Potential degradation	Propsosed action	Comments
Concrete degradation above ground grade	Periodically inspect accessible concrete surfaces for o Freeze-thaw damage o Leaching of calcium hydroxide o Aggressive chemical attack o Reactivity o Corrosion of reinforcing	Assembly of data from available records shouldallow evidence that materials were checked for compatibility, that air entrainment was used, and that concrete cover was adequate. Few adverse observations are anticipated.
	Monitor temperature and radiation around penetrations.	Normal operational temperature and radiation exposures are considered to be incapable of affecting the concrete. Monitoring of local areas near penetrations should establish that exposure levels are too low to cause damage. If temperatures exceed preset value, determine specific action required, such as nondestructive examination, analysis of effects, penetration modification and/or cooling.
Prestressing system degradation	Perform inservice inspections (ISIs) as required by applicable regulatory guide.	Review data from previous inspections to show that currently performed ISI is sufficient to ensure continuously satisfactory performance and that corrective action taken as required was effective.
Liner degradation from containment interior (above intersection with base mat)	Perform inservice inspection of interior and accessible liner surfaces for corrosion and physical abuse.	Incipient stages of interior liner corrosion should be detectable on visual examination. Physical abuse need only be considered after an event such as a maintenance activity.

Table B.3.1-4 (continued)

Potential degradation	Propsosed action	Comments
	Monitor temperatures at penetrations and radiation exposures at strategic areas.	Monitoring these areas will provide data showing that degradation due to temperature or radiation extremes was not possible.
Concrete degradation below ground grade	Monitor groundwater levels, chemistry, and pH when they could affect the lower containment concrete	Groundwater monitoring can be accomplished with relatively limited impact. For containments that have suitable waterproof membranes or where groundwater levels in relation to below grade portions of the containment are controlled, licensees need not consider this action.
	Review existing soil characteristics for potential effects, or conduct tests of soil sample.	Examine concrete if groundwater monitoring or soil characteristics suggest attack is possible.
Reinforcing steel corrosion below grade	Monitor performance of cathodic protection system (if available).	Maintain current levels per system design to ensure protection.
	For those containments that do not have a cathodic protection system for reinforcing, develop and implement a system to remotely monitor corrosion of reinforcing.	Examine specific areas where possible corrosion has been indicated by remote monitoring.
Liner corrosion from concrete side, below grade	Monitor performance of cathodic protection system (if available).	Maintain current levels per system design to ensure protection

Table B.3.1-4 (continued)

Potential degradation	Propsosed action	Comments
	Perform ultrasonic tests (UT) of control areas (frequency of examination depends on existence of cathodic cathodic protection system).	Conduct refined UT in suspect area where corrosion may be ongoing.
	Develop and implement a system to remotely monitor corrosion of liner (from concrete side).	The impact of a potential liner leak suggests that remote monitoring of the liner for corrosion damage is necessary.
Liner corrosion of floor liner plate (from interior of containment)	Monitor performance of cathodic protection system (if available).	Maintain current levels per system design to ensure protection.
	Inspect interior concrete floor surface for signs of attack and corrosion.	
	Inspect condition of joint sealants.	
	Establish areas of control where concrete can be removed for periodic inspection of liner.	
	Develop and implement a system to remotely monitor corrosion of floor liner plate.	Conduct a more extensive removal of concrete for examination of liner in areas where corrosion may be indicated.
Coating degradation	Perform qualification of containment coating with analysis showing that safety system operation is not compromised by coating failure.	If qualification cannot be done, develop an in-place test to assess ability of aged coating to withstand a loss-of-coolant accident such as by adhesion testing.

Table B.3.1-4 (continued)

Potential degradation	Propsosed action	Comments
Pile corrosion	Monitor performance of cathodic protection system and maintain its operation within design parameters.	Pile corrosion in undisturbed soil is not anticipated. Corrosion in zones of disturbed soil is expected to be minor. Cathodic protection of piles should eliminate any concern about pile corrosion.

B.3.2 SECONDARY CONTAINMENT

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - ESGB

I. AREAS OF REVIEW

A. This section addresses the secondary containment system.

1. Description

Some facilities have a secondary containment (the reactor building for some BWRs) that completely encloses the primary containment. The secondary containment aids in minimizing the ground-level release of airborne radioactive materials and provides for the controlled release of radioactive materials under accident conditions.

The secondary containment, if applicable, is described in the most recent revision of the final safety analysis report (FSAR) or updated Safety Analysis Report (USAR) for the facility.

2. System Function

The secondary containment structure and supporting systems are provided to collect and process radioactive material that may leak from the primary containment. Typically, the secondary containment is maintained at a lower pressure than the primary containment and the atmosphere.

3. System Boundaries

The typical secondary containment substructure consists of reinforced concrete exterior walls. The superstructure may be constructed of a structural steel frame with metal siding and roofing. Airlocks and piping, electrical, and instrumentation penetrations are part of the secondary containment system as well as building ventilation isolation dampers and control elements.

B. See Section I, "Area of Review," of SRP-LR B.0.1 for Item I.B.

C. Typical examples of age-related degradation associated with the secondary containment are listed below:

1. Steel corrosion resulting in structural degradation

2. Rebar corrosion resulting in concrete structural degradation

3. Coating degradation which allows corrosion of structural elements to occur

4. Degradation of sealing materials (caulking) used on roofing and siding

The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for Items III.A through III.D.

E. Secondary containment components not specifically addressed in this section may be addressed by the generic aging topic reviews in Part C of the SRP-LR. The reviewer should ensure that structures and components are adequately reviewed for this system. Section C.4.0, "Civil Structure," in SRP-LR Part C should be used for the review. This may require other staff input.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

VII. REFERENCES

B.3.3 CONTAINMENT HEAT REMOVAL SYSTEM

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SPLB

I. AREA OF REVIEW

A. This section addresses the containment heat removal system.

1. Description

Plant-specific designs for the containment heat removal system are subject to considerable variations relative to the systems and equipment used. Some PWR containments are equipped with an ice condenser in which a large mass of ice provides a heat sink for absorbing initial loss-of-coolant-accident (LOCA) heat loads. BWR containments (see SRP-LR B.3.1) a large mass of water in the suppression pool is used as a passive heat sink for initial LOCA heat loads. For either approach, the use of a large heat sink allows the containment volume to be considerably smaller than that of the large dry containments that are used at most PWRs. For most containment designs, the containment spray system (see SRP-LR B.3.8) is used for long term post-LOCA cooling and pressure reduction; in some PWRs, the containment ventilation system (see SRP-LR B.3.9) fan coolers are used. For these PWRs, appropriate components of the containment ventilation system are required to be safety related. In addition, the residual heat removal (RHR) systems (see SRP-LR B.2.2.4) of both PWRs and BWRs are used to transfer heat from the containment sump or suppression pool through the RHR heat exchangers to the ultimate heat sink. The BWR RHR system can also be operated in a suppression pool cooling mode independent of reactor injection. At some older plants, containment sprays are not used for long-term cooling.

During the containment cooling mode of operation, condensate and spray water flow to the containment sump or suppression pool. The water is then circulated through the containment spray (PWR) or RHR (BWR) heat exchangers, which transfer heat to the service water system (SWS) (SRP-LR B.5.4) and then to the ultimate heat sink (UHS) (SRP-LR B.5.5.). At PWRs where the containment ventilation system fan coolers are used, heat is also transferred from these coolers to the UHS via the SWS. Additionally, at some PWRs the component cooling water system is used as an intermediate system between the reactor coolant system and the SWS.

The containment heat removal system is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the facility.

2. System Function

The function of the containment heat removal system is to reduce the containment pressure and temperature following a design-basis accident such as a LOCA and, in some cases, to control sump pH. Containment heat is removed during normal operation and shutdown by the containment ventilation system (see SRP-LR B.3.9)

3. System Boundaries

The containment heat removal system includes the containment spray system, those portions of the RHR system that provide containment cooling (i.e., suppression pool cooling and containment spray), components of the containment ventilation system for certain PWRs, and the ice condenser for PWRs with ice condenser containments. With the exception of the ice condenser, all components of the containment cooling system are addressed in other sections of this SRP-LR as noted in Item I.A.1 above.

System boundaries for the ice condenser include the ice condenser structure, insulation, and doors; the ice and ice baskets; the refrigeration unit; the return air fans, ducts, and dampers; and instrumentation required to ensure proper operation during normal and post-LOCA plant conditions. With the exception of power sources and operator controls, all ice condenser equipment is located within the primary containment.

B. See Section I, "Area of Review," of SRP-LR B.0.1 for Item I.B.

C. The aging concerns and mechanisms for the containment heat removal system are the same as those for other cooling and ventilating systems. The currently recognized problems of bearing wear and replacement, fan blade cracking and repair or replacement, motor failure, and heating and/or cooling coil failures apply. Additionally, aging concerns associated with ice condensers include potential insulation degradation, corrosion and wear of the ice baskets; wear and corrosion effects on the refrigeration system, and corrosion and wear on the large doors and hinges (Ref. 1). The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for Items III.A through III.D.

E. Containment heat removal system components not specifically addressed in this section may be addressed by the generic aging topic reviews in Part C of the SRP-LR. The reviewer should ensure that the structures and components included as part of the generic SRP-LR topics are adequately reviewed for this system. The following sections in SRP-LR Part C should be used for the review: all of Section C.1.0, "Mechanical"; all of Section C.2.0, "Electrical"; and all of Section C.3.0, "Instruments." This may require other staff input.

F. In addition, since the typical containment heat removal system uses other plant systems to perform its functions of post-LOCA containment heat removal and pressure reduction, the reviewer should coordinate this review with the reviews of the other systems, namely:

1. Containment spray (SRP-LR B.3.8).

2. RHR/low-pressure safety (Core) injection (SRP-LR B.2.2.4)

3. Containment ventilation (SRP-LR B.3.9) for PWRs where components of this system are used for post-LOCA heat removal

Ice condensers are not addressed in other SRP-LR sections. For ice condenser containment plants, the reviewer should ensure that the licensee's IPA provides a suitable assessment of industry experience with ice condensers, including research in this area, and that its aging management program specifies appropriate mitigation measures.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "A Review of Information Useful for Managing Aging in Nuclear Power Plants," NUREG/CR-5562 (PNL-7323), June 1990.

B.3.4 CONTAINMENT ISOLATION SYSTEM

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SPLB

I. AREAS OF REVIEW

A. This section addresses the containment isolation system.

1. Description

The containment isolation system consists of sensors, processors, and automatic closing valves, which serve to isolate selected systems under accident conditions. The status of the automatic isolation valves is indicated by lights in the main control room. Containment isolation valves are designed to seismic Category I requirements. Some plants have exemptions for specific containment isolation valves.

The containment isolation system is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the facility.

2. System Function

The containment isolation systems provide the means of isolating fluid systems that pass through containment penetrations so as to confine to the containment any radioactive material that may be released in the containment following an accident. The containment isolation systems are required to function following a design-basis event to isolate non-safety-related fluid systems penetrating the containment.

3. System Boundaries

The components and actions considered within the review of this system include the following:

a. A double barrier at the containment penetration in those fluid systems that are not required to function following a design-basis event

b. Automatic, fast closure of those valves required to close for for maintaining containment integrity following a design-basis event to minimize release of any radioactive material

c. A means of leak testing barriers in fluid systems that are used for containment isolation

d. The capability to test the operability of containment isolation valves periodically

e. The electrical and instrumentation control circuitry required to generate and transmit the actuation signal(s)

B. See Section I, "Area of Review," of SRP-LR B.0.1 for Item I.B.

C. A variety of age-related mechanisms can affect the ability of the containment isolation system to perform its safety function (Ref. 1). Nuclear heat and high thermal temperatures can cause degradation of the cables providing control signals to isolation valves. The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for Items A. through D.

E. Components of the containment isolation system that are not specifically addressed in this section may be addressed by the generic aging topic reviews in Part C of the SRP-LR. The reviewer should ensure that structures and components included as part of the generic SRP-LR topics are adequately reviewed for this system. The following sections should be used for the review: C.1.2, "Valves"; C.2.1, "Cable and Wiring"; C.2.3, "Electrical Penetrations"; C.2.4, "Relays, Circuit Breakers, and Switchgear"; C.2.6, "Solenoid-Operated Valves"; C.2.7, "Electric Motors"; and all of Section C.3.0, "Instrument." This may require other staff support.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1 .

VI. GENERAL INFORMATION

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "A Review of Information Useful for Managing Aging in Nuclear Power Plants," NUREG/CR-5562 (PNL-7323), June 1990.

B.3.5 CONTAINMENT PURGE SYSTEM

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SPLB

I. AREAS OF REVIEW

A. This section addresses the containment purge system.

1. Description

The containment purge system consists of fans, isolation devices, ducting, filters, sensors, and exhaust stacks used to purge containment air under selected circumstances.

The containment purge system is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the facility.

2. System Function

The containment purge system is used to establish the working atmosphere within the containment building for access during planned or unplanned reactor shutdowns. The containment purge system is designed to ensure safe, continuous access to the containment after a planned or unplanned reactor shutdown by reducing the airborne particulates of the containment atmosphere. This system also provides a path for the release of the containment atmosphere to control the pressure increase resulting from normal containment heatup during reactor startup. In addition, for some facilities, portions of the containment purge system are used to aid in the control or removal of hydrogen in the event of hydrogen buildup following a loss-of-coolant-accident. Generally, this is the case for those facilities that do not have a containment combustible gas control system as discussed in SRP-LR B.3.7.

3. System Boundaries

The containment purge system normally consists of supply and exhaust air-handling units, exhaust filter units, ductwork, isolation valves, and exhaust stack. Motor control centers and electrical and instrumentation control circuitry are also a part of the system boundaries to be considered within this review.

B. See Section I, "Area of Review," of SRP-LR B.0.1 for Item I.B.

C. The aging concerns and mechanisms for these systems are the same as those for other ventilation systems. The currently recognized problems of bearing wear and replacement, fan blade cracking and repair or replacement, motor failure, and heating and/or cooling coil failures are expected to continue throughout the life of the plant, including the license renewal term.

The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for Items III.A through III.D.

E. Components of the containment purge system not specifically addressed in this section may be addressed by the generic aging topic reviews in Part C of the SRP-LR. The reviewer should ensure that structures and components included as part of the generic SRP-LR topics are adequately reviewed for this system. The following sections should be used for the review as part of the containment purge system review: C.1.1, "Piping"; C.1.2, "Valves"; C.1.4, "Heat Exchangers"; C.2.1, "Cables and Wiring"; C.2.3 "Electrical Penetrations"; C.2.4, "Relays, Circuit Breakers, and Switchgear"; C.2.7, "Electrical Motors"; and all of C.3.0, "Instruments." This may require other staff input.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1 .

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

VII. REFERENCES

B.3.6 STANDBY GAS TREATMENT SYSTEM (BWR)

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SPLB

I. AREAS OF REVIEW

A. This section addresses the standby gas treatment system (SBGTS).

1. Description

The SBGTS, upon activation by an engineered safety feature (ESF) signal caused by a design-basis accident such as a loss-of-coolant accident (LOCA), usually draws contaminated air and gases from the refueling zones, from the high-pressure core injection area, and from the reactor zones. In the Mark III design, the contaminated gases are collected from the annulus recirculation exhaust, the auxiliary building secondary containment exhaust, and the fuel building exhaust. Ventilation air from the SBGTS rooms and the annulus recirculation exhaust fan room is also discharged to the inlet of the SBGTS. This system also filters the drywell purge flow when the reactor is in the refueling or shutdown mode until there is no indication of high activity in the drywell or in the drywell purge flow.

The SBGTS is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the facility.

2. System Function

The SBGTS removes fission products from the air drawn from the secondary containment under accident conditions to limit radiation dose rates to less than the 10 CFR Part 100 guidelines and purges the drywell and suppression chamber area. An elevated discharge is typically provided by exhausting the gases to the plant stack. The SBGTS is classified as an engineered safety feature.

3. System Boundaries

The SBGTS is located in the secondary containment and typically consists of the following components:

a. Fans, motors and fan housings
b. Electrical and instrumentation control circuitry
c. Heating coils, cooling coils, and moisture separators
d. Prefilters, high-efficiency particulate air filters, and filter housings

e. Activated charcoal adsorbers and adsorber housings
f. Motor-operated valves and dampers
g. Ventilation duct-work, supply and exhaust
h. Plant exhaust stack

B. See Section I, "Area of Re iew," of SRP-LR B.0.1 for Item I.B.

C. The aging concerns and mechanisms for the SBGTS are the same as those for other ventilation systems. The currently recognized problems of bearing wear and replacement, fan blade cracking and repair or replacement, motor failure, and heating and/or cooling coil failures are expected to continue throughout the life of the plant, including the license renewal term (Ref. 1). The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for Items III.A through III.D.

E. SBGTS components not specifically addressed in this section may be addressed by the generic aging topic reviews in Part C of the SRP-LR. The reviewer should ensure that structures and components included as part of the generic SRP-LR topics are adequately reviewed for this system. The following sections in SRP-LR Part C should be used for the review: B.1.1, "Piping"; B.1.4, "Heat Exchangers"; C.2.1, "Cable and Wiring"; C.2.3, "Electrical Penetrations"; C.2.4, "Relays, Circuit Breakers, and Switchgear"; C.2.7, "Electrical Motors"; C.3.1, "Sensors"; C.3.2, Electronic Compone.its"; and C.3.3, "Electronic Devices." This may require other staff input.

IV. EVALUATION FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "A Review of Information Useful for Managing Aging in Nuclear Power Plants," NUREG/CR-5562 (PNL-7323), June 1990.

B.3.7 CONTAINMENT COMBUSTIBLE GAS CONTROL SYSTEM (CCGCS)

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SPLB

I. AREAS OF REVIEW

A. This section addresses the containment combustible gas control system (CCGCS).

1. Description

Some facilities use the CCGCS to control in-containment, postaccident hydrogen buildup to a level below the flammability limit so that uncontrolled hydrogen/oxygen recombination does not occur. Other facilities may use an inert atmosphere inside the containment during operation, or they may use portions of the containment purge system to aid in the control of post-LOCA hydrogen buildup. The CCGCS is capable of controlled hydrogen/oxygen recombination, sampling the containment atmosphere and analyzing the samples for hydrogen. The CCGCS also circulates the containment atmosphere to ensure good hydrogen mixing and may control the hydrogen concentration by dilution or purging. The following systems are considered part of the CCGCS:

a. Postaccident hydrogen venting system
b. Postaccident hydrogen sampling system
c. Postaccident hydrogen mixing system
d. Hydrogen recombiners

The CCGCS is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the specific facility.

2. System Function

In the event of a design-basis accident, the CCGCS provides for the controlled recombination, mixing, venting, and dilution of hydrogen and oxygen to prevent conditions that would result in deflagration or explosion. The system also provides for monitoring the content of oxygen and hydrogen in the containment building atmosphere to alert operators when action levels are reached.

3. System Boundaries

Systems for the control of combustible gas within the containment may be totally inside the containment building or portions may be outside the containment building. The major components of the four systems listed above that constitute the CCGCS are given below.

The postaccident hydrogen venting system consists of a supply and exhaust system, including fans, ducting, a prefilter, a high-efficiency particulated air (HEPA) filter, and a charcoal filter. The postaccident hydrogen sampling system comprises fans, ducting, a sample vessel, and hydrogen monitoring instruments. The postaccident hydrogen mixing system typically includes fans and ducting. The hydrogen recombiners, if external to the containment, also are associated with ducting (large-diameter piping).

B. See Section I, "Areas of Review," of SRP-LR B.0.1 for Item I.B.

C. As a result of aging mechanisms such as fatigue, general wear, thermal and radiation embrittlement, bearing wear, and fan blade cracking, there have been failures of motors, heating coils, and various electrical and instrumentation components (Ref. 1). These areas of concern are expected to continue throughout the life of the plant, including the license renewal period. The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for Items III. A. through III. D.

E. CCGCS components not specifically addressed in this section may be addressed by the generic aging topic reviews in SRP-LR Part C. The reviewer should ensure that structures and components included as part of the generic SRP-LR topics are adequately reviewed for this system. The following sections of SRP-LR Part C should be used for the review: C.2.1, "Cables and Wiring"; C.2.3, "Electrical Penetrations"; C.2.4, "Relays, Circuit Breakers, and Switchgear"; C.2.7, "Electrical Motors"; and all of Section C.3.0, "Instruments." This may require other staff input.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "A Review of Information Useful for Managing Aging in Nuclear Power Plants," NUREG/CR-5562 (PNL-7323), June 1990.

B.3.8 CONTAINMENT SPRAY SYSTEM

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SPLB/EMCB

I. AREAS OF REVIEW

A. This section addresses the containment spray system (CSS)

1. Description

Although specific designs of the CSS are different for the various reactor vendors, their general characteristics are quite similar. In general, the CSS consists of two separate trains of equal capacity with each train independently capable of meeting system requirements. Typically each train includes a pump, heat exchanger, ring header with nozzles, isolation valves and associated piping, and instrumentation and controls. During normal operation, all of the equipment is idle and the associated isolation valves are closed. During a loss-of-coolant accident (LOCA), the CSS may initially draw water from an external source (e.g., borated/refueling water storage tank). Once the external source is depleted, the CSS pump suction is automatically realigned to draw from the containment sump for pressurized-water reactor (PWRs) or the suppression pool for boiling-water reactor (BWRs).

The addition of sodium hydroxide (NaOH) to the containment building spray during injection or trisodium phosphate during recirculation removes and retains radioactive iodine in a non-volatile form, thereby reducing post-LOCA offsite doses. However, these additives are not used with BWR systems, nor are they used with PWR ice condenser systems in which the ice contains sodium tetraborate that accomplishes the same result as iodine control.

The BWR CSS is a subsystem of the residual heat removal (RHR) system. Other post-LOCA RHR subsystems include low pressure coolant injection to the reactor and suppression pool cooling. The BWR RHR system also provides cooling of the reactor during normal shutdown conditions.

The CSS is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the specific facility.

2. System Function

The CSS is an engineered safety features system that is used to reduce containment pressure and temperature following a LOCA. Addition of sodium hydroxide to PWR containment sprays during injection or trisodium phosphate during recirculation helps to remove radioactive fission products from the containment building atmosphere, thereby reducing offsite dose.

BWR containment spray system functions are performed by the RHR system and cannot be placed in operation unless the core cooling requirements of the low-pressure coolant injection subsystem have been satisfied.

The CSS removes heat from primary containment and is, therefore, also part of the containment heat removal system (see SRP-LR B.3.3 title of the SRP-LR).

3. System Boundaries

The CSS extends from the pump suction valves, through the pump, heat exchangers, piping, and containment spray nozzles. All associated instruments, controls, and electrical equipment are included in the CSS. For the BWR CSS, which is a subsystem of RHR, the RHR control logic that places the CSS in service is: considered part of the CSS.

B. See Section I, "Area of Review," of B.0.1 for item I.B.

C. Aging Concerns and Mechanisms

Typical examples of age-related degradation associated with CSS are listed below

1. Valve degradation from wear, foreign material, or vibration damage

2. Pump degradation from wear or vibration damage

3. Heat exchanger degradation, especially the tubes, from corrosion and erosion

4. Piping degradation from corrosion, erosion, and thermal fatigue

The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for Items III.A. through III.D.

E. CSS components not specifically addressed in this section may be addressed by the generic aging topic reviews in SRP-LR Part C. The reviewer should ensure that structures and components included as part of the generic SRP-LR topics are adequately reviewed for this system. Specifically all the sections of Part C should be used for the review except Sections C.1.5, "Tanks and Vessels"; C.2.5, "Transformers"; and C.4.0, "Civil Structures." This may require other staff input.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

Aging assessments of several plant systems similar to the CSS have been performed as part of the nuclear plant aging research (NPAR) program (Ref. 1). Failure data from various national data bases were reviewed and analyzed to identify predominant failure modes, causes, and mechanisms. Time-dependent failure rates for major components were determined to identify aging trends. Plant-specific data were obtained and evaluated to supplement data results.

The data suggested that piping and heat exchangers can become very dominant in later years if failure rates increase at the rates indicated. This is due to the predominant failure mechanisms of corrosion and erosion, which are relatively slow processes. Increased surveillance may be necessary for these components in later years of plant life.

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "A Review of Information Useful for Managing Aging in Nuclear Power Plants," NUREG/CR-5562 (PNL-7323), June 1990.

B.3.9 CONTAINMENT VENTILATION SYSTEM

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SPLB

I. AREA OF REVIEW

A. This section addresses the containment ventilation systems (CVS)

1. Description

The CVS typically uses large fan cooler units (FCUs) with associated ductwork, dampers, and local fans to maintain temperature relatively uniform and within design limits. The system also may include strategically located safety-related fans that are used in the environment following a loss-of-coolant accident (LOCA) to prevent buildup of denotable hydrogen in dead air spaces. The FCUs transfer heat from the containment to the component cooling water system and then to the service water system, or directly to the SWS from the FCUs. Plant-specific designs are subject to considerable variation depending on the equipment used and its importance to safety.

The CVS is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the specific facility.

2. System Function

The CVS cools the atmosphere of containment and related subcompartments, maintaining each within peak and average temperature limits during normal operation and shutdown, including anticipated transients. Some PWRs use elements of the CVS to assist the containment spray system in reducing containment post-LOCA pressure. For these plants, appropriate CVS components are required to be safety-related. Boiling-water reactors (BWRs), and some PWRs, use only the containment sprays for post-LOCA pressure and temperature reduction. For these plants, the CVS is nonsafety-related. Both PWRs and BWRs may use certain elements of the CVS (e.g., fans, ductwork, and dampers) to prevent the potential buildup of post-LOCA hydrogen in dead air spaces. These elements of the CVS are required to be safety-related.

Containment filtration and pressure control during normal operations and shutdown is provided by the containment purge system.

3. System Boundaries

The CVS includes all fan cooler units, ductwork, dampers, fans, and instrument sensors located in the primary containment and related subcompartments for the purpose of heat removal during normal operation and shutdown. Depending on the plant-specific design, the CVS may interface with the containment heat removal system and the containment combustible gas control system or the containment purge system.

B. See Section I, "Areas of Review," of SRP-LR B.0.1 for item I.B.

C. Aging mechanisms such as fatigue, general wear, and thermal and radiation embrittlement, among others, result in bearing wear, fan blade cracking, and failure of motors, heating coils, and electrical and instrumentation control circuitry. These age-related degradation mechanisms are expected to continue throughout the license renewal period. The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the integrated plant assessment (IPA).

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for items III.A. through III.D.

E. CVS components not specifically addressed in this section may be addressed by the generic aging topic reviews in Part C of the SRP-LR. The reviewer should ensure that structures and components included as part of the generic SRP-LR topics are adequately reviewed for this system. The following sections from SRP-LR Part C should be used for the review: C.1.4, "Heat Exchangers"; C.2.1, "Cable and Wiring"; C.2.3, "Electrical Penetrations"; C.2.4, "Relays, Circuit Breakers, and Switchgear"; C.2.7, "Electrical Motors"; and all of section C.3.0, "Instruments." This may require other staff input.

F. The reviewer should coordinate the review of the CVS with that of the containment heat removal system and the containment combustible gas system, SRP-LR B.3.3 and B.3.7, respectively.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

The CVS is subject to aging effects similar to all ventilation systems. Typically, these systems operate in elevated temperatures, receive limited preventive maintenance, and are equipped with minimal instrumentation for detection of performance degradation. Additional information regarding aging of fan coolers may be found in References 1 and 2. Aging effects include:

° Fatigue failures of fan blades, dampers, baffles, and housings

° Bearing failures resulting from wear, misalignment, excessive belt tension or vibration

° Degradation of heat exchanger performance resulting from dirt accumulation on the air side and fouling on the water side

° Corrosion associated with condensation on cool surfaces (ductwork and cooling coils)

° Mechanical damage to ductwork and dampers associated with adjacent maintenance activities

° Electric motor failures resulting from wear, elevated temperature effects, vibration, and radiation degradation

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "A Review of Information Useful for Managing Aging in Nuclear Power Plants," NUREG/CR-5562 (PNL-7323), June 1990.

2. Pacific Northwest Laboratory, "Operating Experience and Aging Assessment of ECCS Pump Room Coolers," PNL-5722, October 1986.

B.4.1 MAIN POWER SYSTEM

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SELB/SICB

I. AREA OF REVIEW

A. This section addresses the main power system

1. Description

The main power system supplies power both to the electrical equipment for balance of plant that is required for normal operation and control of the plant and to the electrical equipment for engineered safety features that is required for the safe shutdown and control of the plant during design-basis accidents. In some plants the same electrical buses that supply nonessential loads also provide power to essential loads. The essential power system is reviewed under SRP-LR B.4.2.1 and the nonessential power system under SRP-LR B.4.2.2. The main power system receives power from redundant sources to enable it to continuously function during adverse conditions. Redundant sources are typically some combination of offsite power, the station switchyard, or the station generator. The main power system may be divided into at least two independent channels to further provide for redundant grouping of loads. Thus, the main power system not only provides for redundant sources of power but also redundant grouping of loads.

The main power system is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the specific facility.

2. System Function

The main power system supplies power both to the electrical equipment for balance-of-plant that is required for normal operation and control of the plant and to the electrical equipment for engineered-safety-features that is required for the safe shutdown and control of the plant during design-basis accidents.

3. System Boundaries

The main power system includes the disconnects, switches, circuit breakers, relays, switchgear, buses, cables, and transformers that are necessary to provide electrical power to both the essential and the nonessential power systems. The system boundary begins with the disconnect switches feeding all power from the switchyard and ends with the circuit breakers feeding essential and nonessential buses. In some plants the system also includes the tertiary offsite power system which provides another source of offsite power to the essential power system.

B. See Section I, "Area of Review," of SRP-LR B.0.1 for Item I.B.

C. The main power system is composed primarily of components that are reviewed in accordance with other sections of SRP-LR. These components and the applicable SRP-LR section are:

Component	Section
Cable and wiring	C.2.1
Junctions	C.2.2
Relays, switchgear, circuit breakers	C.2.4
Transformers	C.2.5

In addition, protective relaying and controls are reviewed in accordance with SRP-LR B.4.1.1.

Typical examples of age-related de radation associated with the main power system are provided in this section. The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

Any aging degradation of these components can potentially affect their ability to function as required. However, aging degradation of some of these components is of particular concern to the main power system because the affected component must function at the time of an event or provide additional power. Some of the components of concern are noted below.

- o Circuit breakers that must disconnect a failed power source, such as loss of power from the switchyard, loss of power generated on site, or total loss of offsite power
- o Protective relaying and control
- o Transformers that must begin to transfer power as result having to switch from one source of power to another
- o Cables
- o Switchgear

Circuit breaker aging stressors may be categorized as either thermal, electrical, mechanical, or environmental. The most likely failures are related to mechanical and electrical effects and primarily result in failure to open or close, improper operation, restrike, shorting, and arcing. A summary of these effects is given in Table B.4.1-1.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1 for Items II.A and II.B.

C. In addition to the above items, the licensee's program should thoroughly address the stressors each circuit breaker will experience and show that the existing or proposed program of surveillance and maintenance will detect and correct aging degradation before loss of function is experienced. Table B.4.1-1 provides a listing of stressors that should be considered.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for Items III.A through III.D.

E. Components of the main power system not specifically addressed in this section may be addressed in the generic aging topic reviews in SRP-LR Part C. The following sections of SRP-LR Part C and should be used for the review: C.2.1, "Cable and Wiring"; C.2.2, "Junctions;" C.2.4, "Relays, Circuit Breakers, and Switchgear;" C.2.5 "Transformers." This may require additional staff input.

F. The reviewer should ensure that the licensee's program identifies and addresses aging degradation of circuit breakers.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

Studies performed as a part of the NRC Nuclear Plant Aging Research (NPAR) Program have shown that of the components that form the main power system, circuit breakers have the highest frequency of events requiring licensee event reports (66.3 percent of all Class 1E power system events) followed by transformers (4.7 percent of all 1E power system events) (Ref. 1). Studies utilizing the Nuclear Plant Reliability Data System (NPRDS) information have shown that about 21 percent of circuit breaker failures and 13 percent of transformer failures are aging related. Aging of electrical conductors accounts for only 8 percent of the reported failures (Ref. 2). Circuit breakers, therefore, have the highest failure rate and also are affected the most by aging-related degradation.

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "Nuclear Plant Aging Research: the 1E Power System," NUREG/CR-5181, April 1990.

2. U.S. Nuclear Regulatory Commission, "An Aging Failure Survey of Light Water Reactor Safety Systems and Components," NUREG/CR-4747, July 1987.

Table B.4.1-1 Aging Degradation of Circuit Breakers

Stressors	Cause	Failure Mechanism	Failure Mode
Thermal	Poor contact	Degraded insulation	Short to ground
	Large current	Degraded contacts	Poor or open contacts
		Degraded arc chutes	Flash over
		Degraded overload mechanism (molded-case)	Failure to extinguish the arc
			Premature trip at low current
Electrical	Over voltage transients	Arcing of contacts causing contamination of components	Restrike
	Spikes		Shorting of components
	Fault interruption		Arcing to ground or between phases
	Lightning		
Mechanical	Routine operation	Degraded contacts	Failure to open or close
	Fault	Fatigue	Improper operation
	Interruptions	Wear	
	Vibration	Loose connections	
	Friction	Reduced force	
		Compound failure	
Environmental	Elevated temperature	Increased friction	Failure to open or close
		Degraded insulation	
	Elevated humidity	Oxidation	Shorting and arcing

Table B.4.1-1 Aging Degradation of Circuit Breakers

Stressors	Cause	Failure Mechanism	Failure Mode
	Dirt	Hardening of lubricant	Improper operation
	Chemicals		
	Rust	Embrittlement of materials	

B.4.1.1 PROTECTIVE RELAYING AND CONTROLS

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SELB/SICB

I. AREA OF REVIEW

A. This section addresses the protective relaying and controls:

1. Description

The protective relaying and controls is composed of components that are required to control and protect the various items of equipment in the electrical distribution system. The components are reviewed in accordance with other sections of SRP-LR. These components and the applicable SRP-LR sections are:

Cable and Wiring	C.2.1
Junctions	C.2.2
Relays, Switchgear, Circuit Breakers	C.2.4
Electronic Devices	C.3.3

The protective relaying and controls specific description is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the facility.

2. System Function

The protective relaying and controls perform the following functions:

a. Continuously monitors power to safety related loads and automatically switches power sources when necessary.

b. Precludes damage to electrical equipment as a result of extended periods of operation at reduced voltage levels.

c. Overrides the effects of short-duration system disturbances as well as the effects of transients due to the starting of large motors fed from the plant distribution system.

d. Provides for remote indication, alarms, and control of the plant power systems.

3. System Boundaries

Protective relaying and controls includes the relays, current and potential transformers, and controls and indicators that are necessary to detect abnormal operating conditions and initial corrective actions. Also included are the cables, switches, and indicators for providing both remote and local control and indication of the state of the equipment. The system boundaries do not include the various items of controlled equipment, such as breakers, or the control panels or enclosures. Also not included are the sources or required power, such as 125 V/dc or 120 V/ac power.

B. See Section I, "Area of Review," of SRP-LP B.0.1 for Item I.B.

C. Any aging degradation of these components can potentially affect their ability to function as required. However, aging degradation of some of these components is of particular concern to the protective relaying and controls system because the affected component must function at the time of an event in contrast to components that are only required to continue to function. Some of significant aging degradations are:

1. Breakdown of relay and transformer coil insulation caused by inductive surges and over temperature. Over temperature may be caused by ohmic heating as a result of overvoltage operation, elevated ambient temperature, and temperature rises in the cabinet housing.

2. Wear of relays and switches as a result of continued use and high cycling rate.

3. Increase friction of relays and switches caused by dust, dirt and contamination.

Relay stressors may be categorized as either thermal, electrical, mechanical, or environmental. The most likely failures are related to mechanical and electrical effects and primarily result in failure to open or close, improper operation, and arcing of the contacts. A summary of these effects is given in Table B.4.1.1-1.

Typical examples of age-related degradation associated with protective relays and controls are provided in this section. The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1.

E. Components of the Protective Relaying and Controls system not specifically addressed in this section may be addressed in the generic aging topic reviews in SRP-LR Part C. Specifically the SRP-LR Part C chapter listed in I.A 1 above should be reviewed in conjunction with SRP-LR B.4.1.1 following sections of Part C are applicable to the Protective Relaying and Controls system and should be reviewed: C.2.1 "Cable and Wiring"; C.2.2 "Junctions"; C.2.4, "Relays, Circuit Breakers, and Switchgear", and C.3.3 "Electronic Devices." This may require additional staff input.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

Studies performed as a part of the NRC Nuclear Plant Aging Research (NPAR) Program have shown that of the components that form the Protective Relaying and Control system, relays and switches account for the majority of the failures. Studies utilizing Nuclear Plant Reliability Data Systems (NPRDS) information have shown that about 25 percent of relay failures and 23 percent of switch failures are aging related. Aging of electrical conductors accounts for only 8 percent of the reported failures (Refs 1 and 2) Relays are a device that must operate during abnormal conditions and yet are significantly affected by aging effects.

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "Nuclear Plant Aging Research: The 1E Power System," NUREG/CR-5181, April 1990.

2. U.S. Nuclear Regulatory Commission, "An Aging Failure Survey of Light Water Reactor Safety Systems and Component," NUREG/CR-4747, July 1987.

Table B.4.1.1-1 Tabulation of stresses and effects on relays

Stress	Effect of Stress on Component	Effect on Operation
Electrical Stresses		
Inductive surge	Breakdown of coil insulation (corona attack and dielectric breakdown of insulation weak points)	Open-circuited coils
Overvoltage operation	Increases ohmic heating of relay	See Thermal Stresses
Mechanical Stresses		
High cycling rate	Wear of moving parts	Binding of relay
	Contact wear	Misoperation of relay
	Increased friction	Coil failure
	Mechanical fatigue	
	Electrical pitting and arcing of contacts	
	Inductive surge	
Loose connections (relay socket/ terminals)	Loosening of pin/socket interface	High resistance paths
	Air gaps between contacts and connections	Arcing across contacts
		Open circuits
Vibration	Material fatigue	Component failures
	Loosening of connections	Open circuits
	Intermittent contact opening (chatter)	Inadvertent operation
	Inadvertent contact closure	
Dormancy (lack of operation)	Organic materials set	Failure to operate
	Organic materials adhere to adjacent material	Binding

Table B.4.1.1-1 (continued)

Stress	Effect of Stress on Component	Effect on Operation
Thermal Stresses		
Continuous energization (ohmic heating)	Accelerates aging of coil insulation and other nonmetallic components	Leads to insulation and component failure
Temperature rises in cabinet housing	Accelerates aging of nonmetallic materials including coil insulation, bobbin, relay base, and contact spacers	Same as above
Elevated ambient temperature	Accelerates aging of nonmetallic components	Same as above
Humidity	Corrosion of contacts	Open circuits/ increased resistances
	Coil and contact leakage paths	
Dust, dirt, contamination	Interferences	Binding
	Increases in friction forces	Slow or sluggish operation
	Increased resistance	Open circuits/ increased ohmic heating

B.4.2.1 ESSENTIAL POWER SYSTEM

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SELB/SICB

I. AREA OF REVIEW

A. This section addresses the essential power system:

1. Description

The essential power system supplies power to the electrical equipment for engineered safety features that is required to provide for the safe shutdown and control of the plant during design-basis accidents. The system receives power from redundant sources so that it can continuously function during adverse and accident conditions. Redundant sources include main power, as described in SRP-LR B.4.1 of SRP-LR, offsite power, and emergency power systems, as described in SRP-LR B.4.4 and B.4.3.1 of SRP-LR. The essential power system is divided into at least two independent channels to further provide for redundant grouping of critical loads. Thus, the essential power system not only provides for redundant sources of power but also for redundant grouping of loads.

The essential power system is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the specific facility.

2. System Function

The essential power system supplies power to the electrical equipment for engineered safety features that is required to provide for the safe shutdown and control of the plant during design-basis accidents.

3. System Boundaries

The essential power system includes the breakers, buses, cabling, and transformers that are necessary to provide electrical power to the electrical equipment for engineered safety features that is required to provide for the safe shutdown and control of the plant during design basis accidents. The system boundary begins with the circuit breakers feeding the various essential power system buses and ends at the motor control centers that provide power to the electrical equipment. The system does not include the sources of electrical power such as the higher voltage buses that feed the essential power system buses, the offsite power, or the diesel generators. In addition, the essential power system does not include the various items of electrical equipment that are connected to the motor control centers.

B. See Section I, "Area of Review," of SRP-LR for Item I.B.

C. The essential power system consists primarily of components that are reviewed in accordance with other sections of the SRP-LR. These components and the applicable SRP-LR section are:

Component	Section
Cable and wiring	C.2.1
Junctions	C.2.2
Relays, switchgear, circuit breakers	C.2.4
Transformers	C.2.5

In addition, protective relaying and controls are reviewed in accordance with SRP-LR B.4.1.1.

Any aging degradation of these components can potentially affect their ability to function as required. However, aging degradation of some of these components is of particular concern to the essential power system because the affected component must function at the time of an event or provide additional power. Some of the components of concern are noted below.

o Circuit breakers that must disconnect a failed power source, such as loss of power from the switchyard, loss of power generated onsite or total loss of offsite power

o The circuit breaker that must close to supply power from a diesel generator

o Circuit breakers that must open and then reclose during load shedding and subsequent energizing on startup of the diesel generator

o Protective relaying and control

o Transformers that must begin to provide power as result having to switch from one source of power to another

Circuit breaker aging stressors may be categorized as either thermal, electrical, mechanical, or environmental. The most likely failures are related to mechanical and electrical effects and primarily result in failure to open or close, improper operation, restrike, shorting, and arcing. A summary of these effects is given in Table B.4.2.1_1.

Typical examples of age-related degradation associated with the essential power system are provided in this section. The areas of concern with regard to aging for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1 for Items II. A and II.B.

C. In addition to the above items, the licensee's program should thoroughly address the stressors each circuit breaker will experience and show that the existing or proposed program of surveillance and maintenance will detect and correct aging degradation before loss of function is experienced. Table B.4.2.1-1 provides a listing of stressors that should be considered.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 Items III. A through III.D.

E. Components of the essential power system not specifically addressed in this section may be addressed in the generic aging topic reviews in SRP-LR Part C. Specifically the following sections of SRP-LR Part C are applicable to the Essential Power system and should be reviewed: C.2.1, "Cable and Wiring"; C.2.2, "Junctions"; C.2.4, "Relays, Circuit Breakers, and Switchgea," C.2.5 "Transformers." This may require other staff input.

F. The reviewer should ensure that the licensee's program identifies and addresses aging degradation of circuit breakers.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

Studies performed as a part of the NRC Nuclear Plant Aging Research (NPAR) Program have shown that, of the components that form the essential power system, circuit breakers have the highest frequency of events requiring licensee event reports (66.3 percent) of all class 1E power system events) followed by transformers (4.7 percent of all 1E power system events). Studies utilizing the Nuclear Plant Reliability Data System (NPRDS) information have shown that about 21 percent of circuit breaker failures and 13 percent of transformer failure are aging related. Aging of electrical conductors accounts for only 8 percent of the reported failures.

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "Nuclear Plant Aging Research: the 1E Power System," NUREG/CR-5181, April 1990.

2. U.S. Nuclear Regulatory Commission, "An Aging Failure Survey of Light Water Reactor Safety Systems and Component," NUREG/CR-4747, July 1987.

3. Institute of Electrical and Electronics Engineers, IEEE Standard 308-1980, October 1980, "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations."

TABLE B.4.2.1-1 Aging degradation of circuit breakers

Stressors	Cause	Failure Mechanism	Failure Mode
Thermal	Poor contact	Degraded insulation	Short to ground
	Large current	Degraded contacts	Poor or open contacts
		Degraded arc chutes	Flash over
		Degraded overload mechanism (molded-case)	Failure to extinguish the arc
			Premature trip at low current
Electrical	Overvoltage transients	Arcing of contacts causing contamination of components	Restrike
	Spikes		Shorting of components
	Fault interruption		Arcing to ground or between phases
	Lightning		
Mechanical	Routine operation	Degraded contacts	Failure to open or close
	Fault	Fatigue	Improper operation
	interruptions	Wear	
	Vibration	Loose connections	
	Friction	Reduced force	
		Compound failure	
Environmental	Elevated temperature	Increased friction	Failure to open or close
	Elevated humidity	Degraded insulation	Shorting and arcing
	Dirt	Oxidation	Improper operation
		Hardening of lubricant	
	Chemicals		
	Rust	Embrittlement of materials	

TABLE B.4.2.1-1 Aging degradation of circuit breakers

Stressors Cause Failure Mechanism Failure Mode

B.4.2.2 NONESSENTIAL POWER SYSTEM

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SELB/SICB

I. AREAS OF REVIEW

A. This section addresses the nonessential power system.

1. Description

The nonessential power system supplies power to the balance of plant electrical equipment that is required for normal operation and control of the plant but not required during design-basis accidents. In some plants, the same electrical buses that supply nonessential loads also provide power to the essential power system that is reviewed under SRP-LR B.4.2.1. The nonessential power system receives power from redundant sources to enable it to continuously function during adverse conditions. Redundant sources are typically some combination of offsite power, the station switchyard, or the station generator. The nonessential power system may be divided into at least two independent channels to further provide for redundant grouping of loads. Thus, the nonessential power system not only provides for redundant sources of power but also for redundant grouping of loads.

The nonessential power system is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the specific facility.

2. System Function

The nonessential power system supplies power to the balance of plant electrical equipment that is required for normal operation and control of the plant but not required during design-basis accidents.

3. System Boundaries

The nonessential power system includes the breakers, buses, cabling, and transformers that are necessary to provide electrical power to the balance of plant electrical equipment that is required for normal operations of the plant. The system boundary begins with the circuit breakers feeding the nonessential power buses and ends at the motor control centers that provide power to the electrical equipment. The system does not include the sources of electrical power, such as the switchyard, that feeds the nonessential power buses. In addition, the nonessential power system does not include the various items of electrical equipment that are connected to the motor control centers.

B. See Section I, "Area of Review," of SRP-LR B.0.1 for item I.B.

C. The nonessential power system is composed primarily of components that are reviewed in accordance with other sections of SRP-LR. These components and the applicable SRP-LR section are:

Component	Section
Cable and wiring	C.2.1
Junctions	C.2.2
Relays, switchgear, circuit breakers	C.2.4
Transformers	C.2.5

In addition, protective relaying and controls are reviewed in accordance with SRP-LR B.4.1.1.

Aging degradation of some of these components is of part r concern to the nonessential power system because the affected component must function at the time of an event in contrast to components that are only required to continue to function or provide additional power. Some components of concern are noted below.

1. Circuit breakers that must disconnect a failed power source, such as loss of power from the switchyard, loss of power generated on-site or total loss of offsite power

2. Protective relaying and control

3. Transformers that must begin to provide power as result having to switch from one source of power to another

Circuit breaker aging stressors may be categorized as either thermal, electrical, mechanical, or environmental. The most likely failures are related to mechanical and electrical effects and primarily result in failure to open or close, improper operation, restrike, shorting, and arcing. A summary of these effects is given in Table B.4.2.2-1. Typical examples of aging-related degradation associated with the nonessential power system are provided in this section. The areas of concern with regard to aging for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for Items III. A through III. D.

E. Components of the nonessential power system not specifically addressed in this section may be addressed in the generic aging topic reviews in SRP-LR Part C. Specifically the following sections of SRP-LR Part C are applicable to the non essential power system and should be used for the review C.2.1, "Cable and Wiring"; C.2.2 "Junctions"; C.2.4, "Relays, Circuit Breakers, and Switchgear"; C.2.5 "Transformers." This may require additional staff input.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.6.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

Studies performed as a part of the NRC Nuclear Plant Aging Research (NPAR) Program have shown that of the components that form the nonessential power system circuit breakers have the highest frequency of events requiring licensee event reports (66.3 percent of all class 1E power system events), followed by transformers (4.7 percent of all 1E power system events). Studies utilizing Nuclear Plant Reliability Data System (NPRDS) information have shown that about 21 percent of circuit breakers failures and 13 percent of transformer failures are aging related. Aging of electrical conductors accounts for only 8 percent of the reported failure. (Ref. 2) Circuit breakers, therefore, have the highest failure rate and are also affected the most by aging-related degradation.

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "Nuclear Plant Aging Research: the 1E Power System," NUREG/CR-5181, April 1990.

2. U.S. Nuclear Regula[illegible] Commission, "An Aging Failure Survey of Light Water Reactor Safety Systems and Component," NUREG/CR-4747, July 1987.

3. Institute of Electrical and Electronics Engineers, IEEE Standard 308-1980, "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations."

TABLE B.4.2.2-1 Aging degradation of circuit breakers

Stressors	Cause	Failure Mechanism	Failure Mode
Thermal	Poor contact	Degraded insulation	Short to ground
	Large current	Degraded contacts	Poor or open contacts
		Degraded arc chutes	Flash over
		Degraded overload mechanism (molded-case)	Failure to extinguish the arc
			Premature trip at low current
Electrical	Over voltage transients	Arcing of contacts causing contamination of components	Restrike
	Spikes		Shorting of components
	Fault interruption		Arcing to ground or between phases
	Lightning		
Mechanical	Routine operation	Degraded contacts	Failure to open or close
	Fault	Fatigue	Improper operation
	interruptions	Wear	
	Vibration	Loose connections	
	Friction	Reduced force	
		Compound failure	
Environmental	Elevated temperature	Increased friction	Failure to open or close
	Elevated humidity	Degraded insulation	Shorting and arcing
	Dirt	Oxidation	Improper operation
	Chemicals	Hardening of lubricant	
	Rust	Embrittlement of materials	

B.4.2.3 HIGH PRESSURE CORE SPRAY POWER SYSTEM (BWR)

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SRXB/SELB/SICB

I. AREAS OF REVIEW

A. This section addresses the high pressure core spray (HPCS) power system.

1. Description

Eight BWR plants utilize HPCS systems to depressurize the reactor and maintain the reactor vessel water level in the event of a small break LOCA. The HPCS Power System supplies power to the HPCS pump and supporting electrical equipment that is needed to depressurize the reactor and maintain the reactor vessel water level in the event of a small break LOCA. The HPCS power system is usually a dedicated bus of the Essential Power System that is reviewed under SRP-LR B.4.2.1, "Essential Power System". The HPCS Power System receives power from redundant sources to enable it to continuously function during adverse conditions. Redundant sources are typically some combination of off-site power, the station switchyard, the station generator, and a dedicated diesel generator for loss of off-site power situations. The sources of power are reviewed under SRP-LR B.4.1, "Main Power", B.4.4, "Emergency Diesel Generators", and B.4.3.1, "DC Power System".

The HPCS power system specific description is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the specific facility.

2. System Function

The HPCS power system supplies power to the HPCS pump and supporting electrical equipment that is needed to depressurize the reactor and maintain the reactor vessel water level in the event of a small break loss-of-coolant-accident (LOCA).

3. System Boundaries

The HPCS power system includes the breakers, busses, cabling, and transformers that are necessary to provide electrical power to the HPCS pump and associated electrical equipment that is required to depressurize the reactor and maintain the reactor vessel water level in the event of a small break LOCA. The system boundary begins with the circuit breakers feeding the HPCS power system busses and ends at the motor control centers that provide power to the electrical equipment. The system does not include the sources of electrical power such as the higher voltage busses that feed the HPCS power buses, the off-site power, or the diesel generators. In addition, the HPCS power system does not include the various items of electrical equipment that are connected to the motor control centers.

B. See Section 1, "Area of Review," of SRP-LR B.0.1 for Item I.B.

C. The HPCS power system is composed primarily of components that are reviewed in accordance with other sections of SRP-LR. These components and the applicable SRP-LR section are:

Component	SRP-LR Section
Cable and Wiring	C.2.1
Junctions	C.2.2
Relays, Switchgear, circuit breakers	C.2.4
Transformers	C.2.5

In addition, protective relaying and controls is reviewed in accordance with SRP-LR B.4.1.1.

Any aging degradation of these components can potentially affect their ability to function as required. However, aging degradation of circuit breakers that must function at the time of the event is of particular concern to the HPCS Power System. Some of the age related degradations are noted below:

1. Degraded insulation that leads to shorts to ground or between conductors. Thermal effects are a leading cause of this degradation.

2. Degraded contacts causing a failure to open or close. Arcing and thermally induced corrosion are primary causes of the degradation.

3. Degraded arc chutes that leads to flash over. Thermal effects caused by large currents are a primary cause of this degradation.

Typical examples of age-related degradation associated with the HPCS power system are provided in this section. The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1 items III.A and III.B.

C. In addition to the above items that should be considered by the licensee, include the following:

1. The licensee's program should thoroughly address the degradation mechanisms each circuit breaker will experience and show that the existing or proposed program of surveillance and maintenance will detect and correct aging degradation before loss of function is experienced. Table B.4.2.3-1 provides a listing of degradation mechanisms that should be considered.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 items III.A through III.D.

E. Components of the HPCS system not specifically addressed in this section may be addressed in the generic aging topic reviews in SRP-LR Part C. Specifically the following sections of SRP-LR Part C are applicable to the HPCS system and should be reviewed: C.2.1 "Cable and Wiring", C.2.2 "Junctions", C.2.4 "Relays, Circuit Breakers, and Switchgear", C.2.5 "Transformers". This may require other staff input.

F. The reviewer should assure that the licensee's program identifies and addresses aging degradation of circuit breakers.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

Studies performed as a part of the NRC Nuclear Plant Aging Research (NPAR) Program have shown that of the components that form the HPCS power system, circuit breakers have the highest frequency of LER events (66.3 percent of all 1E power system events), followed by transformers (4.7 percent of all 1E power system events). Studies utilizing Nuclear Plant Reliability Data Systems (NPRDS) information have shown that about 21 percent of circuit breakers failures and 13 percent of transformer failures are aging related. Aging of electrical conductors accounts for only 8 percent of the reported failures (Refs. 1,2). Circuit breakers, therefore, historically have the highest failure rate of these components and are also affected the most by aging related degradation.

Circuit breaker aging mechanisms may be categorized as either thermal, electrical, mechanical, or environmental. The most likely failures are related to mechanical and electrical effects and primarily result in failure to open or close, improper operation, restrike, shorting, and arcing. A summary of these effects is given in Table B.4.2.3-1.

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "Nuclear Plant Aging Research: The 1E Power System," NUREG/CR-5181, April 1990.

2. U.S. Nuclear Regulatory Commission, "An Aging Failure Survey of Light Water Reactor Safety Systems and Component," NUREG/CR-4747, July 1987.

3. Institute of Electrical and Electronic Engineers, IEEE Std 308-1980, October 1980, "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations."

Table B.4.2.3-1 Aging degradation of circuit breakers

Stressors	Cause	Failure mechanism	Failure mode
Thermal	Poor contact	Degraded insulation	Short to ground
	Large current	Degraded contacts	Poor or open contacts
		Degraded arc chutes	Flash over
		Degraded overload mechanism (molded-case)	Failure to extinguish the arc
			Premature trip at low current
Electrical	Over voltage transients	Arcing of contacts causing contamination of components	Restrike
	Spikes		Shorting of components
	Fault interruption		Arcing to ground or between phases
	Lightning		
Mechanical	Routine operation	Degraded contacts	Failure to open or close
	Fault	Fatigue	Improper operation
	interruptions	Wear	
	Vibration	Loose connections	
	Friction	Reduced force	
		Compound failure	

Table B.4.2.3-1 Aging degradation of circuit breakers

Stressors	Cause	Failure mechanism	Failure mode
Environmental	Elevated temperature	Increased friction	Failure to open or close
	Elevated humidity	Degraded insulation	Shorting and arcing
	Dirt	Oxidation	Improper operation
	Chemicals	Hardening of lubricant	
	Rust	Embrittlement of materials	

B.4.3.1 DC POWER SYSTEM

REVIEW RESPONSIBILITY

Primary - LRPD
Secondary - SELB/SICB

I. AREAS OF REVIEW

A. This section addresses the dc power system.

1. Description

The dc power system includes those dc power sources and their distribution systems and vital supporting systems provided to supply motive or control power to safety-related and important-to-safety equipment.

The dc power system is described specifically in the most recent update of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the individual facility.

2. System Function

The dc power system typically supplies control power to feeder breakers on the main ac feeder buses, 6.9-kV ac (if used), 4.16- kV ac, and 600- or 480-V ac load buses, and load center feeder and load circuit breakers. A separate dc power system may or may not be used for switchyard switching and is not safety related. The emergency safeguards system provides signals for starting the emergency standby power units and control logic for load shedding and subsequent loading of the diesel generators. The dc power system includes batteries, battery chargers, inverters, and associated load centers, switchgear, and buses.

3. System Boundaries

Batteries and battery chargers are used as the power sources for the dc power system. Inverters are used to convert dc power from the dc distribution system to ac instrument and control power as required. The dc power system is unique for each station.

B. See Section I, "Areas of Review," of SRP-LR B.1.0 for Item I.B.

C. A variety of aging mechanisms can affect the ability of the dc power system to continue to operate safely and effectively. The typical dc power system has a complement of meters and alarms in the control room to alert the operators to a rapid degradation of power. With careful observation, the operators can observe long-term degradation of the dc power system by observing the float voltage and current. Observing technical specification required testing for the dc power system will detect degradation of the system. Typical examples of age-related degradation associated with the dc power system are given in this section. The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

The dc power system can be degraded by degradation of the batteries, battery chargers, or bus systems. Batteries are affected by thermally induced grid and connector oxidation, plate and grid swelling, container and cover cracking, separator deterioration, or specific gravity changes. Battery chargers are affected by the aging of electrolytic capacitors, transformers, inductors, solid-state devices, and fuses. The battery chargers are also affected by the quality of the connected source power. Bus systems age with dielectric stress or partial discharge, cooling system degradation, and bus heating.

Circuit breakers, switchgear, and relaying aging are addressed in SLRP-LR C.2.4. Cable aging is addressed in SRP-LR C.2.1.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1 Items II.A and II.B.

C. These criteria should be part of an established ongoing licensee program to assure the availability of the dc power system. The licensee should also have a one time or new periodic inspection of components of the dc power system in conjunction with the license renewal application.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1, Items III.A through III.D.

E. Components of the dc power system not specifically addressed in this section may be addressed in the generic component or structure reviews in SRP-LR Part C. Specifically the following sections of SRP-LR Part C are applicable to the dc power system and should be reviewed: C.2.1, "Cable and Wiring"; C.2.2, "Junctions"; C.2.3, "Electrical Penetrations"; C.2.4, "Relays, Circuit Breakers, and Switchgear"; C.2.5, "Transformers"; and all of C.3.0 "Instrument." This may require additional staff input.

F. 1. The reviewer shall ensure that the licensee has an established ongoing maintenance and surveillance program to ensure the availability of the dc power system. This program should include a mechanism to add new testing and evaluation criteria to monitor newly detected aging deterioration.

2. Has a load study analysis that shows adequate capacity and capability, plus reserves and margins, to power any connected load configuration.

IV. EVALUATION FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

Section 4.4 of IEEE Standard 279-1971 (Ref. 1) requires, for those plants issued a construction permit after 1971, type test data (or reasonable engineering extrapolation of type test data) that verifies this equipment meets, on a continuing basis (emphasis added), the system performance requirements. Older plants may also be committed to IEEE Standard 279-1971.

Regulatory Guide 1.32 (Ref. 2) endorses IEEE Standard 308-1974 (Ref. 3). This industry standard recommends system capabilities and component capabilities for being able to start and operate the required loads during normal and post-accident conditions. It also includes provisions to test the battery capability to IEEE Standard 450-1975 (Ref. 4). This industry standard is endorsed by Regulatory Guide 1.129 (Ref. 5).

A. Batteries

NUREG/CR-4457 (Ref. 6) is an evaluation of the aging effects of safety-related batteries. It also evaluates maintenance, testing, and monitoring practices and discusses the effectiveness of these programs. The most significant aging factor is thermally induced oxidation of grids and top conductors. Oxidation causes the plates and grids to swell. This results in poor conduction between the plate and grid, reducing battery capacity and causing stresses in the container and covers. The stresses can result in cracks in the battery case. Separator deterioration is a cause of cell shorting. Considerable care is needed to maintain batteries in an operable condition. Maintenance practices that conform to IEEE Standard 450 and Regulatory Guide 1.129 ensure reliable battery capacity over its qualified life. Battery maintenance allows a history of the battery to be documented, and can be used to evaluate the state of the battery. Recommended battery maintenance includes battery capacity tests and recommended battery replacement should the battery capacity be less than 80 percent of the manufacturer's rating, among other criteria. Typically, the licensee will have replaced the battery before it has deteriorated to this limit. These other factors include service tests, the original sizing criteria utilized, and the capacity available compared to the load requirements. Other test methods may be developed in the future to better analyze degradation. Physical conditions of individual cells (such as plate condition, cell reversal, specific gravity readings, or electrolyte contamination) often require the replacement of individual cells. A battery nearing the end of its life should not have individual cells replaced.

B. Battery Chargers

NUREG/CR-50511 (Ref. 10) is an evaluation of the aging effects of safety-related battery chargers. Battery chargers are static electrical or electronic devices that convert low voltage ac power (typically 600-V ac or 480-V ac) to dc power (125-V dc or 2-125-V dc systems connected to yield 250-V dc). Battery chargers are the primary source of power to the dc power system. They charge and float charge the batteries that provide power on loss of ac power. Loss of ac power disables the battery chargers.

Electrolytic capacitors, transformers, inductors, and silicon controlled rectifiers (SCRs) are the battery charger components that are most susceptible to aging degradation. Fuse failures, caused primarily by thermal fatigue, can also contribute to battery charger failure, but are difficult to detect until they fail. The life of transformers and inductors is determined by the condition of the insulation. Electrolytic capacitors age as a result of deterioration or outgassing of the electrolyte. SCRs are subject to voltage transients and other semiconductor stress (usually thermal) failures. Operation of the battery chargers during diesel generator testing can stress battery chargers because of voltage and frequency variations. Excessively high equalization charge voltage can damage system loads and cause fuses to open. Some fuses may fail undetected. Excessive ac ripple voltage on the dc output will do this also. In addition, excessive ripple will cause premature aging of the battery. Electrical transients, such as could be caused by a noisy setpoint potentiometer, can cause the same premature aging.

All plants have some battery charger maintenance and capacity testing (Refs. 10 and 11). Some plants follow the battery charger internal temperature. The internal temperature is known to have an effect on component aging. Increased internal temperatures decrease the component life.

VII. REFERENCES

1. Institute of Electrical and Electronics Engineers, IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations."

2. Regulatory Guide 1.32, "Criteria for Safety-related Electric Power Systems for Nuclear Power Plants."

3. Institute of Electrical and Electronics Engineers, IEEE Standard 308-1974, "Criteria for Class 1E Power Systems for Nuclear Power Generating Stations."

4. Institute of Electrical and Electronics Engineers, IEEE Standard 450-1975, "Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

5. Regulatory Guide 1.129, "Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants."

6. U.S. Nuclear Regulatory Commission, "Aging of Class 1E Batteries in Safety Systems of Nuclear Power Plants," NUREG/CR-4457, July 1987.

7. U.S. Nuclear Regulatory Commission, "Nuclear Plant Aging Research: the 1E Power System," NUREG/CR-5181, April 1990.

8. Institute of Electrical and Electronics Engineers, IEEE Standard 535-1979, "IEEE Standard for Qualification of Class 1E Lead Storage Batteries for Nuclear Power Generating Stations."

9. Institute of Electrical and Electronics Engineers, IEEE Standard 484-1975, "Recommended Practice for Installation Design and Installation of Large Lead Storage Batteries for Generating Stations and Substations.

10. U.S. Nuclear Regulatory Commission, "Detecting and Mitigating Battery Charger and Inverter Aging," NUREG/CR-5051, August 1988.

11. U.S. Nuclear Regulatory Commission, "Operating Experience and Aging - Seismic Assessment of Battery Chargers and Inverters," NUREG/CR-4564, June 1986.

B.4.3.2 INSTRUMENT AC POWER SYSTEM

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SELB/SICB

I. AREA OF REVIEW

A. This section addresses instrument ac power systems.

1. Description

The 120-Vac vital instrument power system includes those 120-Vac power sources and their distribution systems and vital supporting systems provided to supply power to safety-related and important-to-safety instruments and control circuits. The dc power system (SRP-LR B.4.3.1) provides power to inverters. These inverters also have a Class 1E ac power source. Depending on plant design, either the ac or the dc source may be the primary source or the backup source of power. A maintenance ac source is also typically provided. In any case, the inverter output is synchronized with the ac source to allow automatic switching between sources in a few milliseconds. The 120-Vac vital instrument power system is typically unique between stations, consisting of inverters, static switches, bypass switches, associated switchgear and buses, and alternate ac power sources.

The ac power system specific description is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the facility.

2. System Function

The 120-Vac vital instrument power system is designed to supply very reliable power to the reactor protection system and other instrumentation and controls that are important-to-safety.

3. System Boundaries

The 120-Vac vital instrument power system interfaces with the dc power system and the essential ac power systems at the inverters and the alternate source transformers. It also interfaces with the reactor protection system and other instrumentation and control circuits.

B. See Section I, "Area of Review," of SRP-LR B.0.1 for Item I.B.

C. There are a variety of aging mechanisms that can affect the ability of the 120-Vac vital instrument power system to continue to operate safely and effectively. The typical 120-Vac vital instrument power system has alarms in the control room to alert the operators to a change in system status. Testing typically required by technical specifications for the 120-Vac vital instrument power system will detect long term system degradation. System redundancy limits the effects of rapid system failure.

Circuit breakers, switchgear, and relaying aging (and by extension static switch and bypass switch) aging is addressed in SRP-LR C.2.4.

The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA FOR THE 120-VAC VITAL INSTRUMENT POWER SYSTEMS

See Section II, "Acceptance Criteria," of SRP-LR B.0.1, Items II.A and II.B.

C. These criteria should be part of an established ongoing licensee program to assure the availability of the 120-Vac vital instrument power system. The licensee should also have a new periodic or a one-time inspection of components of the 120-Vac vital instrument power system in conjunction with the license renewal application.

D. Load study analysis should show that the instrument ac power system has the capacity and capability to power the connected load, including required reserves and margins, for any load configuration.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1, Items A through D.

E. Components of the instrument air system not specifically addressed in this section may be addressed in the generic aging topic reviews in Part C of the SRP-LR. Specifically the following sections of SRP-LR Part C are applicable to the instrument ac power system and should be reviewed: C.2.1 "Cable and Wiring", C.2.2 "Junctions", C.2.3 "Electrical Penetrations", C.2.4 "Relays, Circuit Breakers, and Switchgear", C.2.5 "Transformers", and all of C.3.0 "Instruments". This may require other staff input.

F. 1. The reviewer shall assure that the licensee has an established ongoing maintenance and surveillance program to assure the availability of the 120-Vac vital instrument power system. This program should include a mechanism to add new testing and evaluation criteria to monitor newly detected aging deterioration.

2. The reviewer shall assure that the licensee's load study analysis shows adequate capacity and capability, plus reserves and margins, to power any connected load configuration.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

Section 4.4 of IEEE Standard 279-1971 requires, for those plants issued a construction permit after 1971, type test data (or reasonable engineering extrapolation of type test data) that verifies this equipment meets, leave on a continuing basis the system performance requirements. Older plants may also be committed to IEEE Standard 279-1971.

Regulatory Guide 1.32 (Ref. 2) endorses IEEE Standard 308-1974 (Ref. 3). This industry standard recommends system capabilities and component capabilities for being able to start and operate the required loads during normal and post-accident conditions.

A. Inverters

The primary contributors to inverter failure are overheating, electrical transients, and personnel errors (Refs. 4, 5, and 6). Critical inverter components are stressed by overheating and voltage transients. Electrolytic capacitors, fuses, inductors, transformers, and semiconductors are susceptible to aging degradation that is accelerated by these stresses. Aging deterioration can be detected by component and equipment temperature monitoring, periodic observation of voltage waveform, and component (mostly capacitor) parameter measurements.

An excessively high input voltage (due to an excessively high equalization voltage) can damage an inverter. Damage can occur to fuses, capacitors, semiconductors, and other components. Excessive ac ripple on the dc input will do this also be creating additional heat losses in components.

All plants have some inverter maintenance. This maintenance, according to NUREG/CR-5051 (Ref. 5) has a wide range in level of maintenance. Thus, some units have only minimal maintenance performed. For example, general instructions for inspecting and cleaning the inverters at refueling outages is not an adequate maintenance program. A sufficient program as discussed in NUREG/CR-5051 will include cleaning to remove accumulated debris, dirt, and dust; inspecting cleanliness, electrical and mechanical connections, airflow and evidence of overheating; component replacement (especially for electrolytic capacitors); capacity tests and checks of internal temperature, cable meggering, and fan condition; and calibration of output voltage and frequency and metering instrumentation. The frequency of these tests varies between units. An adequate licensee maintenance and surveillance program for inverters will detect and mitigate the effects of inverter aging (Ref. 6).

VII. REFERENCES

1. Institute of Electrical and Electronics Engineers, IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations."

2. Regulatory Guide 1.32, "Criteria for Safety-related Electric Power Systems for Nuclear Power Plants."

3. Institute of Electrical and Electronics Engineers, IEEE Standard 308-1974, "Criteria for Class 1E Power Systems for Nuclear Power Generating Stations."

4. U.S. Nuclear Regulatory Commission, "Operating Experience and Aging - Seismic Assessment of Battery Chargers and Inverters," NUREG/CR-4564, June 1986.

5. U.S. Nuclear Regulatory Commission, "Detecting and Mitigating Battery Charger and Inverter Aging," NUREG/CR-5051, August 1988.

6. U.S. Nuclear Regulatory Commission, "Nuclear Plant Aging Research: the 1E Power System," NUREG/CR-5181, April 1990.

B.4.4 EMERGENCY DIESEL GENERATORS

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - EMEB

I. AREAS OF REVIEW

A. This section addresses the emergency diesel generators (EDG) system

1. Description

The basic EDG system typically consists of two or more diesel generator sets, each rated at 3,000 to 10,000 hp. For the purpose of this review, the basic EDG consists of the generator, block assembly, one or more heads, baseplate, and air induction and exhaust ducting. Supporting subsystems including the instrumentation and control subsystem, starting subsystem, cooling subsystem, fuel oil subsystem, and lubricating subsystem are covered in SRP-LR B.4.4.1, B.4.4.2, B.4.4.3, B.4.4.4, and B.4.4.5 respectively.

The EDG specific description is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the individual facility.

2. System Function

The EDG system supplies emergency electrical power to maintain cooling and other vital plant functions during loss of offsite power.

3. System Boundaries

The EDG system boundary includes the generator, engine block assembly, heads, baseplate, foundation, and air induction and exhaust systems.

B. See Section I, "Areas of Review," of SRP-LR B.0.1 for Item I.B.

C. EDG operating history includes a significant number of failures and performance degradations as a result of wear and aging as documented by the NRC's Nuclear Plant Aging Research Program (Refs. 1 and 2). The mitigation of identified aging mechanisms and the management of the aging process has also been documented (Refs. 1 and 2). The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

The NRC requires regular EDG testing and some licensees are required by their technical specifications to submit special reports documenting test failures. On-site data should be evaluated, especially for the listed systems which typically are the high failure rate systems within the diesel generator system boundary.

The licensee should evaluate data from EDG failures by reviewing failures and LERs for the station. Maintenance and operational records should be reviewed for repetitive failures and part replacement, failure root-cause analysis, and evidence of operational problems. Surveillance data, data monitoring, and trending data should be reviewed and analyzed. The licensee should evaluate this data for the diesel generator system, especially for the last 10-year period, and assign each failure or important data point to the appropriate EDG subsystem.

Diesel generators are generally very robust mechanisms which have been demonstrated in non-nuclear commercial service to be capable of many decades of reliable continuous service with only routine maintenance attention. When properly operated, maintained, monitored and tested, nuclear service EDGs should be capable of maintaining similar long-term reliability well past the initial licensing period. License-renewal attention should be focused on ensuring the absence of subtle accumulated damage from wear, stress, and metal fatigue which could unknowingly compromise established levels of reliability. The reviewer should not expect the licensee to perform mandatory engine teardowns for detailed inspection, because it has been shown by various studies that such teardowns are counterproductive from the reliability standpoint. Instead, it has been shown to be more appropriate for limited inspections to be performed to investigate abnormal behavior and suspicious trends indicated by engine and generator data.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1 for Items II.A and II.B.

C. The following additional criteria should be applied to one-time tests and engine condition reviews.

1. The EDG goal reliability has been met for the previous 10 years and all operating boundaries are currently within acceptable limits established by the manufacturer.

2. Engine crankshaft and generator alignment is within the manufacturer's recommendations.

3. Main bearing wear should not exceed the manufacturer's recommendation.
4. Fatigue cracking of connecting rod bearings should not exist.
5. No gear fatigue or excessive wear should be found.
6. Turbochargers should be free from signs of ingestion damage, fatigue cracking, and bearing damage.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for Items III.A through III.D.

E. The reviewer should ensure that the licensee has performed the following additional one-time tests and inspections. These one-time tests should be performed within 2 years of the date the application is submitted for license renewal.

1. Historical engine testing and monitoring data have been reviewed for the previous 10 years indicating that goal reliability criteria have been met and that all current engine operating parameters are within the engine manufacturer's recommended limits. Where goals and limits were exceeded, the licensee provided evidence of effective corrective actions.

2. A check was made of the diesel engine crankshaft, pedestal bearing, and generator alignment. If misalignment exceeded manufacturer's recommendations, the unit was realigned to the manufacturer's specifications.

3. If realignment was required or if engine oil analysis showed metallic particulates indicative of excessive bearing wear, the licensee inspected the main bearings and corrected any excessive wear condition.

4. The connecting rod-bearing journal subject to the highest torsional vibration stress was examined with fluorescent dye penetrant for fatigue cracks initiating in the area of highest stresses; that is, at the oil hole and fillets. If cracks were found, additional inspections were performed to determine the extent of damage. If the engine had a history of misfiring or cylinder exhaust temperature variations exceeding manufacturer's specifications, 25 percent to 50 percent of the connecting rod journals were examined for fatigue cracking.

5. The engine gears were inspected for signs of metal fatigue and excessive wear.

6. The turbochargers were inspected for fatigue cracking, bearing wear, and ingestion damage.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

The review should ensure that the applicant has an established effective program for diesel generator reliability and maintenance to address aging. The implementation of a reliability program may be briefly described in the application for license renewal. The licensee's response to the NRC's Generic Letter (90-xx), related to the resolution of Generic Issue B-56, is acceptable for this review.

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "Aging of Nuclear Station Diesel Generators: Evaluation of Operating Experience," Vol. 1, NUREG/CR-4590 (PNL-5832), August 1987.

2. U.S. Nuclear Regulatory Commission, "A Review of Information Useful for Managing Aging in Nuclear Power Plants," NUREG/CR-5562 (PNL-7323), June 1990.

B.4.4.1 EMERGENCY DIESEL GENERATOR INSTRUMENTATION AND CONTROL SUBSYSTEM

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - EMEB/SELB

I. AREAS OF REVIEW

A. This section addresses the emergency diesel generator (EDG) instrument and control (I&C) subsystem.

1. Description

Electrical cables, relays, circuit breakers, limit switches, indicators, and electronics associated with the emergency diesel generators that control starting, monitor operation, and connect the generator to the loads upon automatic or manual start signal.

The EDG I&C subsystem is described specifically in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the individual facility.

2. System Function

The function of the EDG I&C subsystem is to cause the EDG to start in response to a need for emergency electrical power, and to connect automatically or manually to the loads upon an automatic or manual start signal. Other functions are speed control, engine and system protection, and operator information.

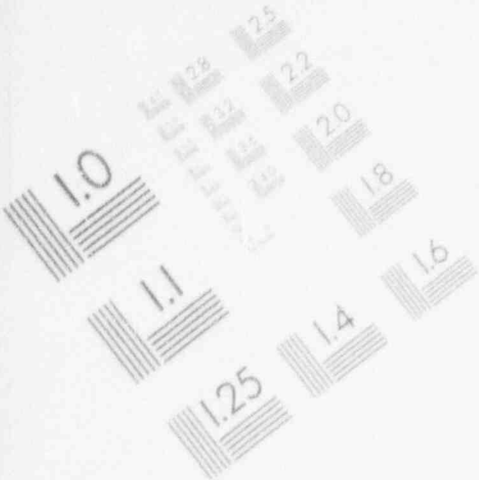
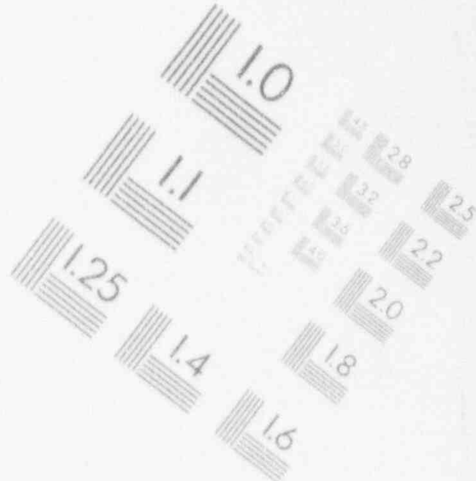
3. System Boundaries

The EDG I&C subsystem boundary consists of all electrical and electronic components associated with the EDGs up to and including the generator breakers and emergency signal input terminals. Also included in the EDG I&C subsystem are local pressure indicators and the governor; however, it does not include the emergency signal source(s) or associated wiring.

B. See Section I, "Areas of Review," of SRP-LR B.0.1 for Item I.B.

1

IMAGE EVALUATION TEST TARGET (MT-3)

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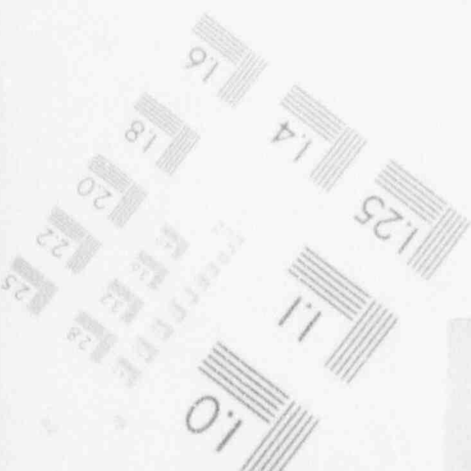
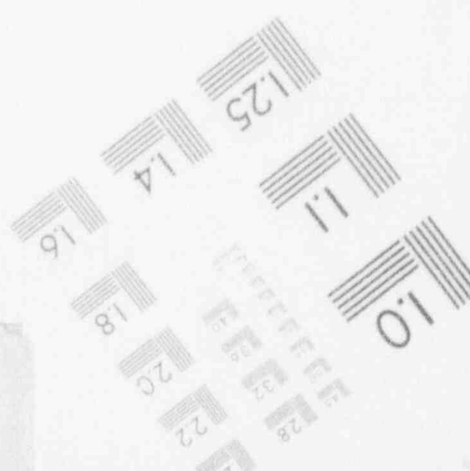
6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION
TEST TARGET (MT-3)

1.0 1.1 1.25 1.4 1.6 1.8 2.0 2.2 2.5 2.8 3.2 3.6 4.0

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

C. References 1-4 show that the instrument and control system is a high failure rate subsystem. Diesel generator aging research data indicated that about 30 of every 100 failures reported could be attributed to failures in the I&C equipment. High individual failure rate components are the governor, sensors, relays, wiring, terminations, and control air devices. Open relays hve often been prevented from operating correctly by dust. Other sensors and components are also environmentally sensitive. More modern components are less environmentally sensitive, as would be expected. The skid-mounted components for the diesel-generator control equipment are exposed to the engine vibration. Engine vibration is a major stressor for certain components including relays. Solid-state devices are not usually degraded by vibration to the same extent.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for Items III.A through III.B.

E. The licensee's maintenance program, evidence of operational problems, and test data should be reviewed for license renewal concerns for the I&C system. The reviewer should also consider the generic aging topics of SRP-LR C.3.0.

F. A general review by the licensee of the EDG control system is appropriate considering its importance, large failure rate, and the perceived obsolescence problem. The review need not address minor system components with acceptable failure histories nor equipment that has been recently upgraded and is expected to be serviceable for the entire license renewal period. The reviewer should ensure that the licensee has reviewed the EDG I&C system to determine:

- o That the expected service life of all major I&C system components is adequate to cover the requested license renewal period. List all components that do not meet this goal. Also list components for which spare parts availability is a problem or is projected to be a significant problem.

- o Note specific plans for I&C module replacement or other actions that are being proposed for the listed equipment.

- o EDG I&C cable is only subjected to a mild environment and ambient conditions. In general, EDG cable is not a concern and should not require replacement. Only visual inspection should be performed, and the results should be documented as part of the license renewal activities.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "Aging of Nuclear Station Diesel Generators: Evaluation of Operating Experience," Vol. 1, NUREG/CR-4590 (PNL-5832), August 1987.

2. U.S. Nuclear Regulatory Commission, "Aging of Nuclear Station Diesel Generators: Evaluation of Operating Experience, Workshop," Vol. II, NUREG/CR-4590 (PNL-5832), August 1987.

3. U.S. Nuclear Regulatory Commission, "Aging Mitigation and Improved Programs for Nuclear Service Diesel Generators," NUREG/CR-5057, December 1989.

4. U.S. Nuclear Regulatory Commission, "A Review of Information Useful for Managing Aging in Nuclear Power Plants," NUREG/CR-5562 (PNL-7323), June 1990.

B.4.4.2 EMERGENCY DIESEL GENERATOR AIR STARTING SUBSYSTEM

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - EMEB

I. AREAS OF REVIEW

A. This section addresses the emergency diesel generator (EDG) air starting subsystem which consists of:

1. Description

Air supply, valves, piping and air motors (if used) used to rotate the engine for starting.

The EDG air starting subsystem specific description is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the individual facility.

2. System Function

The system function is to rotate the EDG at a sufficient speed and within rated time to ensure starting in response to a need for emergency power.

3. System Boundaries

The EDG air starting subsystem consists of one or more compressors, intercoolers, aftercoolers, drain valves, storage tanks, piping, electrically operated valves and air motors (if used). It does not include electrical or electronic circuitry which originates the signal for the electrically operated air start valve.

B. See Section I, "Area of Review," of SRP-LR B.0.1 for item I.B.

C. Internal corrosion attack from condensed air moisture can occur in many air supply system components including compressors, intercoolers, aftercoolers, automatic drain valves, tankage, piping, and both refrigerated and desiccant driers. In addition, extended corrosive attack may occur as desiccant driers become degraded with trace quantities of lubricating oil in the air, deposited over many years, and allow moisture to enter critical starting system components such as starting air valves and air-motors. The effects of this corrosion attack are not easily seen during routine maintenance, but could result in the gradual loss of system reliability over decades of operation from wall-thinning, accelerated wear, and generation of particulates that could cause obstruction of critical flow paths.

Because of their obvious importance to the emergency starting capability of emergency diesel generators, starting air valves and starting motors should, and normally do, receive considerable specific attention in routine maintenance programs. Most electrical and air-powered starting system components will receive adequate attention during routine monitoring and maintenance programs to assure a high level of reliability of the starting system. Issues of concern relating to air-starting systems include: 1) the possible presence of subtle long-term degradation of air supply systems, air storage tanks, and piping as a result of internal corrosion which is difficult to detect during routine maintenance and 2) long-term wear and degradation of valves, moisture separators, and refrigerated (or desiccant) driers that can affect their reliability. The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1 for Items II.A and II.B.

C. The additional following criteria apply to the air start system:

1. The wall thickness of pipes, tanks and other important components should be the minimum code construction or manufacturers design thickness to withstand the design pressure plus the corrosion allowance for the remaining time period (remaining license time and the license renewal period). Generic topics in SRP-LR C.1.1, C.1.2, and C.1.5 also apply to this review.

2. Pipes and air storage tanks constructed of stainless steel should have no indications of stress corrosion, cracking, or other intergranular attack. A minimum of five percent of the weld areas should be sampled by NDE methods.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for Items III.A through III.B.

E. The air starting components not specifically addressed within this section may be addressed by the generic aging topic reviews in SRP-LR Part C. The reviewer should ensure that structures and components included as part of the generic SRP-LR are adequately reviewed for this system. These specific sections from SRP-LR Part C should be reviewed: C.1.1 "Piping"; C.1.2 "Valves"; and C.1.5 "Tanks and Vessels." This may require other staff input.

F. The reviewer should confirm that the licensee has investigated the wall thickness and condition of piping and air storage tanks, if not already a part of the ongoing maintenance and reliability program for the EDG air starting system. The licensee should present evidence and analysis of the following:

1. Approximately five percent of the sensitive locations of piping, tanks, valves and other pressure retaining components should be examined by NDT methods for wall thinning. Locations where water or moisture tend to collect should be included in this testing, as well as intercoolers and aftercoolers.

2. Pressure retaining components constructed of stainless steel should be sampled for stress corrosion cracking. Five percent of the sensitive locations is adequate.

3. Any pipe or tank failure in the LER records for the station due to vibration or other cause should have the same location examined for incipient failures at the time of the license renewal application. An incipient failure should be followed by immediate corrective action. Repeated failures are evidence of incorrect root cause analysis and/or corrective action.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "Aging of Nuclear Station Diesel Generators: Evaluation of Operating Experience," Volume 1, NUREG/CR-4590, (PNL-5832), August 1987.

2. U.S. Nuclear Regulatory Commission, "Aging of Nuclear Station Diesel Generators: Evaluation of Operating Experience, Workshop," Volume II, NUREG/CR-4590, (PNL-5832), August 1987.

3. U.S. Nuclear Regulatory Commission, "Aging Mitigation and Improved Programs for Nuclear Service Diesel Generators," NUREG/CR-5057, December 1989.

4. U.S. Nuclear Regulatory Commission, "A Review of Information Useful for Managing Aging in Nuclear Power Plants," NUREG/CR-5562, (PNL-7323), June 1990.

B.4.4.3 EMERGENCY DIESEL GENERATOR COOLING SUBSYSTEM

REVIEW RESPONSIBILITIES

Primary - LRPD

Secondary - EMEB

I. AREAS OF REVIEW

A. This section addresses the emergency diesel generator (EDG) cooling subsystem.

1. Description

A circulating water system including engine-driven or motor-driven pumps, a heat exchanger, and interconnected piping that circulates water treated with corrosion inhibitors and biocides through the engine cooling jackets and oil cooler. The heat absorbed by the water in the cooling jackets and oil cooler is either rejected to the plant cooling water system by the heat exchanger or it is released to the atmosphere by the engine radiator.

The EDG cooling subsystem is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the specific facility.

2. System Function

The system serves to keep the engine oil and component temperatures within operating limits by transferring combustion and friction heat lost in the engine to the cooling water.

3. System Boundaries

The EDG cooling subsystem includes the circulating pump (engine-driven or motor-driven), cooling water heat exchanger (or radiator), and interconnecting piping. It does not include plant cooling water system piping or the engine jackets (included in basic EDG system).

B. See Section I, "Areas of Review," of SRP-LR B.0.1 for Item I.B.

C. This system is subject to long-term corrosion, fouling of heat exchanger surfaces, and degradation of seals and gaskets which can compromise its function to provide reliable cooling to the emergency diesel generator.

Accumulated degradation may result in reduced diesel generator operating efficiency and reliability in the cooling system heat exchangers. In these heat exchangers, the heat from the engine jackets and oil cooler is rejected to the plant cooling water system. These heat exchangers are subject to corrosion, fouling, and silting principally on the plant cooling water side (the recirculating water is generally treated to reduce corrosion and deposition). This corrosion can result in the leakage of untreated water into the engine cooling systems. Fouling of the heat exchanger tubing can interfere with the heat transfer efficiency of the cooling system.

The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1 for Items II.A and II.B

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for Items III. A through III.D

E. The cooling subsystem components not specifically addressed within this section may be addressed by the generic aging topic reviews in SRP-LR Part C. The reviewer should ensure that structures and components included as part of the generic SRP-LR topics are adequately reviewed for this system. These specific sections from SRP-LR Part C should be reviewed: C.1.1, "Piping"; C.1.2, "Valves"; C.1.3, "Pumps"; and C.1.6, "Equipment and Component Supports." s may require additional staff input.

F. The reviewer should confirm that the licensee has investigated and provided evidence that heat exchangers or radiators are in good condition and free from long-term deterioration that would make them less reliable during the license renewal period. This evidence may come from ongoing programs or from a one-time investigation when preparing for license renewal. The potential problems to be reviewed include wall thinning of tubes and radiators and crud deposits on the secondary side of heat exchangers.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

VII. REFERENCES

B.4.4.4 EMERGENCY DIESEL GENERATOR FUEL OIL SUBSYSTEM

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - EMEB

I. AREAS OF REVIEW

A. This section addresses the emergency diesel generator (EDG) fuel oil subsystem.

1. Description

The EDG fuel oil subsystem comprises fuel oil tanks, pipes, fuel pumps, and filters that supply fuel to the EDGs.

The EDG fuel oil subsystem is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the specific facility.

2. System Function

The system supplies fuel oil to the EDGs during operation and testing.

3. System Boundaries

The EDG fuel oil subsystem includes all fuel storage tanks, above ground and underground fuel piping, fuel oil transfer pumps, and day tanks.

B. See Section I, "Areas of Review," of SRP-LR B.0.1 for Item I.B.

C. The principal license renewal concern is the long-term deterioration of the piping, fuel oil tanks, and any buried piping. The risk of undetected deterioration of the tanks and piping is judged to be high enough to warrant a more complete inspection at license renewal.

Corrosion and corrosion products have been found in the fuel oil tanks and system. Failures of cathodic protection systems have been observed, as well as fractured buried oil transfer lines caused by earth movement problems. However, most fuel oil leakage has been caused by engine vibrations that loosen fittings. The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1 for Items II.A and II.B.

C. Piping and tank wall thickness for the EDG fuel oil subsystem should not be corroded from the inside or the outside to less than the minimum wall thickness for design plus a corrosion allowance for the remaining operational and license renewal period. Ten percent of sensitive areas should be sampled for wall thickness measurements. The criteria of SRP-LR C.1.0 and C.1.5 also apply.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for Items III.A through III.D.

E. The fuel oil components not specifically addressed within this section may be addressed by the generic aging topic reviews in SRP-LR Part C. The reviewer should ensure that structures and components included as part of the generic SRP-LR topics are adequately reviewed for this system. These specific sections from SRP-LR Part C should be reviewed: C.1.1, "Piping"; C.1.2, "Valves"; C.1.3, "Pumps"; and C.1.6, "Equipment and Component Supports." This may require additional staff input.

F. Piping and tank wall thickness measurements should be obtained by a sampling technique as part of the license renewal process, unless these are already part of the surveillance program and it should be shown that these wall thickness are not below code specifications or manufacturer's recommendation. Tank walls should be inspected for local corrosion and pitting and degraded coatings. Wall pitting and coating degradation should be repaired by approved procedures.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

B.4.4.5 EMERGENCY DIESEL GENERATOR LUBRICATING OIL SUBSYSTEM

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - EMEB

I. AREAS OF REVIEW

A. This section addresses the emergency diesel generator (EDG) lubricating oil subsystem.

1. Description

The EDG lubricating oil subsystem comprises lubricating oil storage tanks, pipes, pumps, valves and heat exchangers that supply pressurized and cooled lubricating oil to the EDG including the turbocharger or supercharger. This subsystem also includes the "pre-lube" system consisting of pumps, piping, valves, and heaters that supply warm lubricating oil to the engine during starting and standby conditions.

The EDG lube oil subsystem is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the specific facility.

2. System Function

The EDG lubricating oil subsystem provides oil of proper temperature, pressure, and viscosity to the EDG during starting and normal operation for the purpose of cooling and lubricating internal EDG components.

3. System Boundaries

The EDG lubricating oil subsystem includes all components of the lube oil and pre-lube subsystems external to the EDG block, head(s), baseplate, and generator.

B. See Section I, "Areas of Review," of SRP-LR B.0.1 for Item I.B.

C. The engine lubrication system is one of the higher failure rate systems in the typical licensee event reports (LER) of plant failures. The chief failures occur in pumps, heat exchangers, regulating valves and piping. Oil performance has also been a problem, but is outside of the scope of license renewal concerns.
Typically, the lubrication oil system is well monitored for aging effects through the provisions of NRC guidelines. Except for heat exchanger wall thinning or pitting due to corrosion on the water side, no special license renewal activities or inspections appear to be needed.

A review of the LER reported failures should be compared to the plant maintenance and reliability program activities to ensure that aging concerns were adequately addressed for the lubrication system.

The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1 for Items II.A and II.B

C. In addition to criteria listed above, the licensee should have provided test data to show that heat exchanger wall thickness is no less than 40 percent of the original value. Alternatively, by suitable calculations, the licensee may have demonstrated that this thickness is adequate.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for Items III.A through III.D

E. The lubricating oil components not specifically addressed within this section may be addressed by the generic aging topic reviews in SRP-LR Part C. The reviewer should ensure that structures and components included as part of the generic SRP-LR topics are adequately reviewed for this system. These specific sections from SRP-LR Part C should be reviewed: C.1.1, "Piping"; C.1.2, "Valves"; and C.1.3, "Pumps." This may require additional staff input.

F. The reviewer should ensure that the wall thinning and pitting of the lube-oil cooler was checked for license renewal, or this cooler was replaced within the 40-year license period or at license renewal. The reviewer should be aware that the typical wall thickness of lubricating oil coolers is conservative for corrosion allowance. A minimum of 40 percent remaining wall thickness is usually adequate. The reviewer should also consider the generic topics of SRP-LR C.1.0.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

B.4.5 PLANT ESSENTIAL LIGHTING SYSTEM

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SICB/SELB

I. AREAS OF REVIEW

A. This section addresses the plant essential lighting system.

1. Description

The normal and emergency or essential lighting systems must be operable during loss of offsite power. The plant lighting system is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the specific facility.

2. System Function

The normal lighting system provides lighting during all plant operating conditions; while the essential lighting system, in addition to functioning as a normal lighting system, must also provide lighting under fire, transient and accident conditions.

3. System Boundaries

The plant lighting system comprises motor control centers, transformers, circuit breakers, wires, lighting devices, lighting fixtures, control switches, and enclosures.

B. See Section I, "Areas of Review," of B.0.1 for item I.B.

C. The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

The Illuminating Engineering Society (Ref.1) through its Lighting Handbook as related to systems design and illumination levels sets the standards for the essential lighting system.

VII. REFERENCE

1. Illuminating Engineering Society, Lighting Handbook (latest edition).

B.4.6 PLANT COMPUTER TO BE PROVIDED LATER

B.4.7 SWITCHYARD

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SELB/SICB

I. AREAS OF REVIEW

A. This section addresses the switchyard.

1. Description

The main switchyard with its breakers, transmission lines, dc control power, and protective relays is part of the preferred power supply providing offsite power to the facility. The switchyard is designed to provide a reliable source of auxiliary power from two independent transmission circuits to the plant power distribution system for startup, operation, and shutdown of the plant. It is also designed to be available within a few seconds following a design-basis accident to ensure that vital safety functions are maintained. The power is typically routed to the plant distribution system via a configuration of breakers, transformers, and transmission lines. The dc control power system supplies control power to the high-voltage circuit breakers from independent dedicated batteries. The review of this section should be limited to the equipment required to deliver the power from offsite to the power distribution system. If the switchyard is not the sole source of power to the distribution system and does not tie directly to a vital Class 1E system, the requirements of this section may not be applicable. The switchyard is described in the most recent update of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the specific facility.

2. System Function

The switchyard supplies a reliable source of auxiliary power from independent transmission circuits to the plant power distribution system for startup, operation, and shutdown of the plant.

3. System Boundaries

The switchyard includes the breakers, transmission lines, batteries, housings, and support equipment (towers, arrestors, etc.) that are necessary to provide electrical power to the plant electrical distribution system that is required for operating the plant. The system boundary begins with the transmission lines providing power to the switchyard and ends at the transmission lines leaving the switchyard to provide power to the electrical busses in the plant. The system does not include any equipment not directly associated with providing power to the plant electrical distribution system (such as connecting breakers in the system that only provide supplementary distribution of generated power) or the transformers located in between the switchyard and the internal busses.

B. See Section I, "Areas of Review," of SRP-LR B.0.1 for Item I.B.

C. The offsite power system contains some components that are reviewed in accordance with other sections of the SRP-LR. These components and the applicable SRP-LR section are:

o Cable and wiring, C.2.1
o Circuit breakers, C.2.4

In addition, protective relaying is reviewed in accordance with SRP-LR B.4.1.1.

Any aging degradation of these components can potentially affect their ability to function as required. However, aging degradation of some of these components is of particular concern to the offsite power system because the affected component must function at the time of an event in contrast to components that are only required to continue to function or provide additional power. Some of these are noted below:

1. Circuit breakers that must disconnect a failed power source, such as a fault in the switchyard, loss of onsite generated power, or total loss of offsite power. Circuit breaker aging stressors may be categorized as either thermal, electrical, mechanical, or environmental. The most likely failures are related to mechanical and electrical effects and primarily result in failure to open or close, improper operation, restrike, shorting, and arcing. A summary of these effects is given in Table B.4.7-1.

2. Protective relaying and control.

3. DC batteries that must provide control power to the breakers. The dc control power is dependent upon batteries dedicated to providing this control power. EPRI EL-5885 (Ref. 2), discusses mechanisms and causes of failures in batteries, all of which are directly age related.

4. Supports, towers, and any equipment necessary to maintain the integrity of the incoming power.

Typical examples of age-related degradation associated with plant computers are provided in this section. The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1.

E. Components of the switchyard system not specifically addressed in this section may be addressed in the generic aging topic reviews in SRP-LR Part C. Specifically, the following sections of SRP-LR Part C are applicable to the switchyard and should be reviewed: C.2.1 "Cable and Wiring," and C.2.4 "Relays, Circuit Breakers, and Switchgear." This may require additional staff input.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

As addressed in the SRP-LR, Section 4.2.2, studies performed as a part of the NRC Nuclear Plant Aging Research (NPAR) Program have shown that of the Class 1E power system components, circuit breakers have the highest frequency of LER events (66.3 percent of all 1E power system events), followed by transformers (4.7 percent of all 1E power system events). Studies utilizing Nuclear Plant Reliability Data Systems (NPRDS) information have shown that about 21 percent of circuit breakers failures are aging related (Refs. 1 and 2). Circuit breakers, therefore, have the highest failure rate and are also affected the most by aging-related degradation. Though the breakers used in the switchyard are not the same as those in the plant electrical distribution system, they are subject to many of the same aging factors, as well as being directly exposed to the environment.

Support equipment such as lightning arrestors and support towers needs to maintain its environmental qualification in order to maintain the integrity of the switchyard in events such as severe thunderstorms, hurricanes or earthquakes.

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "Nuclear Plant Aging Research: The 1E Power System", L.C. Meyer, J.L. Edson, NUREG/CR-5181, April 1990.

2. Electric Power Research Institute, "Generic Guidelines for the Life Extension of Plant Electrical Equipment," EPRI EL-5885, July 1988.

Table B.4.7-1 Aging Degradation of Circuit Breakers

Stressor	Cause	Failure Mechanism	Failure Mode
Thermal	Poor contact	Degraded insulation	Short to ground
	Large current	Degraded contacts	Poor or open contacts
		Degraded arc chutes	Flash over
		Degraded overload mechanism (molded-case)	Flash to extinguish the arc
			Premature trip at low current
Electrical	Overvoltage transients	Arcing of contacts causing contamination of components	Restrike
	Spikes		Shorting of components
	Fault interruption		Arcing to ground or between phases
	Lightning		
Mechanical	Routine operation	Degraded contacts	Failure to open or close
		Fatigue	
	Fault interruptions		Improper operation
		Wear	
	Vibration	Loose connections	
	Friction	Reduced force	
		Compound failure	

Table B.4.7-1 Aging Degradation of Circuit Breakers

Stressor	Cause	Failure Mechanism	Failure Mode
Environmental	Elevated temperature	Increased friction	Failure to open or close
	Elevated humidity	Degraded insulation	Shorting and arcing
	Dirt	Oxidation	Improper operation
	Chemicals	Hardening of lubricant	
	Rust	Embrittlement of materials	

B.4.7.1 DC CONTROL POWER SYSTEM

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SELB/SICB

I. AREA OF REVIEW

A. This section addresses the dc control power system

1. Description

The switchyard dc control power system includes those dc power sources and their distribution systems and vital supporting systems required to supply control power to breakers and switchgear in the switchyard. The system utilizes batteries that are kept charged by batteries chargers. The chargers are capable of providing the required power during normal operations when ac power is available. Upon the loss of ac power, the batteries provide the power needed to control the switchyard breakers and switches. Some nuclear plant utilize dc power from the instrumentation and control power system, described in SRP-LR B.4.3.1, "DC Power System"; others utilize a separate system dedicated to the switchyard. The dc control power system is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the specific facility.

2. System Function

The switchyard dc control power system supplies power to the switchyard breakers and switchgear to provide for control during normal operations when ac power is available and during abnormal conditions when ac power is not available.

3. System Boundaries

The switchyard dc control power system includes the batteries, battery chargers, cabling, breakers and switches, and control circuits. The system does not include the ac power sources, the ac distribution system, or the connected dc loads. This system boundary includes portions of the instrumentation and control power system, described in SRP-LR B.4.3.1, "DC Power System", for those plants that utilize the dc power system of the instrumentation and control power system to provide switchyard control power.

B. See Section I, "Areas of Review," of SRP-LR B.0.1 for Item I.B.

C. The switchyard dc control power system is composed primarily of components that are reviewed in accordance with other sections of the SRP-LR. These components and the applicable SRP-LR section are:

o Cable and wiring, C.2.1
o Junctions, C.2.2
o Relays, switchgear, circuit breakers, C.2.4

In addition, protective relaying and controls is reviewed in accordance with SRP-LR B.4.1.1.

A variety of aging mechanisms can affect the ability of the dc control power system to continue to operate safely and effectively. Typical xamples of age-related degradation associated with the dc control power system are provided in this section. The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

The dc Control Power System can be degraded by degradation of the batteries, battery chargers, or bus systems. Batteries are affected by thermally induced grid and connector oxidation, plate and grid swelling, container and cover cracking, separator deterioration, or specific gravity changes (Ref. 1). Battery chargers are affected by the aging of electrolytic capacitors, transformers, inductors, solid-state devices, and fuses. The battery chargers are also affected by the quality of the connected source power (Refs. 2 and 3). Bus systems age with dielectric stress or partial discharge, cooling system degradation, and bus heating (Ref. 4).

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1.

E. Components of the dc control power system not specifically addressed in this section may be addressed in the generic aging topic reviews in SRP-LR Part C. Specifically the following sections of SRP-LR Part C are applicable to the dc contol power system and should be reviewed: C.2.1 "Cable and Wiring," C.2.2 "Junctions," and C.2.4 "Relays, Circuit Breakers, and Switchgear." This may require additional staff input.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

Components of the switchyard dc control power system do experience aging degradation. Studies conducted within the framework of the Nuclear Regulatory Commission's Nuclear Plant Aging Research (NPAR) Program have investigated the aging effect on the various components (Refs. 1-4). The following material discus the results of those studies.

A. Batteries

NUREG/CR-4457 is an evaluation of the aging effects of safety-related batteries. It also evaluates maintenance, testing, and monitoring practices and discusses the effectiveness of these programs. The most significant aging factor is thermally induced oxidation of grids and top conductors. This oxidation causes the plates and grids to swell, resulting in poor conduction between the plate and grid. This reduces battery capacity and causes stresses in the container and covers. The stresses can result in cracks in the battery case. Separator deterioration is a cause of cell shorting. Considerable care is needed to maintain batteries in an operable condition. Maintenance practices that conform to IEEE Standard 450-1975 (Ref. 5) and Regulatory Guide 1.129 ensure reliable battery capacity over its qualified life. Battery maintenance allows a history of the battery to be documented, and can be used to evaluate the state of the battery. Recommended battery maintenance includes battery capacity tests and recommended battery replacement should the battery capacity be less than 80 percent of the manufacturer's rating, among other criteria. These other factors include service tests, the original sizing criteria utilized, and the capacity available compared to the load requirements. Other test methods may be developed in the future to better analyze degradation. Physical conditions of individual cells (such as plate condition, cell reversal, specific gravity readings, or electrolyte contamination) often require the replacement of individual cells. A battery nearing the end of its life should not have individual cells replaced.

B. Battery Chargers

NUREG/CR-5051 (Ref. 3) is an evaluation of the aging effects of safety-related battery chargers.

Electrolytic capacitors, transformers, inductors, and silicon controlled rectifiers (SCRs) are the battery charger components that are most susceptible to aging degradation. Fuse failures, caused primarily by thermal fatigue, can also contribute to battery charger failure, but are difficult to detect until they fail. The life of transformers and inductors is determined by the condition of the insulation. Electrolytic capacitors age as a result of the deterioration or outgassing of the electrolyte. SCRs are subject to voltage transients and other semiconductor stress (usually thermal) failures. Operation of the battery chargers during diesel generator testing can stress battery chargers because of voltage and frequency variations. Excessively high equalization charge voltage can damage system loads and cause fuses to open. Some fuses may fail undetected. Excessive ac ripple voltage on the dc output will do this too. In addition, excessive ripple will cause premature aging of the battery. Electrical transients, such as could be caused by a noisy setpoint potentiometer, can cause the same premature aging.

All plants have some battery charger maintenance and capacity testing. It varies from capacity tests to inspection and calibration to cleaning, adjustment, component replacement, connection tightness verification, and component mounting torque verification. Some plants follow the battery charger internal temperature. The internal temperature is known to have an effect on component aging. Increased internal temperatures decrease the component life.

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "Aging of Class 1E Batteries in Safety Systems of Nuclear Power Plants," J. L. Edson and J. E. Hardin, NUREG/CR-4457, July 1987.

2. U.S. Nuclear Regulatory Commission, "Operating Experience and Aging - Seismic Assessment of Battery Chargers and Inverters," W, E. Gunther, R. Lewis, and M. Subudhi, NUREG/CR-4564, June 1986.

3. U.S. Nuclear Regulatory Commission, "Detecting and Mitigating Battery Charger and Inverter Aging," W. E. Gunther, R. Lewis, and M. Subudhi, NUREG/CR-5051, August 1988.

4. U.S. Nuclear Regulatory Commission, "Nuclear Plant Aging Research: the 1E Power System," L. C. Meyer and J. L. Edson, NUREG/CR-5181, April 1990.

5. Institute of Electrical and Electronics Engineers, IEEE Standard 450-1975, "Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

6. U.S. Nuclear Regulatory Commission, "Aging Evaluation of Class 1E Batteries: Seismic Testing," J. L. Edson, NUREG/CR-5448, July 1990.

7. Institute of Electrical and Electronics Engineers, IEEE Standard 535-1979, "IEEE Standard for Qualification of Class 1E Lead Storage Batteries for Nuclear Power Generating Stations."

B 4.8 INFORMATION SYSTEMS

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SICB

I. AREAS OF REVIEW

A. This section addresses the information systems that are important to license renewal.

1. Description

The information systems important to license renewal consist of the safety parameter display system (SPDS) and the Regulatory Guide (RG) 1.97 displays (Ref. 1).

a. The SPDS is usually a part of a centralized computer-based information and display system and, for the most part, consists of electronic isolation devices, fiber-optic cables, multiplexers, input/output modules, and analog and/or digital displays.

b. The RG 1.97 display system can be a combination of both an analog and a digital information and display system. The system may have dedicated sensors or it may extract its input signal from existing data channels through the use of electronic isolation devices. The RG 1.97 display system will have both analog and digital displays.

The information systems is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the specific facility.

2. System Function

a. The SPDS provides a display of critical plant variables to the control room operators to aid them in rapidly and reliably determining the safety status of the plant, even though the system in not required to operate during or after an accident. The SPDS enhances operator ability to comprehend plant conditions and interact in situations that may require human intervention.

b. RG 1.97 displays indications of plant variables which are needed by the control room operators during and after accident situations to determine (1) the operating status of plant safety systems and other systems important to safety including those that give the operating status of the reactor and (2) the potential for the release of radioactive materials to the environment.

3. System Boundaries

a. The SPDS will consist of electronic isolation devices, fiber-optic cables, multiplexers, input/output modules, analog and/or digital displays, power sources, racks, and panels.

b. The RG 1.97 display system will consist of sensors, electronic isolation devices, fiber-optic cables, input/output modules, analog and/or digital displays, power sources, racks, and panels.

B. See Section I, "Areas of Review," of SRP-LR B.0.1 for Item I.B.

C. The information systems and the electronic devices used in them are or can be subjected to the aging characteristics of their individual components and therefore, the maintenance or replacement schedules should include considerations of the specific aging characteristics of the component materials (see SRP-LR C.3.2).

The components of both systems which are susceptible to aging and/or age degradation include both analog and digital integrated circuits, resistors, capacitors, wiring, terminal blocks, semiconductors, potentiometers, and PC boards. Typical examples of age-related degradation associated with the information systems are provided in this section. The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for Items III.A through III.D.

E. Components of the information system not specifically addressed in this section may be addressed in the generic aging topic reviews in Part C of the SRP-LR. Specifically, Section C.3.2, "Electronic Devices," of Part C is applicable to the information system and should be reviewed. This may require additional staff input.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

VII. REFERENCE

1. U.S. Nuclear Regulatory Commission, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Regulatory Guide 1.97, May 1983, Revision 3.

B.5.1 OFFGAS SYSTEM (BWR)

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SPLB

I. AREAS OF REVIEW

A. This section addresses the offgas system (OGS).

1. Description

The major components of the OGS are the main condenser air ejector, offgas pre-cooler and after condenser to remove water vapors from the gas stream, catalytic recombiners to reduce concentration of detonable hydrogen, and the gas holdup piping for decay of radioactive gases. The system also includes HEPA prefilters and charcoal adsorber filters along with an electric preheater.

The offgas system is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the specific facility.

2. System Function

The OGS performs the following functions:

- o Reduces radioactive releases below established limits to protect the public.
- o Supports the main condenser function by removing and providing a safe path for processing radioactive noncondensibles and other noncondensibles.
- o Provides recombination of radiolytic hydrogen and oxygen to prevent detonation and subsequent equipment damage.
- o Collects and processes miscellaneous contaminated gas streams from other plant processes.

3. System Boundaries

The OGS extends from the main condenser to the offgas stack. The radioactive gases, including hydrogen and oxygen, generated in the reactor and carried over to the main condenser are collected, processed, and held up for radioactive decay by the OGS before discharging to the environment. Besides controlling the release of radioactive material, the offgas system is provided with necessary equipment for recombining hydrogen with oxygen to prevent hydrogen detonation.

The main components of the OGS are:

a. Steam jet air ejector

b. Holdup (delay) pipe for radioactive decay

c. Hydrogen recombiner, moisture condenser

d. Hydrogen analyzer

e. Gas compressor

f. Offgas HEPA filters, including pre-filters, pre-heaters, and charcoal delay bed

The OGS radiation monitors are part of the process radiation monitoring system which is addressed in SRP-LR B.5.2.

B. See Section I, "Areas of Review," of SRP-LR B.0.1 for Item I.B.

C. The OGS is prone to the same problems and failures as most ventilation systems. The presently recognized problems of bearing wear and replacement, fan blade cracking and repair/replacement, motor failure, heating and/or cooling coil failures, and instrumentation and control failures are expected to continue throughout the life of the plant, including the license renewal period (Ref. 1).

The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures, " of SRP-LR B.0.1 for items III.A through III.D.

E. Components of the OGS not specifically addressed in this section may be addressed by the generic aging topic reviews in SRP-LR Part C. The reviewer should ensure that structures and components included as part of the generic SRP-LR topics are adequately reviewed for this system. These specific sections from SRP-LR Part C should be reviewed: C.1.1, "Piping"; C.1.4, "Heat Exchangers"; C.1.6, "Equipment and Component Supports"; C.2.1, "Cable and Wiring"; C.2.3, "Electrical Penetrations"; C.2.4, "Relays, Circuit Breakers and Switchgear"; C.2.7, "Electrical Motors", and all of section C.3.0, "Instrument". This may require additional staff input.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

VII. REFERENCE

1. U.S. Nuclear Regulatory Commission, "A Review of Information Useful for Managing Aging in Nuclear Power Plants," NUREG/CR-5562 (PNL-7323), 1990.

B.5.2 RADIATION MONITORING SYSTEM

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SICB

I. AREAS OF REVIEW

A. This section addresses the radiation monitoring system (RMS).

1. Description

The sub-systems of the RMS are hard-wired stand-alone systems that interface with other information and control systems and the plant's control room. The RMS contains all of the equipment necessary to perform its functions with the actual descriptions of the subsystems contained in the plant-specific FSAR.

The radiation monitoring system is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the specific facility.

2. System Function

The RMS provides indication of direct radiation levels within and outside plant areas, radioactive concentrations in liquid effluent, and continuous sampling of plant gaseous effluent for releases of radioactive iodides and particulate during normal operation and in the event of an accident. The RMS consists of three major sub-systems which perform the functions of monitoring radiation levels of (1) the working and storage areas within the facility (area monitors), (2) the process fluid flows that may discharge radioactive materials (process monitors), and (3) the atmospheric environment surrounding the facility (environmental monitors).

a. The plant area monitors survey the gamma radiation within the facility and warn of any excessive level changes. The system provides operating personnel with a record of gamma radiation levels within the facility.

b. The process monitors survey the radiation levels of liquid and gaseous processes throughout the facility. The system assists in controlling the release of radioactive byproducts within set limits and provides for personnel safety by warning of abnormal radiation levels. It also provides operating personnel with a record of radiation levels.

c. The environmental monitors measure and establish natural background and other radiation levels, determine the facility's contribution to the environmental radiation levels, and assist the facility in complying with public health and safety regulations. The monitoring stations are located in areas selected to obtain representative samples of the atmosphere, fallout, vegetation, water of the plant environs, and radioactivity of the atmosphere.

3. System Boundaries

The review of the RMS is limited to that equipment and interfaces as shown on the plant P&IDs as they relate to the RMS.

B. See Section I, "Areas of Review," of SRP-LR B.0.1 for Item I.B.

C. The RMS is subject to a wide variety of age-related mechanisms that can affect the ability of the system and its components to operate reliably. Many of the components are discussed further in SRP-LR C.3.2.

Typical examples of age-related degradation associated with the RMS are provided in this section. The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for items III.A through III.D.

E. Components of the RMS not specifically addressed in this section may be addressed in the generic aging topic reviews in Part C of the SRP-LR. Specifically, Section C.3.2 "Electronic Components," of Part C is applicable to the RMS and should be reviewed: This may require additional staff input.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

VII. REFERENCES

B.5.3 COMPONENT COOLING WATER SYSTEMS

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SPLB/EMCB

I. AREAS OF REVIEW

A. This section addresses the component cooling water systems.

1. Description

The component cooling water (CCW) systems are those closed-loop systems required for safe shutdown during normal operation, transient, and accident conditions. CCW systems are closed-loop cooling systems that transfer the heat from vital loads to the service water systems.

The loads cooled or supplied by CCW systems vary from plant to plant. Typical safety-related loads are decay heat or residual heat removal coolers, containment building coolers, and injection pump coolers.

The component cooling water system is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the specific facility.

2. System Function

The primary function of the CCW systems is to provide a closed loop cooling system as an additional barrier from potentially contaminated reactor systems.

3. System Boundaries

The boundaries of each CCW system include the head tank, pumps, heat exchangers, piping, and isolation valves. The electrical supply and makeup water supply are not considered part of the CCW system boundary. The systems are different from plant to plant, but in general, a CCW system includes all of the equipment in the system recirculation flow path.

B. See Section I, "Areas of Review," of SRP-LR B.0.1 for Item I.B.

C. The CCW age-related degradation mechanisms include those common to pumps, heat exchangers, vessels, valves, and piping as discussed in SRP-LR Part C. The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. <u>ACCEPTANCE CRITERIA</u>

See Section II, "Acceptance Criteria," of SRP-LR B.0.1.

III. <u>REVIEW PROCEDURES</u>

See Section III, "Review Procedures," of SRP-LR B.0.1 for Items III.A through III.D.

E. The CCW components not specifically addressed in this section may be addressed by the generic aging topic reviews in SRP-LR Part C. The reviewer should ensure that structures and components included as part of the generic SRP-LR topics are adequately reviewed for this system. These specific sections from SRP-LR Part C should be reviewed: C.1.1, "Piping"; C.1.2, "Valves"; C.1.3, "Pumps"; C.1.4, "Heat Exchangers"; and all of C.2.0 , "Electrical." This may require additional staff input.

F.

1. The reviewer should verify that the licensee has evaluated of the CCW operating parameters that are monitored and alarmed to ensure that these parameters represent the best choices as indicators for incipient failures.

2. Established a program to closely monitor components of the CCW systems, including such passive components as heat exchangers and piping.

IV. <u>FINDINGS</u>

See Section IV, "Findings," of SRP-LR B.0.1.

V. <u>IMPLEMENTATION</u>

See Section V, "Implementation," of SRP-LR B.0.1.

VI. <u>GENERAL INFORMATION</u>

VII. <u>REFERENCE</u>

1. U.S. Nuclear Regulatory Commission, "A Review of Information Useful for Managing Aging in Nuclear Power Plants," NUREG/CR-5562 (PNL-7323), 1990.

B.5.4 SERVICE WATER SYSTEM

REVIEW RESPONSIBILITIES
Primary - LRPD
Secondary - SPLB/EMCB

I. AREA OF REVIEW

A. This section addresses the station service water system (SWS).

1. Description

The service water system (SWS) typically includes two subsystems. These are: essential (or standby) service water (ESW) and nonessential service water. The latter is not safety related and may not be addressed in the IPA for license renewal. Although plant-specific ESW systems are subject to wide variations in design details, the basic functional aspects during essential system operation are quite similar. During conditions requiring essential system operations, the SWS takes cooling water from the ultimate heat sink, circulates it through the plant piping system to equipment heat exchangers to control the temperatures of reactor coolant and critical components, and returns the cooling water to the ultimate heat sink.

The station service water system is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the facility.

2. System Function

The ESW system consists of those coolers and heat exchangers that are required to operate for a safe reactor shutdown or to mitigate the consequences of postulated accidents. The ESW system circulates water from the ultimate heat sink through equipment and closed cooling water system heat exchangers to accomplish this goal.

3. System Boundaries

The ESW system extends from the inlet cooling water source (the ultimate heat sink) through any screens or filters, the SWS pumps, the equipment and component cooling water system heat exchangers, and back to the ultimate heat sink. All pumps, valves, screens, filters, piping, heat exchangers and interconnections are included.

The I&C boundaries of the SWS include equipment that is required to operate the system. Most of these components are addressed as generic I&C equipment and include, but are not limited to: control switches, relays, controllers, sensors, transmitters, status displays, recorders, computational modules, and circuit breakers.

B. See Section I, "Areas of Review," of SRP-LR B.0.1 for Item I.B.

C. Of the wetted systems in the NPPS, the SWS may have some of the more aggressive combinations of degradation factors, even though the temperatures experienced by the materials is relatively low (typically, 35°F to 120°F). The following is a listing of important aging degradation mechanisms which have been manifested in ESW systems. Characterization and localization of the listed degradation mechanisms can be found in References 1-3.

1. Corrosion

    a. General surface

    b. Galvanic

    c. Pitting

    d. Under deposit

    e. - Microbiologically stet corrosion (MIC)

    f. Induced chemical

2. Biological Attack

    a. Surface biofouling

    b. Macro-blockage

    c. MIC

3. Deposition

    a. Siltation (micro-impurity)

    b. Debris (macro-impurity)

4. Erosion

    a. Cavitation

    b. Solids impingement

5. Mechanical Agitation

    a. Vibration (low amplitude)

    b. Impact (high amplitude)

The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1. for Items II.A and II.B.C. In addition to the above, the licensee's program should include:

1. Implementation and maintenance of a surveillance program to control flow blockage due to biofouling.
2. Verification of heat transfer capability of safety-related heat exchanges cooled by service water.
3. Implementation of an inspection and maintenance program to ensure that corrosion, erosion, protective coating failure, silting, and biofouling cannot degrade performance of safety related systems.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for Items III.A through III.D.

E. Components of the SWS not specifically addressed in this section may be addressed by the generic aging topic reviews in SRP-LR Part C. The reviewer should ensure that structures and components included as part of the generic SRP-LR topics are adequately reviewed for the SWS. Specifically all of the sections from SRP-LR Part C except for C.1.5, "Tanks and Vessels" and C.4.0, "Civil Structures" should be reviewed in conjunction with the SWS review. This may require additional staff input.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "Biovalve Fouling of Nuclear Power Plant Service-Water Systems. Current Status of Biofouling Surveillance and Control Techniques," Vol. 2, NUREG/CR-4070 (PNL-5300), March 1985.

2. U.S. Nuclear Regulatory Commission, "Nuclear Plant Service Water System Aging Degradation Assessment," Vol. 1 NUREG/CR-5379 (PNL-6560) June 1989. September 1990.

3. U.S. Nuclear Regulatory Commission, "Nuclear Plant Service Water System Aging Degradation Assessment, "Vol. 2, NUREG/CR - 5379 (PNL-), September 1990.

B.5.5 ULTIMATE HEAT SINK

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SPLB/ESGB

I. AREAS OF REVIEW

A. This section addresses the ultimate heat sink (UHS).

1. Description

The UHS provides a highly reliable continuous supply of cooling water to equipment in the engineered safety features system and, for many plants acts as a backup water supply for the auxiliary (emergency) feedwater system. Regulatory Guide 1.27 describes several types of UHSs, including a large river, a large lake, an ocean, two spray ponds, a spray pond and a reservoir, a spray pond and a river, two mechanical draft cooling towers with basins, a mechanical draft cooling tower with a basin and a river, a mechanical draft cooling tower with a basin and a lake, a lake with a cooling pond, two wet/dry forced draft cooling towers, or two dry forced draft cooling towers.

The UHS is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the facility.

2. System Function

The UHS has two principal safety functions: (1) the dissipation of residual heat after reactor shutdown and (2) the dissipation of residual heat after an accident for a period of 30 days (typically) to allow time to evaluate the situation and to take corrective actions.

For a single nuclear power plant unit, the UHS is required to provide sufficient cooling water to accomplish each of these safety functions. For a multiple-unit station, the UHS is required to provide sufficient cooling water to permit simultaneous safe shutdown and cooldown of all units it serves and to maintain them in a safe shutdown condition. In the event of an accident to one unit, the UHS is required to dissipate the heat for that unit, to permit the concurrent safe shutdown and cooldown of the remaining units, and to maintain all units in a safe shutdown condition.

3. System Boundaries

The UHS boundary generally includes a complex of water sources with necessary structures (e.g., a pond with a dam or a river with a dam), mechanical water cooling systems (e.g., spray systems, mechanical draft cooling towers, or dry cooling towers), and the canals or conduits connecting the water sources and/or mechanical water cooling systems with, but not including, the cooling water system intake structure for the plant. The intake structure is covered in SRP-LR Section 4.0, "Civil Structure."

B. See Section I, "Areas of Review," of SRP-LR B.0.1 for Item I.B.

C. The UHS typically includes mechanical, electrical, and instrumentation and control (I&C) components plus civil structures, (e.g., dams, basins, or dikes). As a result, a broad number of age-related degradation mechanisms exist for the UHS. Typical examples of these mechanisms are the following:

1. Fatigue, for example, as applied to rotating fans or pumps

2. General wear of mechanical moving parts

3. Thermal embrittlement of plastic seals, etc., and of electrical and I&C components

4. Corrosion and erosion of metallic piping, in-line piping components, and cooling tower equipment (e.g., heat transfer surfaces used in dry cooling towers and spray nozzles)

5. Erosion and embrittlement of plastic piping used in cooling towers and spray systems and of plastic fill used in wet cooling towers

6. Plugging and/or fouling of cooling tower coils

7. Potential long-term biological effects on water basins and/or cooling towers and related equipment

8. Erosion and freeze damage of concrete dams, concrete basin walls, dikes, intake structures, ponds, or water basin soil shapes

9. Degradation of concrete and rebar corrosion in the intake structure associated with the UHS

Most UHS equipment, components, and subsystems subject to deterioration from aging are considered generically in the generic aging topic reviews in SRP-LR Part C of the SRP-LR as described above. Aging-related issues of particular concern in regard to the wet cooling tower, the dry cooling tower, and spray system components include the following. Depending on the type of packing used (e.g., wood, plastic, or asbestos), the structure and wetability of the packing in wet cooling towers can change with time, possibly resulting in local flow blockages and degraded performance.

Dry cooling tower coils are subject to not only internal fouling and corrosion (analogous to tube-side fouling and corrosion in a conventional heat exchanger), they are also subject to air-side corrosion, fouling, and plugging of the extended heat transfer surfaces. Corrosion of a fin-tube interface has a particularly devastating effect on coil performance and cannot always be detected by visual examination.

Spray pond nozzles are subject to degradation over time from abrasive wear by particulates suspended in the water and from erosion/corrosion. This degradation results in a modification of spray droplet size and dispersal and can result in degraded cooling pond performance.

The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1 for Items II.A and II.B.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for Items III.A through III.D.

E. UHS components not specifically addressed in this section may be addressed by the generic aging topic reviews in SRP-LR Part C. The reviewer should ensure that structures and components included as part of the generic SRP-LR topics are adequately reviewed for this system. Specifically all the sections of SRP-LR Part C with the exception of C.1.5, "Tanks and Vessels," should be used for the review in conjunction with the review of the UHS. This may require other staff support.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

VII. REFERENCES

B.5.6 REFUELING SYSTEM

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary -SPLB

I. AREAS OF REVIEW

A. This section addresses the refueling system.

1. Description

The refueling system provides for movement and/or replacement of the fuel assemblies in the reactor core. The refueling cavity is flooded with water so that the movement of fuel by the refueling system is accomplished under a sufficient depth of water to minimize radiation exposure to operators. Fuel is lifted and maneuvered remotely by using the cask handling and polar cranes.

The refueling system is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the specific facility.

2. System Function

The refueling system is designed to transfer new and spent fuel between the spent fuel pool (SFP) and the reactor, to shuffle fuel assemblies within the SFP and the reactor, and to load spent fuel in the shipping cask. The design of the refueling system includes features to prevent criticality and damage to fuel assemblies during fuel handling operations with a resulting release of radioactive material to the environment. The refueling system is also used to relocate control rod assemblies and burnable poison assemblies.

3. System Boundaries

The refueling system consists of a fuel bridge located at the reactor refueling cavity inside the containment and another similar bridge inside the SFP building. The refueling system also includes the fuel transfer subsystem which transfers fuel between the SFP and the reactor refueling cavity across the containment wall.

The equipment used to lift reactor vessel head and reactor internals are reviewed as components of the refueling system, to ensure that these components do not impair the function of the refueling system.

The major refueling system components are listed below:

- Fuel bridge (or manipulator) inside the containment
- Fuel bridge (or long-handled tool hooked to a crane) inside the SFP building
- Fuel transfer carriages and winch devices
- Fuel assembly upenders (typically hydraulically operated)
- Instruments and controls, including limit switches, sensors, and transmitters
- Reactor vessel head stud tensioners and lifting devices and associated crane equipment
- Reactor internal lifting devices
- Radiation detectors associated with the refueling system are addressed in SRP-LR B.5.2.

B. See Section I, "Areas of Review," of SRP-LR B.0.1 for Item I.B.

C. The refueling system has components that could wear with prolonged use and result in malfunction of the system. Components like limit switches, and locking devices on the grapples could degrade with age to the extent to cause or allow dropping or misalignment of fuel assemblies. Typical examples of age-related degradation associated with refueling system are listed below:

1. Age-related wear of limit switches

2. Corrosion and loose connections on instruments, control components and electric components

3. Electric cables cracking and moisture intrusion

4. Grapple locking devices failing to lock

5. Excessive backlash in indexing devices on fuel bridges, carriages, and upenders

6. Binding of sliding components, such as telescopic fuel masts and fuel transfer carriages.

The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for Items III. A. through III. D.

E. Refueling system components not specifically addressed in this section may be addressed by the generic aging topic reviews in SRP-LR Part C. The reviewer should ensure that structures and components included as part of the generic SRP-LR topics are adequately reviewed. The sections of SRP-LR Part C applicable to the refueling system are: C.1.2, "Valves"; C.1.6, "Equipment and Component Supports"; all of Section C.2.0, "Electrical" (except C.1.5, "Transformers"); and all of Section C.3.0, "Instrument." This may require additional staff input.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of B.0.1.

VI. GENERAL INFORMATION

Information on age-related degradation of components of the type installed in the Refueling System, and the recommended methods of managing age-related degradation are provided in reference 1.

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "A Review of Information Useful for Managing Aging in Nuclear Power Plants," NUREG/CR-5562 (PNL-7323), 1990.

B.5.7 SPENT FUEL STORAGE

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SPLB/EMCB

I. AREAS OF REVIEW

A. This section addresses the spent fuel storage facility, consists of:

1. Description

Nuclear reactor facilities include storage facilities for the wet storage of irradiated fuel assemblies. The spent fuel storage facility consists of all those facilities and tools designed for the safe removal of the irradiated fuel from the core, the transfer of the fuel to storage racks in the spent fuel storage pool, and the subsequent transfer of the irradiated fuel into shipping casks. Also included in the facility is the monitoring equipment designed to ensure safe storage of the irradiated fuel.

The spent fuel storage facility is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the specific facility.

2. System Function

The function of the spent fuel pool and storage racks is to maintain the spent fuel assemblies in a safe and subcritical array during all credible storage conditions and to provide a safe means of loading the assemblies into shipping casks.

3. System Boundaries

Included in the spent fuel storage facility are the spent fuel storage pool, storage racks, the spent fuel pool liner plate, fuel transfer canal, cask loading area, spent fuel pool cooling and cleanup, and all monitoring equipment associated with spent fuel storage. Also included within the boundaries of this facility are all tools used in the manipulation of the irradiated fuel from the time it leaves the reactor vessel until such time it is placed in the shipping cask.

B. See Section I, "Areas of Review," of SRP-LR B.0.1 for Item I.B.

C. Typical examples of age-related degradation associated with spent fuel storage are provided in this section. The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for Items III. A. through III. D.

E. The spent fuel storage facility components not specifically addressed in this section may be addressed by the generic aging topic reviews in SRP-LR Part C. The reviewer should ensure that structures and components included as part of the generic SRP-LR topics are adequately reviewed for this system. These specific sections from SRP-LR Part C should be reviewed: C.1.6, "Equipment and Component Supports"; C.2.4, "Relays, Circuit Breakers, and Switchgear"; C.3.1, "Sensors"; C.3.2, "Electronic Components"; and C.4.0, "Civil Structure." This may require additional staff input.

F. Specific areas of concern that are to be addressed by the licensee program for monitoring aging in the spent fuel storage facility are:

1. The reviewer should verify that the licensee has an established program of inspection and surveillance practices to detect aging and wear-related degradation in the structures housing the facility and the facility itself, including the monitoring system.

2. The reviewer should verify that the licensee has an established program of inspection and surveillance to determine the effects of aging mechanisms on the capability of the spent fuel storage facility to function under both normal operating and accident conditions.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

VII. REFERENCE

B.5.8 COMPRESSED AIR SYSTEM

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SPLB

I. AREA OF REVIEW

A. This section addresses the compressed air system (CAS).

1. Description

The CAS consists of the air compressors and their auxiliaries, and the piping necessary to supply clean, dry air for plant use. The CAS may include a compressed air system which supplies safety-related equipment and an additional compressed air system for non-safety-related equipment (normally referred to as instrument air and service air, respectively). Facilities with containments inerted with nitrogen may also have a nitrogen compressor which takes suction from the containment and supplies containment loads. As used in this section, the CAS will also include the compressed nitrogen subsystem.

The compressed air system is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the specific facility.

2. System Function

The CAS provides air/nitrogen to safety-related equipment and also to plant equipment used only for normal facility operation. Compressed air/nitrogen is, therefore, vital for maintaining stable plant operation. Its loss often results in a reactor trip and, on occasion, the actuation of engineered safety feature systems.

3. System Boundaries

The CAS consists of all components from the air compressors and their drive motors to the air supplies for air operated valves and instruments and controls. This includes the air compressor controls, coolers, receivers, and filters, as well as the air dryers, piping distribution system, and safety-related accumulators. The compressed nitrogen subsystem includes the containment penetration intake piping, compressor(s) receiver, discharge piping and containment penetration, and associated isolation valves. Some facilities may include a bank of compressed nitrogen bottles as an automatic backup to, or in place of, the nitrogen compressors.

B. See Section I, "Areas of Review," of SRP-LR B.0.1 for Item I.B.

C. Compressors and air system dryers and filters cause most CAS component failures, followed by particulate contamination and valve failures. A large fraction of these failures are attributed to the age-related degradation of these components. Typical examples of age-related degradation mechanisms are listed below:

1. Contaminants

2. Corrosion

3. Wear

4. Material deterioration of seals and gaskets

Additional information regarding age-related degradation of CAS components can be found in NUREG/CR-5419 (Ref. 1).

The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1 for Items II.A and II.B.

C. Additional criteria associated with the CAS that should be considered by the licensee, and that should be evaluated by the reviewer, includes

a one-time evaluation of the wall thickness of a representative sample (10 percent) of the CAS piping and storage tanks should be performed to detect the possible, subtle, long-term degradation of CAS components due to internal corrosion.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for Items III.A through III.D.

E. The reviewer should review the licensee's data pertaining to the performance and response of the CAS during demand events to ensure age-related degradation has not reduced the capability of the CAS to function properly.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

Despite the fact that plant maintenance includes regular monitoring of some parameters and inspection of many components, CASs have experienced more failures than were expected (Ref. 2). A testing and maintenance program could provide indications of degradation due to aging. Table B.6.8-1 (Ref. 2) provides typical testing and maintenance actions.

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "Aging Assessment of Instrument Air Systems in Nuclear Power Plants," NUREG/CR-5419.

2. U.S. Nuclear Regulatory Commission, "A Review of Information Useful for Managing Aging in Nuclear Power Plants," NUREG/CR-5562 (PNL-7323), 1990.

Table B.6.8-1 Recommended testing and maintenance for compressed air system

Subsystem	Testing & maintenance	Frequency
Compressor and receiver	Inspect for visible signs of degradation (physical damage, corrosion, erosion, loss of integrity)	Refueling outage
	Nondestructive testing at the bottom of vessel for thickness check (ASME Code)	Refueling outage
	Receiver relief valve functional testing	Once in 2 years
	Bearing monitoring (vibration)	Quarterly
	Oil sample check	Quarterly
	Inlet filter pressure drop	Monthly
	Compressor loaded and unloaded hours	Weekly
	Heat exchanger approach temperature; compressor outlet temperature; compressor oil pressure, and level function of moisture separators and automatic drains	Each shift
Dryer and filter	Dew point for air dryer	Daily
	Pressure drop of filter, cooler air flow measurement	Weekly
	Purge flow for desiccant dryer	Daily
	Function of separator automatic drain for refrigeration dryer	Each shift
	Function check on prefilter (drainage)	Each shift
	Cartridge changeout in prefilters and afterfilters	Semiannually

Table B.6.8-1 (continued)

Subsystem	Testing & maintenance	Frequency
Distribution network	Flow rate	Weekly
	System pressure	Quarterly
	Air quality (dew point, contamination, and operating pressure)	Semiannually
	Leak test on accumulators and check valve operability	Refueling outage

B.6.1 FIRE PROTECTION SYSTEM

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SPLB

I. AREA OF REVIEW

A. This section addresses the fire protection system (FPS).

1. Description

The FPS includes fire detection and suppression equipment which will sense conditions indicative of a fire, actuate alarms, and initiate fire suppression equipment as appropriate. Fire is detected by a network of sensors (thermal, light, smoke) located throughout the plant which actuate alarms in the plant control room identifying the location of the fire indicator. Fire suppression is provided by a fire water system, a carbon dioxide system for specific applications at specific plants, and a halon system for specific applications. Fire suppression systems may be automatically actuated or may require fire brigade action (manual), depending on the location and equipment being protected.

The fire water system (FWS) consists of piping and pumps serving areas throughout the entire site to provide a means of fire suppression. The FWS is usually the largest and most versatile of the automatic fire suppression systems at any plant site. Some facilities may use the FWS as a contingency source of reactor coolant injection during site blackout conditions, as directed in emergency operating procedures.

The carbon dioxide system is used to extinguish fires where the application of water is undesirable. The system may consist of carbon dioxide at high pressure, low pressure, or a combination of these two methods.

The halon system utilizes the total flooding principle to disrupt the combustion process. Two types of halon are typically used: Halon 1301 or Halon 1211. System actuation is determined by design requirements and specifications.

The FPS is described in the most recent revision of the final safety analysis report (FSAR) or updated Safety Analysis Report (USAR) for the specific facility.

2. System Function

The system is designed to automatically detect fires quickly, actuate audible and visual alarms, and, when necessary, provide a sufficient supply of water from the FWS to suppress fire in any of the areas it serves and to supply simultaneous hose station operation. The system is provided for those areas that contain or present potential fire exposure to plant equipment. Although this review should be limited to that part of the FPS which provides protection to or could impair SSCs important to license renewal, integrity of the FPS must also be assured.

3. System Boundaries

The boundary of the FWS consists of a reliable supply of water, fire water pumps, a jockey pump, a fire main loop, supply lines, manual hose stations, standpipes, and automatic sprinkler devices. There are variations to this arrangement; however, the factors that affect the aging process are independent of the type of system design.

The carbon dioxide system includes all storage tanks, valves, distribution piping, and all detection and actuation equipment. Similar types of equipment are included within the system boundary for the halon system.

B. See Section I, "Areas of Review," of SRP-LR B.0.1 for Item I.B.

C. Age-related degradation mechanisms affecting the FWS are typical of low-temperature water systems and include erosion, corrosion, cavitation, plugging, and fatigue.

Concerns related to aging, or degradation of halon system components may include external effects, or the decomposition product effects of halon in the presence of available hydrogen (from water vapor). The main decomposition products of Halon 1301 are hydrogen fluoride, hydrogen bromide, and free bromine. The decomposition products of Halon 1211 include hydrogen chloride and free chlorine. When present in a small quantity, free water can provide a site for concentrating acid impurities into a corrosive liquid. These highly corrosive liquids, if allowed, can influence material corrosion mechanisms in materials like steel, rubber, and synthetic elastomer material that may be used by the system, or in other systems.

The water source for FWSs is frequently from wells. In some cases, precipitation of dissolved solids from the well water supply creates a chronic plugging problem.

FWS piping has potential for erosion/corrosion which advance with age of the system. The resulting degradation (pipe thinning) could create a potential for a break in the later part of the plant's life, even if the problem is not indicated at the time the licensee's application for license renewal is reviewed.

Since the FWS piping is subjected to stagnant conditions or to operation with low or intermittent flow, it is susceptible to microbiologically induced corrosion (MIC). In addition to causing FWS leakage, MIC is known to cause excessive corrosion that can lead to reduced piping flow as well as complete flow blockage.

Corrosion and pitting are also common concerns when dealing with pumps. These problems are expected to continue throughout the plants's life, including the license renewal term.

Typical examples of age-related degradation associated with the fire protection system are discussed above. The specific areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for Items III.A through III.D.

E. FPS components not specifically addressed in this section may be addressed by the generic aging topic reviews in SRP-LR Part C. The reviewer should ensure that structures and components included as part of the generic SRP-LR topics are adequately reviewed for this system. Specifically, the following sections from SRP-LR Part C should be reviewed: all of C.1.0, "Mechanical," except C.1.4, "Heat Exchangers"; all of C.2.0, "Electrical," except C.2.5, "Transformers"; and all of C.3.0, "Instrument." This may require additional staff input.

F. The licensee's IPA shall include an assessment of the performance characteristics and physical condition of the FPS. This assessment should include an evaluation of performance trends over the preceding several years. The performance trends should be used to project any required corrective action for the license renewal period.

The physical condition of the FPS should be evident from the plant maintenance/repair records. These records should address observed degradation, including leaks, cracks, corrosion, erosion, plugging, and so forth. Where the records do not allow a clear assessment of the physical condition, a special inspection should be performed. As a minimum, the following special inspections should be addressed by the licensee's IPA:

- NDE of welds and pipe fittings where deadend piping (e.g., standpipes) connects to circulating piping

- NDE for pipe wall thickness in regions of high fluid velocity and turbulence to assess erosion/corrosion, and in areas of stagnant flow for corrosion

- Physical inspection of fire protection pumps in accordance with NFPA requirements

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, June 1987.

B.6.2 COMMUNICATIONS

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SICB

I. AREAS OF REVIEW

A. This section addresses the communications system.

1. Description

The communications systems of any facility usually consists of (a) hard-wired systems such as telephone, paging system, and headsets and of (b) radio frequency systems such as walkie-talkies, citizen band (CB) radios, and other types of two-way radios.

The communications systems is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the facility.

2. System Function

The sections of the communications systems that are of concern in this review are limited to those portions used in intra-plant and plant-to-offsite communications during transient, fire, and accident conditions. The portions of the system that are under consideration and important to license renewal must remain operable during loss of offsite power.

B. See Section I, "Areas of Review," of SRP-LR B.0.1 for Item I.B.

C. Typical examples of age-related degradation associated with the communications systems are provided in this section. The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of B.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of B.0.1.

IV. FINDINGS

See Section IV, "Findings," of B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of B.0.1.

VI. GENERA INFORMATION

The communications systems and the electronic devices used in them are or can be subject to the aging characteristics of their individual components and, therefore, the maintenance or replacement schedules should include considerations of the specific aging characteristics of the component materials (see SRP-LR C.3.2).

VII. REFERENCES

B.6.3 CONTROL ROOM HABITABILITY SYSTEM

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - EMCB/SPLB

I. AREAS OF REVIEW

A. This section addresses the control room habitability system.

1. Description

The typical system consists of two 100 percent capacity units that filter, cool, heat, and humidify air supplied to the control room, and often the computer and relay rooms, and sometimes other areas. Under normal conditions, fresh air is mixed with recirculated air. The air is cooled by a component cooling water system.

During the special operating mode, the outside air is routed through particulate, absolute, and charcoal filters before mixing with the recirculated air. For post accident recirculation, typically two fan and filter units filter the recirculated control room air. The units may be used to filter fresh air drawn from the outside. There are radiation monitors and toxic gas monitors in the supply ducts that, when activated, will close the intake ducts and establish total recirculation with cleanup flow.

The control room habitability system is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the specific facility.

2. System Function

The control room habitability system is designed to provide a reliable means of cooling and filtering air supplied to the control room area under both normal and post accident conditions. The system cools recirculated air and cools or heats of fresh air used to ventilate the control room for both personnel comfort and equipment cooling.

3. System Boundaries

The control room habitability system extends from the air intakes, normal and remote, to the discharge points. It includes ducts, piping, fans, motors, filters, chiller, electrical and instrumentation control circuitry, dampers, valves, chlorine and other toxic gas monitors where applicable, smoke and radiation detectors, and motor control centers.

B. See Section I, "Areas of Review," of SRP-LR B.0.1 for Item I.B.

C. The control room habitability system is prone to the same problems and failures as most ventilation systems. The presently recognized problems of bearing wear and replacement, fan blade cracking and repair/replacement, motor failure, heating and/or cooling coil failures, and instrumentation and control failures are expected to continue throughout the life of the plant, including the license renewal period. In addition, the door and other penetration seals for the control room habitability system are prone to the typical problems of hardening, cracking, and limited life associated with the various pliable sealing materials.

The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 for Items III.A through III.D.

E. Components of the control room habitability system not specifically addressed in this section may be addressed by the generic aging topic reviews in SRP-LR Part C. The reviewer should ensure that structures and components included as part of the generic SRP-LR topics are adequately reviewed for this system. These specific sections from SRP-LR Part C should be reviewed: C.1.4, "Heat Exchangers"; C.2.1, "Cable and Wiring"; C.2.3, "Electrical Penetrations"; C.2.4, "Relays, Circuit Breakers and Switchgear"; C.2.7, "Electrical Motors"; and all of section C.3.0, "Instruments." This may require additional staff input.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

VII. REFERENCES

B.6.4 AUXILIARY HVAC SYSTEMS

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SPLB

I. AREA OF REVIEW

A. This section addresses the auxiliary heating, ventilation, and air conditioning (HVAC) systems.

1. Description

Typical auxiliary HVAC systems important to license renewal include, but are not limited to, the diesel building, spent fuel pool storage area, and the turbine building.

Typically, the BWR turbine building (area) HVAC system is a safety-grade system. The air is exhausted through a filter bank and discharged to the plant vent stack. This operation occurs especially in the event of a main steamline break outside the containment and during maintenance periods when the turbine covers are removed.

The typical diesel building HVAC system provides a controlled environment in the diesel building area. The ventilation air is heated or cooled as required for the controlled environment. The ventilation system may, but does not necessarily, provide the combustion air for the diesel engines.

For the typical auxiliary building HVAC system, the path of ventilation air is from clean, or low-activity areas toward areas of progressively higher activity. Ventilation air is drawn from outside, through two makeup air units. The system is balanced to maintain the auxiliary building at a pressure slightly negative with respect to atmospheric and adjacent turbine building pressures. The air is exhausted through activated charcoal beds and high efficiency particle absorber (HEPA) filters from areas subject to possible radioactive contamination.

The typical spent fuel pool area HVAC includes fans with ductwork and dampers for distribution and filters to remove airborne radioactivity within the spent fuel pool and related equipment areas. The system may include a non-safety-related subsystem that provides ventilation during normal conditions. For high radiation conditions, the system typically includes redundant 100 percent capacity safety-related trains, each of which typically includes roughing filters, moisture separators, HEPA filters, activated charcoal filters, and associated ductwork and dampers for distribution. In the event of high radiation detected in the pool area, the safety-related system is automatically actuated.

For BWRs, normal ventilation of the spent fuel pool area is provided by the reactor building ventilation system (see SRP-LR C.3.9). Ventilation and filtration during high radiation conditions is provided by the standby gas treatment system (see SRP-LR B.3.6).

Each of these HVAC systems is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the facility.

2. System Function

The turbine area HVAC system is typically designed to provide maximum safety and comfort for operating personnel, with equipment arranged in a noncontaminated area for easy access for testing and maintenance. The system provides cooling in summer and heating in winter and also ventilates gases and fumes from the area. In the BWR plants, this system typically cleans the area atmosphere of potential gaseous and airborne particulate radioactive contamination.

The diesel building HVAC system typically provides the following functions:

1. Maintains the room ambient temperature low enough so that the diesel generator life is not degraded during normal operating and shutdown periods.

2. Maintains room temperatures in a tolerable range for personnel to perform maintenance and surveillance work.

3. Prevents building heat from accumulating.

The auxiliary building HVAC system is typically designed to provide maximum safety and convenience for operating personnel, with equipment arranged so that potentially contaminated areas are separated from clean areas. Redundant equipment is provided for those systems where, in case of malfunctions, public health and safety may be endangered or where safeguard equipment may be impaired.

The typical spent fuel pool area HVAC maintains ventilation in the spent fuel pool equipment areas to permit personnel access and to control airborne radioactivity in the area during normal operation, anticipated operational transients, and following postulated fuel handling accidents.

3. System Boundaries

The components considered within the review of the auxiliary HVAC systems include the following:

a. Ventilation system ductwork

b. Ventilation system inlet dampers, exhaust dampers and flow distribution dampers

c. Ventilation system supply fans, drive motors, and fan housings

d. Ventilation system exhaust fans, drive motors, and fan housings

e. Ventilation system motor control centers for the supply and exhaust fan drive motors

f. Electrical and instrumentation control circuitry for the supply and exhaust ventilation fan motors and damper operators

g. Exhaust filter units, housings, supports, and heating and cooling coils

h. Plant ventilation exhaust stack

i. Radioactive gaseous and particulate radiation samplers, monitors and control circuitry

B. See Section I, "Areas of Review," of SRP-LR B.0.1 for Item I.B.

C. The concerns and mechanisms for age-related degradation within these HVAC systems are the same as for other ventilation systems either safety-related or non-safety-related. The presently recognized problems and maintenance issues include:
   - o Bearing wear and replacement
   - o Fan blade cracking and repair/replacement
   - o Motor failure
   - o Heating coil repair/replacement
   - o Air handling unit supports repair/replacement
   - o Electrical and instrumentation control circuitry repair/replacement
   - o Cooling coil repair/replacement

These issues are expected to continue to be concerns throughout the life of the plant, including the license renewal term.

The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR B.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR B.0.1 Items III.A through III.D.

E. CVS components not specifically addressed within this section may be addressed by the generic aging topic reviews in SRP-LR Part C.

The reviewer should ensure that structures and components included as part of the generic SRP-LR topics are adequately reviewed for this system. These specific sections from SRP-LR Part C should be reviewed: C.1.4, "Heat Exchangers"; C.1.6, "Equipment and Component Supports"; C.2.1, "Cable and Wiring"; C.2.3, "Electrical Penetrations"; C.2.4, "Relays, Circuit Breakers, and Switchgear"; C.2.7, "Electrical Motors"; and all of Section C.3.0, "Instrument." This may require additional staff input.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR B.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR B.0.1.

VI. GENERAL INFORMATION

VII. REFERENCES

C.0.1 GENERIC COMPONENTS AND STRUCTURES REVIEW CRITERIA

SRP-LR Part C addresses the review of the various generic components and structures requiring evaluation of age-related degradation. Standard acceptance criteria, review procedures, evaluation findings, and implementation applicable to all of the generic topics are provided below. Specific requirements, descriptions and information unique to a particular generic component or structure are provided in that specific generic section.

I. AREA OF REVIEW

A. Description
See individual generic topic section for description.

B. Limits of Review:

The SRP-LR addresses aging degradation related to the subject topic that must be understood and controlled with sufficient certainty to permit the staff to consider issuing an operating license for the requested renewal period while maintaining the current licensing basis. The licensee has conducted an integrated plant assessment (IPA) to identify potential age-related degradation of systems, structures, and components and to evaluate the adequacy of the licensee's programs to identify and mitigate age-related degradation for the renewal term. This SRP-LR is not intended to be a review of the existing licensing basis.

The areas of aging Concern should be reviewed in accordance with site-specific conditions and experience as documented in the IPA.

C. Aging concerns and Mechanisims: See individual generic topic section for sepcific age related degradation concerns related to this component or structure.

II. ACCEPTANCE CRITERIA

Acceptance and performance criteria for specific structures or components are typically contained, in part, in such sources as technical specifications; ACI, AISC, ASME, and IEEE codes and standards; root-cause analyses; failure-mode analyses; equipment performance history; branch technical positions; approved topical and other industry reports; vendor criteria; and regulatory guides. For specific components, the vendor recommendations for extending their life through the renewal period could be critical in areas such as (1) applicability of current maintenance practices, (2) applicability of the current technical manual, and (3) design limitations for the specific component which may require replacement of selected parts.

The licensees may have review previously completed analysis if aging issues identify questions concerning the analysis (including its assumptions). The acceptability of the licensee's proposed program for identifying age-related degradation (as described in Item I.C for each specific chapter in SRP-LR Part C), monitoring aging degradation, and mitigating the effects of age-related degradation related to specific structures and components will be based on the following criteria:

A. The licensee has performed and documented an IPA which demonstrates that degradation related to the aging of specific structures and components has been identified, evaluated, and accounted for as necessary to ensure that the licensee's current licensing basis will be maintained throughout the term of the renewed license. As part of the IPA, the licensee has described the methodology for identifying SSCs important to license renewal and structures and components requiring evaluation of age-related degradation, and has provided a list of these SSCs and structures and components. This section of the SRP-LR is restricted to the review of specific components or structures.

B. As part of the IPA, the licensee has evaluated and accounted for age-related degradation related to specific structures and components to ensure that the licensing basis will be maintained throughout the term of the renewed license. The review focuses on the following items:

   1. The licensee has listed the structures and components from Item II. A above that are subject to an established effective program. This program must continue to ensure the capability of the specific structures and components to either perform their safety functions during the renewal term or not to impair the safety functions of other SSCs. In accordance with the requirements of 10 CFR 54.3(a), an established effective program shall include as appropriate, but is not limited to, inspection, surveillance, maintenance, trending, recordkeeping, replacement, refurbishment, and assessment of operational life for timely mitigation of the effects of age-related degradation. An established effective program must satisfy the following three criteria:

      a. The program is documented in the FSAR, approved by the onsite review committee, and implemented by the facility operating procedures.

      b. The program ensures that all SSC safety functions affected by age-related degradation of the specific structures or components are properly evaluated by the program procedures.

      c. The program establishes acceptance criteria against which the need for corrective action is to be evaluated and requires that timely corrective action be taken when these criteria are not met.

Programs and practices acceptable to the staff are discussed in Regulatory Guide DG-1009, "Standard Format and Content of Technical Information for Applications to Renew Nuclear Power Plant Operating Licenses." Such programs and practices have the following important elements: (1) use of state-of-the-art knowledge of age-related degradation in nuclear power plants; (2) integration of relevant materials science concepts, which describe degradation processes, with plant-specific design and operational information; and (3) use of state-of-the-art monitoring methods that reflect the mechanistic and empirical assessments performed by the licensee to understand age-related degradation and mitigate its effects.

Some existing programs will require modification to be classified as established effective programs for the renewal period. For example, for selected electrical components the licensee may claim the equipment qualification (EQ) program required by 10 CFR 50.49 is an established effective program. But for a subset of these components, either extensive additional testing is required or a reanalysis (with appropriate justification documented or selective verification tests, as appropriate) performed for the EQ program to apply for the renewal period.

2. For those specific structures or components identified as requiring evaluation of age-related degradation but not included in an established effective program, the licensee has described and provided the bases for actions taken, or to be taken, to manage the age-related degradation or has demonstrated, by evaluation, that the age-related degradation is not significant with respect to the current licensing basis. This action will include one of the following:

   A. Discuss specific aging management actions including inspection, maintenance, surveillance testing, condition monitoring, replacement, refurbishment, recordkeeping, and any adjustments made to the operating environment of the SSCs, as appropriate; or

   B. Demonstrate that age-related degradation is not significant and that the specific structure or component will continue to meet the current licensing basis without additional action during the term of the renewed license.

   C. The licensee has identified plant-specific exemptions granted pursuant to 10 CFR 50.12, "Specific Exemptions," and reliefs granted pursuant to 10 CFR 50.55a(a)(3), "Codes and Standards." The licensee should justify continuing those exemptions and reliefs that were granted on the basis of an assumed service life or period of operation bounded by the original license term of the facility, or otherwise related to SSCs subject to age-related degradation.

D. Additional criteria are discussed in the sections for specific structures and components, as applicable.

III. REVIEW PROCEDURES

Upon request from the primary reviewer (LRPD), the secondary review branches will provide material for the areas of review identified in Item I above. The primary reviewer obtains and uses such information as required to ensure that this review procedure is complete.

These review procedures should be followed for the review of the specific structure or component to determine whether or not: (1) the structures and components of the type addressed in the generic topic section have been appropriately identified as requiring evaluation of age-related degradation, (2) the potential aging mechanisms have been identified by the licensee for the specific components and structures (typical example, are provided in Item I.C in each chapter of SRP-LR Part C), (3) the established or new programs for managing age-related degradation are adequate, (4) exemptions and reliefs based upon assumed service life will continue to be appropriate during the renewal term, and (5) proposed modifications to the administrative procedures are adequate to manage age-related degradation.

The reviewer should perform the following steps to evaluate the licensee's program for license renewal based on the acceptance criteria given in Item II above.

A. The reviewer should confirm that an IPA has been documented and submitted which demonstrates that age-related degradation of the specific structures and components has been identified and evaluated in conformance with 10 CFR 54.21(a). The methodology for selecting SSCs important to license renewal and structures and components requiring evaluation of age-related degradation and the lists of SSCs and structures and components should be reviewed to ensure that all structures and components of the type addressed in this section have been appropriately identified. This section of the SRP-LR is limited to specific components or structures.

B. The reviewer should verify that the licensee has presented information which demonstrates acceptable performance from an aging perspective for each structure and component in an established effective program. The reviewer should confirm that the licensee identifies the method for evaluating age-related degradation and the adequacy of the aging-management program for each structure and component. Typical degradation mechanisms of concern for a specific component or structure are discussed in Item I.C of each chapter of SRP-LR Part C. However the actual mechanisms of concern for a particular facility should be addressed in its IPA. For structures or components identified as being routinely replaced or refurbished at defined intervals, the reviewer should ensure the licensee demonstrates ongoing programs are adequate for timely mitigation of age-related degradation. The support for this determination could focus on evaluation of operational experience, replacement or refurbishment intervals, and, as appropriate, design and manufacturer information, known aging mechanisms, and other relevant information. For structures and components not routinely replaced or refurbished, the reviewer should ensure the licensee's support for the conclusion that the structure or component is subject to an established effective program includes a detailed mechanistic analysis of age-related degradation mechanisms. The reviewer should confirm that:

1. The established program is documented in the FSAR, approved by the onsite review committee, and implemented by the facility operating procedures.

2. All SSC safety functions affected by age-related degradation are evaluated.

3. The program establishes acceptance criteria against which the need for corrective action is to be evaluated and requires that timely corrective action be taken when these criteria are not met. Replacement, refurbishment, and inspection schedules that may be necessary to manage age-related degradation are implemented by ensuring the plant program defines inspection methods used, inspection frequency, replacement and refurbishment frequency, and meets current licensing-basis requirements.

   The reviewer should ensure that the acceptance criteria are based on an industry standard or technically acceptable report and that the action to be taken is timely and will restore the component or structure to an acceptable performance condition in accordance with the facility's current licensing basis.

C. For specific structures and components not subject to established effective programs, the reviewer should verify one of the following:

   1. Current programs have been or will be revised to provide for timely mitigation of age-related degradation for this structure or component, or a new program will be developed specifically for this structure or component. The reviewer should confirm that the licensee's evaluation of the adequacy of the aging-management program includes detailed mechanistic analyses for all structures and components not routinely replaced or refurbished. These analyses may also be required for structures and components that are routinely replaced or refurbished if analysis of operational experience is not sufficient to demonstrate adequacy of the replacement or refurbishment program to provide for timely mitigation of age-related degradation.

2. Evaluation is provided to demonstrate that age-related degradation is not significant with respect to the current licensing basis for this structure or component and to justify the reason for the structure or component not being required to be part of an aging-management program.

D. Exemptions and reliefs granted on the basis of assumed service life have been reviewed to determine if they will continue to be valid for the term of the license renewal.

E. Additional review procedures are discussed in the sections for specific structures and components, as applicable.

IV. FINDINGS

The reviewer should determine and verify that the licensee has provided sufficient information and the review supports the following conclusions to be included in the staff's SER regarding license renewal.

A. The licensee's analysis acceptably identified the specific structures and components requiring evaluation of age-related degradation.

B. The licensee demonstrated compliance with the requirements of 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," and demonstrated through the IPA that degradation related to the aging of specific structures and components has been identified, evaluated, and accounted for as necessary to ensure that the licensee's current licensing basis will be maintained throughout the term of the renewed license.

C. The licensee proposed or is implementing an effective program for license renewal which uses existing programs and any necessary new procedures and methods to identify the plant-specific age-related degradation mechanisms, to manage degradation related to the aging of specific structures and components, and to ensure the activities authorized by the renewed license can be conducted in accordance with current licensing basis over the renewed term. For components and structures not covered by an existing or new program, the licensee has provided acceptable justification that the degradation experienced over the renewal term will not be significant.

D. The licensee provided a list of exemptions related to the specific structures and components granted pursuant to 10 CFR 50.12, "Specific Exemptions," and reliefs granted pursuant to 10 CFR 50.55a(a)(3), "Codes and Standards." The justifications for continuing the exemptions and reliefs are acceptable for the renewal term.

E. The licensee adequately identified any proposed modifications to the facility or its administrative control and plant procedures.

With respect to potential degradation that could result from degradation related to the aging of specific structures and components, the staff concludes that this issue is adequately addressed and the potential effects will be monitored, evaluated, and corrected.

V. IMPLEMENTATION

Except in those cases in which the licensee proposes an acceptable alternative method for complying with specified portions of the SRP-LR, the staff plans to use the methods described herein during its evaluation.

VI. GENERAL INFORMATION

Addressed in individual generic topic sections.

VII. REFERENCES

Addressed in individual generic topic sections.

C.1.1 PIPING

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - EMEB/EMCB

I. AREAS OF REVIEW

A. Description

This section addresses that piping important to license renewal. Design codes typically contain corrosion allowance, which is based on the service life of the piping system. This corrosion allowance is added to the wall thickness needed to withstand pressure and to other design calculation considerations. Further corrosion allowances may be added in the selection of piping schedules for a specific system during the construction process.

B. See Section I, "Areas of Review," of SRP-LR C.0.1 for Item I.B.

C. Aging Concerns and Mechanisms

A variety of age-related degradation mechanisms can affect the safe, continued operation of safety-related and high-energy piping. These mechanisms include fatigue, embrittlement, stress-corrosion cracking, wall thinning by various erosion and corrosion processes, pitting, and microbiologically induced corrosion (MIC). Typical degradation mechanisms and degradation sites are as follows (Ref 1).

1. High cycle thermal and mechanical fatigue acting on PWR nozzles and thermal sleeves on charging, safety injection, surge, spray, and coolant lines; terminal and dissimilar metal welds; feedwater piping, nozzles, and thermal sleeves/spargers; spray, surge, charging, safety injection, and residual heat removal lines; BWR high-thermal stress regions of recirculation piping; feedwater nozzles; and main steam and feedwater piping near fittings and other piping discontinuities (i.e., tee's, orifices, elboews, valves, etc.).

2. Thermal embrittlement acting on PWR cast stainless steel piping and BWR cast stainless steel recirculation piping.
3. Intergranular stress corrosion cracking (IGSCC), on BWR cast austenitic piping and safe-end welds. (The NRC recognizes that, in general, the materials now in use for BWR recirculation piping are less susceptible to IGSCC.)

4. Corrosion fatigue acting on BWR recirculation piping and high thermal stress regions.

5. Low-cycle fatigue caused primarily by thermal stress transients as a result of heatup and cooldown cycles.

6. Erosion and corrosion acting on PWR feedwater piping, and BWR main steam and feedwater piping near piping discontinuities (i.e., elbows, pipe joints, flow control valves, orifices, etc.).

7. Corrosion acting on BWR main steam and feedwater piping near structural discontinuities.

8. Other degradation mechanisms include crevice corrosion, erosion, pitting, galvanic corrosion, microbiologically influenced corrosion, intergranular attack, transgranular stress corrosion cracking, hydrogen embrittlement, and oxidation.

The areas of aging that affect a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR C.0.1 for Items II.A through II.C.

D. Additional Criteria

1. Thinning of Pipe Wall

Piping systems are examined for reduced wall thickness caused by the following corrosion processes: general and uniform corrosion, and erosion/corrosion.

A representative sample (10 percent) of the piping that is exempt from ACME Code, Section XI, inspections should be nondestructively examined to ensure that the pipe wall is sufficiently thicnk to satisfy design requirements through the renewal period. A one-time examination should be performed within 5 years of the license renewal application, as a minimum. In accordance with NRC Generic Letter 89-08 (May 1989), licensees are required to implement a long term erosion/corrosion monitoring program that provides assurances that procedures-administrative controls are in place to assure that an effective program is implemented and the structural integrity of all high-energy (including two phase as well as single phase) carbon steel systems is maintained.

For those cases in which the wall thicknesses do not comply with these criteria, wall thickness should be tracked to identify trends and action levels should be specified when acceptance criteria are not met.

2. Fatigue-Thermal, Vibratory, and Pressure

Licensees shall verify using plant-specific fatigue analyses for all ASME Class I piping that the ASME Section III cumulative usage factor allowable of 1.0 will not be exceeded during the projected lifetime of the plant.

3. Stress Corrosion Cracking

The license should examine a representative sample (10 percent) of the piping exempt from ASME Code, Section XI, inspections for evidence of intergranular and transgranular-assisted stress corrosion cracking (IGSCC and TGSCC) and any other forms of cracking. A one-time examination should be performed within 5 years of the license renewal application as a minimum.

4. Potential Flow Reduction

The licensee should examine a representative sample (10 percent) of those piping systems exposed to untreated water for potential buildup of silt or corrosion products that could restrict or limit flow through the piping. A one-time examination should be performed within 5 years of the license renewal application as a minimum.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR C.0.1 for Items III.A through III.D.

E. The methods of examination for piping systems important to license renewal are listed in the requirements of ASME Codes, Section XI (Ref. 2), IWB-2000, IWC-2000, and IWD-2000.

The reviewer should ensure that the examination techniques and procedures used by the licensee are in agreement with the following:

1. The methods, techniques, and procedures for visual, surface, or volumetric examination are in accordance with ASME Code, Section XI, IWA-2000.

   Alternative examination methods, combination of methods, or newly developed techniques to those given in IWA-2000 are acceptable provided that the results are equivalent or superior. The acceptance standards for these alternate methods are given in ASME Code, Section XI, IWB-3000, IWC-3000, and IWD-3000.

   The samples should be obtained from the following areas of piping systems exempt from ASME Code, Section XI, inspections: discharge of pumps, downstream of elbows, downstream of pressure reducing orifices, downstream of control valves, flow measurement devices, high-velocity areas of pipe, stagnant areas, low-pressure areas, screwed fittings (crevice corrosion), high-vibration areas, areas where vibration can cause cold working of pipe.

Approximately 10 percent of these piping areas should be inspected. However, the entire circumference of those pipes in areas susceptible to erosion and corrosion or two-phase flow should be inspected to ASME Code, Section XI, inspection methods.

If the licensee discovers areas where the original design corrosion allowance has already been utilized or is expected to be utilized before the end of the renewal period, begin to track wall thickness to identify trends upon license renewal on a basis commensurate with the wall corrosion allowance utilization. Also take additional samples throughout the system to track for trends. The action is to specify levels when selected criteria have been met.

If pipe thinning is detected in excess of wall thinning allowances for the current life, the inspection program should be expanded consistent with the severity of the thinning. For example, another 10 percent of the pipes should be examined for slight thinning up to 25 percent for significant thinning.

2. The licensee should list exemptions that have been permitted by ASME Code, Section XI, IWC-1220. For license renewal, the licensee should describe the sample of exempt piping that is selected by the plant staff for an additional one-time examination.

3. Fatigue data need to be developed to assess fatigue damage to piping. The stressors include heatups, cooldowns, operational transients, water hammers, steam hammers, thermal shocks, stratified flows, and flow-induced and equipment vibrations. A complete accounting of actual in-plant thermal loadings is needed to accurately predict the residual life of those components. Cyclical piping analyses need to be reverified, using the operational history data for all cyclic loadings.

   An acceptable conservative approach to satisfy the staff's fatigue concerns would include the following.

   a. List the original design basis calculated cumulative usage factors for all components. These calculations should have been based on the estimated number of plant transients and cycles for a plant life of 40 years.

b. Provide the additional number of transients and cycles to be used as the design basis for the extended life of the plant, e.g., if the projected life is an additional 20 years for a total life of 60 years, the original design basis transients from a. above should be increased by 50 percent. For all components, calculate the cumulative usage factors for this additional increment of time.

c. List any known cycles due to unanticipated plant transients which were not considered as design basis events in a or b above. For all components, calculate cumulative usage factors for these additional transients.

d. Add the cumulative usage factors calculated from a, b and c above to arrive at the total end of life fatigue usage factors for all components.

e. If the rate of actual plant cycles indicate that the design basis cycles will be exceeded at the end of life of the plant, the procedures in a, b and c above should be adjusted to account for these additional cycles.

f. The above analyses should be in accordance with ASME Section III, Subsections NB-3222.4 (e) (5) and NB-3228.5. If the total number of stress cycles is estimated to be greater than 10^6, the licensee should provide the basis for appropriate design fatigue curves.

g. In the above analyses, the licensee should include an evaluation of environmental effects on fatigue crack initiation to the extent needed to show that the analyses are conservative.

h. All of the above evaluations should be based on elastic analyses. The use of elastic-plastic or fully plastic approaches as a means to remove conservatism in fatigue analyses may be acceptable if a detailed description of the analysis techniques and the basis on which these techniques have been qualified are submitted to the staff for review and evaluation prior to using such procedures.

i. Each licensee should list any plant-specific history of failure due to fatigue in any piping.

4. For erosion/corrosion, ASME Code Case N-480, dated May 10, 1990, should be used for analytical evaluation, inservice inspection, repair, and replacement of Class 1, 2, and 3 carbon and low alloy steel piping susceptible to wall thinning.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR C.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR C.0.1.

VI. GENERAL INFORMATION

A. Aging Concerns

1. PWR Primary System Piping and BWR Recirculation Piping

The degradation processes of concern for PWR primary system piping and BWR recirculation piping are summarized in Tables C.1.1-1 and C.1.1-2, respectively (Ref. 1). For the current generation of reactors, these piping systems are designed to withstand several hundred heatup and cooldown cycles, during which time thermal stress transients produce low-cycle fatigue damage in the pipes.

Improved methods for modeling of fatigue damage will supplant those currently incorporated in ASME Code, Sections III (Ref. 3) and XI (Ref. 2). Before small fatigue cracks penetrate the pipe, they can be identified by nondestructive examination techniques; the pipes can then be repaired or replaced before the cracks penetrate a pipe.

Intergranular stress corrosion cracking (IGSCC) has occurred extensively in BWR recirculation piping. Most of that piping has been replaced with steels that are less susceptible to IGSCC as discussed in NUREG 0313 (Ref. 4). Hydrogen water chemistry has been implemented on several BWR plants to mitigate IGSCC. However, the potential for IGSCC degradation in crevices and in cold-worked areas still exists. Extensive long-term testing and examination may be required to ensure that IGSCC does not reemerge as a problem in BWR piping.

Long-term thermal embrittlement of cast stainless steels is a potential problem that is yet to be resolved. The results of accelerated aging tests are not conclusive, because of differences in the mechanisms controlling embrittlement in different temperatures.

2. Feedwater and BWR Main Steam Piping

Table C.1.1-3 summarizes the degradation processes of concern for BWR feedwater and main steam lines (Ref 1). Table C.1.1-4 summarizes the degradation processes of concern for PWR feedwater piping, feedwater nozzles, safety injection nozzles, and RCS nozzles and, in some plants, feedwater spargers and thermal sleeves (Ref. 1).

The primary degradation mode of concern for the piping is erosion and corrosion, a process whereby carbon or low alloy steel components lose their protective oxide layer and dissolve at an accelerated rate. Erosion and corrosion result in wall thinning, especially in the vicinity of flow discontinuities, which are caused by elbows, pipe joints, flow control valves, etc. Low pH, low oxygen content, and temperatures in the range from 250°F to 400°F increase the rate of attack, which can occur over a large area, or be as highly localized as a narrow axial groove.

In addition to the normal temperature transients experienced by PWR primary system piping and BWR recirculation piping, feedwater piping can experience thermal shock when cold feedwater is injected. These thermal shocks can initiate cracking at the interior surface of the piping and may contribute substantially to crack propagation.

Water-hammer or steam-hammer events can produce fatigue damage, or, in more severe cases, deformation and cracking of piping and pipe support structures. In addition, flow-induced and equipment-induced vibrations can cause fatigue damage.

3. Surge and Spray Piping, and Nozzles

In addition to the thermal cycling expected under normal operating conditions, the surge and spray lines can be subjected to stratified flow conditions wherein a layer of hot water or steam in the spray lines flows over a stagnant layer of cold water. In addition to circumferential stresses, stratified flow produces beam-bending forces in the pipe. The stress conditions resulting from stratified flows have a more severe effect on the pipes because they tend to persist for longer periods of time and tend to propagate any existing cracks. Thermal shock loadings of the spray lines can also occur when water flow is initiated in a steam-laden line. Table C.1.1-5 summarizes the degradation processes of concern for the pressurizer surge and spray lines, and nozzles.

Note that thermal shock loading can cause crack initiation, but usually not enough cycles exist for continued crack propagation. Thermal fatigue can initiate cracking and can cause the crack to propagate by a small increment during each successive cycle. Mechanical vibration also may produce only small increments of crack propagation per cycle; however, cycle repetition rates may be quite high, compared to thermal cycle rates. Even more rapid crack propagation can be caused by stresses induced by stratified flows and seismic events, which persist over longer time periods than do the thermal cycles. The potential for flow stratification could be eliminated by relocating pipes to avoid horizontal runs of piping. Likewise, thermal fatigue and thermal shock could be minimized by continuous operation of the spray system.

The aging degradation mechanism that is pervasive throughout PWR pressurizers and associated piping is fatigue. Low-cycle fatigue damage is caused by plant heatup and cooldown cycles, plant uploading and loading at power, step-load increases and decreases, reactor trips, hydrotests, so forth. The surge-line nozzle and thermal sleeve are particularly affected by the insurge of pressurizer fluid associated with power changes. Key fatigue degradation sites are calculated to have high cumulative fatigue usage factors and include the spray and surge lines.

B. Aging Mitigation

1. Inspection

Formal guidelines for examination of pressure piping are described in ASME Code, Section XI, (Ref. 2). Only the weld areas of the Class 1 systems, such as the BWR feedwater piping inside containment, must now be inspected. Inspection of areas of piping away from weld zones is not now required. However, in response to an NRC bulletin developed because of the Surry pipe break incident, all utilities have instituted inspection programs that include their feedwater system piping. Various computer programs have been developed to assist in identifying potential problem areas. These programs use plant piping material alloy content, chemistry, and flow data to determine areas in the piping most susceptible to erosion and corrosion. The ASME Code, Section XI, Sub-Committee ASME Code Case N-480, dated May 10, 1990, has developed inspection procedures to detect wall thinning in piping. NUMARC has also developed guidance for selecting examination techniques for specific plant situations and has provided suggestions for additional detailed examinations if erosion and corrosion is detected (Ref. 5).

EPRI has developed CHEC and CHECMATE computer programs to assist in selecting inspection locations for single phase and tow phase piping respectively.

Inspection procedures are being developed to detect erosion and corrosion damage in piping. Manual ultrasonic techniques can be used to measure average thicknesses, but these techniques could not be satisfactorily used in the past to determine minimum thicknesses. However, the accuracy of ultrasonic inspection has improved in recent years.

Note that these methods do not provide 100 percent inspection coverage of the susceptible sites, and they may therefore not be able to detect the minimum wall thickness, that is, the maximum erosion and corrosion damage. Development has begun on a new ultrasonic inspection method, a modified portable automated remote inspection system (PARIS) that uses a flexible transducer array. The flexible transducer array can conform and acoustically couple to the complex geometries of elbows and tees. Laboratory results show that this new ultrasonic method can examine carbon steel piping rapidly, with 100 percent coverage. Inspection of wrought stainless steel can be aided by performance demonstration qualification. However no present inspection method has been found completely acceptable for cast stainless steel.

Two other nondestructive examination (NDE) methods available to detect wall thinning are (a) high-energy radiography, through the insulation of a water-filled pipe, and (b) high-energy or isotope radiography, through the insulation of an empty pipe. The tangential radiographic technique has been used to measure wall thicknesses to within 0.076 mm (0.003 in.) in small-diameter, thick-walled pipe. High-energy radiation sources are used to inspect large-diameter (202-mm [8-in.]) pipe. The perpendicular radiographic technique can detect abrupt changes in thickness within 2 percent of the wall thickness. A calibration curve of thickness versus density is required for accurate measurements.

2. Recordkeeping and Trending

Thickness measurement data from the plant can be used to identify sites susceptible to erosion/corrosion.

Monitoring of water chemistry, including oxygen level, pH level, and impurities can assist in estimating erosion/corrosion rates.

An accounting of actual in-plant thermal loadings is needed to accurately predict the residual life of those components. Once the loadings are more accurately defined, an appropriate prediction of fatigue life can be made, and an appropriate inservice inspection program can be implemented with state-of-the-art techniques. This increased accuracy is particularly important for the horizontal portions of the surge line, which are subject to stratified flow. Stratified flows can cause significant fatigue damage to the horizontal portions of the surge line, which the original design may not have accounted for.

3. Managing Aging Degradation

Countermeasures for managing age-related degradation of reactor coolant piping are summarized in Table C.1.1-6, and are discussed in the following paragraphs of this item (Ref. 4). Research is continuing in this area to assess the effectiveness of these countermeasures.

a. Heat sink welding, induction heating stress improvement, and mechanical stress improvement are three stress improvement methods that effectively mitigate IGSCC, by introducing residual compressive stresses in the heat-affected zone (HAZ) on the inside surface of the recirculation piping.

b. The stress improvement methods introduce compressive stresses at the tip of shallow cracks in the HAZ, and are effective in inhibiting the growth of short cracks that do not exceed 30 percent of the wall thickness. However, a higher inspection frequency and a larger sample size are required for these welds.

c. For BWRs, use of hydrogen water chemistry has been successful in suppressing IGSCC crack initiation and IGSCC work growth, provided it is combined with very low levels of ionic impurities. Hydrogen water chemistry is effective when the level of dissolved oxygen is reduced below 20 parts per billion (ppb), and the coolant conductivity is kept below 0.3 micro per centimeter (mhos) (and the electro-chemical potential is kept below -230 mV - standard hydrogen electrode). Online crack arrest verification testing is desirable.

d. BWRs on hydrogen water chemistry inject oxygen into the condensate system to maintain 20 to 50 ppb in the feed-water and condensate syste-- to control carbon steel corrosion.

e. Weld overlays introduce compressive stresses in the weldment that inhibit IGSCC crack initiation and growth. Analytical results indicate that weld overlays will inhibit the growth of cracks that do not extend beyond 60 percent of wall thickness. The major barrier to extended use of weld overlays is the difficulty in performing reliable inspections of the weldment under the overlays. Improved ultrasonic methods are under development for this purpose. Welds repaired by weld overlays are generally inspected within two refueling cycles following the repair.

f. Mechanical clamping devices introduce axial and circumferential compressive stresses in the piping and retard crack growth. In addition, such clamping devices could provide an alternative load path around any degraded weldment to ensure its structural integrity.

g. Solution heat treatment of piping shop welds eliminates sensitization in the heat affected zone (HAZ), and thus provides protection against IGSCC. This treatment is applicable to new piping; presently approximately 40 percent of the welds in the recirculation piping are solution heat-treated.

h. Types 304NG and 316NG stainless steels are much more resistant to IGSCC, and have been qualified as alternative materials for BWR piping. However, Type 316NG does not have the same weldability as Type 304 stainless steel, and it is susceptible to transgranular stress corrosion cracking (TGSCC). Laboratory results show that the use of hydrogen water chemistry and strict control of impurities in the coolant can mitigate TGSCC.

i. Application of corrosion-resistant cladding on the inside surface of the piping helps to protect any sensitized surfaces from exposure to BWR coolant. Corrosion resistant cladding may be applied to the new piping weldments in the shop or field.

Acoustic emission methods are being developed for detecting fatigue crack growth in both the base metal and the welds of surge and spray lines. Acoustic emission methods can potentially provide global information regarding defects in piping and may be capable of detecting the location and growth of small flaws that are not detectable by other NDE methods. However, acoustic emission methods should be viewed as complementary to inservice inspection methods, not as their replacement.

Fatigue crack detection by acoustic emission methods dependse upon the ability of the instrumentation to detect the accustic signals caused by crack growth under reactor operating conditions, specifically in the presence of reactor coolant flow noise. The acoustic signal produced by crack growth consists of discrete burst-type sounds with a duration ranging from a few microseconds to a few milliseconds. The source of the signal is determined from the times of signal arrives at several different sensors installed at various locations. However, some test results indicate that the acoustic signal produced during tensile crack growth in Type 304 stainless steel may not be detectable at certain stages of the crack growth. On the other hand, the preliminary results from the inservice acoustic emission monitoring of a Peach Bottom Unit 3 recirculation-bypass line, core spray line, and feedwater nozzle indicate that pipe cracking can be detected using acoustic emission techniques.

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "A Review of Information Useful for Managing Aging in Nuclear Power Plants," NUREG/CR-5562 (PNL-7323), June 1990.

2. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."

3. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components."

4. U.S. Nuclear Regulatory Commission, "Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," NUREG-0313, Revision 2, 1986.

5. Electric Power Research Institute, NP-4947-SR, "BWR Hydrogen Water Chemistry Guidelines," 1987, revision October 1988.

Table C.1.1-1 Summary of degradation processes for PWR reactor coolant system piping

Degradation site	Stressors	Degradation mechanisms	Potential failure modes	Inservice inspection methods
Main coolant pipe nozzles *	System operating transients	Fatigue crack initiation and propagation	Through-wall leakage	Volumetric inspection for diameter of 24-in.
Thermal end dissimilar metal weld *	System operating transients	Fatigue crack initiation and propagation	Through-wall leakage	Surface inspection for diameter of 4-in.
Cast stainless steel	Temperature	Thermal embrittlement	Through-wall leakage	Volumetric inspection

* The nozzles in the reactor coolant piping are ranked highest among the degradation sites. Additional specificity is avoided because nozzles which are the most severely loaded are difficult to determine. Reported results are heavily dependent on the type of analysis performed. Actual damage will completely depend on the actual transient usage that occurs in the plant which is the basis for the recommendation.

** In Westinghouse plants, the dissimilar metal welds are at the reactor vessel and steam generator nozzles.

Table C.1.1-2 Summary of degradation process for BWR recirculation piping

Degradation site	Stressors	Degradation mechanisms	Potential failure modes	Inservice inspection Methods
Weld heat-affected zone furnace-sensitized safe ends	Tensile stress, oxygen environment sensitized heat-affected zone	IGSCC	Cracks, leaks	Ultrasonic examination, moisture-sensitive tape, acoustic emission
High thermal stress regions predicted by stress rule index analysis	Cyclic tensile stress, corrosive environment	Thermal fatigue, corrosion fatigue	Cracks, leaks	Ultrasonic examination, moisture-sensitive tape, acoustic emission
Austenitic-ferritic stainless steel casting with high delta ferrite levels	High temperature tensile stress, shock loads	Thermal embrittlement	Cracks, leaks	Ultrasonic examination, moisture-sensitive tape, acoustic emission

Table C.1.1-3 Summary of BWR feedwater and main steam system degradation processes

Degradation site	Stressors	Degradation mechanisms	Potential failure modes	Inservice inspection methods
Feedwater nozzles into reactor, thermal sleeves/ spargers	Thermal stress, corrosive water, water-nammer, vibration	Fatigue rupture	Cracks, leaks	Volumetric and surface examination at welds
Main steam piping near fittings and at geometric discontinuities	Wet steam, steam hammer, temperature gradients, vibration	Corrosion, erosion and corrosion, fatigue	Rupture, leaks, cracks, large deformations	Volumetric and surface examination at welds
Feedwater piping near fittings and at geometric discontinuities	Corrosive high-velocity water, temperature gradients, water-hammer, vibration	Corrosion; erosion/ corrosion; fatigue	Rupture, leaks, cracks, large deformations	Volumetric and surface examination at welds

Table C.1.1-4 Summary of PWR feedwater piping and nozzle degradation processes

Degradation site	Stressors	Degradation mechanisms	Potential failure modes	Inservice inspection methods
Feedwater piping inside containment, sites in horizontal piping runs in vicinity of mixing layer	Flow velocity, O^2 content and pH level in feedwater, impurities, stratified flows, thermal shocks, water-hammers	Erosion and corrosion, high- and low-cycle thermal fatigue, mechanical fatigue	Rupture caused by wall thinning, leakage through fatigue cracks, rupture caused by water hammer	Ultrasonic testing, radiography *
Feedwater piping near fittings	High flow velocity, O^2 content and pH level in feedwater, impurities, water-hammer	Erosion and corrosion mechanical fatigue	Rupture caused by wall thinning, leakage through cracks	Ultrasonic testing, radiography *

* Currently being performed but not required by ISI requirements.

Table C.1.1-4 (continued)

Degradation site	Stressors	Degradation mechanisms	Potential failure modes	Inservice inspect methods
Geometric discontinuities on inside surface of piping	Flow velocity, 0^2 content and pH level in feedwater, impurities, water hammer	Erosion and corrosion, mechanical fatigue	Rupture caused by wall thinning	Ultrasonic testing, radiography *
Charging nozzle	Thermal transient and shock stress loadings, flow-induced vibration	High- and low-cycle thermal fatigue, mechanical fatigue	Crack initiation and propagation leading to possible through-wall leak	Piping and nozzle welds inspected volumetrically at each of the four 10-year intervals

* Currently being performed but not required by ISI requirements.

Table C.1.1-5 Summary of degradation process for PWR pressurizer surge and spray line and nozzles

Degradation site	Stressors	Degradation mechanisms	Potential failure modes	Inservice inspection methods
Feedwater and safety injection nozzles	Thermal transient and shock stress loadings, stratified flow stress loadings, flow-induced vibration	High- and low-cycl thermal fatigue, mechanical fatigue	Crack initiation and propagation leading to possible through-wall leaks	Piping and nozzle welds inspected volumetrically at each of the four 10-year intervals
Pressurizer spray line and nozzle	Thermal transient stress loadings	Low- and high-cycle thermal fatigue	Crack initiation and propagation leading to possible through-wall leak	Piping and nozzle welds inspected volumetrically at each of the four 10-year intervals
	Stratified flow stress loadings, thermal stripping			
	Thermal shock		Thermal sleeve cracking	
	Flow induced mechanical vibration	Mechanical fatigue	Thermal sleeve cracking, crack initiation in nozzle	
Pressurizer surge line and nozzle	Thermal transient stress loadings	Low- and high-cycle fatigue	Crack initiation and propagation leading to possible through-	Piping and nozzle welds inspected volumetrically at each of the four 10-year intervals
intervals				
			wall leak	
	Stratified flow loadings (pipe only) thermal striping		Thermal sleeve cracking	

Table C.1.1-5 Summary of degradation process for PWR pressurizer surge and spray line and nozzles

Degradation site	Stressors	Degradation mechanisms	Potential failure modes	Inservice inspection methods
	Flow-induced mechanical vibration	Mechanical fatigue	Thermal sleeve cracking, crack initiation in	

Table 1.1-6 Summary of countermeasures for managing degradation of reactor coolant pipe

Mechanism	Countermeasure	Mitigation	Repair	Replacement
IGSCC	Inductive heating stress improvement	X	X	X
	Heat sink welding	X	X	X
	Mechanical stress improvement	X	X	X
	Solution heat treatment			X
	Corrosion-resistant cladding			X
	Nuclear-grade material			X
	Hydrogen water chemistry	X		
	Weld overlay	X	X	
	Clamping device	X	X	
TGSCC	Hydrogen water chemistry	X		
	Minimize cold working in fabrication			X
Thermal embrittlement	Use of less susceptible material			X

C.1.2 VALVES

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - EMEB/EMCB

I. AREAS OF REVIEW

A. Description

This section addresses the valves identified as important to license renewal. The valves typically are those valves required to meet the ASME Code, Section XI, (Ref. 1) Class 1, 2, or 3. In addition, other valves the staff considers important to license renewal are discussed in this section.

B. See Section I.B., "Area of Review," of SRP-LR C.0.1 for Item B.

C. Aging Concerns and Mechanisms

The valve components affected by age-related degradation are as follows:

Valve Bodies: external and internal surfaces, materials of construction, mounting and mating surfaces and components,

Valve Internals: all surfaces; materials of construction; operating components; gaskets and lubricants; springs, clips, fasteners, pins, shims, and spacers; liners, protectants, and surface hardeners; spaces, channels, gaps, and orifices

Fasteners: nuts, bolts, studs, bushings and clamps

Operators: handwheels and cranks; gear boxes, transmissions and their components and lubricants; all parts and components of motor, pneumatic, and hydraulic operators unless covered elsewhere in the SRP-LR.

Age-related mechanisms affecting valves, valve components, and operators, which include corrosion, erosion, fatigue, and physical damage are discussed below. Typical examples of age-related degradation concerns associated with valves are listed below:

1. Valve Bodies
   Corrosion; stress, chemical and microbioligically influenced corrosion (MIC) erosion; high fluid velocities, inappropriate application, environmental changes, cavitation, high temperature high differential pressure, and throttling fatigue; thermal and mechanical

2. Valve Internals
   Physical damage: wear; breaks, cracks, and chips; warped, bent, binding corrosion and erosion of valve internals and seats; high fluid velocities, inappropriate application, high temperature, and high differential pressure, throttling fatigue; thermal, mechanical, and cavitation

3. Fasteners
   Corrosion: stress and chemical attack

4. Operators physical damage:
   Wear; breaks, and cracks at mounting, couplings and connector locations; warped, bent, and bound
   Corrosion; stress and chemical attack
   set point drift

   Pneumatic and hydraulic operators: air and oil leaks and blow-by; excess drag and friction; elastomers and synthetics abraded, torn, brittle, inflexible; orifices, vents and filters obstructed; debris and foreign material present; grease aging; and additional aging concerns identified in SRP-LR B.5.8.

   Solenoid (electro-pneumatic) valves: liquid contaminants; debris and foreign material; orifices, vents, and filters obstructed; additional aging concerns identified in SRP-LR B.5.8 and SRP-LR C.2.6.

   Electric motor operators: any aging concerns identified in SRP-LR C.2.7.

   Instrumentation and controllers: any aging concerns identified in SRP-LR C.3.2, and SRP-LR C.3.3.

The aging mechanisms stated, though comprehensive, are by no means an all encompassing listing of valve aging and degradation phenomena. Because of the myriad of valve applications, incipient and unique valve aging and degradation mechanisms can exist that are not included here.

The areas of concern about aging for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR C.0.1 for Items II.A through II.C.

D. The ASME Code, Section XI, provisions for inservice inspection and inservice testing (ISI/IST) of valves is continued through the license renewal period.

The acceptance criteria for fatigue as described in SRP-LR C.1.1 Item II.D is applicable to all ASME class 1 valves.

For those valves important to license renewal that are not included in the facility ISI/IST program, a 10-percent minimum sample shall be tested to ensure that design adequacy is maintained throughout the license renewal period.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR C.0.1 for Items III.A through III.D.

E. The reviewer should verify that wall thinning mechanisms such as erosion are specifically addressed for valves that are routinely placed in a throttled position or are located in lines with high-velocity fluids.

The reviewer should verify that the licensee at least once performs an inspection on a 10-percent sample of valves that are important to license renewal and that are not part of the ASME Code, Section XI, ISI/IST program. This testing shall be in accordance with ASME Code, Section XI, or its equivalent.

The review procedure for fatigue as discussed in SRP-LR C.1.1 Item III.E is applicable to all ASME class 1 valves.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR C.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR C.0.1.

VI. GENERAL INFORMATION

VII. REFERENCES

1. ASME Code Section XI

C.1.3 PUMPS

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - EMEB

I. AREAS OF REVIEW

A. Description

This section addresses the area of review for those pumps identified as being important to license renewal. These pumps are typically, but are not limited to, all pumps required to meet the inservice inspection and testing (ISI/IST) requirements of the ASME Code, Section XI, Class 1, 2, and 3.

B. See Section I, "Area of Review," of SRP-LR C.0.1 for Item I.B.

C. Aging Concerns and Mechanisms

The aging concerns and mechanisms listed below have been identified as applicable to the components of reactor coolant and recirculation pumps (Refs. 1-8) and may apply to other pumps covered by the scope of this section.

1. Pump Casings

° thermal embrittlement
° thermal and mechanical fatigue
° stress corrosion cracking
° high residual stress owing to no postweld heat treatment
° erosion and erosion/corrosion
° crevice corrosion

2. Closure Studs

° corrosion
° stress corrosion cracking

3. Shafts

° mechanical and thermal fatigue
° corrosion

Cavitation erosion is a concern for many pumps although it has not been identified as a concern for reactor coolant or recirculation pumps. The areas of about aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR C.0.1 for Items II.A. through II.C.

D. The licensee's IPA should address those aging mechanisms identified in Item I.C above such as corrosion, fatigue, and embrittlement. The assessment shall ensure that design adequacy is maintained throughout the license renewal period.

Pump shaft inspections done during shutdown should include surface and volumetric examinations.

The ASME Code, Section XI, inservice inspection requirements, which are currently limited to volumetric examinations, should be supplemented to include visual inspections.

ASME Code, Section XI, provides for ISI/IST of pump components throughout the license renewal period.

The acceptance criteria for fatigue as described in SRP-LR C.1.1 Item II.D is applicable to all ASME Class 1 pumps.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR C.0.1 for Items III.A. through III.D.

E. The reviewer should ensure that the licensee's program addresses those effects of aging on pumps that are not detected by existing inspection and monitoring programs. Examples of potential problem areas that may not be detected by existing programs include small flaws caused by thermal embrittlement and stud corrosion.

The reviewer should verify that pump shaft inspections done during shutdown include surface and volumetric examinations.

The reviewer should verify that the ISI/IST program will continue throughout the license renewal period.

The review procedure for fatigue as discussed in SRP-LR C.1.1 Item III.E is applicable to all ASME Class 1 pumps.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR C.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR C.0.1.

VI. GENERAL INFORMATION

The following is a discussion of degradation mechanisms and inspection mechanisms for pump casings, closure studs, and pump shafts.

A. Pump Casings

1. Degadation

Some pump casings are cast stainless steel that have austenitic ferritic microstructures. These casings are subject to thermal embrittlement resulting from prolonged exposures at the typical operating temperatures of 288°C (550°F). Thermal embrittlement of the base metal results in a slow loss of fracture toughness over extended periods of time and is influenced by the coolant temperature and the time of exposure at that temperature.

The pump casing is subjected to thermal and mechanical fatigue damage caused by the operating transients and pump vibrations system. In some pumps, the welds were made using the electroslag technique, and these welds are susceptible to fatigue damage because of the high residual stresses. The susceptible sites for fatigue damage are likely to include some portion of both the base metal and weld region.

2. Inspection Methods

The pumps that are within the scope of this section of the SRP-LR will be disassembled for inspection and maintenance at intervals specified in the program.

ISI requirements for the pump body include surface and volumetric examination of identified repair and fabrication welds. The pump bodies made of cast stainless steel are difficult to inspect with conventional ultrasonic testing methods because of the elastic anisotropy caused by the different grain structures in the castings and because of the severe attenuation of the ultrasonic wave caused by the coarse grains in the steel. Therefore, radiography is generally used for volumetric examination of welds for pump body made of cast stainless steel. In accordance with ASME Code, Section XI, any indication detected by radiography must be considered a worst-case flaw because normal radiography can only detect the presence of a flaw, not its location or size (a worst-case flaw is a surface flaw with an aspect ratio of 0.5). However, triangulation radiography may be used to characterize, size, and locate a flaw.

The extent of the thermal embrittlement of cast stainless steel components may be such that the critical flaw size decreases to the size of an existing flaw or that the size of the critical flaw may become too small to be reliably detected using current inservice inspection methods. However, advanced ultrasonic testing (UT) methods are being developed that can better detect flaws and determine their size, orientation, and location (Refs. 2 and 3). These advanced UT methods should be more effective than radiography.

B. Closure Studs

1. Degradation

Typically, two concentric Type 304 stainless steel-graphite-asbestos gaskets are used for sealing between the PWR coolant pump cover and casing. A leakoff line is installed between the gaskets to allow for detection of any leakage of reactor coolant. Only one gasket is used in BWR coolant pumps. Leaking reactor coolant, if not checked, may cause corrosion of the closure studs, which are made of low-alloy steel-either SA193 Grade B7 or SA540 Grade B23. If the leakoff line installed between the gaskets is plugged or not instrumented, no indication of reactor coolant leakage from the inner gasket will be evident.

2. Inspection Methods

ASME Code, Section XI, requires volumetric examination of all the bolts, studs, nuts, and bushings during each inspection interval. However, the conventional UT volumetric examination techniques do not effectively measure stud corrosion wastage. Therefore, the ASME inservice inspection requirements, which are currently limited to volumetric examinations, should be supplemented to include visual examinations. Removal of the insulation and paint covering the studs will facilitate visual examinations to determine whether reactor water leakage has caused corrosion of the studs. It may also be necessary to either remove the bolt for inspection or consider a leak-before-break analysis (Ref. 5). ASME Code, Section XI, requires examination of the flange surfaces when a mechanical joint is disassembled.

C. Pump Shafts

1. Degradation

Pump shafts in light-water reactors are susceptible to damage from high-cycle mechanical and thermal fatigue, which is caused by alternating mechanical bending stresses and also by the rapidly varying thermal stresses in the thermal barrier region. The bending stresses are caused by any asymmetric distribution of the pressure. These alternating bending stresses, along with any stress risers and high residual stresses at the local welds on the shaft surface, can initiate circumferential cracks and propagate them in a plane perpendicular to the shaft axis. These cracks usually occur in grooves on the shaft surface and propagate in a transgranular manner.

Reactor coolant pumps use thermal barriers or heat exchangers, or both, to limit the reactor coolant heat reaching the mechanical seal cavity. In earlier pumps, the hot reactor coolant was mixed with cold cooling water at the top of the thermal barrier. The resulting turbulent mixing introduced high-cycle (1 to 25 Hz) thermal fatigue loads on the pump shaft surface.

2. Inspection Methods

LWR coolant pumps are generally equipped with two monitors mounted at the top of the motor stand in a horizontal plane to detect radial vibrations of the pump. Monitoring of pump motor frame vibrations has been successfully used to detect damage to the pump shaft (Refs. 6 and 7). Proximity probes have also been used for vibration monitoring to detect circumferential cracks in the pump shaft. However, vibration monitoring can not detect axial cracks caused by thermal fatigue.

The inspections done during shutdown should include surface and volumetric examinations of the pump shaft. Several utilities have used the conventional UT technique to inspect pump shafts, but the results have been inconclusive and misleading. A new UT technique, the modified cylindrically guided wave technique, is being developed for shaft inspection; the initial results of its use are promising.

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "A Review of Information Useful for Managing Aging in Nuclear Power Plants," NUREG/CR-5562 (PNL-7323), June 1990.

2. G. R. Egan et al., "Inspection of Centrifugally Cast Stainless Steel Components in PWRs, "Electric Power Research Institute, EPRI NP-5131, June 1987.

3. P. Jeong and F. Ammirato, "Nondestructive Examination of Coolant Pump Welds--Ultrasonic Examination of Cast Stainless Steel Components," paper presented at the Pressure Vessel and Piping Conference, Pittsburgh, Pennsylvania, June 1988.

4. U.S. Nuclear Regulatory Commission, Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," Generic Letter 88-05, March 1988.

5. R. E. Nickell, "Degradation and Failure of Bolting in Nuclear Power Plants, Electric Power Research Institute, EPRI" RP 2520-7, June 1987.

6. S. D. Leshnoff and R. C. Riccardella, "Reactor Coolant Pump Shaft Crack Investigations at TMI-1," Proceedings of the Conference on Main Coolant Pump Diagnostics, December 1988, EPRINP-6116, pp. 11-1 through 11-34.

7. S. M. Stoller Corporation, "RCP Shaft Fractured, Cap Screws Cracked, Broken-Fabrication, Thermal and Weld Stresses, IGSCC, Insufficient Preload," Nuclear Power Experience, PWR-2, April 1986, p. 52.

8. U.S. Nuclear Regulatory Commission, "Residual Life Assessment of Major Light Water Reactor Components - Overview," Vol. 1, NUREG/CR-4731 (EGG-2469), June 1987.

C.1.4 HEAT EXCHANGERS

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - EMEB/EMCB

I. AREA OF REVIEW

A. Description

This section addresses those heat exchangers identified as being important to license renewal. This typically includes, but is not limited to, all heat exchangers that are required to meet the inservice inspection and testing (ISI/IST) requirements of ASME Code, Section XI, Class 1, 2, and 3. Heat exchangers and coolers provide heat removal capability and pressure boundaries. The license renewal process must include an assessment of the role that each heat exchanger/cooler/chiller plays in the overall safety of the plant.

Steam generators, that are affected by many of the aging issues, are also addressed in Section B.1.2, "Reactor Coolant System," of the SRP-LR. Heat exchangers identified as being important to license renewal potentially include, but are not limited to:

o Heat exchangers (HEX)

Component cooling water HEX
Service water HEX
Shutdown cooling HEX
Regen HEX
Non-Regen HEX
Residual heat removal (RHR) HEX

o Coolers

Containment cooler/chiller
Instrument air cooler
Jacket water cooler
Lube oil cooler
Service air cooler

B. See Section I, "Area of Review," of SRP-LR C.0.1 Item I.B.

C. Aging Concerns and Mechanisms

Conditions that could impair the function of any heat exchanger include high temperatures; high pressures; exposure to filtered, demineralized, or raw (untreated) water; fluid flow; radiation exposure; foreign material intrusion; and deficient maintenance (Refs. 1-3).

The typical effects on heat exchanger performance from exposure to these conditions include:

1. Fouling of heat transfer surfaces
2. Cracking of shell, tubes or welds
3. Distortion of internal parts
4. Erosion/corrosion of internals
5. Blockage of flow passages
6. Fatigue of nozzles, tubes and supports
7. MIC pitting
8. Galvanic corrosion

The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of C.0.1 for Items A through C.

D. Heat exchanger performance characteristics, accounting for observed and projected degradation, shall be within the design envelope as represented by the design calculations, safety analyses, and industry codes and standards to which the licensee is committed.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of C.0.1 for Items A through D.

E. Historical trends may be used to reflect the useful lifetime remaining for each component. The amount of new performance testing should be inversely proportional to the acceptable data available.

The licensee's IPA shall include an assessment of the performance characteristics and physical condition of all heat exchangers important to license renewal. This assessment should include an evaluation of performance trends over the preceding several years. The performance trends may be used to project the remaining useful life for each heat exchanger.

- Temperature, pressure, and flow data required to assess heat exchanger performance should be available from existing plant records for most of the heat exchangers of interest. Where the existing plant information is not sufficient or does not include performance at heat loads near design, new performance information is required.

- The physical condition of most heat exchangers of interest should be evident from the plant maintenance/repair records. These records should address observed degradation including at least leaks, cracks, corrosion (general, pitting, galvanic, etc.), erosion, fouling/plugging, and tube support damage. If the records do not allow a clear assessment of the physical condition of a heat exchanger, a special inspection should be performed.

IV. FINDINGS

See SRP-LR Section IV, "Findings," of C.0.1.

V. IMPLEMENTATION

See SRP-LR Section V, "Implementation," of C.0.1.

VI. GENERAL INFORMATION

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "Residual Life Assessment of Major Light Water Reactor Components - Overview," Vol. 1, NUREG/CR-4731 (EGG-2469), June 1987.

2. U.S. Nuclear Regulatory Commission, "Residual Life Assessment of Major Light Water Reactor Components - Overview," Vol. 2, NUREG/CR-4731 (EGG-2469), October 1989.

3. A. S. Amar and V. N. Shah, "Remedies for PWR Recirculating Steam Generator Tube Aging," presented at the 10th International Conference on Structural Mechanics in Reactor Technology, Anaheim, California, August 14-18, 1989, EGG-M-88422 (CONF-890855).

C.1.5 TANKS AND VESSELS

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - EMCB/EMEB

I. AREA OF REVIEW

A. Description

This section addresses the generic structural and mechanical considerations for tanks and vessels identified as important to license renewal. Pressure vessel design codes, as applicable and as found in Article 3000 of ASME Code, Section III (Ref. 1), for Class 2 and 3 vessels, and Section VIII have been typically used to establish the required material thickness of these components. However, some steel tanks considered important to license renewal may be designed in accordance with other industry codes such as the American Petroleum Institute (API) or the American Water Works Association (AWWA). In the plant design specification, the licensee or the designer identifies the loadings anticipated or postulated to occur during the intended service life of each component. These loadings, which include temperature, pressure, and other anticipated service and test conditions, are used as the basis for establishing the appropriate design, service, and test limits for each component.

This section does not cover the reactor pressure vessel, nor the pressurizer and steam generators, which are covered in Section B.1.1, "Reactor Pressure Vessel," and Section B.1.2, "Reactor Recirculation System," respectively, of the SRP-LR. Concrete tanks will be covered in SRP-C.4.0 Section C.4.0, "Civil Structures," of the SRP-LR.

B. See Section I, "Area of Review," of SRP-LR C.0.1 Item I.B.

C. Aging Concerns and Mechanisms

Erosion/corrosion, embrittlement, fatigue, oxidation, pitting, microbiologically influenced corrosion (MIC), and stress corrosion cracking are typical examples of age-related degradation mechanisms that should be reviewed for tanks and vessels. The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR C.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR C.0.1 for Items A through D.

E. The licensee's aging-management program should include reconciliation of the design calculations and analyses (e.g., ASME Code stress reports) with the applicable age-related degradation concerns identified in I.C above for the license renewal term.

1. Examination Categories, Methods, and System Pressure Tests

   The descriptive examination categories and methods included in ASME Code, Section XI (Ref. 2), Articles IWA-1000, IWC-2000, and IWD-2000, should be reviewed as they pertain to tanks and vessels important to license renewal. Section XI, Article IWF-1000 and IWF-2000, pertain to component supports. System pressure testing programs are reviewed against criteria found in Section XI, Article IWC-5000 for Class 2 components, or Article IWD-5000 for Class 3 components.

2. Evaluation of Examination Results

   Disposition of examination results should be reviewed as applicable for compliance with applicable sections of ASME Code, Section XI, Article IWC-4000 for Class 2 repair procedures, Article IWC-7000 for Class 2 replacements, Article IWD-4000 for Class 3 repair procedures, and Article IWD-7000 for Class 3 replacements. Component support examination results should be reviewed as applicable with ASME Code, Section XI, Article IWF-3000.

3. Code and Licensee Exemptions

   The licensee's exemptions, as permitted by ASME Code, Section XI, Article IWC-1220 for Class 2 components and systems which are exempted for the licensee's aging mitigation program, should be reviewed for acceptability for the renewal term.

IV. FINDINGS

See Section IV, "Findings," of C.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of C.0.1.

VI. GENERAL INFORMATION

VII. REFERENCES

1. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components."

2. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."

C.1.6 EQUIPMENT AND COMPONENT SUPPORTS

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - ESGB/EMEB

I. AREA OF REVIEW

A. Description

This section addresses reactor pressure vessel supports piping/equipment snubbers, piping/equipment supports, anchor bolts, and reactor shield walls.

There are five major types of reactor pressure vessel supports:

° Neutron shield tank supports

° Column supports

° Cantilever supports

° Bracket-type supports

° Skirt-type supports

Snubbers are mechanical or hydraulic devices which allow free thermal movement of piping or equipment during normal operating conditions but which control dynamic displacements during abnormal dynamic conditions (such as earthquakes). The majority of piping snubbers are mechanical devices, with load ratings of 130,000 lb or less, whereas equipment snubbers are almost exclusively hydraulic devices and are manufactured with load ratings up to 2,000,000 lb.

Piping/equipment supports (other than snubbers) are structural members that support dead weight, accommodate thermal displacements and accommodate design-basis dynamic loads for the piping/equipment supported. These supports include features that may allow free movement along certain axes while restricting motion along other axes depending on the specific application.

Anchor bolts provide connections between equipment/supports and concrete structures. They may be cast in the concrete, grouted in the concrete, or retained through expansion features.
Reactor shield walls are large concrete/steel structures which surround portions of the RPV to shield instrumentation and electrical equipment. The shield wall may also perform structural functions such as for pipe supports.

B. See Section I, "Area of Review," of SRP-LR C.0.1 for Item I.B.

C. Aging Concerns and Mechanisms

The aging mechanism important to a given reactor vessel support structure will vary, depending on the type of support structure and its location in the plant. The age-related degradation mechanisms applicable to RPV supports are briefly summerized in Table C.1.6-1 (Ref. 1).

Both hydraulic and mechanical snubbers are susceptible to failure from a variety of mechanisms, including those related to aging. The major aging concern for hydraulic snubbers is seal degradation which may be influenced by environmental conditions such as radiation or temperature. For mechanical snubbers, the major aging concern relates to loss of free movement characteristics due to environmental conditions such as corrosion, temperature, and vibration (Ref. 2).

Many plants have experienced failures of piping/equipment supports under normal plant operating conditions. These failures are the result of unanticipated or improperly characterized loading phenomena such as water-hammer or cavitation loads. Metal fatigue is often the apparent cause of failure, although the root cause is likely a failure to properly characterize and design for cyclic loading phenomena.

Piping/equipment supports are also subject to corrosion damage if not properly protected.

Anchor bolts are subject to corrosion and may experience relaxation of preload (initial torque) which could degrade performance during a design-bases accident (DBA).

Material properties of reactor shield walls and doors are subject to degradation associated with radiation exposure (primarily neutrons). Corrosion may also be a significant degradation mechanism.

Typical examples of age-related degradation concerns associated with support structures are provided above. The specific areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of C.0.1. for items A through C.

D. Observed and projected age-related degradation shall be within the design envelope as represented by the design calculations, safety analyses, and industry codes and standards to which the licensee is committed.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of C.0.1 for items A through D.

E. The following review procedures apply:

1. The reviewer should verify that age-related mechanisms applicable RPV supports, and their effects, have been adequately addressed. These include the mechanisms stated below:

   - Ferritic components of support structures exposed to neutron irradiation will have degraded mechanical properties and should be examined/evaluated to ensure that they will maintain their structural integrity for all design loads.

   - The long-term effects of radiation-enhanced corrosion of RPV supports should be examined.

   - The need for replacement of non-metallic elements of support, from exceeding radiation thresholds, should be assessed.

   - The increased number of thermal cycles on the skirt supports should be addressed.

2. The reviewer should verify that the licensee's program incorporates the results of further NRC snubber aging research.

3. The reviewer should verify that where piping/equipment support failures have occurred as a result of fatigue or dynamic loads, these and similar supports are assessed for potential increased cycles of the license renewal period.

   - The licensee should visually examine a statistical sample of piping/equipment supports to assess corrosion damage.

4. The reviewer should verify that the licensee inspects a statistical sample of all types of anchor bolts for torque setting (preload) and corrosion.

5. The reviewer should verify that the licensee reconciles the design calculations for the reactor shield wall and doors with any material property degradation associated with increased neutron fluence over the license renewal period.

6. The reviewer should verify that the licensee performs a visual inspection for corrosion damage should be performed.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR C.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR C.0.1.

VI. GENERAL INFORMATION

VII. REFERENCES

1. "U.S. Nuclear Regulatory Commission, "A Review of Information Useful for Managing Aging in Nuclear Power Plants," NUREG/CR-5562 (PNL-7323), 1990.

2. U.S. Nuclear Regulatory Commission, "Aging and Service Wear of Hydraulic and Mechanical Snubbers Used on Safety Related Piping and Components of Nuclear Power Plants," NUREG/CR-4279 (PNL-5479), Vol. 1, February 1986.

Table C.1.6-1 Summary of degradation processes for PWR and BWR RPV Supports

Degradation site	Stressor	Degradation mechanism	Potential failure modes	ISI method
Neutron shield tank at the core horizontal midplane evaluation	Neutron irradiation, tensile stresses, operating temperature, water chemistry	Neutron embrittlement, corrosion	Catastrophic brittle failure	Monitoring
Column support at the horizontal mid-plane elevation	Gamma neutron irradiation, mechanical and thermal stresses, operating temperature	Embrittlement	Catastrophic brittle failure	Monitoring and core sampling
Cantilever support in the active zone of the core	Gamma neutron irradiation, tensile stresses, operating temperature	Embrittlement	Catastrophic brittle failure	Monitoring and sampling
Threaded parts in sliding foot assembly	Tensile stresses, operating temperature, high humidity	Stress corrosion cracking	Binding that causes possibly excessive stresses in the primary system during heatup and cooldown	

Table C.1.6-1 continued

Degradation Site	Stressor	Degradation Mechanism	Potential Failure Modes	ISI Method
Dry lubricant in sliding foot assembly	Neutron irradiation, operating temperature	Degradation caused by neutron irradiation	Binding that causes possibly excessive stresses in the primary system during heatup and cooldown	
Skirt support	Mechanical and thermal stresses	Fatigue	Overlaod failure on fatigue cracked component	

C.2.1 CABLE AND WIRING

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SELB/SICB

I. AREA OF REVIEW

A. Description

Cables provide the path for signals between sensors and electronics used for the protection and control of the reactor, and for the control and powering of equipment used during normal operation and in mitigating the effects of accidents. Thus, cables are important to plant safety both during normal operation and under accident conditions. The material for conductors and shields of most cables is stranded copper, often with tin or some other coating to prevent copper/insulation interaction and to provide good corrosion-resistant connections. The insulations of cables are generally polymer-based compounds, except for mineral-insulated cables (Ref. 1). Depending upon circuit requirements, insulations serve to isolate the electrical conductors from ground, and sometimes to maintain high dc resistivity, low ac losses, or proper concentricity of conductor and shield. The dielectric properties of breakdown strength and insulation resistance are particularly important. The jacket is a material on the outside of the cable and is often considered vital in maintaining the hermetic integrity of the cable, as well as in furnishing outer protection. Jacket material is usually extruded polymers, except for silicone cables where a textile braid is woven over the insulation for mechanical protection.

B. See Section 1, "Area of Review," of SRP-LR C.0.1 for Item I.B.

C. Aging Concerns and Mechanisms

There are a variety of age-related mechanisms that can affect the ability of cables to continue to operate reliably. While the aging processes are presently in progress and some failures have been observed, it is expected that the aging degradation will be more evident as cables and wiring approach the end of their documented qualified life. Typical examples of age-related degradation concerns associated with cable and wiring are given in this section. The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

1. Thermal and radiation aging leads to embrittlement of the jacket and insulation. Embrittlement allows the jacket and insulation to crack or break with the subsequent intrusion of moisture or chemicals. The moisture or chemicals can cause reduced insulation values, as well as corrosion of the conductor material (Ref. 1).

2. Thermal stress and damage, radiation, and moisture lead to increased leakage of current and large changes in ac losses or capacitance. These effects can degrade performance of sensitive circuits (Ref. 1).

These aging degradations are a particular concern for design-basis accidents in which severe temperature, moisture, and mechanical stresses may all be present at the same time. The most common design-basis accidents are LOCA or main steamline break.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of C.0.1. Items A through C.

D. In addition to the above items that should be considered by the licensee, include the following specifically for cable concern.

   1. The licensee has or will implement a program to identify degradation due to aging. This program should include a combination of tests, analyses and inspections to (a) define the current material condition of cables and (b) establish a replacement interval if appropriate.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR C.0.1 Items A through D.

E. The reviewer will verify that the current material condition of cables has been identified and that replacement intervals are established where appropriate.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR C.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," ofSRP-KR C.0.1.

VI. GENERAL INFORMATION

Studies performed as a part of the NRC Nuclear Plant Aging Research (NPAR) Program have shown that of the components that form cabling and wiring (conductors and shields, insulations, and jackets) conductors and shields are expected to experience the least amount of aging degradation and that insulations and jackets are more prone to aging degradation. The most severe aging stressors are usually found in the containment. The most common aging stressors and failure mechanisms are thermal and radiation aging leading to embrittlement of the jacket and insulation. Embrittlement allows the jacket and insulation to crack or break with the subsequent intrusion of moisture or chemicals. The moisture or chemicals can cause reduced insulation values, as well as corrosion of the conductor material. In addition, temperature, radiation, and moisture leads to increased dc leakage, dc and ac capacitance, and ac losses in the insulation. These effects can degrade performance of sensitive circuits (Ref.1). Table C.2.1-1 summarizes the aging-related degradation mechanisms, stressors, and degradation sites for cables (Ref. 2).

Several bulletins have alerted utilities to check for weaknesses that have occurred at one or more plants. Examples of those weaknesses are physical damage to silicon rubber-insulated cables, misapplication of shrink tube splice covers, degradation by local heat sources, and abuse/degradation of Kapton-insulated wires. To date, no continuing inspections have been required for cables in the containment. Routine visual examinations of open runs, terminal areas, and areas of known local high stressors are another element of an effective program of maintenance to detect signs of abuse or degradation. Aging changes that lead to eventual circuit failures or near-failures are normally not observable until the troubled component is isolated for dissection and analysis. Until more experience has been accumulated or new measurement techniques have been developed, it is not possible to electrically monitor the aging of in situ cable systems in a way that relates electrical measurements to the preaging carried out as a part of equipment qualification (EQ) programs.

A number of physical, electrical, and chemical laboratory test methods that have been used for in situ monitoring of aging changes in low-voltage cables. None of those test methods have been used in evaluating cables in place because the methods are destructive and must be applied to samples, using laboratory equipment. Several programs are being developed for using electrical and mechanical techniques cables to track gradual aging effects on cables; such techniques must have the potential for relating to qualification preaging programs. Several new and/or evolving aging assessment test methodologies for monitoring cable systems include the following:

1. A mechanical cable indenter that may be effective for surveying overall cable environmental aging severity in a containment and for tracking the aging of cable jackets or exposed insulations with reference to a preaged condition in qualification is being developed

2. Time domain spectroscopy is a technique of applying a dc step voltage to a cable, analyzing the frequency spectrum of the resulting current flow, and deriving the cable insulation capacitance and loss characteristics as a function of a wide frequency range. It is presently being evaluated in laboratory testing.

3. Partial discharge detection methods for finding insulation defects are customarily used for medium- and high-voltage cables. New concepts of these techniques that are applicable to nuclear plants are being investigated by NIST and EPRI.

4. Chemical analysis of surface scrapings from cable jackets is a method of possible value in life assessments.

5. NRC and EPRI are sponsoring projects at Sandia National Laboratory to tests which of the many previous electrical and mechanical cable monitoring tests produced data that correlate with cable performance under LOCA/MSLB stressor conditions (Ref. 3).

Table C.2.1-2 summarizes results of NPAR aging research and presents a listing of materials, aging concerns, recommended inspection and monitoring, and recommended maintenance for cables (Ref. 2).

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "Pressurized Water Reactor and Boiling Water Reactor Cables and Connections in Containment, Residual Life Assessment of Major Light Water Reactor Components - Overview," Vol. 2, NUREG/CR-4731, November 1989.

2. U.S. Nuclear Regulatory Commission, "A Review of Information Useful for Managing Aging in Nuclear Power Plants," NUREG/CR-5562 (PNL-7323), 1990.

3. U.S. Nuclear Regulatory Commission, "An Aging Assessment of Cables, Connectors, and Electrical Penetration Assemblies Used in Nuclear Power Plants," NUREG/CR-5461 (Draft), March 1990.

TABLE C.2.1-1 SUMMARY OF POTENTIALLY IMPORTANT FAILURE MODES AND DEGRATION FACTORS FOR LWR CONTAINMENT CABLES AND CONNECTIONS

Failure Modes	Age Degradation Mechanisms	Dominant Aging Stressors	Components	Degradation Sites
Circuit ground or short when subject to condensing steam, spray or water (CCF)[b]	Jacket embrittlement and cracking-propagating through insulation. Bare insulation cracking.[c]	High temperature, O_2 presence, radiation in a few cases, sometimes moisture.	Single and multiconductor nonshielded jacketed cables. Kapton-exposed wires.	Hot spots, terminal areas at hot equipment, proximity to hot pipes, fire stops, exposed susceptible insulation.
Corrosion causes opens in, total loss of, or multiple grounds on shields (CCF, RF).	Jacket cracking or moisture diffuses through jacket and condenses.	Moisture, high temperature.	Shielded coaxial or multi-conductor paired cables.	Moist warm areas, high humidity; near water, steam leaks, or seepage.
Corrosion of contacts. Circuit opens, grounds, or shorts (CCF, RF).	Diffused moisture collects in cable and migrates into connection internals.	High temperature, moisture, and water contamination.	Connections not permanently sealed against cable internal moisture.	Moist warm areas, high humidity, near water, or steam leaks, or seepage.
DBA pressure/steam/spray passes into or through connection. Contacts corrode or circuit grounds or shorts (CCF).	Polymer seals (O-rings) or cable polymers cold flow so that seals are not hermetic.[c]	High temperature and/or radiation dose, cable movements, vibration, thermal cycling.	Connections with compression seals.	Hot spots-thermal and radiation, connections where cable is moved.
Peak temperature and radiation during DBA cause excess leakage or losses to disable circuit function or lead to insulation breakdown (CCF).	Thermal and radiation aging leave remanent electrolytes to increase leakage or losses.	Heat, radiation, and moisture diffusion in normal service. Temperature, dose rate, and moisture after DBA.	Cables insulated with halogenated or filled polymers.	General-where exposed to harsh accident environments.
Same as above, except steam condensation and ionizing radiation are prime factors.	Gradual increases in surface contamination.[c]	Accumulations of wettable or conductive surface contamination.	Terminal strips.	Nonhemetric junction or terminal boxes with external or internal dust or contamination generators.
Excessive leakage disables MI cable circuit operation (CCF, RF).	Metal cold flow or loosened threads open hermetic seals to moisture intrusion.	Vibration, repeated movement, thermal cycling.	MI cable connections.	Connections subject to vibration or flexing.

a. The problems listed may have been anticipated and adequately addressed in the original Class 1E nuclear qualification program practices. However, they are ones that should be considered if qualification practices were not complete or rigorous in their application or in considering the extension of the original life.

b. Notations in parentheses indicate potential for common-cause failures after a DBA or submersion (CCF) and for random failures during normal or abnormal service (RF).

c. The degraded condition noted is probably not electrically detectable when conditions are dry.

TABLE C.2.1-2 MANAGING AGING OF INSTRUMENTATION CABLES, CONNECTIONS, AND PENETRATIONS FOR LIGHT WATER REACTORS

UNDERSTANDING AGING (Materials, Stressors, & Environment Interactions)			
Component	Subcomponents	Typical Material	Aging Concern
Cable	Insulation	Crosslinked polyethylene	Thermal and radiation embrittlement, oxidation, cracking, and moisture intrusion.
	Jacket	Chlorosulfonated polyethene	Mechanical stress, thermal and radiation embrittlement, and cracking, and moisture intrusion.
	Conductor & Shields	Stranded copper	Fatigue or corrosion.
Penetration	R-ring seal	Elastomer	Pressure leak, cracking
	Contact socket	Gold plated copper	Wear with use
	Interfacial seal	Dow corning syigard	Cracking
	Insulator	Polysolfone	Cracking
Multi-pin connector	Pins and sockets	Gold plated copper	Wear with use, gold-solder chemical reaction
	Inserts (insulation)	Thermal plastic polymer	Embrittlement, wear
	Seals and grommets	Fluorosillicone elastomers	Cracking
	Shells and rings	Aluminum or stainless steel	Cracking
Terminal strips	Terminal board	Glass filled phenolic	Embrittlement
	Cable clamp lug and screw	Stainless steel	Broken or loose screws or dirty connection
	Shrink tubing	Polyolelin	Cracking
Junction box	Seals	Elastomer	Embrittlement

TABLE C.2.1-2 MANAGING AGING OF INSTRUMENTATION CABLES, CONNECTIONS, AND PENETRATIONS FOR LIGHT WATER REACTORS (CONTINUED)

MANAGING AGING		
Inspection and Monitoring		Maintenance
Preservice	Inservice	
<u>10 CFR 50.49</u> Calls for artificial or natural aging prior to E.Q. testing <u>Reg. Guide 1.89</u> Qualified life may be demonstrated based on arrhenius theory and a surveillance and maintenance program <u>IEEE-383</u> Provides industry guidance for qualifying Class 1E cables and connections <u>IEEE-317</u> Covers design, installation, and testing of electrical penetrations in containment structures	à No requirement for inservice inspection à Inspection of connections following maintenance à Monitor redundant channels for discrepancies à End-to end system tests during refueling outage <u>Recommendations</u> à Develop inservice surveillance criteria à Perform periodic inspections à Perform temperature and radiation mapping in containment cable locations	à Performed when system performance has degraded à Performed when testing identifies specific problems à Replace components at end of qualified life <u>Recommendations</u> à Develop advanced remote monitoring for detecting deterioration à Develop criteria for replacing cable in severe environment

C.2.2 JUNCTIONS

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SPLB and SELB

I. AREA OF REVIEW

A. Description

The most common types of junctions/connections used in nuclear safety-related applications are splices (butt or bolted), crimp-type ring lugs, and terminal blocks. Splices and lugs may be insulated or uninsulated. Some splices are covered with tape or heat shrink tubing when used in potentially harsh environments.

Terminal blocks are used throughout plants in many low-voltage power (less than 480 V) and control applications. In response to equipment qualification (EQ) concerns such as those outlined in Information Notice 84-47, a number of plants have removed either all inside containment terminal blocks in safety circuits or all inside containment terminal blocks in instrumentation circuits. Terminal blocks are especially convenient where access to equipment leads is necessary for maintenance or calibration.

Coaxial connections are in limited use in safety-related circuits in harsh enviornment areas; the most critical application (in terms of required function) is for radiation monitoring circuits, where very high insulation resistance may be required during accident conditions.

Other types of connections are used in nuclear plants, such as thermocouple connectors, but they are less popular and are generally specialized connections (Ref. 1).

B. See Section I, "Area of Review," of SRP-LR C.0.1 for Item I.B.

C. Aging Concerns and Mechanisms

The simplicity of typical connections limits the number of age-vulnerable materials they contain. Terminal blocks are often constructed of phenolic materials that are very age resistant. However contaminates may cause excessive leakage in instrumentation circuits. Butt and bolt splices may have insulation that could be vulnerable to aging, usually nylon or Kynar. Raychem heat shrink tubing and the tape are polymeric materials that could be degraded by aging (Ref. 1).

The possible failure modes of connections are either loss of dielectric isolation sufficient to disrupt a circuit or loose connections. Loose connections can cause open circuits, or in some cases, electrical fires. However, the large number of terminations in a nuclear plant and the relatively few reports of loose connections indicate that loose connections are not a significant aging effect. Loss of dielectric isolation is most likely during accident conditions and is rarely reported during normal operation.

Coaxial connections are typically constructed of metal with an organic insulator that might be Teflon. In a coaxial connection, the insulator is in a confined location and has mechanical separation, which provides electrical separation. Thus, although Teflon is known to be age sensitive, its application in coaxial connections appears to render the aging effect minimal.

Coaxial connections, while relatively immune to aging effects by themselves, might be vulnerable to accident environments as a result of aging effects on coaxial cable jackets. This situation could arise, for example, if coaxial cable jacket integrity were lost before or during an accident and moisture were to travel along the cable shield into the connection. This could result in decreased insulation resistance or induced voltages, with possible failure of the circuit.

Because of the materials used in terminal block construction, it is unlikely that aging would have a significant impact on accident performance. The one possible exception to this statement is that corrosion and/or dirt accumulation on the blocks might affect their performance. It should be noted that corrosion and dirt accumulation are largely ignored under current qualification requirements; the assumption is that normal maintenance would identify and correct any such degradation mechanism.

Typical examples of age-related degradation concerns associated with junctions are provided in this section. The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," SRP-LR of C.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," SRP-LR of C.0.1.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR C.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," SRP-LR of C.0.1.

VI. GENERAL INFORMATION

Most NRC information that has been disseminated regarding connections resulted from design, selection, installation, and quality assurance inadequacies, not from any aging effects.

Information Notice 82-03 discussed the requirements for keeping equipment clean, particularly terminal blocks. Dust and chemical attack are the two major ways terminal blocks become contaminated.

The major cause of failure of connectors under accident conditions is moisture-induced leakage currents to other electrical equipment or to ground. A second possible cause of failure is loosening of connections resulting in open circuits. In a harsh environment, temperature effects could cause a loose connection or make an already loose connection worse.

Heat shrink tubing and tapes are made from materials similar to cable materials and their degradation can be expected to be similar to cable materials to a significant extent. One advantage that these connections have over cable insulation is that they normally have significantly thicker insulation. However, their big disadvantage as compared to cable is that they must bond to existing insulation to form a moisture-tight seal (Ref. 1).

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "An Aging Assessment of Cables, Connectors, and Electrical Penetration Assemblies Used in Nuclear Power Plants," NUREG/CR-5461 (SAND89-2369 RV), (Draft) March 1990.

2. Institute of Electrical and Electronics Engineers, IEEE Standard 323-1974, "Qualifying Class 1 Equipment for Nuclear Power Generating Station."

C.2.3 ELECTRICAL PENETRATIONS

REVIEW RESPONSIBILITIES

Primary - LRPD

Secondary - SPLB/SELB

I. AREA OF REVIEW

A. Description

The electrical penetration assemblies (EPAs) covered by SRP-LR C.2.3 Electrical Penetrations are devices performing a safety function. They are used (to 1) extend conductors through the reactor containment structure and (2) provide a hermetic seal between the inside environment of the containment structure and the outside environment. The integrity of the hermetic seal is usually determined by monitoring the internal pressure of an inert gas placed in the area of the two sealing bulkheads. Hermetic seals between each conductor and between the inside and outside containment environments are obtained through the use of such materials as gaskets, O-rings, metals, plated metals, polymer-based rubbers, ceramic materials, high-strength and high-temperature glass (Ref. 1).

B. See Section I, "Area of Review," SRP-LR of C.0.1 for Item I.B.

C. Aging Concerns and Mechanisms

Electrical penetrations are subjected to many of the same stressors and aging mechanisms as electrical cables. The most common (normal) aging stressors for EPAs are thermal, radiation, vibration, humidity, chemical (corrosion), electrical load cycling, and maintenance damage. The abnormal aging stressors are imposed by accident conditions; the major stressor is moisture related and occurs when high-temperature, high-pressure steam causes reduced electrical isolation in circuits (Ref. 1). The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE-CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR C.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR C.0.1.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR C.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR C.0.1.

VI. GENERAL INFORMATION

VII. REFERENCE

1. U.S. Nuclear Regulatory Commission, "Aging of Cables, Connections, and Electrical Penetration Assemblies Used in Nuclear Power Plants," NUREG/CP-5461, July 1990.

C.2.4 RELAYS, CIRCUIT BREAKERS, AND SWITCHGEAR

REVIEW RESPONSIBILITY

Primary - LRPD
Secondary - SELB/SICB

I. AREA OF REVIEW

A. Description

Circuit breakers, load centers, motor control centers, power panel boards, and relays, regardless of nomenclature, are included within the scope of this review. For the purposes of this review, relays are used and associated with switchgear and circuit breaker control and protective circuits. Relays are also discussed in SRP-LR C.3.2.

B. See Section I, "Area of Review," of SRP-LR C.0.1 for Item I.B.

C. Aging Concerns and Mechanisms

This review is restricted in scope to only age-related concerns of relay, switchgear, and circuit breaker equipment. Typically, this equipment is located outside the containment in a mild environment, as defined in 10 CFR 50.49. Potential problems include protective relay failure, control relay failure, auxiliary relay failure, instrument relay failure, timing relay failure (such as used in time overcurrent applications), indication and annunciation failure, wiring and connection failures, corrosion (either electrolytic or moisture based), dielectric breakdown, internal corona, surface tracking or arcing, and circuit breaker failure. A circuit breaker failure can be failure to clear a fault, or failure to energize a load. The failure can often be attributed to root causes, such as welded contacts, overcurrent relay failure, or a jammed mechanism. Typical aging concerns related to relays, circuit breakers, and switchgear are discussed below.

1. Relays

Relays are normally located in mild environments, yet they experience significant aging stresses (Ref. 1). These stresses are generally grouped into four categories: electrical, mechanical, thermal, and environmental. Relays are subject to aging via, among other causes, coil failures, contact wear, oxidation and pitting, mechanical binding, and setpoint drift. Setpoint drift affects time-delay relays and protective relays. High thermal stresses, which can be caused by overvoltage and undervoltage conditions, cause coil burnout on continuously energized relays. Shrinkage and misalignment of plastic frames are also caused by thermal conditions. Most relays, except for those mounted next to such large machinery as diesel generators and compressors, are not subject to high level vibrations. Such vibration should have been reviewed as part of the original SRP review.

Electrical stressors applicable to relays are in two groups: inductive surges (transient) and continuing under- or overvoltage. Inductive surges cause the breakdown of coil insulation via corona and dielectric breakdown. Ohmic heating is caused by either under- or overvoltage conditions.

Mechanical stressors applicable to relays are in four groups: high cycle rate, loose connections, vibration, and lack of operation. Higher-than-design cycle rates cause excessive wear on moving parts and contacts that cause binding and misoperation. These conditions also cause coil failure, increased friction, and mechanical fatigue. Loose connections cause high resistance circuits, open circuits, and arcing. Vibration causes material fatigue, loose connections, and intermittent relay operation (open or close) when not intended. Lack of operation can cause mechanical binding by component outgassing, material set, or adherence causing binding or a stuck relay.

Thermal stresses applicable to relays lead to insulation deterioration and component failure. Excessive component heating-caused by ohmic heating, enclosure temperature rises, and elevated ambient temperatures-causes accelerated aging of coil and contact lead (if any) insulation and other nonmetallic components. Such accelerated aging is not readily determined until the component fails. However, safety systems are designed to function with such a single detectable failure.

Environmental stresses applicable to relays, such as humidity, dust, dirt, and contamination, cause open circuits, increased resistances, contact and ohmic heating, binding, and sluggish or slow (or non) operation. These stresses come from contact corrosion, current leakage, mechanical binding, and friction forces.

2. Circuit Breakers

Circuit breakers are normally located in mild environments, yet they also experience significant aging stress (Ref. 1).

These stresses are generally grouped into four categories: electrical, mechanical, thermal, and environmental. Circuit breakers are subject to aging via, among other causes, restrikes, internal short circuits, arcing to ground or between phases, improper (failure to open or close) operation, premature trips at a low current, failure to trip at high currents, and flashover.

Shrinkage and misalignment of nonmetallic components are also caused by thermal conditions. Lubricant migration can be caused by elevated temperatures. Most circuit breakers, except for those mounted next to such large machines, as diesel generators and compressors, are not subject to high vibration levels. Such vibrations may have been reviewed as part of the original licensing review. If mounted in an area subject to high vibration levels, the circuit breaker could age prematurely.

Electrical stressors applicable to circuit breakers are in two groups: overvoltage transients, including voltage spikes and lightning strikes, and fault interruption. Overvoltage transient and inductive surges can cause corona and resultant carbon tracking. Fault interruption also causes an arc which should be extinguished as designed. However, flashover can degrade the arc chutes and contacts. Carbon tracking can contaminate the circuit breaker enabling restrike and additional arcing, and potential low impedance between phases or phase to ground. Arcing also causes ohmic heating.

Mechanical stressors applicable to circuit breakers are the result of routine operation, fault clearing operation, vibration and friction. Wear, component fatigue, degraded contact area, and reduced mechanism operating force (and resultant change in operating characteristics) can result. Lack of operation can cause binding by component outgassing, material set, lubricant gelling, or adherence.

Thermal stresses lead to degraded insulation, degraded contacts, degraded arc chute, and, for molded case circuit breakers, degraded overload mechanisms.

Environmental stresses applied to circuit breakers, such as humidity, dust, dirt, and contamination, cause increased friction, oxidation, corrosion, degraded insulation, hardening or migration of lubricants, and material embrittlement.

3. Switchgear

Switchgear are assembled pieces of equipment including, but not limited to, one or more of the following: function switching, interrupting, control, instrumentation, metering, protective and regulating devices or relays, together with their supporting structures, enclosures, conductor, electrical interconnections, and accessories. The component aging mechanisms discussed above are applicable to switchgear that contains these components.

The following conditions can cause switchgear to fail prematurely (Ref. 1): ambient temperature excursions beyond design, excessive relative humidity, continuous loading factors close to maximum load, overloads, repetitive circuit breaker operation beyond design, presence of corrosive or conductive contaminants, abnormal vibration or shock, and excessive fault interrupting duty.

Electrical stressors applicable to switchgear are in four groups: inductive surges and continuing under- and over-voltage (discussed above) and overvoltage transients and fault interruption (discussed above).

Mechanical stressors applicable to switchgear are in three groups: loose connections, vibrations, and friction. Loose connections cause high resistance circuits, open circuits, and arcing. Vibration causes material fatigue, loose connections, and intermittent operation when not intended. Friction can cause reduced mechanism operating forces and resultant change in operating characteristics and binding.

Thermal stresses applicable to switchgear lead to insulation and component degradation. Exc ss component heating, enclosure temperature rises, and elevated ambient temperature accelerates aging of insulation and other nonmetallic components. Such accelerated aging is not readily determined until the component fails. However, safety-systems are designed to function with such a single detectable failure.

Environmental stresses applicable to switchgear, such as humidity, dust, and contamination cause open or short circuits, increased circuit resistance, contact and ohmic heating, binding, increased mechanism friction, oxidation, corrosion, degraded insulation, hardening or migration of lubricants, and material embrittlement.

Typical examples of age-related degradation concerns associated with relays, circuit breakers, and switchgear are provided in this section. The areas of aging concern for a facility should be reviewed to evaluate the cacceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR C.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR C.0.1.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR C.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR C.0.1.

VI. GENERAL INFORMATION

Relays, switchgear, and circuit breakers are components of the electric power system. Many components of the electric power system receive adequate attention during routine surveillance programs to ensure a high level of reliability (Ref. 1).

Infrared heat scanners can be used to identify overheated connections, contacts, or components. Audible discharges can be used to identify potential age-related degradation of the insulation material. Cleanliness of the switchgear cubicle is important. Dry-type transformers used in load centers are addressed in the SRP-LR Section C.2.6.

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "Nuclear Plant Aging Research: The 1E Power System," LeRoy C. Meyer and Jerald L. Edson, NUREG/CR-5181.

2. Institute of Electrical and Electronics Engineers, IEEE Standard 308-1974, "Criteria for Class 1E Power Systems for Nuclear Power Generating Stations."

C.2.5 TRANSFORMERS

REVIEW RESPONSIBILITY

Primary - LRPD
Secondary - SELB

I. AREA OF REVIEW

A. Description

Transformers are used extensively in nuclear power stations to transfer power by electromagnetic induction between circuits of different voltage levels. Transformers vary in size from the large station service transformers, main transformers, and unit auxiliary transformers to small instrument and control transformers. The scope of this review does not include instrument and control transformers, which are reviewed under the appropriate system SRP-LR review. This review covers oil-immersed transformers and dry-type transformers, where the secondary voltage is equal to or greater than 120-V ac.

B. See Section I, "Area of Review," of SRP-LR C.0.1 for Item I.B.

C. Aging Concerns and Mechanisms

Typical examples of age-related degradation concerns associated with transformers are covered in this section. The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

1. Oil-Immersed Transformers

An insulating oil, such as mineral oil or Askarel, is used for both insulation and cooling in an oil-immersed transformer. Askarel is a group of synthetic, fire-resistant, chlorinated, aromatic hydrocarbons that are used for insulating properties in electrical liquid-cooled equipment. The service life of oil-immersed transformers is affected by the condition of the insulating oil, long-term core exposure to moisture or oxygen, external short circuits, overloading, arcing or flashovers, bushing design, and hot spots. System transient faults and transformer maintenance also affect the life of the transformer.

2. Dry-Type Transformers

Dry-type transformers can be cooled by natural or forced air circulation. An insulating gas, such as air, nitrogen, or fluorocarbon, may be used. The service life of dry-type transformers is affected by the integrity of the magnetic case, bushing design, moisture seal cracking, corrosion, overheating, insulation breakdown, and lead fracture. System transient faults and transformer maintenance also affect the life of the transformer.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR C.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR C.0.1.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR C.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR C.0.1.

VI. GENERAL INFORMATION

Regulatory Guide 1.32 (Ref. 2) addresses Class 1E power systems for nuclear power stations. Transformers are an essential part of these power systems. Regulatory Guide 1.32 endorses IEEE Standard 308-1974 (Ref. 1). Section 7.4 of this industry standard recommends testing to detect the deterioration of equipment toward an unacceptable condition.

VII. REFERENCES

1. Institute of Electrical and Electronics Engineers, IEEE Standard 308-1974, "Criteria for Class 1E Power Systems for Nuclear Power Generating Stations."

2. U.S. Nuclear Regulatory Commission, "Criteria for Safety-related Electric Power Systems for Nuclear Power Plants," Regulatory Guide 1.32.

3. U.S. Nuclear Regulatory Commission, "Nuclear Plant Aging Research: The 1E Power System," LeRoy C. Meyer and Jerald L. Edson, NUREG/CR-5181.

C.2.6 SOLENOID-OPERATED VALVES

REVIEW RESPONSIBILITY

Primary- LRPD
Secondary- SELB

I. AREAS OF REVIEW

A. Description

A solenoid-operated valve, for the purposes of this review, is defined as an electric-actuated device that consists of an electromagnet, with an energizing coil approximately cylindrical in form, an armature, whose motion is reciprocating within and along the axis of the coil, a linkage, and a valve. For the purposes of this review, there are two types of solenoid-operated valves. The first is a direct operating valve. The second is a pilot valve that controls air or another pressurized fluid which then controls the process valve. They are essentially identical in form.

B. See Section I, "Area of Review," of SRP-LR C.0.1 for Item I.B.

C. Aging Concerns and Mechanisms

This review is restricted in scope to only age-related concerns of solenoid-operated valves. This equipment is located both outside the containment in a mild environment during accident conditions and inside the containment in a potentially harsh environment. Potential problems include solenoid failure; valve failure; seal failure; mechanical binding of the linkage, plunger, or valve mechanism; wiring and connection failures; corrosion, either electrolytic or moisture based; dielectric breakdown; internal corona; and surface tracking or arcing.
The failure can be a failure to open, a failure to return to normal position, an incomplete stroke, an oscillating motion, or a pressure boundary failure. Typical examples of age-related degradation concerns associated with solenoid-operated valves are given in this section. The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR C.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR C.0.1.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR C.0.1.

V. IMPLEMENTATION

See Section V, "Implementation, of SRP-LR C.0.1.

VI. GENERAL INFORMATION

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "Nuclear Plant Aging Research: The 1E Power System," LeRoy C. Meyer and Jerald L. Edson, NUREG/CR-5181.

C.2.7 ELECTRIC MOTORS

REVIEW RESPONSIBILTIES

Primary - LRPD

Secondary - SELB

I. AREA OF REVIEW

A. Description

Electric motors that are important to license renewal are used in systems needed for safe operation of the plant during both normal and accident situations. They vary in size from thousands of horsepower to a fraction of horsepower and are used as the motive force for, as examples: normal and emergency core cooling, feedwater and condensate water, ventilation, positioning various devices, movement of control rods, driving compressors, makeup and exhaust air, makeup and letdown water, transfer of various other liquids and gases throughout the plant both inside and outside of the containment, and also for such tasks as driving strip-chart recorders and small cooling fans for control devices (Ref. 1).

B. See Section 1, "Area of Review," of SRP-LR C.0.1 for Item I.B.

C. Aging Concerns and Mechanisms

The main aging concerns and failure mechanisms for electric motors are electrical, mechanical, chemical, thermal, humidity, and radiation (Ref. 2). The electrical aging of motors affects the dielectric strength of the insulating material covering the conductors and windings of the motor. The mechanical degradation affects the bearings, the mechanical strength of the windings and the insulating material, the inherent strength of the material, the loosening of rotor bars and the lamination in the rotor and stator. Chemically induced degradation causes lube oil decomposition, degradation of the insulating properties of materials, and overall corrosion of the motor.

The thermal, humidity, and radiation aging mechanisms affect the performance of motors over a long period of time. If these mechanisms occur at excessive levels, the insulating material is affected the most.

The electrical motor components may degrade through wear exacerbated by internal vibration, at rotating surfaces (e.g., brushes, commutators, bearings, shafts, seals), and at bushings. Vibration may also affect mechanical components in motor installations which includes shafts, supports, bolts, anchors, braces, brackets, terminals, lugs, connectors, and wiring.

The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR C.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP LR C.0.1.

IV. FINDINGS

See Section V, "Findings," of SRP-LR C.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR C.0.1.

VI. GENERAL INFORMATION

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "Improving Motor Reliability in Nuclear Power Plants," NUREG/CR-4939, November 1987.

2. U.S. Nuclear Regulatory Commission, "Operating Experience and Aging-Seismic Assessment of Electric Motors," NUREG/CR-4156, June 1985.

C.3.1 SENSORS

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SICB

I. AREA OF REVIEW

A. Description

The electronic sensors that are important to license renewal are those used in instrumentation and control (I&C) systems that have been determined to be necessary for the continued safe operation of the plant. The sensors are used in safety systems which are relied upon for maintaining the integrity of the reactor coolant pressure boundary, safe shutdown capability, and accident prevention and mitigation. The sensors of concern are commonly known as process sensors and are used to measure pressure, fluid flow, fluid level, and temperature. They are directly responsive to the value of the measured quantity.

B. See Section I, "Area of Review," of SRP-LR C.0.1 for Item I.B.

C. Aging Concerns and Mechanisms

Pressure sensors and differential pressure (flow, level) sensors may exhibit age or performance degradation through sense line blockage, seal failure, sensor failure, electronic device failures (e.g., power supplies, amplifiers, signal converters), and electronic component failure (e.g., resistors, capacitors, semiconductors, printed circuit (PC) boards, and potentiometers). Temperature sensors may exhibit age and or performance degradation through broken connectors, lead-in wire damage, resistance changes, and electronic device failure. Typical examples of age-related degradation concerns associated with sensors are given here. The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR C.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR C.0.1.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR C.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR C.0.1.

VI. GENERAL INFORMATION

The sensors that are installed in harsh environments are qualified to a specific lifetime (Ref. 1) and are replaced at or before their qualified life expires; therefore, sensors in harsh environments are affected less by the life extension program than sensors used in mild environments.

The sensors and the electronic devices used with them are or can be subjected to the aging characteristics of their individual components. Therefore, the maintenance or replacement schedules normally include considerations of the specific aging characteristics of the component materials used (Ref SRP-LR C.3.2).

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants," Regulatory Guide 1.89, June 1984.

C.3.2 ELECTRONIC COMPONENTS

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SICB

I. AREAS OF REVIEW

A. Description

The electronic components are the items from which the electronic devices are assembled or fabricated. The predominant components are capacitors, resistors, semiconductors, potentiometers, printed circuit (PC) boards, relays and switches (instrument and control), instrument transformers, and connectors (PC board and small signal). Relays are also discussed in SRP-LR C.2.4.

B. See Section I, "Area of Review," of SRP-LR C.0.1 for Item I.B.

C. Aging Concerns and Mechanisms

Some time-dependent stresses that can cause degradation within the electronic components, either separately or in combination with other stressors, include vibration, electrical stressors, thermal stressors, corrosion, erosion, embrittlement, wear, maintenance, testing, and fatigue. Another factor involved in stress is: are the main stressors electrical and/or thermal. Typical examples of age-related degradation concerns associated with electronic components are given below. The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA. Some details of component degradation follow (Ref. 1):

1. Capacitors may suffer a loss of capacitance and dielectric breakdown by overheating from internal stresses caused by high ambient and internal temperatures and excessive electrical conditions. They may also be subjected to an open circuit or lead wire breakage caused by continued vibration.

2. Resistors may suffer a change in their ohmic value by overheating from internal stresses caused by high ambient and internal temperatures and excessive electrical conditions. They may also be subjected to an open circuit or lead wire breakage caused by continued vibration.

3. PC boards (modules) may experience a change in their output as a result of temperature cycling and excessive electrical voltage. They may also experience lifted tracks and degraded solder joints upon continued vibration.

4. Semiconductors may experience a semiconductor barrier breakdown resulting in short or open circuits caused by high ambient or internal temperatures and excessive electrical voltage. They may also be subjected to lead wire breakage caused by continued vibration.

5. Potentiometers may experience a short circuit, an open circuit, a short between adjacent spool wires, or lead wire breakage as a result of thermal degradation, corrosion, or excessive vibration.

6. The instrument and control relays and switches may fail because of failed-open or failed-closed contacts caused by oxidation and pitting of contact surfaces, or excessive vibration. The relay coil can be subjected to open-circuit conditions and insulating material degradation caused by corrosion and/or high thermal stresses.

7. The instrument transformers may experience a primary to secondary short circuit, a short between adjacent wires in the same winding, or failure of the dielectric material caused by high operating temperatures, excessive electric field intensity, vibration, thermal shock, or any combination of the stressors.

8. The PC board and small signal connectors may suffer from separation of contacts, degradation of insulating material, increase in contact resistance, and lost signals caused by corrosion, peeling of the plating, or high temperature and humidity conditions.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR C.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR C.0.1.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR C.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR C.0.1.

VI. GENERAL INFORMATION

The main stressors in these components are electrical stress and/or thermal stress. But some time-dependent stresses that can cause degradation within the electronic components either separately or in combination with other stressors include vibration, electrical stressors, thermal stressors, corrosion, erosion, embrittlement, wear, maintenance, testing, and fatigue.

Electrical stressors are induced in the insulating materials used in the fabrication of electronic and electro-mechanical components and are at their worst during switching operations and during accident situations.

Thermal stressors are induced in electronic components because the various materials of construction have properties that vary with temperature at different rates.

VII. REFERENCE

1. NP-1558, EPRI Project 890-1, "A Review of Equipment Aging Theory and Technology," September 1980.

C.3.3 ELECTRONIC DEVICES

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - SICB

I. AREAS OF REVIEW

A. Description

The electronic devices used in systems that are important to license renewal include electronic isolation devices, signal processors, controllers, signal converters, and bistables.

Isolation devices used in instrumentation and control (I&C) circuits function in such a way that voltage and current faults applied to the device's non-Class 1E side will not degrade below an acceptable level the operation of the safety-related circuit connected to the device's Class 1E side. Isolation devices are used to separate safety circuits from non-safety circuits (isolation between control circuits and protection circuits), redundant circuits within the same safety division (isolation between safety circuits A and C), and one safety division from another safety division (isolation between Divisions A and B). Signal processors and controllers act upon a signal in such a way that it is recognizable by other devices, scaled so that the signal has a meaningful relationship with the measured variable and thus can be used to control other equipment. These devices can also provide for sensor excitation.

Signal converters and bistables act a upon a signal to cause a change of state of the signal, that is, a signal may be changed from ac to dc, from a high to a low signal, or from an analog to a digital signal.

Indicators and recorders are display devices used to display the status of a signal. The display may be in analog or digital form, temporary (indicators), or permanent (recorders).

B. See Section I, "Area of Review," of SRP-LR C.0.1 for Item I.B.

C. Aging Concerns and Mechanisms

Random failure is the major failure mode of electronic devices. This is mainly caused by the aging characteristics of the specific components used in the design of the devices. Therefore, the maintenance or replacement schedules should include considerations of the specific aging characteristics of the component materials used (See SRP-LR C.3.2). Typical examples of age-related degradation concerns associated with electronic devices are provided in this section. The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR C.0.1.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of SRP-LR C.0.1.

IV. FINDINGS

See Section IV, "Findings, of SRP-LR C.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR C.0.1.

VI. GENERAL INFORMATION

The main stressors associated with these components are electrial stress and/or thermal stress. But some time-dependent stresses that can cause degradation within the electronic devices either separately or in combination with other stressors include vibration loads, electrical stressors, thermal stressors, corrosion, erosion, embrittlement, wear, maintenance, testing, fatigue, shrinkage, and creep.

Electrical stressors are induced in the insulating materials used in the fabrication of electronic and electromechanical devices and are at their worst during switching operations and during accident situations.

Thermal stressors are induced in electronic devices because the various materials of construction have properties that vary with temperature at different rates.

VII. REFERENCES

C.4.0 CIVIL STRUCTURE

REVIEW RESPONSIBILITIES

Primary - LRPD
Secondary - ESGB

I. AREAS OF REVIEW

A. Description

This section addresses the aging factors and environmental stressors that can degrade structures of importance to license renewal (Category I structures). These structures include: buildings that house and support systems/components important to license renewal such as reactor buildings, control rooms, control buildings, radwaste building, diesel generator buildings, etc.; primary containment (addressed in SRP-LR B.3.1); major load carry features such as the RPV support structure (addressed in SRP-LR C.1.6 coolant reservoirs and intake structures (e.g., spray ponds); refueling canal and fuel storage facilities; concrete tanks; and elevated release stacks. Structures important to license renewal are typically constructed of reinforced concrete, post-tensioned concrete, structural steel, or a combination of these, and are, in general, designated as those structures required to withstand a design-basis event such as an earthquake or large-scale loss-of-collant-accident (LOCA). For specific applications at certain plants, masonry block walls must also be considered.

Potential age-related degradation mechanisms for the various structural components include (Refs. 1 and 2):

Concrete

- o Freeze-thaw
- o Leaching of calcium hydroxide
- o Aggressive chemicals
- o Reactions with aggregates
- o Corrosion of embedded steel

Elevated temperature

- o Irradiation
- o Creep
- o Shrinkage
- o Abrasion and cavitation
- o Cracking of masonry block walls

Reinforcing Steel

- o Corrosion
- o Elevated temperature
- o Irradiation

Piles

- o Corrosion

Structural Steel

- o Corrosion
- o Elevated temperature
- o Irradiation

Stainless Steel Liner Plate

- o Corrosion, IGSCC
- o Elevated temperature
- o Irradiation

Miscellaneous Degradation Issues

- o Fatigue
- o Cathodic protection effects on bond strength
- o Settlement

While the majority of these potential degradation mechanisms are not significant at all the plants, each should be assessed on a plant-by-plant basis before dismissint it.

B. See Section 1, "Areas of Review," of SRP-LR C.0.1 for Item I.B.

C. Aging Concerns and Mechanisms

The long-term performance of concrete structures may be influenced by the presence of aggressive environments and the occurrence of degradation. The relevant degradation factors are those that affect the material systems and manifest themselves in such a manner as to reduce structural integrity or decrease structural margins, for example, loss of concrete strength due to leaching of the calcium-containing products, cracking of concrete due to alkali-silica reactions or sulfate attack, and corrosion of the mild steel reinforcement or post-tensioning systems. Synergism of these factors is also important as it can accelerate the degradation process, for example cracking of concrete due to alkali-silica reaction and reinforcement corrosion due to chloride penetration. Concrete materials are susceptible to degradation through chemical and physical attack. Masonry block walls can develop cracks as a result of such movements, as foundation settlement, thermal expansion. Metallic materials can occur as a result of corrosion, elevated temperature, irradiation, or fatigue.

SRP-LR C.1.6 provides review guidance for radiation embrittlement of structural steels. Stainless steel liner plates (e.g., fuel pool liners) are subject to general corrosion and IGSCC.

Typical examples of age-related degradation concerns associated with civil structures are discussed above. The areas of aging concern for a facility should be reviewed to evaluate the acceptability of the licensee's program to manage potential aging mechanisms. Site-specific conditions and experience are documented in the IPA.

II. ACCEPTANCE CRITERIA

See Section II, "Acceptance Criteria," of SRP-LR C.0.1. Items A through C.

D. Observed and projected age-related degradation shall be such that despite its effects, the performance of the structure will remain within the design envelope as represented by the design calculations, safety analyses, and industry codes and standards to which the licensee is committed.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of C.0.1 Items A through D.

E. The reviewer should confirm that the licensee performs a one-time inspection of civil structures important to license renewal which shall include, as a minimum:

1. Visually inspect above-grade buildings, structures, tanks, and embankments. Assess observed and projected cracks, corrosion, settlement, and other forms of age-related degradation on the capability of the structure to accommodate its design loads.

2. Inspect condition of structures, anchors, and protective coatings submerged in spray ponds, cooling tower basins, intake structures, BWR suppression pools, and other water pools. Assess effects of observed and projected corrosion and/or abrasion/cavitation damage on the capability of structures to accommodate design loads.

3. Assess chemical environment of foundations and structures below grade. Visually inspect at grade elevation for indications of settlement (may be covered by existing program). Assess observed and projected degradation on capability of structures to accommodate design loads.

4. Assess condition of prestressing elements of all prestressed structures that are not part of the primary containment structure.

The licensee's IPA should address all of the potential age-related degradation mechanisms discussed in paragraphs I.C and VI of this section and any potential age-related degradation mechanisms experienced during the current license period for civil structures important to license renewal. Results of the above one-time inspection shall be addressed and ongoing monitoring programs, analyses and proposed corrective actions defined to provide assurance that these structures will perform their required functions during the license renewal period. For those potential degradation mechanisms that the licensee proposes to eliminate from consideration, an adequate technical justification shall be provided.

IV. FINDINGS

See Section IV, "Findings," of SRP-LR C.0.1.

V. IMPLEMENTATION

See Section V, "Implementation," of SRP-LR C.0.1.

VI. GENERAL INFORMATION

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "Class I Structures License Renewal Industry Report," NUMARC 90-006, June 1990.

2. U.S. Nuclear Regulatory Commission, "Concrete Component Aging and Its Significance Relative to Life Extension of Nuclear Power Plants", D.J. Naus, Oak Ridge National Laboratory, NUREG/CR-4652, (ORNL/TM-10059), September 1986.