NUREG-1299

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

Draft Report for Comment

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

November 1990

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ABSTRACT

The Standard Review Plan for the Review of License Renewal Application for Nuclear Power Plants (SRP-LR) is to be used by the NRC staff when performing safety reviews of applications for the renewal of power reactor licenses. The use of the SRP-LR when reviewing license renewal applications provides a framework for the staff to determine whether or not (1) the application is sufficent to allow the timely renewal provisions of 10 CFR 2.109 to apply, (2) systems, structures, and components important to license renewal have been identified, (3) significant age-related degradation has been identified and its effects evaluated, and (4) programs for age-related degradation management have been or will be so implemented that the current licensing basis will be maintained during the renewal term. The draft SRP-LR has been developed to enable the staff to identify areas and issues requiring review, and provides acceptance criteria to assist the reviewers.

CONTENTS

PART C: GENERIC LICENSE RENEWAL TOPICS

GLOSSARY OF ABBREVIATIONS

ACI - American Concrete Institute
ADS - automatic depressurization system
AFW - auxiliary feedwater
AFWS - auxiliary feedwater system
AISC - American Institute of Steel Construction
API - American Petroleum Institute
ASME - American Society of Mechanical Engineers
ATWS - anticipated transient without scram
AWWA - American Water Works Association

BWR - boiling-water reactor

CAS - compressed air system
CB - citizen band
CCF - common cause failure
CCGCS - containment combustible gas control system
CCW - component cooling water
CFR - Code of Federal Regulations
CFS - core flood system
CLB - current licensing basis
CRD - control rod drive
CRDM - control rod drive mechanism
CRDS - control rod drive system
CST - condensate storage tank
CVCS - chemical and volume control system
CVS - containment ventilation system

DBA - design-basis accident

EAS - essential auxiliary supporting system
ECCS - emergency core cooling system
EDG - emergency diesel generator
EEP - established effective program
EMCB - Materials & Chemical Engineering Branch
EPA - electrical penetration assembly
EPRI - Electric Power Research Institute
EPS - essential power system
EQ - equipment qualification
ESF - engineered safety features
ESFAS - engineered safety features actuation system
ESGB - Structural & Geosciences Branch
ESW - essential service water

FCU - fan cooler unit
FPS - fire protection system
FSAR - final safety analysis report
FWS - fire water system

HAZ - heat-affected zone
HCU - hydraulic control unit
HEPA - high-efficiency particle absorber
HEPA - high-energy particulate air
HPCI - high-pressure coolant injection
HPCS - high-pressure core spray
HVAC - heating, ventilation, and air conditioning

I&C - instrumentation and control
IEEE - Institute of Electrical and Electronics Engineers
IGA - intergranular attack
IGSCC - intergranular stress-corrosion cracking
ILR - important to license renewal
IPA - integrated plant assessment
ISI - inservice inspection
IST - inservice testing

LER - licensee event report
LOCA - loss-of-coolant accident
LPCS - low-pressure core spray
LRPD - License Renewal Project Directorate
LWR - light-water reactor

MIC - microbiologically influenced corrosion
MOV - motor-operated valve
MPS - main power system
MSLB - main steamline break

NDE - nondestructive examination
NFPA - National Fire Protection Association
NIST - National Institute of Standards and Technology
NMS - neutron monitoring system
NPAR - nuclear plant aging research
NPRDS - nuclear plant reliability data system
NPS - nonessential power system
NRC - Nuclear Regulatory Commission

OGS - offgas system

PC - printed circuit
PCP - primary coolant pump
PCS - primary containment structure
P&ID - piping and instrumentation diagram
PTS - pressurized thermal shock
PWR - pressurized-water reactor

RCIC - reactor core isolation cooling
RCS - reactor control system
RCS - reactor coolant system
RF - random failure
RG - regulatory guide
RHR - residual heat removal
RMS - radiation monitoring system
RPS - reactor protection system
RPV - reactor pressure vessel
RRS - reactor recirculation system
RWCS - reactor water cleanup system

SBGTS - standby gas treatment system
SCs - structures and components
SCC - stress-corrosion cracking
SCR - silicon-controlled rectifier
SELB - Electrical Systems Branch
SER - safety evaluation report
SFP - spent fuel pool
SICB - Instrumentation & Control Systems Branch
SLCS - standby liquid control system
SP - suppression pool
SPDS - safety parameter display system
SPLB - Plant Systems Branch
SRP-LR - Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants
SRV - safety relief valve
SRXB - Reactor Systems Branch
SSCs - systems, structures, and components
SWS - service water system

TGSCC - transgranular-assisted stress-corrosion cracking

UHS - ultimate heat sink
USAR - updated safety analysis report
UT - ultrasonic test(ing)

STANDARD REVIEW PLAN FOR THE REVIEW OF LICENSE RENEWAL
APPLICATIONS FOR NUCLEAR POWER PLANTS (SRP-LR)

PART A: GENERAL INFORMATION

BACKGROUND

The Nuclear Regulatory Commission (NRC) regulations 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 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 based 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 again at the time of license renewal. Therefore, 10 CFR Part 54 focuses 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 these principles, the Standard Review Plan for the Review of License Renewal Applications (SRP-LR) is based on the staff's position that reasonable assurance must be provided to demonstrate that license renewal will not lead to age-related degradation 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).

PURPOSE AND SCOPE

This review plan 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 in 10 CFR Part 54. The plan parallels Regulatory Guide DG-1009, "Standard Format and Content of Technical Information for Applications to Renew Nuclear Power Plant Operating Licenses," which guides 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 renewal of an operating license.

Regulatory Guide DG-1009 gives 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 this review plan 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 this plan 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, increasing 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. Although guidance given here represents approaches that are acceptable to the staff, 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 "Requirements of the License Renewal Rule," the section that follows. Criteria for reviewing environmental concerns to satisfy 10 CFR 54.23 will be addressed in revisions to 10 CFR Part 51.

The staff review of an application for renewal of an operating license is not intended to be a review of the current licensing basis. Therefore, guidance offered in this plan differs from that given in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants." The emphasis here is on providing guidance to staff reviewers 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 period requested in the application.

The manner in which the staff applies this plan 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 that serves as a frame on which to produce a more detailed document. This review plan is a living document to be revised as experience is gained during the review of the lead plant applications and industry technical reports. The staff will consider comments and suggestions for improvement; these 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 be sent to the same address.

ORGANIZATION

The SRP-LR is divided into three parts. Part A provides the general context for the purpose and scope of the SRP-LR. In the next section of Part A, "Requirements of the License Renewal Rule," the staff 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. The final section of Part A, "Sufficiency of the License Renewal Application," assigns review responsibilities and describes the criteria for making the preliminary determination of whether or not a license renewal application is sufficiently detailed for the staff to conduct the technical review. Appendix A.1 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. A systems level review is necessary to properly integrate the total facility review. The first section SRP-LR (B.0.1) covers generic license renewal review guidance for systems important to license renewal in general. The remaining SRP-LRa 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 licensee's integrated plant assessment will determine the actual systems important to license renewal for its particular facility and appropriate management programs for age-related detradation.

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, will 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 interfere with 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:

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

- 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 mechanisms that must be evaluated as part of the particular system being reviewed.

- 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 specific 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.

- 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.

- Findings: This subsection presents the types 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.

- 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 the SRP-LR.

- General 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 (NPAR) Program that may be useful in understanding aging mechanisms and ways for managing aging for a particular system during the period of license renewal. This subsection serves solely as background material and should not be used as review criteria or review procedures.

- References: This subsection lists documents 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.

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 that demonstrates that age-related degradation of the facility's SSCs has been identified, evaluated, and accounted for as needed to ensure 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. 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. Also identify the SCs that are constituent elements of the SSCs important to license renewal included in the initial list.

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 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 54.3(a). 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 degradation, 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 degradation is found to be a significant factor, then the applicant must describe and provide the basis for actions taken or to be taken to respond to aging degradation.

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 on 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. Safety functions are generally the result of a system's operation and 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 identifying SCs that need to be evaluated for age-related degradation.

REVIEW OF THE LICENSE RENEWAL APPLICATION FOR SUFFICIENCY

Title 10 CFR 2.109 allows the licensee's current license to remain in effect during the time the staff is reviewing the license renewal application, provided the licensee files a sufficient application at least 3 years before the current operating license is scheduled to expire. Although the staff expects to complete its review of a renewal application before the existing license expires, in some instances the staff may need more time to fully evaluate the technical adequacy of the application and make a final determination on the application. Therefore, the staff will review the application initially to determine if the application is sufficient to commence the detailed review for operating license renewal.

It is important to note that the review for sufficiency is neither a detailed, in-depth review of the technical aspects of the application nor a simple check-off sheet. A sufficiency review determines if the applicant has made a reasonable effort to provide adequate justification for its application. Acceptance of the application at this point does not preclude requesting additional information as the review proceeds; nor does it predict the NRC's final determination (acceptance or rejection). The staff expects to complete its sufficiency review generally within 60 days of receipt of the application. The end result of the sufficiency review will be a letter to the applicant which either (1) rejects the application as containing insufficient detail to initiate staff review or (2) accepts the application as sufficiently complete to allow the provisions of 10 CFR 2.109 to apply during the review. In this letter the staff may also reject portions of the application that are outside the scope of license renewal (i.e. not associated with age-related degradation). If it decides to pursue further these additional requests, the applicant must submit an application for a separate license amendment.

Review Responsibilities

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

Review Procedures

Regulatory Guide DG-1009 offers guidance to the applicant on the content of the technical information that should be submitted as part of the license renewal application. The project manager should use the checklist in Figure A-2 as guidance in determining sufficiency. 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-1. Overall perspective of the review process

NOTES:

1. LRPD is the primary reviewer listed in all SRP-LRs of this plan. 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 individual SRP-LRs that apply to the system or component under review.

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

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

CHECKLIST FOR REVIEWING SUFFICIENCY OF AN 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 application for license renewal. A check in the "No" column without sufficient explanation may be justification for rejecting the application. A sufficiency review is not a detailed technical review; however, an applicant should have made a reasonable effort to justify its 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		
i. Where incorporated	____	____
ii. Board of directors and principal officers	____	____
iii. Ownership details	____	____
E. Class of license	____	____

Figure A-2 Checklist for reviewing sufficiency of an application for license renewal

	Yes	No
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 describes the integrated plant assessment (IPA) [54.21(a)] and includes the following:		
A. Methodology to identify all SSCs important to license renewal [54.21(a)(1)]	___	___
B. List of all SSCs important to license renewal [54.21(a)(1)]	___	___
C. List of structures and components (SCs) that are constituent elements of SSCs important to license renewal (optional) [54.21(a)(2)]	___	___
D. Methodology, including selection criteria, to identify those SCs from 9.C that contribute to the performance of the safety function of an SSC important to license renewal or whose failure could interfere with an SC performing its intended safety function [54.21(a)(2)]	___	___
E. List of SCs identified in 9.D [54.21(a)(2)].	___	___
F. Methodology to identify those SCs identified in 9.E that are subject to an established effective program as defined in 54.3(a) [54.21(a)(3)]	___	___

Figure A-2 (continued)

	Yes	No
G. List of those SCs identified in 9.F [54.21(a)(3)]	____	____
H. Established effective programs for the SCs identified in 9.G [54.21(a)(3)]	____	____
I. Information of the following type for SCs on the list from 9.E and not on the list from 9.G [54.21(a)(4)]:		
i. Describe and provide basis for action taken to manage age-related degradation; or	____	____
ii. Describe and provide basis for action to be taken to manage age-related degradation; or	____	____
iii. Demonstrate by evaluation that the age-related degradation is not significant with respect to the current licensing basis.	____	____
10. List is provided of all plant-specific exemptions granted pursuant to 10 CFR 50.12 and reliefs granted pursuant to 10 CFR 50.55a [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)]	____	____
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, or limitations on plant operations during the renewal term due to age-related degradation concerns.	____	____

Figure A-2 (continued)

	Yes	No
14. List of documents is provided identifying portions of current licensing basis that are relevant to the IPA and a brief description is provided of the administrative controls for and l, ation of these documents [10 CFR 54.37]	____	____
15. Environmental report complies with the requirements of subpart A of 10 CFR Part 51	____	____
16. List is provided of the elements of the application requesting NRC approval that are not germane to license renewal (i.e., are not covered by the provisions of 10 CFR Part 54). These elements must be rejected and the licensee told to file a separate application if it still desires these changes to its license.	____	____

Figure A-2 (continued)

APPENDIX A.1

SELECTING STRUCTURES AND COMPONENTS FOR EVALUATING 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 facility is 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 this review plan. Review guidance to the staff is provided below.

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - All technical review branches

I. AREAS OF REVIEW

The applicant 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 offers guidance to the staff in reviewing methodologies for preparing the required lists.

II. ACCEPTANCE CRITERIA

The acceptability of the applicant'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. SSCs Important to License Renewal

The applicant has submitted a list of SSCs important to license renewal and has described the methodology for identifying these SSCs. If the applicant 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 methodology has been used, the applicant should provide specific detail concerning how SSCs were selected.

B. SCs That Are Constituent Elements of the SSCs Important to License Renewal

The applicant 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 Futher Evaluation of Age-Related Degradation

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

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

2. The failure of the structure or component could interfere with an SSC important to license renewal performing its intended safety function.

The applicant 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 adhere to the following review procedures to assess the adequacy of the applicant'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 in one way for applicants referencing methodologies that have already been approved by the NRC, and in another way for applicants 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 applicant's program for selecting SSCs and SCs based on the acceptance criteria in Section II "Acceptance Criteria" above.

A. SSCs Important to License Renewal

The reviewer should confirm that the 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):

1. safety-related SSCs

2. all SSCs used in a safety analysis or plant evaluation for the licensing basis

3. any, including non-safety-related, SSCs whose failure could keep safety-related equipment from satisfactorily performing required safety functions

4. 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 applicant paying particular attention to deviations from previously approved approaches or to new approaches
developed by the applicant. 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 offers reasonable confidence that all SSCs important to license renewal have been properly identified. The reviewer should examine the plant's final safety analysis report (FSAR) and other licensing-basis documents to the extent necessary. The review should ensure that all safety-related systems as well as non-safety-related systems, such as overfill protection systems and systems to mitigate anticipated transients without scram have been identified as important to license renewal. For methodologies not previously approved by the 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 applicant's list. Although this generic list is likely to contain some SSCs 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 applicant has described the methodology for converting SSCs important to license renewal into SCs 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 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 have not been approved by the NRC.

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

C. SCs Requiring Evaluation of Age-Related Degradation

The reviewer should confirm the applicant 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 applicants 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 applicant 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 Parts B and C of this review plan.

The reviewer should confirm the applicant has identified SCs which 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 Part A of this review plan, 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 interfere with SSCs important to license renewal performing their intended safety functions. Although SCs of this nature need not be included in the list of SCs requiring further evaluation, the reviewer should confirm that the applicant has supplied the justification for eliminating them from further consideration. The reviewer should check the implementation of the applicant'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 to which they supplement the list of SCs for further evaluation.

IV. FINDINGS

The reviewer should determine and verify the applicant has provided sufficient information and the review supports the conclusion to be included in the staff's SER that: (1) the applicant's methodology for identifying SSCs important to license renewal and for identifying SCs requiring aging evaluation is acceptable and (2) the applicant'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 this review plan, the staff plans to use the methods described herein in conducting its review.

VI. GENERAL INFORMATION

VII. REFERENCES

PART B: PLANT SYSTEMS

Part B of this review plan addresses the review of various systems requiring evaluation of age-related degradation. A systems level review is necessary to properly integrate a total review of the facility. Potential degradation mechanisms depend on the environment, which is a function in the system design. Additionally, safety functions are generally achieved on a system basis. To determine if an established program is effective for managing age-related degradation, the effects on the system must be evaluated. Key plant systems that are likely to be identified as important to operating license renewal have been included in Part B. Plant-specific analysis could identify additional systems as important to license renewal that are not treated separately in Part B, such as main steam, turbine generator, and turbine bypass systems. These additional systems are to be reviewed in accordance with SRP-LR B.0.1. Part C of this review plan provides guidance for reviewing the constituent structures and components of all systems important to license renewal.

SRP-LR

B.0.1 SYSTEM REVIEW CRITERIA

SRP-LR B.0.1, provides information and criteria that are applicable to all systems important to license renewal. The standard acceptance criteria, review procedures, findings, and implementation applicable to each system are given below. The remaining SRP-LRs contain the information and criteria that are applicable to selected specific systems and include system descriptions, requirements, and information unique to a particular system.

Systems not addressed in a section on an individual SRP-LR should be reviewed according to the generic guidance given here in SRP-LR B.0.1.

I. AREAS OF REVIEW

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

B. The SRP-LR addresses the 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 applicant has conducted an integrated plant assessment (IPA) to identify potential age-related degradation of systems, structures, and components and to evaluate the adequacy of its programs to identify and mitigate age-related degradation for the renewal period.

The FSAR supplement supporting license renewal lists (1) systems, structures, and components identified as important to license renewal and (2) structures and components requiring evaluation of age-related degradation. The reviewer's safety evaluation for the subject system will be found in the corresponding section of the safety evaluation report (SER) supporting license renewal.

The review of issues contained in this review plan is not intended to constitute a review of the existing licensing basis. The actual licensing basis for an individual plant is contained, in part, in the FSAR specific to that facility, and the NRC staff documented its review of the FSAR in the SER it prepared to support 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 SRP-LR on the specific system for a discussion of the typical age-related degradation concerns 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; industry codes and standards*; root-cause analysis; failure-mode analysis; equipment performance history; NRC branch technical positions; approved topical and other industry reports; vendor criteria; and NRC 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 that may require replacement of selected parts. The applicants may have to review a previously completed analysis if aging issues raise questions concerning the analysis (including its assumptions). The acceptability of the applicant's proposed program for identifying, monitoring and mitigating the effects of age-related degradation in the subject system will be based on the following criteria:

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

1. The applicant has listed all structures and components within the boundary of this system that contribute to the performance of a safety function of a structure, system, or component (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 of Part A of this review plan).

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*"Industry codes and standards" refer to the codes, standards, and specifications of such groups as the American Concrete Institute (ACI), the American Institute of Steel Construction (AISC), the American Society of Mechanical Engineers (ASME), and the Institute of Electrical and Electronics Engineers (IEEE).

2. The applicant has listed the structures and components within the boundary of this system from Item II.A.1 (above) that are subject to an established effective program. This program must continue to ensure either that the structures and components are capable of performing their safety functions during the renewal period or that, if degraded they will not interfere with 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 not be 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 applicant 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, the applicant may claim the equipment qualification (EQ) program required by 10 CFR 50.49 is an established effective program for selected electrical components. But, for a subset of these components, either extensive additional testing is required or a reanalysis (with appropriate justification documented or selected verification testing required, 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 applicant 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 period of the renewed license.

B. The applicant has identified plant-specific exemptions granted pursuant to 10 CFR 50.12, "Specific Exemptions," and reliefs granted pursuant to 10 CFR 50.55a, "Codes and Standards." The applicant should justify the continued appropriateness of 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 are otherwise related to SSCs subject to age-related degradation.

C. Additional criteria are discussed in the SRP-LRs for on specific systems, as applicable.

III. REVIEW PROCEDURES

Upon request from the primary reviewer (LRPD), the secondary review branch(es) will provide material for the areas of review identified in Section I above ("Areas of Review"). The primary reviewer obtains and uses such information as required to ensure that the review procedure is complete.

The reviewer should adhere to 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 applicant 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 period, 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 applicant's program for license renewal based on the acceptance criteria given in Section II above ("Acceptance Criteria").

A. The reviewer should confirm that an IPA has been documented and submitted which demonstrates that age-related degradation in 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 SRP-LR chapter of 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 Part A of this review plan to ensure it has been adequately applied to this system.

B. The reviewer should verify that the applicant has presented information which demonstrates acceptable performance from an aging perspective for each specific components and structures within the system in an established effective program. The reviewer should confirm the applicant has identified 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 the applicant demonstrates the adequacy of ongoing programs for timely mitigation of age-related degradation. The support for this determination could focus on review of operational experience, replacement or refurbishment intervals, and, as appropriate, design and manufacturer information, known aging mechanisms, and other relevant information. For structures or components not routinely replaced or refurbished, the reviewer should ensure that the applicant'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 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 for managing age-related degradation are implemented by ensuring that the plant program defines inspection methods used, inspection frequency, and replacement refurbishment frequency and meets current licensing-basis requirements.

   The reviewer should ensure the acceptance criteria are based on an industry standard or a technically acceptable report and the action taken or to be taken is timely and will restore the component or structure to a condition of acceptable performance in accordance with the current licensing basis.

C. For structures and components in the system 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 applicant'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 SRP-LRs on specific systems, as applicable.

IV. FINDINGS

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

A. The applicant's analysis of the subject system acceptably identified the structures and components requiring evaluation of age-related degradation. The generic components and structures reviewed under Part C of this review plan that are applicable to the system under review are included in this finding.

B. The applicant 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 process was identified, evaluated, and accounted for as necessary to ensure that the applicant plant's current licensing basis will be maintained throughout the period of the renewed license.

C. The applicant has proposed or is implementing an effective program for license renewal for this specific system 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, and to ensure that the activities authorized by the renewed license can be conducted in accordance with the current licensing basis over the renewed license period.

D. The applicant has 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, "Codes and Standards." The justifications for continuing the exemptions and reliefs are acceptable for the renewal period.

E. The applicant has adequately identified and justified any proposed modifications to the facility or its administrative control and plant procedures necessary to manage age-related degradation.

V. IMPLEMENTATION

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

VI. GENERAL INFORMATION

Addressed in SRP-LRs on specific systems

VII. REFERENCES

Addressed in SRP-LRs on specific systems.

B.1.0 NUCLEAR SYSTEMS

B.1.1 REACTOR PRESSURE VESSEL

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Mechanical Engineering Branch (EMEB)
Materials & Chemical Engineering Branch (EMCB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the reactor pressure vessel (RPV).

1. Description

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

The RPV support structure is discussed in SRP-LR C.1.6, "Equipment and Component Supports." RPV boundaries include all penetrations, i.e.:

a. For pressurized-water reactors (PWRs), head penetrations include those for control rod drive mechanism (CRDM) adapters, head vent, and, for those plants so equipped, upper head injection adapters. The bottom penetrations are 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, in some early RPV designs, the inlet and outlet nozzles are located near the top of the fuel assemblies.

b. For boiling-water reactors (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 and reactor internal components 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 RPV serves as a barrier to the release of fission products, forms part of the primary coolant boundary, and supports and aligns the reactor core, assemblies, and other RPV internal components. The main functions of the RPV internal components are to provide orientation and support for the reactor core and to guide and protect the control rod drive assemblies. They also (a) provide a passageway, support, and protection for any in-vessel instrumentation and (b) direct water flows as necessary.

3. System Boundaries

The RPV boundary extends to the end of each penetration nozzle, housing, or adapter. The BWR jet pumps and the normal BWR and PWR internal components are included in the system boundaries.

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 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 principal RPV degradation mechanisms and degradation sites include the following:

1. 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.

2. 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 internal attachments. The beltline region is subject to irradiation embrittlement and thermal fatigue. Closure studs are susceptible to mechanical fatigue and corrosion.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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 for Items II.A and II.B.

C. The applicant has verified using plant-specific fatigue analyses for the RPV and all safety-related RPV internal components, that the ASME Section III (Ref. 1) allowable cumulative usage factor of 1.0 will not be exceeded during the projected lifetime of the plant.

D. All of the current programs for monitoring or mitigating age-related degradation of the RPV (or both) shall remain in effect throughout the license renewal period.

Tables B.1.1-1 and B.1.1-2 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 Referer ≀ 2, that may assist in managing or mitigating the aging conc rns.

E. The following criteria for the RPV are applicable during the renewal period:

1. The acceptance criteria of 10 CFR 50.61 applies for pressurized thermal shock (PWRs only) throughout the renewal period.

2. The acceptance criteria of 10 CFR Part 50, Appendix G apply for Charpy upper-shelf energy throughout the renewal period.

3. The acceptance criteria of 10 CFR Part 50, Appendix G, and ASME Code, Section XI (Ref.3), apply for beltline inspections throughout the renewal period.

4. The acceptance criteria of ASTM Standard Practice E-185-82 (Ref. 4) apply for surveillance programs throughout the renewal period.

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 applicant is adequate for managing the effects of aging degradation. The program should include reviews, current status, evaluations, and engineering analysis to offer reasonable assurance that the applicant understands and is managing aging degradation for this important component. As a concurrent review, SRP-LR C.1.6, "Equipment and Component Supports," should be used for reviewing the RPV supports. The following elements that should be included in the applicant's RPV aging-management program, as appropriate:

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 content 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 applicant has done the following:

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

   2. Provided the additional number of transients and cycles to be used as the design basis for the extended life of the plant; for example, if the projected life is an additional 20 years for a total of 60 years, the original design-basis transients from Item III.F.1 (above) should be increased by 50 percent. For all components, calculated the cumulative usage factors for the additional increment of time.

3. Listed all cycles due to unanticipated plant transients that were not considered as design-basis events in Item III.F.1 or III.F.2 (above). For all components, calculated cumulative usage factors for these additional transients.

4. Added the cumulative usage factors calculated from Items III.F.1, III.F.1, or III.F.3 (above) to arrive at the total end-of-life fatigue-usage factors for all components. The allowable cumulative usage factor of 1.0 shall not be exceeded during the projected lifetime of the plant.

5. If the rate of actual plant cycles indicates that the design-basis cycles will be exceeded at the end of plant life, the procedures in Items III.F.1, III.F.2, and III.F.E (above) should be adjusted to account for these additional cycles. If the rate of actual plant cycles indicates that the design-basis cycles will not be exceeded at the end of plant life and Item III.F.4 (above) is satisfied, no further analysis is required for fatigue.

6. These analyses should be performed in accordance with ASME codes Section III, Subsections NB-3222.4(e) and NB-3228.5 or NG-3222.4(e) and NG-3228.3 (Ref. 1). If the total number of stress cycles is estimated to exceed 10^6, the applicant should provide the basis for appropriate design fatigue curves.

7. In these analyses, the applicant should include an evaluation of environmental effects on the fatigue crack initiation to the extent needed to show that the analyses are conservative.

8. All of these analyses should be based on elastic approaches. The use of elastic-plastic or fully plastic approaches 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 before such procedures are used.

9. The applicant should list any plant-specific history of failure due to fatigue in the reactor vessel or any internal component.

G. The reviewer should verify that the acceptance criteria for the elements of Item II.E are satisfied for the renewal period.

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

Emphasis has been placed on the study of RPV aging. Therefore, more of the following discussion pertains to the vessel rather than to the internal components.

A. Aging Concern

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 (Ref. 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. 2 and 5):

- o high-cycle thermal fatigue and mechanical fatigue at the feedwater 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 internal 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 for PWR RPVs; the greater amount of water between the reactor core and the vessel wall generally reduces the expected neutron fluence accumulated

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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 for Items II.A and II.B.

C. The applicant has verified using plant-specific fatigue analyses for the RPV and all safety-related RPV internal components, that the ASME Section III (Ref. 1) allowable cumulative usage factor of 1.0 will not be exceeded during the projected lifetime of the plant.

D. All of the current programs for monitoring or mitigating age-related degradation of the RPV (or both) shall remain in effect throughout the license renewal period.

   Tables B.1.1-1 and B.1.1-2 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 2, that may assist in managing or mitigating the aging concerns.

E. The following criteria for the RPV are applicable during the renewal period:

   1. The acceptance criteria of 10 CFR 50.61 applies for pressurized thermal shock (PWRs only) throughout the renewal period.

   2. The acceptance criteria of 10 CFR Part 50, Appendix G apply for Charpy upper-shelf energy throughout the renewal period.

   3. The acceptance criteria of 10 CFR Part 50, Appendix G, and ASME Code, Section XI (Ref.3), apply for beltline inspections throughout the renewal period.

   4. The acceptance criteria of ASTM Standard Practice E-185-82 (Ref. 4) apply for surveillance programs throughout the renewal period.

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 applicant is adequate for managing the effects of aging degradation. The program should include reviews, current status, evaluations, and engineering analysis to offer reasonable assurance that the applicant understands and is managing aging degradation for this important component. As a concurrent review, SRP-LR C.1.6, "Equipment and Component Supports," should be used for reviewing the RPV supports. The following elements that should be included in the applicant's RPV aging-management program, as appropriate:

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 content 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 applicant has done the following:

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

   2. Provided the additional number of transients and cycles to be used as the design basis for the extended life of the plant; for example, if the projected life is an additional 20 years for a total of 60 years, the original design-basis transients from Item III.F.1 (above) should be increased by 50 percent. For all components, calculated the cumulative usage factors for this additional increment of time.

3. Listed all cycles due to unanticipated plant transients that were not considered as design-basis events in Item III.F.1 or III.F.2 (above). For all components, calculated cumulative usage factors for these additional transients.

4. Added the cumulative usage factors calculated from Items III.F.1, III.F.1, or III.F.3 (above) to arrive at the total end-of-life fatigue-usage factors for all components. The allowable cumulative usage factor of 1.0 shall not be exceeded during the projected lifetime of the plant.

5. If the rate of actual plant cycles indicates that the design-basis cycles will be exceeded at the end of plant life, the procedures in Items III.F.1, III.F.2, and III.F.E (above) should be adjusted to account for these additional cycles. If the rate of actual plant cycles indicates that the design-basis cycles will not be exceeded at the end of plant life and Item III.F.4 (above) is satisfied, no further analysis is required for fatigue.

6. These analyses should be performed in accordance with ASME codes Section III, Subsections NB-3222.4(e) and NB-3228.5 or NG-3222.4(e) and NG-3228.3 (Ref. 1). If the total number of stress cycles is estimated to exceed 10^6, the applicant should provide the basis for appropriate design fatigue curves.

7. In these analyses, the applicant should include an evaluation of environmental effects on the fatigue crack initiation to the extent needed to show that the analyses are conservative.

8. All of these analyses should be based on elastic approaches. The use of elastic-plastic or fully plastic approaches 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 before such procedures are used.

9. The applicant should list any plant-specific history of failure due to fatigue in the reactor vessel or any internal component.

G. The reviewer should verify that the acceptance criteria for the elements of Item II.E are satisfied for the renewal period.

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

Emphasis has been placed on the study of RPV aging. Therefore, more of the following discussion pertains to the vessel rather than to the internal components.

A. Aging Concern

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 (Ref. 2):

- irradiation embrittlement in the beltline region
- thermal and mechanical fatigue at the inlet and outlet nozzles (penetrations)
- thermal and mechanical fatigue at the CRDM penetrations and incore instrument penetrations

The major areas of concern for the BWR RPVs are (Refs. 2 and 5):

- high-cycle thermal fatigue and mechanical fatigue at the feedwater inlet nozzles (penetrations)
- 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)
- thermal and mechanical fatigue and potential for IGSCC at the CRDM and incore instrumentation penetrations and internal component attachments
- irradiation embrittlement in the beltline region and in the reactor internal components
- 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 for 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 $5x10^{17}/cm^2$ [E-1 MeV] for BWR RPVs, compared with $4x10^{19}/cm^2$ [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 and Charpy upper-shelf energy. 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 increases 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. 5) 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 VI.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 may result 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. 1), 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. 3). Additional research is being conducted, under the NRC pressure vessel research program, to improve the life-prediction procedures. NUREG/CR-4731 (Ref. 2) 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 (Refs. 1 and 3).

3. Internal Components

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. 2). 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.

The BWR jet pumps have experienced some aging failures and have been repaired. However, the failure rate is relatively low and operational parameters indicate when degraded performance occurs.

B. Managing Aging Degradation

1. PWR RPVs

Inservice inspection (ISI) is required by ASME Code, Section XI (Ref. 3). 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. Successive inspection intervals require fewer beltline region, head, and repair weld examinations. However, the staff anticipates implementing a required 100-percent volumetric examination for all successive ISIs. 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. Integrally welded attachments are required to have surface or volumetric inspections of welds during each inspection interval.

The inspection plan 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. 4) 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 (Ref. 4). 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 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 (Ref. 3), 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 they are inaccessible. 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. Their configuration 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 for 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. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Division 1, New York, 1983.

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

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

4. 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.

5. 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.

6. 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

Site	Aging concerns	Managing aging	Recommendation	Mitigation
Beltline region	Irradiation embrittlement ° Chemical composition of vessel materials (Cu, Ni, P)	Surveillance program to assess irradiation damage, i.e., shift in RT_{NDT}* and drop in upper-shelf energy (10 CFR Part 50 Appendix H; Regulatory Guide 1.99, Revision 2)	Include fracture toughness and tensile test specimens in surveillance programs	Neutron flux reduction
	° Drop in upper-shelf energy	Pressurized thermal shock screening criteria (10 CFR 50.61)	Develop use of reconstituted and miniature specimens	Inservice annealing (ASTM E509-86)
	° Shift in RT_{NDT}*	Damage evaluation (10 CFR Part 50, Appendix G)	Perform accelerated irradiation of reconstituted specimens	Determination of effects of annealing and reembrittlement rate
	Environmental fatigue (thermal and mechanical)	Volumetric examination of all welds during each inspection interval (10 CFR 50.55a; ASME Code, Section XI, Article IWB-2500)	Revise Regulatory Guide 1.99, Revision 2, to account for phosphorus when copper content is low	

*RT_{NDT} = reference temperature, nil ductility transition

Table B.1.1-1 (continued)

Site	Aging concerns	Managing aging	Recommendation	Mitigation
		Flaw detection and evaluation (10 CFR 50.55a; ASME Code, Section XI, Article IWB-3000)	Use state-of-the-art nondestructive examination techniques for improved reliability with regard to detection, sizing, and characterization	
		Leakage and hydrostatic pressure tests (10 CFR 50.55a; ASME Code, Section XI, Article IWA-5000)	Use fatigue crack growth curves (ASME Code, Section XI, Appendix A)	
			Develop acoustic emission monitoring to detect crack growth (nonmandatory appendix is being developed for ASME Code, Section XI)	
Outlet and inlet nozzles	Environmental fatigue	Volumetric examination of all nozzle-to-vessel welds and nozzle inside radius section during each inspection interval ASME Code, Section XI, Article IWB-2500	Use online fatigue monitoring (monitoring of pipe wall temperatures and coolant flow, temperature, and pressure)	
	Irradiation embrittlement ° Function of nozzle elevation	Volumetric and surface examination of all dissimilar metal welds during each inspection interval ASME Code, Section XI, Article IWB-2500	Evaluate irradiation embrittlement damage	

Table B.1.1-1 (continued)

Site	Aging concerns	Managing aging	Recommendation	Mitigation
	° Potential effect of Regulatory Guide 1.99, Revision 2	Visual examination of external weld surface of 25% of nozzles during each inspection interval ASME Code, Section XI, Article IWB-2500		
Instrumentation nozzles/control rod drive mechanism housing nozzles	Environmental fatigue	Volumetric and surface examination of all studs and threads in flange stud holes during each inspection interval ASME Code, Section XI, Article IWB-2500		
Flange closure Studs	Boric acid corrosion (if leakage occurs)			

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

Site	Aging concerns	Managing aging	Recommendation	Mitigation
Feedwater nozzle safe-end welds	High-cycle thermal fatigue caused by feedwater leakage	Volumetric examination of all nozzle-to-vessel welds and nozzle inside-radius sections during each inspection interval ASME Code, Section XI, Article IWB-2500	Use online fatigue monitoring (monitoring of pipe wall temperature and pressure)	Revision of design and operating procedures and removal of feedwater nozzle cladding to prevent fatigue cracking
	Environmental fatigue (thermal and mechanical)		Develop criteria for assessing high-cycle fatigue damage	
Recirculation/ inlet and outlet nozzles and dissimilar metal welds	Possible propagation in base metal of intergranular stress corrosion cracking (IGSCC) initiated in heat-affected zone (HAZ)	Volumetric and surface examination of all dissimilar metal welds during each inspection ASME Code, Section Article (IWB-2500)	Develop online corrosion monitoring	Implementation of hydrogen water chemistry to mitigate corrosion fatigue
	Environmental fatigue		Evaluate long-term effects of hydrogen water chemistry	
Welds ° Control rod drive stub tubes ° Interior attachments	Possible propagation in base metal by corrosion and/or environmental fatigue of IGSCC initiated in HAZ	Visual examination of all accessible interior attachment welds during each inspection interval ASME Code, Section XI, Article IWB-2500	Develop remote inspection technique for interior attachment welds	

Table B.1.1-2 (continued)

Site	Aging concerns	Managing aging	Recommendation	Mitigation
Beltline region	Irradiation embrittlement ° Chemical composition of materials (Cu, Ni, P)	Surveillance program to assess shift in RT_{NDT}* and drop in upper-shelf energy Part 50, Appendix H; Regulatory Guide 1.99, Revision 2	Revise Regulatory Guide 1.99, Revision 2, to account for phosphorous when copper content is low	Inservice annealing (ASTM E509-86)
	° Drop in upper-shelf energy (USE)	Damage evaluation (10 CFR Part 50 Appendix G)	Use state-of-the art nondestructive examination inspection techniques for improved reliability with regard to defect detection, sizing, and characterization	Determination of effects of annealing and reembrittlement rates
	° Shift in RT_{NDT}*	Volumetric examination of all welds during each inspection interval (10 CFR 50.55a; ASME Code, Section XI, Article IWB-2500)	Develop robotics system for remote probe inspection positioning and scanning	Implementation of neutron flux reduction program program
	° Welds are more susceptible than base metal	Flaw detection and evaluation of all shell welds during each inspection interval (10 CFR 50.55a; ASME Code, Section XI, Article IWB-3000; Regulatory Guide 1.150, Revision 1)	Include fracture toughness and tensile test specimens in surveillance program	

*RT_{NDT} = reference temperature, nil ductility transition.

Table B.1.1-2 (continued)

Site	Aging concerns	Managing aging	Recommendation	Mitigation
	° Flux is lower than that in PWR vessel	Leakage and hydrostatic pressure tests (10 CFR 50.55a; ASME Code, Section XI, Article IWB-5000)	Develop use of reconstituted and miniature specimens and perform accelerated irradiation of reconstititued specimens Use fatigue crack growth curves (ASME Code Section XI, Appendix A) Develop acoustic emission monitoring to detect crack growth (nonmandatory appendix is being developed for ASME Code, Section XI)	
Closure studs	Fatigue, fretting	Volumetric and surface examination of all studs and threads in flange stud holes during each inspection interval (ASME Code, Section XI, Article IWB-2500)		

Table B.1.1-3 Summary of degradation processes for PWR reactor pressure vessels

Rank of degradation site*	Degradation site	Stressor	Degradation mechanism	Potential failure mode	Inservice inspection surveillance method
1	Beltline region	Neutron irradiation, mechanical and thermal stresses	Irradiation embrittlement (degree depends on individual vessel materials and flux spectrum history)	Ductile high-energy tearing, leading to leakage (net section overload)	First inspection, 100% volumetric; one weld inspected each subsequent inspection
			Stress corrosion cracking	Brittle fracture (i.e., pressurized thermal shock (PTS)	Surveillance program for assessing irradiation damage required
				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 and Inlet nozzles	Mechanical and thermal stresses	Fatigue crack initiation and propagation	Ductile overload, leading to a leak; possible brittle fracture if PTS occurs, with some irradiation embrittlement	All nozzle welds inspected volumetrically at each interval

Table B.1.1-3 (continued)

Rank of degradation site*	Degradation site	Stressor	Degradation mechanism	Potential failure mode	Inservice inspection surveillance method
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 and remaining 75% spread out over next three intervals
4	Flange closure studs	Mechanical and thermal stresses	Fatigue crack initiation 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 site*	Degradation site	Stressor	Degradation mechanism	Potential failure mode	Inservice inspection surveillance method
1	Nozzles (including instrument and control rod drive penetrations) and safe-end welds	Mechanical and thermal stresses	Fatigue crack initiation and propagation, intergranular stress-corrosion cracking	Ductile overload, leading to a leak	All large nozzle welds inspected volumetrically at each 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, (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)	A 100% volumetric inspection; surveillance as required
4	External attachments	Mechanical and thermal stresses	Fatigue	Ductile overload fatigue	Volumetric and surface inspections

* Rank of degradation site: 1 is highest in ranking priority.

B.1.2 REACTOR COOLANT SYSTEM

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Reactor Systems Branch (SRXB)
Mechanical Engineering Branch (EMEB)
Materials & Chemical Engineering Branch (EMCB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the reactor coolant system (RCS).

1. Description

a. The RCS in a pressurized-water reactor (PWR) 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 RCS in a boiling-water reactor (BWR) 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 are typically vertical shell and U-tube or 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 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 transfers the heat produced in the reactor to steam generators, where steam is produced and transported to the turbine generator.

b. In BWRs the function of the RCS is to provide a forced flow of coolant through the reactor core and to control the level of power in the core.

c. Another function of the RCS is to bar the release of fission products 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 SRP-LR B.1.1, "Reactor Pressure Vessel."

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 pressurizer and steam generators. Typical examples of age-related degradation associated with the RCS listed below (Refs. 1-3).

- 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

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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 for Items II.A and II.B.

C. The applicant has verified using plant-specific fatigue analyses for all ASME Class 1 components in the RCS, that the ASME Code, Section III allowable cumulative usage factor of 1.0 will not be exceeded during the projected lifetime of the plant.

D. The provisions of ASME Code, Section XI (Ref. 4), for inservice inspection (ISI) and inservice testing (IST) of RCS components are implemented throughout the license renewal period.

For those components important to license renewal that are not included in the ISI/IST program (i.e., because they are too 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 SRP-LR chapter may be addressed by the generic aging topic reviews in Part C of this review plan. The reviewer should ensure that structures and components included as part of the generic topics are adequately reviewed for this system. The following SRP-LR chapters are applicable to the RCS and should be used in the review: all of C.1.0, "Mechanical"; all of C.2.0, "Electrical"; and all of C.3.0, "Instrumentation." The various control systems associated with the RCS (i.e., steam generator level control, pressurizer pressure control, recirculation flow control, etc.) are addressed in SRP-LR B.1.3, "Reactor Control System." These reviews may require additional staff support.

F. The reviewer should verify that:

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

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

3. on a one-time basis, the applicant 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 concerns regarding fatigue would be to verify that the applicant has done the following:

1. Listed 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.

2. Provided the additional number of transients and cycles to be used as the design basis for the extended life of the plant;

for example, if the projected life is an additional 20 years for a total life of 60 years, the original design-basis transients from Item III.G.1 (above) should be increased by 50 percent. For all components, calculated the cumulative usage factors for this additional increment of time.

3. Listed all cycles due to unanticipated plant transients that were not considered as design-basis events in Items III.G.1 or III.G.2 (above). For all components, calculated cumulative usage factors for these additional transients.

4. Added the cumulative usage factors calculated from Items III.G.1, III.G.2, and III.G.3 (above) to arrive at the total end-of-life fatigue usage factors for all components. The allowable cumulative usage factor of 1.0 shall not be exceeded during the lifetime of the plant.

5. If the rate of actual plant cycles indicates that the design-basis cycles will be exceeded at the end of plant life, the procedures in Items III.G.1, III.G.2, and III.G.3 (above) should be adjusted to account for these additional cycles. If the rate of actual plant cycles indicates that the design-basis cycles will not be exceeded at the end of plant life and Item III.G.4 (above) is satisfied, no further analysis is required for fatigue.

6. Performed the above analysis in accordance with ASME Code, 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 applicant should provide the basis for appropriate design fatigue curves.

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

8. All of these evaluations should be based on elastic analyses. The use of elastic-plastic or fully plastic approaches 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 before such procedures are used.

9. Each applicant 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," 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. U.S. Nuclear Regulatory Commission, "Life Assessment Procedures for Major LWR Components," Vol. 4, NUREG/CR-5314, November 1989.

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

B.1.3 REACTOR CONTROL SYSTEM

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Instrumentation & Control Systems Branch (SICB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the reactor control system (RCS).

1. Description

The RCS consists of the instrumentation and control elements that sense plant conditions and are used for normal operations. The RCSs are not relied on to perform safety functions following anticipated operational occurrences or accidents but are relied on to control those plant processes that have 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 SRP-LR chapter.

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 it will perform its intended function during these plant conditions for which it is required.

Typical reactor control systems are:

- o reactivity control systems
- o reactor coolant pressure control systems
- o reactor coolant temperature control systems
- o reactor coolant flow control systems
- o condensate and feedwater control systems
- o environmental control systems for safety-related instruments and instrument sensing lines
- o other reactor control systems may also be used

Typical secondary control systems are:

- o secondary system pressure control systems
- o secondary system flow control systems
- o environmental control systems for safety-related instruments and instrument sensing lines
- o other secondary control systems may also be used

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

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

- o age-related degradation due to setpoint drift
- o age-related degradation due to functional testing cycles and trips
- o age-related degradation due to improper maintenance repair
- o age-related degradation of sensors, connectors, cables and wiring, circuit breakers, relays and electronic components, etc

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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.C.

D. Components of the RCS not specifically addressed in this SRP-LR may be addressed in the generic aging topic reviews in Part C of this review plan. The following SRP-LR chapters 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.3.0, "Instrumentation." This review 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

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

VII. REFERENCE

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 - License Renewal Project Directorate (LRPD)
Secondary - Instrumentation & Control Systems Branch (SICB)
Mechanical Engineering Branch (EMEB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the control rod drive system (CRDS).

1. Description

The control rod drive mechanisms (CRDMs) in pressurized-water reactors (PWRs) 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 control rod drive (CRD) rods can also be released to drop into the core by gravity for maximum negative reactivity insertion (scram).

In boiling-water reactors (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:

- control rod drive mechanisms
- hydraulic control units
- hydraulic system
- control rod drive supports

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

- drive water pumps
- filters/strainers
- flow control station
- pressure control station
- scram discharge volume

The CRD supports are horizontal beams installed directly below the bottom of the reactor vessel between the rows of CRD housings. These supports ensure that control rod movement, following a CRD 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:

- The water used for hydraulic functions 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.

- o 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.

- o 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 given in Tables B.1.4-1 and B.1.4-2.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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. CRDS components not specifically addressed in this SRP-LR may be addressed by the generic aging topic reviews in Part C of this review plan. 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 SRP-LR chapters are applicable to the CRD system and should be used for the review: C.1.1, "Piping"; C.1.2, "Valves", C.2.1, "Cable and Wiring"; C.2.4, "Relays, Circuit Breakers, and 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 review 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

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

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

Degradation site	Stressor	Degradation mechanism	Potential failure mode	Inservice inspection surveillance method
Pressure housing stub tube	Corrosive water, thermal stress, residual stress	Intergranular stress corrosion cracking (IGSCC)	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, over pressure misalignment	Valve diaphragm embrittlement and cracking	Stuck rods, unintentional rod movement	
Piston seal	Temperature, corrosive water	Embrittlement, wear	Stuck rods	
Latch assembly	Loose parts, impacting	Fretting, wear, spalling	Binding, stuck rods	
Coil stack	Moisture, temperature, radiation	Insulation breakdown, electrical shorting	Dropped rods, stuck rods	
Drive rod	Rubbing, impacting	Wear, low-cycle fatigue	Uncoupling of control assembly	
External components	Boric acid (if leak is present)	Boric acid corrosion	Leaks	

B.1.5 REACTOR PROTECTION SYSTEM

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Instrumentation & Control Systems Branch (SICB)

I. AREAS OF REVIEW

A. This SRP-LR chapter 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 to 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 facility.

2. System Function

The RPS ensures that specified plant safety limits are not exceeded by automatically deenergizing 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 SRP-LR. The RPS also actuates emergency equipment in the event of a loss of primary or secondary system coolant inventory. This function is carried out 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.

3. System Boundaries

The RPS, 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 SRP-LR chapters.

In regard to the remaining system boundaries, the RPS 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 RPS satisfies the acceptance criteria and guidelines for age-related degradation applicable to the protection system and that it will perform its intended function during the plant conditions for which it is required.

Actuation of the RPS to scram the control rods is initiated by the following typical elements:

- high reactor power
- nuclear overpower based on reactor coolant flow and axial imbalance
- loss of power to reactor coolant pumps
- high T-hot
- high reactor coolant system pressure
- low reactor coolant system pressure
- high reactor building pressure
- anticipatory loss of main feedwater
- turbine trip
- overtemperature delta temperature
- overpower delta temperature
- low steam generator level
- low feedwater flow
- safety injection signal actuation
- turbine stop valve closure
- turbine control valve fast closure
- reactor vessel low water level
- main steamline isolation
- scram discharge volume high water level
- main steamline high radiation
- main condenser low vacuum
- mode switch in shutdown
- manual operator action

Other elements may also be used.

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 RTS are the following:

- age-related degradation due to setpoint drift

- age-related degradation due to functional testing cycles and trips

- age-related degradation due to improper maintenance or repair

- o age-related degradation of sensors, connectors, cables and wiring, circuit breakers, relays, electronic components, and so forth.

The areas of aging concern regarding a facility's aging should be reviewed to evaluate the acceptablity of the applicant's program for managing 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. Components of the RPS not specifically addressed in this SRP-LR may be addressed in the generic aging topic reviews in Part C of this review plan. The following SRP-LR chapters 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 "Instrumentation." This review 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

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

VII. REFERENCE

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 - License Renewal Project Directorate (LRPD)
Secondary - Instrumentation & Control Systems Branch (SICB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the neutron monitoring system (NMS).

1. Description

The NMS contains 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 boiling-water reactors (BWRs), incore detectors are used for monitoring the core and supply 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 pressurized-water reactors (PWRs), excore detectors are used for monitoring the core and for monitoring all automatic safety functions. The excore nuclear instrumentation system provides the detectors and electronic circuitry needed to monitor the leakage neutron flux 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 NMS 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 devices 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 level of power in the core. 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 given below.

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 within the framework of calibration programs.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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 for Items II.A and II.B.

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

1. 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 verify that this instrumentation meets, <u>on a continuing basis</u> [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 1 (Ref. 3), requires neutron flux instrumentation that is installed and maintained

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

3. These criteria should be part of an established ongoing applicant program to ensure the availability of the neutron flux instrumentation. The applicant may also propose a one-time or new periodic inspection of components of the neutron flux instrumentation. For example, the applicant may institute an in-containment cable check and evaluation (Ref. 4). In addition, the applicant'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-LR B.0.1 for Items III.A through III.D.

E. Components of the NMS not specifically addressed in this SRP-LR chapter may be addressed in the generic aging topic reviews in Part C of this review plan. The following SRP-LR chapters are applicable to the NMS 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, "Instrumentation." This review may require additional staff support.

The reviewer should confirm the existence of a well-planned and well-implemented surveillance and maintenance program. This program should include the periodic replacement of the neutron flux detectors. The reviewer should also establish that the applicant has shown the acceptability of or has replaced the in-containment cables associated with the neutron flux monitoring instrumentation.

The reviewer should verify that the applicant's program for circumventing age-related degradation of the NMS includes 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.

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 is capable of performing during and following an accident. Cables which run through the containment and in auxiliary building areas are addressed as part of SRP-LR C.2.1. 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. 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, Revision 3, May 1983.

3. U.S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements," NUREG-0737, Supplement 1, February 1989.

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 (TI88 007920), January 1988.

B.1.7 REACTOR WATER CLEANUP SYSTEM (BWR)

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Plant Systems Branch (SPLB)
Reactor Systems Branch (SRXB)
Materials & Chemical Engineering Branch (EMCB)

I. AREAS OF REVIEW

A. This SRP-LR chapter 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 final 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-mho and undissolved solids at less than 0.01 parts per million (PPM); and chlorides and sulphates less than 15 parts per billion (PPB). The RWCS is normally operated continuously during all phases of operation, startup, shutdown, and refueling.

3. System Boundaries

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

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

C. No specific concerns regarding a facility's aging are associated with the RWCS other than those generic aging concerns discussed in Part C of this review plan 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 for Items III.A through III.D.

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

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

8.1.8 STANDBY LIQUID CONTROL SYSTEM (BWR)

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Instrumentation & Control Systems Branch (SICB)
Reactor Systems Branch (SRXB)
Materials & Chemical Engineering Branch (EMCB)

I. AREAS OF REVIEW

A. This SRP-LR chapter 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 high concentration of boron 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, as well as relief valves to prevent piping damage due to overpressurization. A mini-flow line provides a flow path from the test tank, through the pumps, back to the test tank.

Each pumping train consists of a piston-type, positive displacement pump with an associated explosive discharge valve and a 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 [i.e., an 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 safety function of the SLCS. 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, "Areas 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 in 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 Specifications 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 concern regarding a facility's aging should be reviewed to evaluate th[illegible]cceptability of the applicant's program for managing potential [illegible]g mechanisms. Site-specific conditions and experience are documente[illegible] 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. SLCS components not specifically addressed in this SRP-LR may be addressed by the generic aging topic reviews in Part C of this review plan. 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 SRP-LR chapters are applicable to the SLCS and should be used 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"; and all of C.3.0, "Instrumentation." This review 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

The four types of failures 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 appear in NUREG/CR-5562, "A Review of Information Useful for Managing Aging in Nuclear Power Plants," when it is published at a later date.

VII. REFERENCES

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

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Reactor Systems Branch (SRXB)
Materials & Chemical Engineering Branch (EMCB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the chemical and volume control system (CVCS) and the emergency boration system.

1. Description

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

The CVCS or the emergency boration system is described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for a 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 perform the following functions:

- adjusts the reactor coolant system (RCS) boric acid concentration
- maintains the proper water inventory in the RCS
- provides seal water flow to the reactor coolant pump shaft seals
- maintains the proper concentration of corrosion-inhibiting chemicals in the reactor coolant
- purifies the reactor coolant by removing impurities to keep RCS ionic impurities and fission products within operating limits
- provides borated water for emergency core cooling
- processes reactor coolant for reuse of boric acid and reactor makeup water

- o degases the RCS

- o provides a means of emergency boration of the RCS

- o provides a hydrostatic pressure capability for those systems that use a positive displacement pump

- o provides dissolved hydrogen to control and scavenge 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 of 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 reactors only), and at the check valves leading back to the boric acid tanks in the emergency boration system (called the rapid boration system at some plants).

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

C. No system-specific areas regarding a facility's aging 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 SRP-LR chapter may be addressed by the generic aging topic reviews in Part C of this review plan. 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 SRP-LR chapters in Part C (C.1.0, "Mechanical"; C.2.0, "Electrical"; and C.3.0, "Instrumentation") are applicable to the CVCS and the emergency boration systems and should be used for the review. This review may require additional staff support.

   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

B.2.0 ENGINEERED SAFETY FEATURES

8.2.1 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (PWR)

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPB)
Secondary - Instrumentation & Control Systems Branch (SICB)

I. AREAS OF REVIEW

A. This SRP-LR chapter 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 chapter 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 is addressed in the SRP-LRs 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 elements. Other elements may also apply.

- o reactor coolant system pressure low/pressurizer pressure low
- o reactor building pressure high/containment pressure high
- o steam generator pressure low
- o steam generator differential pressure high
- o main steamline flow high
- o manual operator action

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

- o containment and reactor vessel isolation systems
- o emergency core cooling systems
- o containment heat removal and depressurization systems
- o auxiliary/emergency feedwater systems
- o containment air purification and cleanup systems
- o 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, "Areas of Review," of SRP-LR B.0.1 for Item I.B.

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

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

- o age-related degradation due to setpoint drift

- o age-related degradation due to functional testing cycles and trips

- o age-related degradation due to improper maintenance or repair

- o age-related degradation of such equipment as sensors, connectors, cables and wiring, circuit breakers, relays, electronic components

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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. Components of the ESFAS not specifically addressed in this SRP-LR chapter may be addressed in the generic aging topic reviews in Part C of this review plan. The following SRP-LR chapters 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 "Instrumentation." This review 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

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-4747 (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 SAFETY INSPECTION SYSTEMS

B.2.2.1 REACTOR CORE ISOLATION COOLING SYSTEM (BWR)

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Reactor Systems Branch (SRXB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the reactor core isolation cooling (RCIC) system of boiling-water reactors (BWRs)

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 an 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, "Areas 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. Table B.2.2.1-2 lists aging processes typical for the RCIC system.

The failures listed above are typical examples of age-related degradation associated with the RCIC system.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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 for Items II.A and II.B.

C. Specific areas of concern that should be addressed by the applicant for monitoring aging in the RCIC system include 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 1) to guide the applicant in targeting components that are susceptible to age-related failures.

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 RCIC system not specifically addressed in this SRP-LR chapter may be addressed in the generic aging topic reviews in Part C of this review plan. The following SRP-LR chapters 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, "Instrumentation." This review may require additional staff support.

F. The reviewer should ensure that the applicant'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 applicant'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 survey on aging in light-water-reactors (LWRs), the NRC contractor analyzed the contributions of components of safety systems to time-dependent failures using data compiled from the Nuclear Plant Reliability Data System (NPRDS) data base (Ref. 1). In that report, data on failure are identified by system and are grouped in five categories, one of which includes age-related failures. Table B.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, which identifies the component failure fractions by failure category and orders them by aging fractions. Of the recorded failures associated with aging, valve failures occurred most frequently; approximately 50 percent of these failures were attributed to aging. Although valve operators and instrumentation switches also failed relatively frequently, 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, aging mechanisms have been identified in turbines.

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

VII. REFERENCE

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

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	Stressor	Degradation mechanism	Potential failure mode	Inservice Inspection Surveillance Method
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 circuits, 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 connections	Breaking loose	Visual inspection
Piping	Vibration, water-hammer, thermal cycles	Thermal, fatigue, abrasive water	Through-wall leaking or cracks	Visual inspection, volumetric inspections

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

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Reactor Systems Branch (SRXB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the high-pressure injection system (HPIS) and the intermediate-pressure injection system of pressurized-water reactors (PWRs).

1. Description

The HPIS and safety injection system (SIS) are emergency core cooling systems (ECCSs). Plant-specific designs are subject to considerable variation with respect to the equipment used and the role of that equipment in the 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 SRP-LR B.2.2.4). The B&W HPIS pumps also supply 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 supply 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 pump borated water from a storage tank to the RPV cold-leg piping under small-break 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, "Areas of Review," of SRP-LR B.0.1 for Item I.B.

C. Table B.2.2.2-1 lists 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 concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant'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. HPIS and SIS components not specifically addressed in this SRP-LR chapter may be addressed by the generic aging topic reviews in Part C of this review plant. The reviewer should ensure that structures and components included as part of the generic topics are adequately reviewed for this system. The following SRP-LR chapters 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, " Instrumentation." This review 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

The HPIS and SIS are designed with redundant channels, which interact with many of the other reactor systems making them vulnerable to common-mode failures. However, for failures associated with the HPIS and SIS, maintenance error is the leading cause of system failure. As components age, the systems deteriorate. The HPIS components that fail most frequently are valves, followed by instrumentation and control (I&C) components, pumps, and pipes. The pipe failures usually take the form of leaks through cracks. For example, one of the piping concerns involves cracks in nozzles and thermal sleeves that had occurred at B&W and Westinghouse plants. In this case, the cracks were attributed to fatigue that occurred as a result of 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 cooling or poorly controlled boration concentration.

Because of redundancy, only about 0.7 percent of the HPIS and SIS failures caused total loss of system function, and 21.3 percent of the component failures were age related. Plant records reveal that many small leaks and problems with I&C must occur before major failures develop. Thus, one method for identifying incipient failures is to consult plant records. Personnel at the plants usually follow the manufacturer's recommendations for maintenance of major components. About 28 percent of abnormal occurrences at PWRs can be attributed to faulty maintenance and surveillance testing. Current inspection methods include visual inspection, volumetric inspection, and operational tests.

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," Vol. 1, Proceedings of the Topical Meeting on Nuclear Power Plant Life Extension, July 31-August 3, 1988, Snowbird, Utah 1988.

B.2.2.3 CORE FLOOD SYSTEM (PWR)

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Reactor Systems Branch (SRXB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the core flood system (CFS) of pressurized-water reactors (PWRs)

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-coolant accident (LOCA), the pressure in the vessel 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 pressure in the reactor vessel 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, a passive system that requires no external signal or power source to operate, is designed to inject cooling water rapidly into the reactor vessel when the pressure in the vessel 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 a connection with the radioactive liquid waste system for draining the tanks.

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 CFS are:

- o corrosion stressors - electrical, mechanical, and thermal
- o environmental factors - chemical, atmospheric, and underground
- o corrosion/fouling phenomena - uniform and pitting corrosion
- o intergranular stress-corrosion cracking - alloy selection and treatment

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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 Section B.0.1 for Items III.A through III.D.

E. CFS components not specifically addressed in this SRP-LR chapter may be addressed by the generic aging topic reviews in Part C of this review plan. The reviewer should ensure that structures and components included as part of the generic topics are adequately reviewed for this system. The following SRP-LR chapters are applicable to the core flood system of PWRs and 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 all of C.3.0, "Instrumentation". This review 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.2.2.4 RESIDUAL HEAT REMOVAL SYSTEM

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Reactor Systems Branch (SRXB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the residual heat removal (RHR) system.

1. Description

a. The typical RHR system in a boiling-water reactor (BWR) performs one or more of the following functions, depending on plant design:

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

b. For a pressurized-water reactor (PWR), 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 RHR 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 RHR system brings the reactor to the safe shutdown condition and mitigates 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 the long-term cool-down mode following a LOCA. The RHR system can also provide cooling to the spent fuel pool if a cooling capacity beyond the normal system capacity is needed.

3. System Boundaries

Typically, the RHR system works in concert 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 a BWR and the refueling water storage tank for a PWR. Some BWR facilities may have RHR cross-ties to the recirculation system for shutdown cooling of the fuel pool and cleanup features for supplemental fuel pool cooling.

B. See Section I, "Areas 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 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 this review plan. In addition to the generic concerns, regarding a facility's aging, spray nozzles and spargers are subject to erosion or corrosion, or both, and radiation embrittlement. Table B.2.2.4-1, "Summary of Aging Processes for the High-Pressure Injection System" lists aging processes typical for the RHR.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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 II, "Review Procedures," of SRP-LR B.0.1 for Items III.A through III.D.

E. Components of the RHR system not specifically addressed in this SRP-LR chapter may be addressed by the generic aging topic reviews in Part C of this review plan. The reviewer should ensure that structures and components included as part of the generic topics are adequately reviewed. The following SRP-LR chapter are applicable to the RHR system and should be used for the review: C.1.0, "Mechanical" all of C.2.0 "Electrical" (except C.2.5, "Transformers); and all of C.3.0 "Instrumentation." This review 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.2.2.5 CORE SPRAY SYSTEMS (BWR)

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Reactor Systems Branch (SRXB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the high-pressure and low-pressure core spray (HPCS and LPCS) systems of boiling-water reactors (BWRs).

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 SRP-LR B.2.2.6). The more recent BWR5 and BWR6 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 provides spray cooling to the reactor core and maintains 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 provides spray cooling to the reactor core and to help 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 share boundaries 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-V dc system provides logic power and control and instrument power. The 24-V dc system provides analog instrument power to the logic. These systems are addressed in SRP-LR B.4.3.1.

B. See Section I, "Areas 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 this review plan. In addition to the generic concerns regarding a facility's aging, spray nozzles and spargers are subject to erosion or corrosion, or both, and radiation embrittlement. Table B.2.2.5-1 l[illegible]ging processes typical for the HPCS and LPCS systems.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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. The HPCS and LPCS components not specifically addressed in this SRP-LR chapter may be addressed by the generic aging topic reviews in Part C of this review plan. The reviewer should ensure that structures and components included as part of the generic topics are adequately reviewed for this system. The following SRP-LR chapters are applicable to core spray systems of BWRs and 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, "Instrumentation". This review 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.2.2.6 HIGH-PRESSURE COOLANT INJECTION SYSTEM (BWR)

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Reactor Systems Branch (SRXB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the high pressure coolant injection (HPCI) system of boiling-water reactors (BWRs).

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 of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the facility.

2. System Function

The HPCI system provides 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 through 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 shares boundaries with plant electrical and reactor instrumentation systems. The electrical system supplies ac and dc power to operate associated valves, turbine controls, 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, "Areas 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:

- o Fatigue crack initiation and propagation have been found in nozzles and thermal sleeves.
- o Wear, foreign material, and faults have resulted in valve leaks and operability problems.
- o Wear, vibration, and fatigue have resulted in pump reliability problems.
- o Vibration, waterhammer, and thermal cycles have affected the system piping and pipe supports.
- o Errosion and corrosion of the steamlines of the turbine-driven pump have resulted in wall thinning.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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. The HPCI components not specifically addressed in this SRP-LR chapter may be addressed by the generic aging topic reviews in Part C of this review plan. The reviewer should ensure that structures and components included as part of the generic topics are adequately reviewed for this system. The following SRP-LR chapters are applicable to the HPCI system of BWRs and 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 "Instrumentation." This review may require additional staff support.

F. The reviewer should ensure that the applicant's IPA addressees the cumulative effects of applicable aging processes 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

Table B.2.2.6-1 Summary of aging process for the High-pressure injection system

Major component	Stressor	Degradation mechanism	Potential failure mode	Inservice inspection surveillance method
Nozzle 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, vibration	Corrosion, loose connections, failure (catastrophic)	Open circuits, 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 connections	Breaking away	Visual inspection
Piping	Vibration, water-hammer, thermal cycles	Thermal fatigue, abrasive water	Through-wall leaking, cracks	Visual inspection, volumetric inspections

B.2.3 AUXILIARY FEEDWATER SYSTEM (PWR)

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Plant Systems Branch (SPLB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the auxiliary feedwater system (AFWS) of pressurized-water reactors (PWRs).

1. Description

A typical AFWS 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 arrange-ment; 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 heat to be removed from the secondary side of 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, "Areas 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. 1-3):

- o 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 also contribute significantly to system degradation.

- o The single, largest source of historical AFWS degradation is the turbine drive for AFW pumps. It should be noted that the turbine itself has proved to be 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.

- o Instrumentation and control (I&C)-related failures dominated the group of failures 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 not detected by the current monitoring practices were related to the I&C portion of the system.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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. AFWS components not specifically addressed in this SRP-LR chapter may be addressed by the generic aging topic reviews in Part C of this plan. The reviewer should ensure that structures and components included as part of the generic topics are adequately reviewed for the AFWS. The following SRP-LR chapters are applicable to the auxiliary feedwater system of PWRs and 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" (except C.2.5, "Transformers"); and all of C.3.0, "Instrumentation." This review 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

Historical data on failure 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. Although the turbine drive itself has been fairly reliable, 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 (BWR) plants, turbine drives in general, and more specifically the turbine controls will continue to be reviewed within the framework 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 could not be detected by the monitoring methods in place at the reference plant for NPAR studies, indicates the need for improving 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 review of failure data (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 I&C and governor controls).

During the reference-plant review for the NPAR Program, 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 the case particularly where the interaction of multiple components is involved. Decidedly adverse consequences could result from routine testing of some of these areas that currently are not tested (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 verifying pump capability by additional monitoring 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 or certain parts of the AFWS are either never tested or are not tested thoroughly. It appears that enhanced testing requirements are eeded 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, "Aging and Service Wear of Auxiliary Feedwater Pumps for PWR Nuclear Power Plants, Vol. 1, NUREG/CR-4597, July 1986.

2 U.S. Nuclear Regulatory Commission, "Aging and Service Wear of Auxiliary Feedwater Pumps for PWR Nuclear Plants," Vol. 2, NUREG/CR-4597, June 1989.

3. U.S. Nuclear Regulatory Commission, "Auxiliary Feedwater System Aging Study," Vol. 1 "Operating Experience and Current Monitoring Practices," NUREG/CR-5404 Draft, 1989.

B.2.4 AUTOMATIC DEPRESSURIZATION SYSTEM (BWR)

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Plant Systems Branch (SPLB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the automatic depressurization system (ADS) of boiling-water reactors (BWRs).

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

C. The primary concern regarding aging 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, a dirty air supply, and other causes.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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 SPR-LR B.0.1 for Items III.A through III.D.

E. The ADS components not specifically addressed in this SRP-LR chapter may be addressed by the generic aging reviews in Part C of this review plan. The reviewer should ensure that structures and components included as part of the generic topics are adequately reviewed for this system. The following SRP-LR chapters are applicable to the ADS of BWRs and 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, "Instrumentation." In addition, since most of the problems with the ADS are associated with the air supply, the reviewer should review SRP-LR B.5.8, "Compressed Air System," for those portions of the ADS exposed to the air supply. This review 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 AND SAFE SHUTDOWN SYSTEMS

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Instrumentation & Control Systems Branch (SICB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the remote shutdown and safe shutdown systems.

1. Description

Remote and safe shutdown systems achieve and maintain a safe shutdown of the plant.

The remote and safe shutdown systems are described in the most recent revision of the final safety analysis report (FSAR) or updated safety analysis report (USAR) for the 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 and safe shutdown.

2. System Function

The remote shutdown and safe shutdown systems achieve and maintain safe shutdown of the plant. Some engineered safety feature (ESF) systems are used mitigating accidents and for achieving and maintaining a safe shutdown. The review of the systems in this SRP-LR is limited to those system features that are unique to safe shutdown; the review does not include those features 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 (by circuit breakers isolation amplifiers, isolation transformers, actuation logic, fuses, or other approved isolation devices) from the systems components they control and from the normal and emergency ac and dc power systems.

The objectives of this review are to confirm that the remote shutdown and 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 and safe shutdown are:

- o remote shutdown panel
- o auxiliary feedwater system
- o residual heat removal system
- o chemical and volume control system (boration control)
- o reactor protection system
- o neutron monitoring system
- o mode switch

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

C. Typical examples of age-related degrada-tion associated with remote shutdown and safe shutdown are provided below:

- o age-related degradation due to setpoint drift
- o age-related degradation due to functional testing cycles and trips
- o age-related degradation due to improper maintenance and repair
- o age-related degradation of sensors, connectors, cables and wires, circuit breakers, relays and electronic components, and so forth

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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. Components of the remote shutdown and safe shutdown systems not specifically addressed in this SRP-LR chapter may be addressed in the generic aging topic reviews in Part C of this review plan. The following SRP-LR chapters are applicable to the remote shutdown and safe shutdown systems 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, "Instrumentation." This review 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

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 the operability and performance of the reactor protection system (RPS). This, in conjunction with the redundancy built into the RPS makes the RPS and similar systems, that is, the remote shutdown and safe shutdown systems, resistant to many of the age-related degradation concerns.

VII. REFERENCE

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

B.3.0 CONTAINMENT SYSTEMS

B.3.1 PRIMARY CONTAINMENT STRUCTURE

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Structural & Geosciences Branch (ESGB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the primary containment structure (PCS).

1. Description

Primary containments are either of the free-standing steel vessel type that meets the structural and leak-tightness requirements of the ASME Codes, or of the concrete containment type that uses a reinforced or prestressed concrete structure with a carbon steel liner for leak-tightness.

Containments for pressurized-water reactors (PWRs) enclose the steam generators, the reactor coolant pumps, the pressurizer, and the reactor pressure vessel. PWR containment structures therefore tend to be large, providing room for the expansion of steam resulting from a loss-of-coolant accident (LOCA). Some PWR containments include an ice condenser in which a large mass of ice provides passive heat sink for absorbing initial LOCA heat loads. An ice condenser reduces peak post-LOCA containment pressure, thereby allowing for a smaller containment volume. The ice-condenser containment is addressed further in SRP-LR B.3.3, "Containment Heat Removal System."

Containments for boiling-water reactors (BWRs) use a pressure-suppression concept whereby the LOCA fluid is channeled to a large water-filled pool (suppression pool) in which 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 also is 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," has a drywell in the general shape of an 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 has a double tongue-and-groove seal so that periodic leak checks can be made without pressurizing the entire vessel.

The second BWR containment design, referred to as the "Mark II containment," has a cylindrical or 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 and has a cylindrical drywell with a 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 contain weirs to regulate 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, an equipment hatch or hatches 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 serves as 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 includes a sump (PWR) or a suppression pool (BWR) to supply water to the emergency core cooling system during LOCA conditions after the water has been drained from the borated water or 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;" BWR containments include the drywell, suppression pool, and connecting vent pipes.

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

C. Concerns regarding a facility's aging include the potential for loss of structural integrity and leak-tightness of the boundary. One concern with concrete is the loss of bound water and associated degradation of the shielding properties that can be caused by nuclear heat.

The reinforcing steel can corrode causing the concrete to crack and spall which degrades the structural integrity of the containment or its shield. The internal environment can cause the steel vessel or steel liner and suppression pool (BWR) and vent pipes (BWR) to corrode and decreases the stress cycles of thermal cyclic loading expansion joints on the vent pipes. The occasional actuation of the SRVs adds to the thermal and cyclic loads placed on the containment and suppression pool for BWRs. References 1 through 2 furnish additional information on containment aging.

A unique aging concern with prestressed concrete containment structures is loss of 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 about the aging concerns of containment structures can be found in References 1 through 5. In the discussion that follows, many of those concerns are summarized and recommendations are made 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 1 discusses aging and degradation of concrete structures. Concrete generally tends to get stronger 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 moisture and air to penetrate the concrete cause 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 radioactive materials leaking 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 and 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 cracking. 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 protective 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 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 because, in addition to the potential for microbial growth (and increased corrosion) in the suppression pool, the stagnant pool water increases the possibility of corrosion if a 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 2 addresses 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 2 are summarized in SRP-LR Table B.3.1-4.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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 for Items II.A and II.B.

C. The results of the inspection and reviews required in Item III.F (below) shall comply with the requirements of References 6 through 8, as appropriate.

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 PCS not specifically addressed in this SRP-LR chapter may be addressed by the generic aging topic reviews in Part C of this review plan. The reviewer should ensure that structures and components included as part of the generic topics are adequately reviewed for this system. The following SRP-LR chapters are applicable to the PCS and should be used in the review: C.1.1, "Piping"; C.2.3, "Electrical Penetrations"; and all of C.4.0, "Civil Structures." This review may require additional staff support.

F. The reviewer should confirm that the applicant has committed to implement the inservice inspection requirements of Reference 6 during the license renewal period. For prestressed concrete containment structures, the applicant's inservice inspections also should address the provisions of Reference 7 or 8.

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, "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.

2. W.B. Dodson, P.F. McHale, P. Beament, and M. Lapides, "Life Extension Considerations for Pressurized Water Reactor Containment Structures," Vol. 1, pp. 68-75, Proceedings of the Topical Meeting on Nuclear Power Plant Life Extension, July 31-August 3, 1988, Snowbird, Utah.

3. 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.

4. 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.

5. D. Naus, M. Marchbanks, and G. Arndt. "Evaluation of Aged Concrete Structures for Continued Service in Nuclear Power Plants," Vol. 1 pp. 57-67, Proceedings of the Topical Meeting on Nuclear Power Plant Life Extension, July 31 - August 3, 1988, Snowbird, Utah.

6. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section XI.

7. U.S. Nuclear Regulatory Commission, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures," Regulatory Guide 1.35.

8. 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

Major component	Stressor	Degradation mechanism	Potential failure mode
Reinforcing bars	Corrosive environ- stray currents	Corrosion, fatigue	Loss of structural integrity
Post-tensioning system anchors*	Trapped water, steady-state stress	Hydrogen embrittlement, corrosion	Loss of stress
Post-tensioning tendon wire or strand*	Moisture, trapped water, microorganisms, steady-state stress	Pitting, microbiologically influenced 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, micro-organisms	Corrosion caused by 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

Major component	Stressor	Degradation mechanism	Potential failure mode
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 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

Table B.3.1-2 (continued)

Major component	Stressor	Degradation mechanism	Potential failure mode
Stainless steel bellows	Corrosive internal environment, cyclic thermal loading, pressure testing	Intergranular stress corrosion cracking (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, microbiologically influenced corrosion	Leakage of radioactive gases
Transition region from cylindrical to spherical 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

Major component	Stressor	Degradation mechanism	Potential failure mode
Exterior surface of drywell base near sand pocket	Moisture, microorganisms, degraded fill material	Aqueous corrosion, crevice corrosion, microbiologically 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, cyclic thermal loading, pressure testing	Intergranular stress corrosion cracking (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, microbiologically 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

Table B.3.1-3 (continued)

Major component	[illegible]	Degradation mechanism	Potential failure mode
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	Proposed 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 steel	Assembly of data from available records should allow 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 the corrective action taken as required was effective.
Liner degradation from containment interior (above intersection with base mat)	Perform ISI 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	Proposed action	Comments
	Monitor temperatures at penetrations and monitor radiation exposures at strategic areas.	Monitoring these areas will provide data showing that degradation as a result of 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. Licensees need not consider this action for containments that have suitable waterproof membranes or where groundwater levels in relation to below-grade portions of the containment are controlled.
	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 of monitoring according to system design to ensure protection.
	For those containments that do not have a cathodic protection system for reinforcment, develop and implement a system to remotely monitor corrosion of reinforcing steel.	Examine specific areas where remote monitoring indicates possible corrosion exists.
Liner corrosion from concrete side, below grade	Monitor performance of cathodic protection system (if available).	Maintain current levels monitoring according to system design to ensure protection

Table B.3.1-4 (continued)

Potential degradation	Proposed action	Comments
	Perform ultrasonic tests (UTs) of control areas (frequency of examination depends on existence of 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).	Examine those areas in which 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 according to 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 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 performed, develop an in-place test (such as adhesion testing) to assess ability of aged coating to withstand a loss-of-coolant accident

Table B.3.1-4 (continued)

Potential degradation	Proposed 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 SYSTEM

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Structural & Geosciences Branch (ESGB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the secondary containment system.

1. Description

Some facilities have a secondary containment (such as the reactor building for some boiling-water reactors [BWRs]) that completely encloses the primary containment. Under accident conditions, the secondary containment aids in minimizing the ground-level release of airborne radioactive materials and provides for the controlled release of radioactive materials.

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 serve 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 has reinforced -concrete exterior walls. The superstructure may be constructed with a structural steel frame and metal siding and roofing. Airlocks and piping, electrical, and instrumentation penetrations, as well as building ventilation isolation dampers and control elements are part of the secondary containment system.

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 secondary containment are given below:

- o steel corrosion resulting in structural degradation
- o rebar corrosion resulting in concrete structural degradation
- o coating degradation that allows structural elements to corrode
- o degradation of sealing materials (caulking) used on roofing and siding

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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. Secondary containment components not specifically addressed in this SRP-LR chapter may be addressed by the generic aging topic reviews in Part C of this review plan. The reviewer should ensure that structures and components are adequately reviewed for this system. All of SRP-LR chapter C.4.0, "Civil Structure," is applicable to the secondary containment system and should be used in the review. This review 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.3.3 CONTAINMENT HEAT REMOVAL SYSTEM

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Plant Systems Branch (SPLB)

I. AREAS OF REVIEW

A. This SRP-LR chapter 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 containments for pressurized water reactors (PWRs) 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. In containments for boiling-water reactors (BWRs) (see SRP-LR B.3.1), a large mass of water in the suppression pool is used as a passive heat sink for absorbing initial LOCA heat loads. For either approach, the large heat sink allows the containment volume to be considerably smaller than the larger volume dry containments 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 also can 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 those PWRs where the containment ventilation system fan coolers are used, heat also is transferred from these coolers to the UHS through 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 containment heat removal system serves 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 system for other PWRs With the exception of the ice condenser, all components of the containment cooling system are addressed in other SRP-LR chapters 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 conditions. With the exception of power sources and operator controls, all ice condenser equipment is located within the primary containment.

B. See Section I, "Areas 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 or cooling coil failures, or both, 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.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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. Containment heat removal system components not specifically addressed in this SRP-LR chapter may be addressed by the generic aging topic reviews in Part C of this review plan. The reviewer should ensure that the structures and components included as part of the generic topics are adequately reviewed for this system. The following SRP-LR chapters are applicable to the containment heat removal system and should be used in the review: all of C.1.0, "Mechanical"; all of C.2.0, "Electrical"; and all of C.3.0, "Instrumentation." This review may require additional staff support.

F. In addition, since the typical containment heat removal system uses other systems to perform its functions of post-LOCA containment heat removal and pressure reduction, this review should coordinate with the reviews of the other systems, namely:

- containment spray (SRP-LR B.3.8)
- RHR/low-pressure safety (core) injection (SRP-LR B.2.2.4)
- containment ventilation (SRP-LR B.3.9) for PWRs where components of this system are used for post-LOCA heat removal

For plants with ice condenser containments 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

B.3.4 CONTAINMENT ISOLATION SYSTEM

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Plant Systems Branch (SPLB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the containment isolation system.

1. Description

The containment isolation system consists of sensors, processors, and automatic closing valves that 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 system serves to isolate fluid systems that pass through containment penetrations so that any radioactive material that may be released in the containment following an accident is confined to the containment. The containment isolation system is 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:

o a double barrier at the containment penetration in those fluid systems that are not required to function following a design-basis event

o automatic, fast closure of those valves required to close following a design-basis event to minimize release of any radioactive material by maintaining containment integrity

- a means of leak-testing barriers in fluid systems that are used for containment isolation

- the capability to test the operability of containment isolation valves periodically

- the electrical and instrumentation control circuitry required to generate and transmit the actuation signal(s)

B. See Section I, "Areas 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. Nuclear heat and high thermal temperatures can cause degradation of the cables through which control signals are sent to isolation valves. The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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. Components of the containment isolation system not specifically addressed in this SRP-LR chapter may be addressed by the generic aging topic reviews in Part C of this review plan. The reviewer should ensure that structures and components included as part of the generic topics are adequately reviewed for this system. The following SRP-LR chapters are applicable to the containment isolation system and should be used in 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 C.3.0, "Instrumentation." This review 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.3.5 CONTAINMENT PURGE SYSTEM

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Plant Systems Branch (SPLB)

I. AREAS OF REVIEW

A. This SRP-LR chapter 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, following a planned or unplanned reactor shutdown, by reducing the airborne particulates in the containment atmosphere. This system also is used to control the pressure increase of the containment atmosphere resulting from normal containment heatup during reactor startup. In addition, portions of the containment purge system may be 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 an exhaust stack. Motor control centers and electrical and instrumentation control circuitry also are a part of the system boundaries to be considered in this review.

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

C. The concern regarding a facility's aging and mechanisms for this system 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 or cooling coil failures, or both, are expected to continue throughout the life of the plant, including the license renewal period.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the licensee's program for managing 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. Components of the containment purge system not specifically addressed in this SRP-LR chapter may be addressed by the generic aging topic reviews in Part C of this review plan. The reviewer should ensure that structures and components included as part of the generic topics are adequately reviewed for this system. The following SRP-LR chapters are applicable to the containment purge system and should be used in the 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, "Instrumentation." This review 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.3.6 STANDBY GAS TREATMENT SYSTEM (BWR)

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Plant Systems Branch (SPLB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the standby gas treatment system (SBGTS) of boiling-water reactors (BWRs).

1. Description

If the SBGTS is activated by an engineered safety feature (ESF) signal caused by such a design-basis accident as a loss-of-coolant accident (LOCA), it 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 also is discharged to the inlet of the SBGTS. In addition, this system 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 ESF.

3. System Boundaries

The SBGTS is located in the secondary containment and typically consists of the following components:

- fans, motors and fan housings
- electrical and instrumentation control circuitry
- heating coils, cooling coils, and moisture separators
- prefilters, high-efficiency particulate air filters, and filter housings

- activated charcoal adsorbers and adsorber housings
- motor-operated valves and dampers
- ventilation ductwork, supply and exhaust
- plant exhaust stack

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

C. The concerns regarding a facility's aging 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 or cooling coil failures, or both, are expected to continue throughout the life of the plant, including the license renewal period. The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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. SBGTS components not specifically addressed in this SRP-LR chapter may be addressed by the generic aging topic reviews in Part C of this review plan. The reviewer should ensure that structures and components included as part of the generic topics are adequately reviewed for this system. The following SRP-LR chapters are applicable to the SBGTS (BWR) and should be used in the review: C.1.1, "Piping"; 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"; C.3.1, "Sensors"; C.3.2, Electronic Components"; and C.3.3, "Electronic Devices." This review may require additional staff support.

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

B.3.7 CONTAINMENT COMBUSTIBLE GAS CONTROL SYSTEM

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Plant Systems Branch (SPLB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the containment combustible gas control system (CCGCS).

1. Description

Some facilities use the CCGCS to control in-containment, post-accident hydrogen buildup to a level below the flammability limit so that hydrogen and oxygen do not recombine in an uncontrolled manner. Other facilities may use an inert atmosphere inside the containment during operation, or they may use portions of the containment purge system to help control hydrogen buildup after a loss-of-coolant accident (LOCA). The CCGCS is capable of controlling hydrogen and 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 post-accident hydrogen venting, post-accident hydrogen sampling, and post-accident hydrogen mixing systems and the hydrogen recombiners are considered part of the CCGCS.

The CCGCS 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

In the event of a design-basis accident, the CCGCS controls the recombinating, mixing, venting, and diluting of hydrogen and oxygen to prevent conditions that would result in fire or explosion. The system also monitors the content of oxygen and hydrogen in the containment building atmosphere to alert operators when levels requiring action are reached.

3. System Boundaries

Systems for controlling combustible gas within the containment may be totally inside the containment building or portions may be outside the containment building.

The post-accident hydrogen venting system consists of a supply and exhaust system, including fans, ducting, a prefilter, a high-efficiency particulate air (HEPA) filter, and a charcoal filter. The post-accident hydrogen sampling system consists of fans, ducting, a sample vessel, and hydrogen monitoring instruments. The post-accident 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 such aging mechanisms 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.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the licensee's program for managing 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. CCGCS components not specifically addressed in this SRP-LR chapter may be addressed by the generic aging topic reviews in Part C of this review plan. The reviewer should ensure that structures and components included as part of the generic topics are adequately reviewed for this system. The following SRP-LR chapters are applicable to the CCGCS and should be used in 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 C.3.0, "Instrumentation." This review 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.3.8 CONTAINMENT SPRAY SYSTEM

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Plant Systems Branch (SPLB)
Materials & Chemical Engineering Branch (EMCB)

I. AREAS OF REVIEW

A. This SRP-LR chapter 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. The CSS consists of two separate trains of equal capacity. Each train is 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 or 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 reactors (PWRs) or the suppression pool for boiling-water reactors (BWRs).

During recirculation, the addition of sodium hydroxide (NaOH) to the containment building spray during injection of trisodium phosphate 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 which 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 cools 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 facility.

2. System Function

The CSS is an engineered safety feature used to reduce containment pressure and temperature following a LOCA. During recirculation the addition of sodium hydroxide to PWR containment sprays during injection of trisodium phosphate helps to remove radioactive fission products from the containment building atmosphere, thereby reducing the offsite dose.

BWR CSS 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 a part of the containment heat removal system (see SRP-LR B.3.3, "Containment Heat Removal System").

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. In BWRs, the RHR control logic that places the CSS in service is considered part of the CSS.

B. See Section I, "Areas 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 valve degradation from wear, foreign material, or vibration damage; pump degradation from wear or vibration damage; heat exchanger degradation, especially the tubes, from corrosion and erosion; and piping degradation from corrosion, erosion, and thermal fatigue.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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. CSS components not specifically addressed in this SRP-LR chapter may be addressed by the generic aging topic reviews in Part C of this review plan. The reviewer should ensure that structures and components included as part of the generic topics are adequately reviewed for this system. All the SRP-LR chapters in Part C are applicable to the CSS and should be used in the review except C.1.5, "Tanks and Vessels"; C.2.5, "Transformers"; and C.4.0, "Civil Structures." This review 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

Aging assessments of several plant systems similar to the CSS have been performed as part of the Nuclear Plant Aging Research (NPAR) Program Failure data from various national databases 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 suggest that piping and heat exchanger failures can become very dominant in later years if failures 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

B.3.9 CONTAINMENT VENTILATION SYSTEM

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Plant Systems Branch (SPLB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the containment ventilation system (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 (SWS), 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 facility.

2. System Function

The CVS cools the atmosphere of the containment and related subcompartments, maintaining each within peak and average temperature limits during normal operation and shutdown, including anticipated transients. Some pressurized-water reactor 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, only use the containment sprays for reducing pressure and temperature after a LOCA, and, in those cases, the CVS is not required to be safety 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.

The containment purge system controls filtration and pressure during normal operations and shutdown

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 share boundaries 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. The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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 SRP-LR chapter may be addressed by the generic aging topic reviews in Part C of this review plan. The reviewer should ensure that structures and components included as part of the generic topics are adequately reviewed for this system. The following SRP-LR chapters are applicable to the CVS and should be used in 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 C.3.0, "Instrumentation." This review may require additional staff support.

F. The review of the CVS should be coordinated 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 detecting performance degradation. Additional information regarding aging of fan coolers may be found in Reference 1. 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. REFERENCE

1. Pacific Northwest Laboratory, "Operating Experience and Aging Assessment of ECCS Pump Room Coolers," PNL-5722, October 1986.

B.4.0 ELECTRICAL SYSTEMS

B.4.1 MAIN POWER SYSTEM

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Electrical Systems Branch (SELB)
Instrumentation & Control Systems Branch (SICB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the main power system (MPS).

1. Description

The main power system supplies power both to the electrical equipment for the 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 function continuously 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 for 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 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, "Areas 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 appropriate SRP-LRs of Part C of this review plan, SRP-LR as discussed in Item III.E below. In addition, protective relaying and controls are reviewed in accordance with SRP-LR B.4.1.1.

Any aging degradation of components can potentially affect their ability to function as required. However, aging degradation of some components is of particular concern to the main power system because the affected component must function at the time of an event or must provide additional power. Some of the components of concern are noted below:

- 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
- protective relaying and control
- transformers that must begin to transfer power as result having to switch from one source of power to another
- cables
- 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. Circuit breakers are discussed in SRP-LR C.2.4, "Relays, Circuit Breakers, and Switchgear.

Typical examples of age-related degradation associated with the main power system are given in this SRP-LR. The areas of concern regarding a facility;s aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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 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. Components of the main power system not specifically addressed in this SRP-LR chapter may be addressed in the generic aging topic reviews in Part C of this review plan. The following SRP-LR chapters are applicable to the main 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"; and C.2.5 "Transformers." In particular, this system is affected by circuit breaker failures and the review under SRP-LR C.2.4 for circuit breakers should be emphasized. This review 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

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 Class 1E power system events) (Ref. 1). Studies utilizing information from the Nuclear Plant Reliability Data System (NPRDS) show 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 (EGG-2473), July 1987.

B.4.1.1 PROTECTIVE RELAYING AND CONTROLS COMPONENTS

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Electrical Systems Branch (SELB)
Instrumentation & Control Systems Branch (SICB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses protective relaying and controls components.

1. Description

The protective relaying and controls components that are required for controlling and protecting the various items of equipment in the electrical distribution system. The components are reviewed in accordance with appropriate SRP-LRs noted in Item III.E (below).

The protective relaying and controls components are described specifically 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 components perform the following functions:

- Continuously monitor power to safety-related loads and automatically switch power sources when necessary
- Preclude damage to electrical equipment as a result of extended periods of operation at reduced voltage levels
- Override the effects of short-duration system disturbances as well as the effects of transients that result from the starting of large motors fed from the plant distribution system
- Provide for remote indication, alarms, and control of the plant power systems

3. System Boundaries

Protective relaying and controls components include the relays, existing and potential transformers, and controls and indicators that are necessary to detect abnormal operating conditions and initiate 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, "Areas 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 significant aging degradations are

- Breakdown of relay and transformer coil insulation caused by inductive surges and overtemperature (Overtemperature may be caused by ohmic heating as a result of overvoltage operation, elevated ambient temperature, and temperature rises in the cabinet housing.)

- Wear of relays and switches as a result of continued use and high cycling rate

- Increased 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 given in this SRP-LR chapter.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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. Protective Relaying and Controls components not specifically addressed in this SRP-LR chapter may be addressed in the generic aging topic reviews in Part C of this review plan. Specifically, the following SRP-LR chapter are applicable to protective relaying and controls components and should be used in the review in conjunction with SRP-LR B.4.1.1: 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 review 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

Studies performed as a part of the NRC Nuclear Plant Aging Research (NPAR) Program show that of the protective relaying and control components, relays and switches account for the majority of the failures. Studies utilizing Nuclear Plant Reliability Data System (NPRDS) information show 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 devices that must operate during abnormal conditions and yet are significantly affected by aging.

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, "Nuclear Plant Aging Research: the 1E Power System," J.C. Meyer and J. L. Edson NUREG/CR-5181 (EGG-2545), May 1990.

2. U.S. Nuclear Regulatory Commission, "An Aging Failure Survey of Light Water Reactor Safety Systems and Components," Vol. 1, NUREG/CR-4747 (EGG-2514), July 1987.

Table B.4.1.1-1 Stresses and effects on relays

Stress	Effect of stress on component	Effect on operation
ELECTRICAL		
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 below.
MECHANICAL		
High cycling rate	Wear on 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		
Continuous energization (ohmic heating)	Accelerates aging of coil insulation and other nonmetallic components	Leads to insulation and component failure
Temperature rise in cabinet housing	Accelerates aging of nonmetallic materials including coil insulation, bobbin, relay base, and contact spacers	Leads to insulation and component failure
Elevated ambient temperature	Accelerates aging of nonmetallic components	Leads to insulation and component failure
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 AC DISTRIBUTION SYSTEMS

B.4.2.1 ESSENTIAL POWER SYSTEM

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Electrical Systems Branch (SELB)
Instrumentation & Control Systems Branch (SICB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the essential power system (EPS).

1. Description

The EPS supplies power to the electrical equipment for engineered safety features that are required for the safe shutdown and control of the plant during design-basis accidents. The system receives power from redundant sources so that it can function continuously during adverse and accident conditions. Redundant sources include main power, as described in SRP-LR B.4.1, and offsite power, and emergency power systems, as described in SRP-LR B.4.4 and SRP-LR B.4.3.1 respectively. The EPS is divided into at least two independent channels to further provide for redundant grouping of critical loads. Thus, the EPS not only provides for redundant sources of power but also for redundant grouping of loads.

The EPS 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 EPS supplies power to the electrical equipment for engineered safety features that are required for the safe shutdown and control of the plant during design-basis accidents.

3. System Boundaries

The EPS includes the breakers, buses, cabling, and transformers that are needed for providing power to the electrical equipment for engineered safety features that are 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 EPS buses and ends at the motor control centers that provide power to the electrical equipment. The system does not include such sources of electrical power as the higher voltage buses that feed the EPS buses, the offsite power, or the diesel generators. In addition, the EPS does not include the various items of electrical equipment that are connected to the motor control centers.

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

C. The EPS consists primarily of components that are reviewed in accordance with appropriate SRP-LR chapters, as discussed in Item III.E (below). 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 EPS because the affected component must function at the time of an event or must provide additional power. Some of the components of concern are noted below:

- 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
- the circuit breaker that must close to supply power from a diesel generator
- circuit breakers that must open and then reclose during load shedding and subsequent energizing on startup of the diesel generator
- protective relaying and control
- transformers that must begin to provide power as result of 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. These effects are summarized in SRP-LR C.2.4, "Relays, Circuit Breakers, and Switchgear."

Typical examples of age-related degradation associated with the essential power system are given in this LRP-LR chapter.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the licensee's program for managing 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 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. Components of the EPS not specifically addressed in this SRP-LR chapter may be addressed in the generic aging topic reviews in Part C of this review plan. The following SRP-LR chapters are applicable to the essential power system and should be used in the review: C.2.1, "Cable and Wiring"; C.2.2, "Junctions"; C.2.4, "Relays, Circuit Breakers, and Switchgear"; and C.2.5, "Transformers."In particular, this system is affected by circuit breaker failures and the review under SRP-LR C.2.4 for circuit breakers should be emphasized. This review 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

Studies performed as a part of the NRC Nuclear Plant Aging Research (NPAR) Program show that, of the components that form the EPS 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 Class 1E power system events). Studies utilizing the Nuclear Plant Reliability Data System (NPRDS) information show 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, May 1990.

2. U.S. Nuclear Regulatory Commission, "An Aging Failure Survey of Light Water Reactor Safety Systems and Component," Vol.1, 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," October 1980.

B.4.2.2 NONESSENTIAL POWER SYSTEM

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Electrical Systems Branch (SELB)
Instrumentation & Control Systems Branch (SICB)

I. AREAS OF REVIEW

A. This SRP-LR chapter 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 function continuously 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 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, cables, and transformers that are necessary to send electrical power to the balance-of-plant electrical equipment that is required for normal operations. The system boundary begins with the circuit breakers feeding the non-essential 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, "Areas 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 applicable SRP-LR chapters, as discussed in Item III.E (below). 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 particular 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.

- 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
- protective relaying and control
- 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. These effects are summarized in SRP-LR C.2.4, "Relays, Circuit Breakers, and Switchgear".

Typical examples of aging-related degradation associated with the nonessential power system are given in this SRP-LR chapter.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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. Components of the nonessential power system not specifically addressed in this SRP-LR chapter may be addressed in the generic aging topic reviews in SRP-LR Part C of this review plan. Specifically the following SRP-LR chapters are applicable to the nonessential 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"; and C.2.5, "Transformers." In particular, this system is affected by circuit breaker failures and the review under SRP-LR C.2.4 for circuit breakers should be emphasized. This review 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

Studies performed as a part of the NRC Nuclear Plant Aging Research (NPAR) Program show 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 Class 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 failures (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, (EGG-2545), May 1990.

2. U.S. Nuclear Regulatory Commission, "An Aging Failure Survey of Light Water Reactor Safety Systems and Component," Vol.1, 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," October 1980.

B.4.2.3 HIGH-PRESSURE CORE SPRAY POWER SYSTEM (BWR)

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Reactor Systems Branch (SRXB)
Electrical Systems Branch (SELB)
Instrumentation & Control Systems Branch (SICB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the high-pressure core spray (HPCS) power system of boiling-water reactors (BWRs).

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 loss-of-coolant accident (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 function continuously during adverse conditions. Redundant sources are typically some combination of offsite power, the station switchyard, the station generator, and a dedicated diesel generator for loss-of-offsite power situations. The sources of power are reviewed under SRP-LRs B.4.1, "Main Power System"; B.4.4, "Emergency Diesel Generators"; and B.4.3.1, "DC Power System."

The HPCS power 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 HPCS power system supplies power to the HPCS pump and supporting electrical equipment that are needed to depressurize the reactor and maintain the reactor vessel water level in the event of a small-break LOCA.

3. System Boundaries

The HPCS power system includes the breakers, buses, cables, 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 buses and ends at the motor control centers that provide power to the electrical equipment. The system does not include such sources of electrical power as the higher voltage buses that feed the HPCS power buses, the offsite 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 I, "Areas 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 appropriate SRP-LR chapters as discussed in Item III.E below. 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. Circuit breakers are discussed in SRP-LR C.2.4, "Relays, Circuit Breakers, and Switchgear."

Typical examples of age-related degradation associated with the HPCS power system are given in this SRP-LR chapter.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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. Components of the HPCS system not specifically addressed in this SRP-LR chapter may be addressed in the generic aging topic reviews in Part C of this review plan. The following SRP-LR chapters are applicable to the HPCS system and should be used in the review: C.2.1, "Cable and Wiring"; C.2.2, "Junctions"; C.2.4, "Relays, Circuit Breakers, and Switchgear"; and C.2.5 "Transformers." In particular, this system is affected by circuit breaker failures and the review under SRP-LR C.2.4 for circuit breakers should be emphasized. This review 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

Studies performed as a part of the NRC Nuclear Plant Aging Research (NPAR) Program show that, of the components that form the HPCS 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 Class 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 and 2). Circuit breakers, therefore, historically have the highest failure rate of these components 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," L. C. Meyer and J. L. Edson NUREG/CR-5181 (EGG-2545), May 1990.

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

3. Institute of Electrical and Electronic Engineers, IEEE Standard 308-1980. "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations."

B.4.3 INSTRUMENT AND CONTROL POWER SYSTEMS

B.4.3.1 DC POWER SYSTEM

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Electrical Systems Branch (SELB)
Instrumentation & Control Systems Branch (SICB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the dc power system.

1. Description

The dc power system includes those dc power sources and their distribution systems and vital supporting systems which supply motive or control power to equipment that is safety related and important to safety.

The dc power 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 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. Operators can detect long-term degradation of the dc power system by carefully observing the float voltage and current. Testing the dc power system as required by Technical Specifications will reveal degradation of the system.

The dc power system can be adversely affected by deterioration of the batteries, the battery chargers, or the bus systems. Batteries are affected by thermally induced grid and connector oxidation, plate and grid swelling, container and cover cracking, separator deterioration, or changes in specific gravity. Battery chargers are adversely 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.

The aging of relays, circuit breakers, and switchgear is addressed in SLRP-LR C.2.4. Cable aging is addressed in SRP-LR C.2.1. Typical examples of age-related degradation associated with the dc power system are given in this SRP-LR chapter. The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the licensee's program for managing 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 for Items II.A and II.B.

C. These criteria should be part of an established ongoing applicant program to ensure the availability of the dc power system. The applicant should also inspect components of the dc power system anew (once or periodically) in conjunction with the license renewal application.

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 dc power system not specifically addressed in this SRP-LR chapter may be addressed in the generic component or structure reviews in Part C of this review plan. The following SRP-LR chapters are applicable to the dc power system and should be used in 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"; and all of C.3.0 "Instrumentation." This review may require additional staff support.

F. The reviewer shall ensure that the applicant 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.

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, <u>on a continuing basis</u> [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 starting and operating 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) evaluates 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 swelling 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 the qualified life of the battery. Battery maintenance records allow the 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 that recommend replacement of the battery 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 point. 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. The physical condition of individual cells (such as the plate condition, cell reversal, specific gravity readings, or electrolyte contamination) often requires 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. 10) evaluates 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 two 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 upon 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 also cause failure. In addition, excessive ripple will cause the battery to age prematurely. 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). At some plants, operators keep track of the internal temperature of the battery charger. The internal temperature is known to affect degradation. Increased internal temperatures decrease the life of the component.

VII. REFERENCES

1. Institute of Electrical and Electronics Engineers, IEEE Standard 279-1971, "Criteria for Protection 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. 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. U.S. Nuclear Regulatory Commission,"Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," Regulatory Guide 1.129.

6. 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.

7. U.S. Nuclear Regulatory Commission, "Nuclear Plant Aging Research: The 1E Power System," NUREG/CR-5181 (EGG-2545), May 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," W.E. Gunther, R. Lewis, and M. Subudhi, NUREG/CR-5051, August 1988.

11. 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.

B.4.3.2 INSTRUMENT AC POWER SYSTEM

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Electrical Systems Branch (SELB)
Instrumentation and Control Systems Branch (SICB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses instrument ac power systems.

1. Description

The 120-V ac vital instrument power system includes those 120-V ac power sources and their distribution systems and vital supporting systems that 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-V ac 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 is described in the most recent revision of the final safety analysis report (FSAR) or the updated safety analysis report (USAR) for the facility.

2. System Function

The 120-V ac vital instrument power system supplies very reliable power to the reactor protection system and to other instrumentation and controls that are important to safety.

3. System Boundaries

The 120-V ac vital instrument power system shares boundaries with the dc power system and the essential ac power systems at the inverters and the alternate source transformers. It also shares boundaries with the reactor protection system and other instrumentation and control circuits.

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

C. A variety of aging mechanisms can affect the ability of the 120-V ac vital instrument power system to continue to operate safely and effectively. The typical 120-V ac 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-V ac vital instrument power system will detect long-term system degradation. System redundancy limits the effects of rapid system failure.

The aging of relays, circuit breakers, and switchgear (and, by extension, the aging of static and bypass switches) is addressed in SRP-LR C.2.4.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing potential aging mechanisms. Site-specific conditions and experience are documented in the integrated plant assessment (IPA).

II. ACCEPTANCE CRITERIA FOR THE 120-V AC VITAL INSTRUMENT POWER SYSTEMS

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

C. These criteria should be part of an established ongoing applicant program to ensure the availability of the 120-V ac vital instrument power system. The applicant should also implement a new periodic or a one-time inspection of components of the 120-V ac vital instrument power system in conjunction with the license renewal application.

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 instrument ac power system not specifically addressed in this SRP-LR chapter may be addressed in the generic aging topic reviews in Part C of this review plan. The following SRP-LR chapters are applicable to the instrument ac power system and should be used in 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"; and all of C.3.0, "Instrumentation." This review may require additional staff support.

F. The reviewer should verify that the applicant has an established, ongoing maintenance and surveillance program to ensure the availability of the 120-V ac vital instrument power system. This program should include a mechanism to add new testing and evaluation criteria to monitor any newly detected aging deterioration.

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 (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 this equipment meets, 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 starting and operating the required loads during normal and post-accident conditions.

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 monitoring temperature in components and other equipment by observing voltage waveforms periodically, and by measuring the intake and output of components (mostly capacitors).

An excessively high input voltage (as a result of an excessively high equalization voltage) can damage an inverter. Fuses, capacitors, semiconductors, and other components can be damaged. Excessive ac ripple on the dc input will cause damage also by creating additional heat losses in components.

All plants have some inverter maintenance. The level of maintenance, according to NUREG/CR-5051 (Ref. 5), has a wide range. Thus, some units have only minimal maintenance performed. For example, general instructions for inspecting and cleaning the inverters at refueling outages do not constitute an adequate maintenance program. A sufficient program, as discussed in NUREG/CR-5051, will include cleaning the inverters to remove accumulated debris, dirt, and dust; inspecting cleanliness, electrical and mechanical connections, airflow, and evidence of overheating; replacing components (especially electrolytic capacitors); performing capacity tests and checks of internal temperature, cable meggering, and fan condition; and calibrating of output voltage and frequency and metering instrumentation. The frequency of these tests varies among plants 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. U.S. Nuclear Regulatory Commission, "Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants," Regulatory Guide 1.32.

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," W.E. Gunther, R. Lewis, and M. Subudhi, NUREG/CR-4564, June 1986.

5. 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.

6. U.S. Nuclear Regulatory Commission, "Nuclear Plant Aging Research: the 1E Power System," L.C. Meyer and J.L. Edson, NUREG/CR-5181 (EGG-2545), May 1990.

B.4.4 EMERGENCY DIESEL GENERATORS

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Mechanical Engineering Branch (EMEB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the emergency diesel generator (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, the block assembly, one or more heads, the baseplate, and air induction and exhaust ducting. Supporting subsystems, including the instrumentation and control subsystem, the starting subsystem, the cooling subsystem, the fuel oil subsystem, and the 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 is described in the most recent revision of the final safety analysis report (FSAR) or the updated safety analysis report (USAR) for the 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, the engine block assembly, the heads, the baseplate, the foundation, and the 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 that have resulted from wear and aging as documented by the NRC's Nuclear Plant Aging Research Program (Ref. 1). The mitigation of identified aging mechanisms and the management of the aging process has also been documented (Refs. 1 and 2).

The NRC requires regular EDG testing, and some applicants are required by their Technical Specifications to submit special reports documenting test failures. Onsite data should be evaluated, especially for the listed systems, which typically are the high-failure-rate systems within the diesel generator system boundary.

The applicant should evaluate data from EDG failures by reviewing failures and licensee event reports (LERs). Maintenance and operational records should be reviewed for repetitive failures, repetitive part replacement, root-cause analysis of failures, and for evidence of operational problems. Surveillance data, data monitor-ing, and trending data should be reviewed and analyzed. The licensee should evaluate this data for the diesel generator system, especially for the most recent 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. 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. For the license-renewal process, attention should be focused on ensuring the absence of subtle accumulated damage from wear, stress, and metal fatigue that could compromise established levels of reliability. The reviewer should not expect the applicant to tear an engine down for detailed inspection, because various studies show that such teardowns decrease engine reliability. Instead, limited inspections have been shown to be more appropriate for investigating abnormal behavior and suspicious trends indicated by engine and generator data.

Site-specific conditions and experience are documented in the integrated plant assessment (IPA). The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing potential aging mechanisms.

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.

- The EDG reliability goal should have been met for the previous 10 years, and all operating boundaries should be currently within acceptable limits established by the manufacturer.

- Engine crankshaft and generator alignment should be within the manufacturer's recommendations.

- Main bearing wear should not exceed the manufacturer's recommendation.

- No fatigue cracking of connecting rod bearings should exist.

- No gear fatigue or excessive wear should be found.

- Turbochargers should be free of 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 applicant 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 and indicate that goal reliability criteria have been met and that all current engine operating are within the engine manufacturer's recommended limits. Where goals or limits were exceeded, the applicant provided evidence of effective corrective actions.

2. A check was made of the diesel engine crankshaft, pedestal bearing, and generator alignment. If misalignment exceeded the 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 the manufacturer's specifications, 25 to 50 percent of the connecting rod 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 applicant's response to the NRC's Generic Letter (90-xx), related to the resolution of Generic Issue B-56, is acceptable for purposes of this review.

VII. REFERENCE

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.

B.4.4.1 EMERGENCY DIESEL GENERATOR INSTRUMENTATION AND CONTROL SUBSYSTEM

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Mechanical Engineering Branch (EMEB)
Electrical Systems Branch (SELB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the emergency diesel generator (EDG) instrument and control (I&C) subsystem.

1. Description

The EDG I&C subsystem comprises electrical cables, relays, circuit breakers, limit switches, indicators, and electronic components associated with the emergency diesel generators that control starting, monitor operation, and connect the generator to the loads upon an 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 facility.

2. System Function

The EDG I&C subsystem causes 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, the subsystem 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.

C. References 1-3 show that the instrumention and control system has a high failure rate. Research data on diesel generator aging indicate 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 have 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. Vibration does not usually affect solid-state devices to the same extent.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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. The reviewer should address the applicant's maintenance program, evidence of operational problems, and test data should be reviewed for license renewal concerns for the I&C subsystem. The reviewer should also consider the generic aging topics of SRP-LR C.3.0, "Instrumentation."

F. The reviewer should assess the applicant's general review of the EDG control system because of its importance, large failure rate, and the perceived obsolescence problem. The applicant 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 applicant has reviewed the EDG I&C subsystem to:

- o Determine that the expected service life of all major I&C subsystem 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 Inspect EDG I&C cable. In general, EDG I&C cable is only subjected to a mild environment and ambient conditions; it 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. 2, 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.

B.4.4.2 EMERGENCY DIESEL GENERATOR AIR STARTING SUBSYSTEM

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Mechanical Engineering Branch (EMEB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the emergency diesel generator (EDG) air starting subsystem.

1. Description

The EDG air starting subsystem comprises the air supply, valves, piping, and air motors (if used) used to rotate the engine for starting.

The EDG air starting subsystem is described specifically 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 EDG air starting subsystem system rotates 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, "Areas of Review," of SRP-LR B.0.1 for Item I.B.

C. Internal corrosion attack from condensed air moisture can occur in many components of the air supply system, including compressors, intercoolers, aftercoolers, automatic drain valves, tanks, piping, and both refrigerated and desiccant drier. 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 such critical components of the starting system 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 obstruct 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 ensure 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 concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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 for Items II.A and II.B.

C. The following additional criteria apply to the EDG air starting subsystem:

1. The wall thickness of pipes, tanks, and other important components should meet the minimum code construction or manufacturer's design thickness to withstand the design pressure as well as the corrosion allowance for the remaining time period (remaining license time and the license renewal period). Generic topics in SRP-LRs 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 5 percent of the weld areas should be sampled by nondestructive examination (NDE) methods.

III. REVIEW PROCEDURES

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

E. The air-starting components not specifically addressed within this SRP-LR chapter may be addressed by the generic aging topic reviews in Part C of this review plan. The reviewer should ensure that structures and components included as part of the generic review are adequately assessed for this system. The following specific SRP-LR chapters are applicable to the EDG air starting subsystem and should be used for the review: C.1.1, "Piping"; C.1.2, "Valves"; and C.1.5, "Tanks and Vessels." This review may require additional staff support.

F. The reviewer should confirm that the applicant 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 applicant should present evidence and analysis of the following:

1. Approximately 5 percent of the sensitive locations of piping tanks, valves, and other pressure-retaining components should be examined by NDE methods for wall thinning. Locations at which water or moisture tends 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. A 5-percent sample of the sensitive locations is adequate.

3. Any pipe or tank failure in the licensee event reports attributed to vibration or an 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 or corrective action, or both.

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.4.4.3 EMERGENCY DIESEL GENERATOR COOLING SUBSYSTEM

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Mechanical Engineering Branch (EMEB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the emergency diesel generator (EDG) cooling subsystem.

1. Description

The EDG cooling subsystem is a circulating water system that includes 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 cooling water system by the heat exchanger or is released to the atmosphere through 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 EDG cooling subsystem keeps 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 (these are included in the 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; these 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 cooling water system. These heat exchangers are subject to corrosion, fouling, and silting principally on the 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 concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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. The components of the EDG cooling subsystem not specifically addressed within this SRP-LR chapter may be addressed by the generic aging topic reviews in Part C of this review plan. The reviewer should ensure that structures and components included as part of the generic topics are adequately reviewed for this system. The following SRP-LR chapters are applicable to the EDG cooling subsystem and should be used in the review: C.1.1, "Piping"; C.1.2, "Valves"; C.1.3, "Pumps"; C.1.4, "Heat Exchangers"; and C.1.6, "Equipment and Component Supports." This review may require additional staff support.

F. The reviewer should confirm that the applicant 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 - License Renewal Project Directorate (LRPD)
Secondary - Mechanical Engineering Branch (EMEB)
Materials & Chemical Engineering Branch (EMCB)

I. AREAS OF REVIEW

A. This SRP-LR chapter 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 facility.

2. System Function

The EDG fuel oil subsystem supplies fuel oil to the EDGs during operation and testing.

3. System Boundaries

The EDG fuel oil subsystem includes all fuel storage tanks, aboveground 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. However, most fuel oil leakage has been caused by engine vibrations that loosen fittings.

The areas of concern regarding a facility's aging should be reviewed to evaluate the accept-ability of the applicant's program for managing potential aging mech-anisms. 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 for Items II.A and II.B.

C. Piping and tank walls for the EDG fuel oil subsystem should not be corroded from the inside or the outside to less than the minimum wall thickness for their 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-LRs 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 SRP-LR chapter may be addressed by the generic aging topic reviews in Part C of this review plan. The reviewer should ensure that structures and components included as part of the generic topics are adequately reviewed for this system. The following SRP-LR chapters are applicable to the EDG fuel oil subsystem and should be used in the review: C.1.1, "Piping"; C.1.2, "Valves"; C.1.3, "Pumps"; and C.1.6, "Equipment and Component Supports." This review may require additional staff support.

F. The reviewer should ensure that piping and tank wall thickness measurements were obtained by a sampling technique as part of the license renewal process, unless these are already part of the surveillance program. These wall thicknesses should be shown to meet below code specifications or the 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

VII. REFERENCES

B.4.4.5 EMERGENCY DIESEL GENERATOR LUBRICATING OIL SUBSYSTEM

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Mechanical Engineering Branch (EMEB)

I. AREAS OF REVIEW

A. This SRP-LR chapter 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 lubricating oil subsystem 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 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 lubricating 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 (LERs) 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 lubricating oil system is well monitored for aging effects through the applicant's observation 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 concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant'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 for Items II.A and II.B.

C. In addition to criteria listed above, the applicant has provided test data to show that heat exchanger wall thickness is no less than 40 percent of the original value. Alternatively, by suit-able calculations, the applicant may have demonstrated this thickness to be 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 SRP-LR chapter may be addressed by the generic aging topic reviews in Part C of this review plan. 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 SRP-LR chapters are applicable to the EDG lubricating oil subsystem and should be used in the review: C.1.1, "Piping"; C.1.2, "Valves"; and C.1.3, "Pumps." This review may require additional staff support.

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

VII. REFERENCES

B.4.5 ESSENTIAL LIGHTING SYSTEM

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Instrumentation & Control Systems Branch (SICB)
Electrical Systems Branch (SELB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the essential lighting system.

1. Description

Both the normal and emergency (or essential) lighting systems must be operable during loss of offsite power. Both lighting 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 normal lighting system supplies lighting during all operating conditions. The essential lighting system, in addition to functioning as a normal lighting system, must also supply lighting under fire, transient, and accident conditions.

3. System Boundaries

The essential 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 SRP-LR B.0.1 for Item I.B.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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.

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, sets the standards for the essential lighting system in its Lighting Handbook (Ref. 1) as related to systems design and illumination levels.

VII. REFERENCE

1. Illuminating Engineering Society, Lighting Handbook (latest edition).

B.4.6 COMPUTER SYSTEM

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Instrumentation & Control Systems Branch (SICB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the computer systems.

1. Description

The computer system is a microprocessor-based digital system. The system receives its data from both safety-related and non-safety-related sensors, and directs its outputs to safety-related and non-safety-related controls, indicators, and recorders. The system is isolated at input and output from the safety-related functions of the plant. The computer system is controlled by its software and by the station operators.

The computer 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 computer system is mainly used for data acquisition and display for post-accident monitoring, safety-grade signal processing for instrumentation, and Class 1E to non-Class 1E isolation. A significant portion of the computer system falls outside the scope of license renewal activities. However, some computer systems have been installed as Class 1E safety-related systems, and are being used as Class 1E control systems to monitor and control safety-related systems. Such computer systems may possibly meet one of the four definitions in 10 CFR 54.3 as an SSC important to license renewal and, if so, must be evaluated in the facility integrated plant assessment (IPA).

3. System Boundaries

The computer system is configured using such devices as converters, amplifiers, function generators, bistables, controllers, microprocessors, multiplexors, power supplies, and equipment racks.

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

C. Random failure is the major failure mode of the electronic equipment used in the computer systems. This is caused mainly by the aging characteristics of the specific components used in the fabrication of the devices. Therefore, the maintenance or replacement schedules should consider the specific aging characteristics of the component materials (see SRP-LR C.3.2).

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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

VII. REFERENCES

B.4.7 SWITCHYARD

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Electrical Systems Branch (SELB)
Instrumentation & Control Systems Branch (SICB)

I. AREAS OF REVIEW

A. This SRP-LR chapter 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 power distribution system for startup, operation, and shutdown. 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 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 the switchyard should be limited to the equipment required to deliver the power from off site 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 given here may not be applicable. The switchyard 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 switchyard supplies a reliable source of auxiliary power from at least two independent transmission circuits to the power distribution system for startup, operation, and shutdown.

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 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 buses. The system does not include any equipment not directly associated with providing power to the 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 buses.

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 applicable SRP-LR chapters as discussed in Item III.E (below). 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, as well as improper operation, restrike, shorting, and arcing. These effects are summarized in SRP-LR C.2.4, "Relays, Circuit Breakers, and Switchgear.")

2. protective relaying and control

3. 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 given in this SRP-LR chapter.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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. Components of the switchyard system not specifically addressed in this chapter may be addressed in the generic aging topic reviews in Part C of this review plan. Specifically, the following SRP-LR chapters are applicable to the switchyard and should be used in the review: C.2.1 "Cable and Wiring," and C.2.4 "Relays, Circuit Breakers, and Switchgear." In particular, this system is affected by circuit breaker failures and the review under SRP-LR C.2.4 for circuit breakers should be emphasized. This review 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

As addressed in SRP-LR, 4.2.2, studies performed as a part of the NRC Nuclear Plant Aging Research (NPAR) Program show that of the Class 1E power system components, circuit breakers have the highest frequency of LER events (66.3 percent of all Class 1E power system events), followed by transformers (4.7 percent of all Class 1E power system events). Studies utilizing Nuclear Plant Reliability Data System (NPRDS) information show 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, and J.L. Edson, NUREG/CR-5181 (EGG-2545), May 1990.

2. Electric Power Research Institute, "Generic Guidelines for the Life Extension of Plant Electrical Equipment," EPRI EL-5[illegible]85, July 1988.

B.4.7.1 DC CONTROL POWER SYSTEM

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Electrical Systems Branch (SELB)
Instrumentation & Control Systems Branch (SICB)

I. AREAS OF REVIEW

A. This SRP-LR chapter 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 battery 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 supply the power needed for controlling the switchyard breakers and switches.

Some nuclear plants 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 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 3.0.1 for Item I.B.

C. The switchyard dc control power system is composed primarily of components that are reviewed in accordance with applicable SRP-LR chapters as discussed in Item III.E (below). In addition, protective relaying and controls is reviewed in accordance with SRP-LR B.4.1.1.

The dc control power system can be degraded by deterioration 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 power source (Refs. 2 and 3). Bus systems age with dielectric stress or partial discharge, cooling system degradation, and bus heating (Ref. 4).

A variety of aging mechanisms can affect the ability of the dc control power system to continue to operate safely and effectively. Typical examples of age-related degradation associated with the dc control power system are given in this SRP-LR chapter.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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. Components of the dc control power system not specifically addressed in this SRP-LR chapter may be addressed in the generic aging topic reviews in Part C of this review plan. The following SRP-LR chapters are applicable to the dc contol power system and should be used in the review: C.2.1, "Cable and Wiring"; C.2.2, "Junctions"; and C.2.4, "Relays, Circuit Breakers, and Switchgear." This review 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

Components of the switchyard dc control power system deteriorate with age. Studies conducted within the framework of the Nuclear Regulatory Commission's Nuclear Plant Aging Research (NPAR) Program have investigated the effects of aging on various components (Refs. 1-4). The material that follows discusses the results of those studies.

A. Batteries

NUREG/CR-4457 presents an evaluation of the aging effects of aging on safety-related batteries. It also gives an evaluation of maintenance, testing, and monitoring practices and 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; that record can be used for evaluating the state of the battery. Recommended battery maintenance includes testing battery capacity and replacing a battery should its capacity be less than 80 percent of the manufacturer's rating. Other recommended maintainence includes performing service tests, determining the original sizing criteria utilized, and comparing available capacity to the required load. Other test methods may be developed to improve the analysis of 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) presents an evaluation of the effects of aging on safety-related battery chargers. Electrolytic capacitors, trans-formers, 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 such failures are difficult to predict. 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. Operating 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 age the battery prematurely. Electrical transients, such as could be caused by a noisy setpoint potentiometer, can also age the battery prematurely.

All battery chargers have some maintenance and capacity testing performed on them, varying from capacity tests to inspection and calibration to cleaning, adjustment, component replacement, connection tightness verification, and component mounting torque verification. Some licensees observe and record the internal temperatures of battery chargers. The internal temperature is known to affect 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, (EGG-2545), May 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 - License Renewal Project Directorate (LRPD)
Secondary - Instrumentation & Control Systems Branch (SICB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the information systems.

1. Description

The information systems important to license renewal consist of the safety parameter display system (SPDS) and the displays required by Regulatory Guide (RG) 1.97 (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 or digital displays, or both.

b. The display system required by RG 1.97 combines 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.

Information systems 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. The SPDS displays 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 is not required to operate during or after an accident. The SPDS enhances operator ability to comprehend plant conditions and to act in situations that may require human intervention.

b. RG 1.97 displays indications of plant variables that are needed by 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 or digital displays or both, power sources, racks, and panels.

   b. The RG 1.97 display system will consist of sensors, electronic isolation devices, fiber-optic cables, input and output modules, analog or digital displays (or both), 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 consider the specific aging characteristics of the component materials (see SRP-LR C.3.2).

   The components of both systems, which are susceptible to aging or age degradation, or both, include both analog and digital integrated circuits, resistors, capacitors, wiring, terminal blocks, semiconductors, potentiometers, and printed circuit (PC) boards.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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. Components of the information system not specifically addressed in this SRP-LR chapter may be addressed in the generic aging topic reviews in Part C of this review plan. SRP-LR C.3.2, "Electronic Devices," is applicable to the information system and should be used in the review. This review 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. 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, Revision 3, May 1983.

B.5.0 PROCESS AUXILIARY SYSTEMS

B.5.1 OFFGAS SYSTEM (BWR)

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Plant Systems Branch (SPLB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the offgas system (OGS) of boiling-water reactors (BWRs).

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 pre-heater.

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 facility.

2. System Function

The OGS reduces radioactive releases below established limits to protect the public, supports the main condenser function by removing and providing a safe path for processing radioactive noncondensibles and other noncondensibles, provides recombination of radiolytic hydrogen and oxygen to prevent detonation and subsequent equipment damage, and 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 for radioactive decay by the OGS before discharging to the environment. Besides controlling the release of radioactive material, the offgas system is equipped to recombine hydrogen with oxygen to prevent hydrogen detonation.

The main components of the OGS are the steam jet air ejector; holdup (delay) pipe for radioactive decay; hydrogen recombiner, moisture condenser; hydrogen analyzer; gas compressor; offgas HEPA filters, including pre-filters, pre-heaters, and a charcoal delay bed.

The OGS radiation monitors are part of the process radiation monitoring system 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 or replacement, motor failure, heating and cooling coil failures, or both, and instrumentation and control failures are expected to continue throughout the life of the plant, including the license renewal period.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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. Components of the OGS not specifically addressed in this SRP-LR chapter may be addressed by the generic aging topic reviews in Part C of this review plan. The reviewer should ensure that structures and components included as part of the generic topics are adequately reviewed for this system. The following SRP-LR chapters are applicable to the OGS and should be used in the review: 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 C.3.0, "Instrumentation." This review 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.5.2 RADIATION MONITORING SYSTEM

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Instrumentation & Control Systems Branch (SICB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the radiation monitoring system (RMS).

1. Description

The subsystems of the RMS are hard-wired stand-alone systems that work with other information and control systems and the plant's control room. The RMS contains all of the equipment necessary to perform its functions.

The RMS and its subsystems 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 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 has three major subsystems that monitor radiation levels of (a) the working and storage areas of the facility (plant area monitors), (b) the process fluid flows that may discharge radioactive materials (process monitors), and (c) the atmospheric environment surrounding the facility (environmental monitors).

The area monitors survey the gamma radiation within the facility and warn of any excessive changes. The system gives operating personnel a record of gamma radiation levels within the facility.

The process monitors survey the radiation levels of liquid and gaseous processes throughout the facility. The system helps control the release of radioactive byproducts within set limits and warns personnel of abnormal radiation levels. It also gives operating personnel with a record of radiation levels.

The environmental monitors measure and establish natural background and other radiation levels, determine the facility's contribution to the environmental radiation levels, and help the licensee comply with public health and safety regulations. The monitoring stations in selected areas provide representative samples of the atmosphere, fallout. vegetation, and water around the plant to determine levels of radioactivity.

3. System Boundaries

The review of the RMS is limited to that of equipment and interfaces as shown on the piping and instrumentation diagrams (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, "Electronic Components".

Typical examples of age-related degradation associated with the RMS are provided in this SRP-LR chapter.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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. Components of the RMS not specifically addressed in this SRP-LR chapter may be addressed in the generic aging topic reviews in Part C of this review plan. SRP-LR C.3.2 "Electronic Components," is applicable to the RMS and should be used in the review: This review 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.5.3 COMPONENT COOLING WATER SYSTEM

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Plant Systems Branch (SPLB)
Materials & Chemical Engineering Branch (EMCB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the component cooling water (CCW) system.

1. Description

The CCW system is required for safe shutdown during normal operation, transient, and accident conditions. The CCW system is a closed-loop cooling system that transfers the heat from vital loads to the service water system.

The loads cooled or supplied by the CCW system 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 CCW 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 CCW system to provides a closed-loop cooling system as an additional barrier from potentially contaminated reactor systems.

3. System Boundaries

The CCW system includes 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 system is different from plant to plant, but in general, it 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 Part C of this review plan.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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. The CCW components not specifically addressed in this SRP-LR chapter may be addressed by the generic aging topic reviews in Part C of this review plan. The reviewer should ensure that structures and components included as part of the generic topics are adequately reviewed for this system. The following SRP-LR chapters are applicable to the CCW system and should be used in the review: 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 review 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.5.4 SERVICE WATER SYSTEM

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Plant Systems Branch (SPLB)
Materials & Chemical Engineering Branch (EMCB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the service water system (SWS).

1. Description

The SWS typically includes two subsystems: essential (or standby) service water (ESW) and nonessential service water. The latter is not safety related. 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, in order to control the temperatures of reactor coolant and critical components, and returns the cooling water to the ultimate heat sink.

The SWS 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.

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

C. Of the wetted systems in the power plant, the SWS may have some of the more aggressive combinations of degradation factors, even though the temperatures experienced by the materials are relatively low (typically, 35 °F to 120 °F). The following is a listing of important aging degradation mechanisms that have been manifested in the ESW system. Characterization and localization of the listed degradation mechanisms can be found in References 1 through 3.

1. Corrosion

   - general surface
   - galvanic
   - pitting
   - under deposit
   - microbiologically influenced corrosion (MIC)
   - induced chemical

2. Biological attack

   - surface biofouling
   - macro-blockage
   - MIC

3. Deposition

   - siltation (micro-impurity)
   - debris (macro-impurity)

4. Erosion

   - cavitation
   - solids impingement

5. Mechanical agitation

   - vibration (low amplitude)
   - impact (high amplitude)

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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. for Items II.A and II.B.

C. In addition, the applicant's program should include:

- implementation and maintenance of a surveillance program to control flow blockage resulting from biofouling
- verification of heat transfer capability of safety-related heat exchangers cooled by service water
- 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 SRP-LR chapter may be addressed by the generic aging topic reviews in Part C of this review plan. The reviewer should ensure that structures and components included as part of the generic topics are adequately reviewed for the SWS. All of the SRP-LR chapters in Part C are applicable to the SWS and should be used in the review, except for C.1.5, "Tanks and Vessels" and C.4.0, "Civil Structures". This review 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

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.

3. U.S. Nuclear Regulatory Commission, "Nuclear Plant Service Water System Aging Degradation Assessment," Vol. 2, NUREG/CR-5379 September 1990.

B.5.5 ULTIMATE HEAT SINK

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Plant Systems Branch (SPLB)
Structural & Geosciences Branch (ESGB)

I. AREAS OF REVIEW

A. This SRP-LR chapter 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, acts as a backup water supply for the auxiliary (emergency) feedwater system at many plants. Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants" (for comment), 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 or dry forced-draft cooling towers, and 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 dissipates residual heat after reactor shutdown and after an accident for typically 30 days to allow time to evaluate the situation and to take corrective actions.

For a single nuclear power plant unit, the UHS is required to supply sufficient cooling water to accomplish each of these two safety functions. For a multiple-unit plant, the UHS must 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 at 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 the multiple 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 or mechanical water cooling systems, or both, with, but not including, the cooling water system intake structure. The intake structure is discussed in SRP-LR 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:

- fatigue, for example, as applied to rotating fans or pumps
- general wear of mechanical moving parts
- thermal embrittlement of certain parts such as plastic seals, and of electrical and I&C components
- 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)
- erosion and embrittlement of plastic piping used in cooling towers and spray systems and of plastic fill used in wet cooling towers
- plugging or fouling, or both, of cooling tower coils
- potential long-term biological effects on water basins, cooling towers, and related equipment
- erosion and freeze damage of concrete dams, concrete basin walls, dikes, intake structures, ponds, or water basin soil shapes
- 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 Part C of this review plan. Aging-related issues of particular concern in regard to the wet cooling tower, the dry cooling tower, and spray system components include the following:

1. 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.

2. 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), but also to air-side corrosion, fouling, and plugging of the extended heat transfer surfaces. Corrosion of a fin-tube connection has a particularly devastating effect on coil performance and cannot always be detected by visual examination.

3. Spray pond nozzles are subject to degradation over time from abrasive wear by particulates suspended in the water and from erosion and corrosion. This degradation results in a modification of spray droplet size and dispersal and can degrade cooling pond performance.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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 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.

- UHS components not specifically addressed in this SRP-LR chapter may be addressed by the generic aging topic reviews in Part C of this review plan. The reviewer should ensure that structures and components included as part of the generic topics are adequately reviewed for this system. All the SRP-LR chapters with the exception of C.1.5, "Tanks and Vessels," are applicable to the UHS and should be used in the review. This review 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.5.6 REFUELING SYSTEM

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Plant Systems Branch (SPLB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the refueling system.

1. Description

The refueling system is designed for moving and for replacing the fuel assemblies in the reactor core. The refueling cavity is flooded with water so that the fuel can be moved under a sufficient depth of water to minimize radiation exposure to operators. Fuel is lifted and maneuvered remotely by 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 facility.

2. System Function

The refueling system transfers new and spent fuel between the spent fuel pool (SFP) and the reactor, shuffles fuel assemblies within the SFP and the reactor, and loads spent fuel in the shipping cask. The refueling system design 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 that transfers fuel between the SFP and the reactor refueling cavity across the containment wall.

The equipment parts used to lift the 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:

- o fuel bridge (or manipulator) inside the containment
- o fuel bridge (or long-handled tool hooked to a crane) inside the SFP building
- o fuel transfer carriages and winch devices
- o fuel assembly upenders (typically hydraulically operated)
- o instruments and controls, including limit switches, sensors, and transmitters
- o reactor vessel head stud tensioners and lifting devices and associated crane equipment
- o reactor internal lifting devices
- o radiation detectors associated with the refueling system (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. Limit switches and locking devices on the grapples could degrade with age to cause or allow dropping or misalignment of fuel assemblies. Typical examples of age-related degradation associated with the refueling system are listed below:

- o age-related wear of limit switches
- o corrosion and loose connections on instruments, control components, and electric components
- o cracked electric cables and moisture intrusion
- o grapple-locking devices failing to lock
- o excessive backlash in indexing devices on fuel bridges, carriages, and upenders
- o binding of sliding components, such as telescopic fuel masts and fuel transfer carriages

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant'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. Refueling system components not specifically addressed in this SRP-LR chapter may be addressed by the generic aging topic reviews in Part C of this review plan. The reviewer should ensure that structures and components included as part of the generic topics are adequately reviewed. The following SRP-LR chapters are applicable to the refueling system and should be used in the review: C.1.2, "Valves"; C.1.3. "Equipment and Component Supports"; all of C.2.0, "Electrical" (except C.1.5, "Transformers"); and all of C.3.0, "Instrumentation." This review may require additional staff support.

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

VII. REFERENCES

B.5.7 SPENT FUEL STORAGE FACILITY

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Plant Systems Branch (SPLB)
Materials & Chemical Engineering Branch (EMCB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the spent fuel storage facility.

1. Description

Nuclear plants have facilities for the wet storage of irradiated fuel assemblies. The spent fuel storage facility consists of all facilities and tools designed for: the safe removal of 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. The facility also contains 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 facility.

2. System Function

The spent fuel pool and storage racks maintain the spent fuel assemblies in a safe and subcritical array during all credible storage conditions, and 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 equipment, all monitoring equipment associated with spent fuel storage, and all tools used in the manipulation of the irradiated fuel from the time it leaves the reactor vessel until 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. The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant'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. The spent fuel storage components not specifically addressed in this SRP-LR chapter may be addressed by the generic aging topic reviews in Part C of this review plan. The reviewer should ensure that structures and components included as part of the generic topics are adequately reviewed for this system. The following SRP-LR chapters are applicable to the spent fuel storage facility and SRP-LRs should be used in the review: 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 all of C.4.0, "Civil Structures." This review may require additional staff support.

F. Specific areas of concern to be addressed by the applicant's program for monitoring aging in the spent fuel storage facility are

1. The reviewer should verify that the applicant has an established program of inspection and surveillance practices to detect aging and wear-related degradation in the structures housing the facility and in the facility itself, including the monitoring system.

2. The reviewer should verify that the applicant 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. REFERENCES

B.5.8 COMPRESSED AIR SYSTEM

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Plant Systems Branch (SPLB)

I. AREAS OF REVIEW

A. This SRP-LR chapter 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. In some plants, the CAS may encompass two systems, one of which supplies compressed air to safety-related equipment and the other of which supplies compressed air system to non-safety-related equipment (these are normally referred to as instrument air and service air, respectively). Facilities with containments inerted with nitrogen may also have a nitrogen compressor that takes suction from the containment and supplies containment loads. As used in this SRP-LR chapter, 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 facility.

2. System Function

The CAS provides air or nitrogen to safety-related equipment and to equipment used only for normal operation. Compressed air or nitrogen is vital for maintaining stable plant operation; its loss often results in a reactor trip and, on occasion, in 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 receiver(s), discharge piping and containment penetration, and associated isolation valves. Some facilities may have 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:

- contaminants
- corrosion
- wear
- material deterioration of seals and gasket

Additional information regarding age-related degradation of CAS components can be found in NUREG/CR-5419 (Ref. 1).

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the licensee's program for managing 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 for Items II.A and II.B.

C. The applicant has performed a one-time evaluation of the wall thickness of a representative sample (10 percent) of the CAS piping and storage tanks 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 assess the applicant'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 maintenance includes regular monitoring of some parameters and inspection of many components, CASs have failed more often than was expected. A testing and maintenance program could provide indications of degradation due to aging. Table B.5.8-1 provides typical testing and maintenance actions.

VII. REFERENCE

1. U.S. Nuclear Regulatory Commission, "Aging Assessment of Instrument Air Systems in Nuclear Power Plants," NUREG/CR-5419, January 1990.

Table B.5.8-1 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 pre filters and afterfilters	Semiannually

Table B.5.8-1 (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.6.0 AUXILIARY SYSTEMS

B.6.1 FIRE PROTECTION SYSTEM

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Plant Systems Branch (SPLB)

I. AREAS OF REVIEW

A. This SRP-LR chapter 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 control room identifying the location of the fire indicator. Fire is suppressed by means of a fire water system, a carbon dioxide system for specific applications at specific plants, and a halon system, also for specific applications. Fire suppression systems may be automatically actuated or may require manual action, depending on 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 or low pressure, or both.

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 facility.

2. System Function

The fire protection 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. The PPS is also designed to supply simultaneous operation of the hose station. The system is provided for those areas in which equipment could be exposed to fire. 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 ensured.

3. System Boundaries

The FWS is bounded by 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.

Wells are often the water source for FWSs. In some cases, precipitation of dissolved solids from the well water supply creates a chronic plugging problem.

FWS piping has the potential for erosion or corrosion or both. This potential increases as the system ages. The resulting degradation (pipe thinning) could create a potential for a break in the later part of plant life, even if the problem is not indicated at the time the staff reviews the application for license renewal.

Since the FWS piping is subjected to stagnant conditions or to operation at low or intermittent flow, it is susceptible to microbiologically influenced 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 period.

Typical examples of age-related degradation associated with the fire protection system are given in this SRP-LR chapter.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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. FPS components not specifically addressed in this SRP-LR chapter may be addressed by the generic aging topic reviews in Part C of this review plan. 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 SRP-LR chapters are applicable to the fire protection system and should be used in the review: 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, "Instrumentation." This review may require additional staff support.

F. The reviewer should confirm that the IPA includes 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 or repair records or both. 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 applicant's IPA:

- nondestructive examination (NDE) of welds and pipe fittings where dead-end piping (e.g., standpipes) connects to circulating piping

- NDE for pipe wall thickness in regions of high fluid velocity and turbulence to assess erosion or corrosion, and in areas of stagnant flow to assess corrosion

- physical inspection of fire protection pumps in accordance with National Fire Protection Association (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

B.6.2 COMMUNICATIONS SYSTEM

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Instrumentation & Control Systems Branch (SICB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the communications system.

1. Description

The communications system 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

Those parts of the communications system 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. Those portions of the system that are under consideration and important to license renewal must remain operable during a loss of offsite power.

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

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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.

IV. FINDINGS

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

V. IMPLEMENTATION

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

VI. GENERAL INFORMATION

The communications system and the electronic devices used in it are or can be subject to the aging characteristics of their individual components and, therefore, the maintenance or replacement schedules should consider the specific aging characteristics of the component materials (see SRP-LR C.3.2, "Electronic Devices").

VII. REFERENCES

B.6.3 CONTROL ROOM HABITABILITY SYSTEM

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Materials & Chemical Engineering Branch (EMCB)
Plant Systems Branch (SPLB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the control room habitability system.

1. Description

The typical control room habitability 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. Radiation monitors and toxic gas monitors in the supply ducts 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 facility.

2. System Function

The control room habitability system provides 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 fresh air used to ventilate the control room for both the comfort of 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. The system includes ducts, pipes, fans, motors, filters, a 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 or replacement, motor failure, heating and 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 concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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. Components of the control room habitability system not specifically addressed in this SRP-LR chapter may be addressed by the generic aging topic reviews in Part C of this review plan. The reviewer should ensure that structures and components included as part of the generic topics are adequately reviewed for this system. The following SRP-LR chapters are applicable to the control room habitability system and 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 C.3.0, "Instrumentation." This review 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.6.4 AUXILIARY HEATING, VENTILATION, AND AIR CONDITIONING SYSTEM

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Plant Systems Branch (SPLB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the auxiliary heating, ventilation, and air conditioning (HVAC) system.

1. Description

Typical auxiliary HVAC systems important to license renewal include, but are not limited to, the diesel building, the spent fuel pool storage area, and the turbine building.

Typically, the boiling-water reactor (BWR) turbine building (area) HVAC system is a safety-grade system. The air is exhausted through a filter bank and discharged to the vent stack. This operation occurs 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 maintains 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, supply the combustion air for the diesel engines.

For the typical auxiliary building HVAC system, the path of ventilation air goes from clean, or low-activity areas toward areas of progressively higher activity. Ventilation air is drawn from the outside through two makeup air units. The system is balanced to keep the auxiliary building at a slightly negative pressure 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 supplies ventilation during normal conditions. For high-radiation conditions, the system 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. If high radiation is 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). The standby gas treatment system (see SRP-LR B.3.6) supplies ventilation and filtration during high radiation conditions.

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 provides maximum safety and comfort for operating personnel. Equipment is arranged in an uncontaminated area for easy access during testing and maintenance. The system supplies cooling in summer and heating in winter, and ventilates gases and fumes from the area. In BWR plants, this system cleans the area atmosphere of potential gaseous and airborne particulate radioactive contamination.

The diesel building HVAC system maintains room ambient temperatures low enough so that the diesel generator life is not degraded during normal operating and shutdown periods. It also maintains room temperatures in a tolerable range for personnel to perform maintenance and surveillance work, and prevents the accumulation of building heat.

The auxiliary building HVAC system is designed for the maximum safety and convenience of operating personnel; equipment is 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 safeguards equipment may be impaired.

The typical spent fuel pool area HVAC maintains ventilation to permit personnel access, and to control airborne radioactivity in the area during normal operation and anticipated operational transients, and following postulated fuel-handling accidents.

3. System Boundaries

The components considered in the review of the auxiliary HVAC systems include the following:

- ventilation system ductwork

- ventilation system inlet dampers, exhaust dampers, and flow distribution dampers

- ventilation system supply fans, drive motors, and fan housings

- ventilation system exhaust fans, drive motors, and fan housings

- o ventilation system motor control centers for the supply and exhaust fan drive motors

- o electrical and instrumentation control circuitry for the supply and exhaust ventilation fan motors and damper operators

- o exhaust filter units, housings, supports, and heating and cooling coils

- o ventilation exhaust stack

- o 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 of mechanisms for age-related degradation within these HVAC systems are the same as for other safety-related or non-safety-related ventilation systems. 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 and replacement
- o cooling coil repair and replacement

These concerns are expected to continue throughout the life of the plant, including the license renewal period.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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. HVAC components not specifically addressed in this SRP-LR chapter may be addressed by the generic aging topic reviews in Part C of this review plan. The reviewer should ensure that structures and components included as part of the generic topics are adequately reviewed for this system. The following SRP-LR chapters are applicable to the auxiliary HVAC system and 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 C.3.0, "Instrumentation." This review 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

PART C: GENERIC LICENSE RENEWAL TOPICS

Part C addresses the review of the various generic components and structures that need to be evaluated for age-related degradation.

SRP-LR

C.0.1 GENERIC COMPONENTS AND STRUCTURES REVIEW CRITERIA

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 SRP-LR.

I. AREAS OF REVIEW

A. See individual generic topic SRP-LR chapter for a description of the structure or component, its function, and its boundaries.

B. The SRP-LR addresses aging degradation of the subject component or structure 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 applicant has conducted an integrated plant assessment (IPA) to identify potential age-related degradation of systems, structures, and components and to evaluate the adequacy of its programs to identify and mitigate age-related degradation for the renewal period. The FSAR supplement supporting license renewal lists (1) systems, structures, and components identified as important to license renewal and (2) structures and components requiring evaluation of age-related degradation. The reviewer's safety evaluation for the subject system will be found in the corresponding section of the safety evaluation report (SER) supporting license renewal.

The reviewer of issues contained in this review plan is not intended to be a review of the existing licensing basis. The actual licensing basis for an individual plant is contained, in part, in the FSAR specific to that facility, and the NRC staff documented its review of the FSAR in the SER it prepared to support the original operating license.

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

See individual generic topic SRP-LR chapter for a discussion of the specific age-related degradation concerns related to this component or structure.

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*"Industry codes and standards" refers to the codes, standards, practices and specifications of such groups as the American Concrete Institute (ACI), the American Institute of Steel Construction (AISC), the American Society of Mechanical Engineers (ASME), and the Institute of Electrical and Electronic Engineers (IEEE).

II. ACCEPTANCE CRITERIA

Acceptance and performance criteria for specific structures or components are typically contained, in part, in such sources as technical specifications; industry codes and standards*; root-cause analyses; failure-mode analyses; equipment performance history; NRC branch technical positions; approved topical and other industry reports; vendor criteria; and NRC regulatory guides. For specific components, the vendor recommendations for extending the life of a component 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 that may require replacement of selected parts. Applicants may have to review a previously completed analysis if aging issues raised questions concerning the analysis (including its assumptions). The accept-ability of the applicant's proposed program for identifying monitoring, and mitigating the effects of age-related degradation related to specific structures and components will be based on the following criteria:

A. The applicant has performed and documented an IPA demonstrating that degradation related to the aging of specific structures and components has been identified, evaluated, and accounted for as necessary to ensure that the plant's current licensing basis will be maintained throughout the period of the renewed license. As part of the IPA, the applicant has described the methodology for identifying systems, structures, and components (SSCs) important to license renewal, as well as structures and components requiring evaluation of age-related degradation, and has listed these SSCs and structures and components. Part C is restricted to the review of specific components and structures.

B. As part of the IPA, the applicant 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 period of the renewed license. The review focuses on the following items:

   1. The applicant 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 either that the specific structures and components are either capable of performing their safety functions during the renewal period or that, if disgraded, they will not interfere with 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 not be 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:

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*"Industry codes and standards" refers to the codes, standards, practices and specifications of such groups as the American Concrete Institute (ACI), the American Institute of Steel Construction (AISC), the American Society of Mechanical Engineers (ASME), and the Institute of Electrical and Electronic Engineers (IEEE).

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 be 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 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 applicant 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, the applicant may claim the equipment qualification (EQ) program required by 10 CFR 50.49 is an established effective program for selected electrical components. But, for a subset of these components, either extensive additional testing is required or a reanalysis (with appropriate justification documented or selective verification testing required, as appropriate) must be performed in order for the EQ program applied to the renewal period.

2. For those specific structures or components identified as requiring evaluation of age-related degradation, but which are not included in an established effective program, the applicant 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 age-related 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,

   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 period of the renewed license.

C. The applicant has identified plant-specific exemptions granted pursuant to 10 CFR 50.12, "Specific Exemptions," and reliefs granted pursuant to 10 CFR 50.55a, "Codes and Standards." The applicant 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 period of the facility or that were otherwise related to SSCs subject to age-related degradation.

D. Additional criteria are discussed in the SRP-LR chapters for specific structures and components, as applicable.

III. R''IEW PROCEDURES

Upon request from the primary reviewer (LRPD), the secondary review branch(s) will provide material for the areas of review identified in Section I above ("Areas of Review"). The primary reviewer obtains ar such information as required to ensure that this review procedure complete.

The reviewer should adhere to these procedures for reviewing of t. pecific structure or component to determine whether: (1) the structures a 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 applicant for the specific components and structures (typical examples are provided in Item I.C in each SRP-LR in 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 period, 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 applicant's program for license renewal based on the acceptance criteria given in Section II above (Acceptance Criteria").

A. The reviewer should confirm that an IPA has been documented and submitted which demonstrates that age-related degradation in 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. Part C of this review plan is limited to specific components or structures.

B. The reviewer should verify the applicant 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 the applicant has identified 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 1.C of each SRP-LR in 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 applicant demonstrates ongoing programs are adequate for timely mitigation of age-related degradation. The support for this determination could focus on review 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 applicant'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 for managing age-related degradation are implemented by ensuring the plant program defines inspection methods used, inspection frequency, and 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 a condition of acceptable performance in accordance with the 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 applicant'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. The reviewer should review 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 period of the license renewal.

E. Additional review procedures are discussed in the SRP-LR chapters for specific structures and components, as applicable.

IV. FINDINGS

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

A. The applicant analysis acceptably identified the specific structures and components requiring evaluation of age-related degradation.

B. The applicant 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 was identified, evaluated, and accounted for as necessary to ensure that the plant's current licensing basis will be maintained throughout the period of the renewed license.

C. The applicant has proposed or is implementing an effective program for license renewal for this specific component or structure 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 the current licensing basis over the renewed license period. For components and structures not covered by an existing or new program, the applicant has provided acceptable justification that the degradation experienced over the renewal period will not be significant.

D. The applicant has 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, "Codes and Standards." The justifications for continuing the exemptions and reliefs are acceptable for the renewal period.

E. The applicant has adequately identified and justified any proposed modifications to the facility or its administrative control and plant procedures necessary to manage age-related degradation.

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 applicant proposes an acceptable alternative method for complying with specified portions of this review plan, the staff plans to use the methods described herein during its evaluation.

VI. GENERAL INFORMATION

Addressed in individual generic topic SRP-LR chapters.

VII. REFERENCES

Addressed in individual generic topic SRP-LR chapters.

C.1.0 MECHANICAL

C.1.1 PIPING

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Mechanical Engineering Branch (EMEB)
Materials & Chemical Engineering Branch (EMCB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses piping identified as being important to license renewal. Design codes for this piping typically contain corrosion allowances, which are based on the service life of the piping system. The 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. 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 influenced corrosion (MIC). Typical degradation mechanisms and degradation sites are discussed below.

1. High-cycle thermal and mechanical fatigue can affect pressurized-water reactor (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 and spargers; spray, surge, charging, safety injection, and residual heat removal lines; boiling-water reactor (BWR) high-thermal stress regions of recirculation piping; feedwater nozzles; and main steam and feedwater piping near fittings and other piping discontinuities (such as tees, orifices, elbows, and valves).

2. Thermal embrittlement can affect PWR cast stainless steel piping and BWR cast stainless steel recirculation piping.

3. Intergranular stress corrosion cracking (IGSCC) can affect 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 can affect 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 can affect PWR feedwater piping and BWR main steam and feedwater piping near piping discontinuities (such as elbows, pipe joints, flow control valves, and orifices).

7. Corrosion can affect BWR main steam and feedwater piping near structural discontinuities.

8. Other degradation mechanisms include crevice corrosion, erosion, pitting, galvanic corrosion, MIC, intergranular attack, transgranular stress corrosion cracking, hydrogen embrittlement, and oxidation.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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 C.0.1 for Items II.A through II.C.

D. Additional Criteria

1. Thinning of Pipe Wall

The applicant has examined piping systems for reduced wall thickness caused by general and uniform corrosion and erosion/corrosion.

A representative sample (10 percent) of the piping that is exempt from the inspections required by ASME Code, Section XI (Ref. 1), has been nondestructively examined to ensure that the pipe wall is sufficiently thick to satisfy design requirements through the license renewal period. A one-time examination has been performed within 5 years of the license renewal application as a minimum. In accordance with NRC Generic Letter 89-08 (May 1989), applicants are required to implement a long-term erosion/corrosion monitoring program that provides assurances that procedures and administrative controls are in place to ensure that an effective program is implemented to maintain the structural integrity of all high-energy (including two-phase and single-phase flow) carbon steel systems.

For those cases in which the wall thicknesses do not comply with these criteria, the applicant has tracked wall thickness to identify trends, and has specified action levels.

2. Fatigue--Thermal, Vibratory, and Pressure

The applicant has verified, using plant-specific fatigue analyses for all ASME Class 1 piping, that the ASME Code, Section III (Ref. 2), cumulative usage factor allowable of 1.0 will not be exceeded during the projected lifetime of the plant.

3. Stress-Corrosion Cracking

The applicant has examined a representative sample (10 percent) of the piping exempt from inspections required by ASME Code, Section XI, for evidence of intergranular and transgranular 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 applicant has examined 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 Code, Section XI, Articles IWB-2000, IWC-2000, and IWD-2000 (Ref. 1).

The reviewer should ensure that the examination techniques and procedures used by the applicant 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, Ar) IWA-2000.

Alternative examination methods, combination of methods, or newly developed techniques to those given in Article 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, Article IWB-3000, IWC-3000, and IWD-3000.

The samples were obtained from the following areas of piping systems exempt from the inspections required by ASME Code, Section XI: 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 was inspected However, the entire circumference of those pipes in areas susceptible to erosion and corrosion or two-phase flow should be inspected in accordance with the methods of ASME Code, Section XI.

If the applicant discovers areas where the original design corrosion allowance has already been used or is expected to be used before the end of the renewal period, wall thickness should be tracked during the license renewal period on a basis commensurate with the wall corrosion allowance utilization to identify trends. Additional samples should be taken throughout the system to track for trends. Action levels should be specified when selected criteria have not 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 and up to 25 percent for significant thinning.

2. The applicant should list exemptions that have been permitted by ASME Code, Section XI, Article IWC-1220. The sample of exempt piping that is selected by the applicant for an additional one-time examination should be described.

3. Fatigue data should be developed to assess fatigue damage to piping. The stress factors include heatups, cooldowns, operational transients, water hammers, steam hammers, thermal shocks, stratified flows, and flow-induced and equipment vibrations. Actual in-plant thermal loadings should be completely accounted for to accurately predict the residual life of those components. Cyclical piping analyses should 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. For example, if the projected life is an additional

20 years for a total life of 60 years, the original design-basis transients from Item III.E.3.a above should be increased by 50 percent. For all components, calculate the cumulative usage factors for this additional increment of time.

c. List all cycles that have resulted from unanticipated plant transients that were not considered as design basis events in Items III.E.3.a or b above. For all components, calculate cumulative usage factors for these additional transients.

d. Add the cumulative usage factors calculated from Items III.E.3.a, b and c above to arrive at the total end-of-life fatigue usage factors for all components. The cumulative usage factor allowable of 1.0 shall not be exceeded during the projected lifetime of the plant.

e. If the rate of actual plant cycles indicates that the design-basis cycles will be exceeded at the end of plant life, the procedures in Items III.E,3.a, b and c above should be adjusted to account for these additional cycles. If the rate of actual plant cycles indicate that the design-basis cycles will not be exceeded at the end of plant life and step III.E.3.d above is satisfied, no further analysis is required for fatigue.

f. The above analyses should be performed in accordance with ASME Code, Section III, Subsections NB-3222.4(e)(5) and NB-3228.5. If the total number of stress cycles is estimated to exceed 10, the applicant should provide the basis for appropriate design fatigue curves.

g. In the above analyses, the applicant 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 before using such procedures.

i. Each applicant should list any plant-specific history of piping failure resulting from fatigue.

4. For erosion/corrosion, ASME Code, Section XI, 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. 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 fatigue damage will supplant those currently incorporated in ASME Code, Sections III (Ref. 2) and XI (Ref. 1). Nondestructive examination techniques should be used to identify small fatigue cracks so that the pipes can be repaired or replaced before the cracks penetrate the 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. 3). Hydrogen water chemistry has been implemented at several BWR plants to mitigate IGSCC (Ref. 4). 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 remerge again as a problem in BWR piping.

Long-term (greater than 40 years) thermal embrittlement of cast stainless steels is a potential problem, and research on this topic is still in progress. However, the staff has approved a Westinghouse topical report concerning thermal embrittlement of cast stainless steels for the first 40 years of operation.

2. Feedwater and BWR Main Steam Piping

SRP-LR Table C.1.1-3 summarizes the degradation processes of concern for BWR feedwater and main steam lines. SRP-LR 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.

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, and so forth. 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 in 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, can deform and crack 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 affect the pipes more severely because they tend to persist for longer periods and tend to propagate any existing cracks. Thermal shock loadings of the spray lines also can 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.

Thermal shock loading can cause crack initiation, but usually not enough cycles exist for cracks to continue to propagate. However, thermal fatigue can initiate cracking and can cause the crack to propagate a little 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.

Fatigue is the aging degradation mechanism that is pervasive throughout PWR pressurizers and associated piping. 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, and 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. Managing Aging Degradation

1. Inspection

Formal guidelines for examining pressure piping are described in ASME Code, Section XI (Ref. 1). 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 required at this time. 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 help identify potential problem areas. These programs use the alloy content, chemistry, and flow data of piping materials to determine areas in the piping most susceptible to erosion and corrosion. The subcommittee for ASME Code Case N-480 has developed inspection procedures to detect wall thinning in piping. NUMARC also has developed guidance for selecting examination techniques for specific plant situations and has provided suggestions for additional detailed examinations if erosion and corrosion is detected. EPRI has developed CHEC and CHECMATE computer programs to assist in selecting inspection locations for single phase and two-phase piping, respectively.

Utilities are developing inspection procedures to detect erosion and corrosion damage in piping. Although manual ultrasonic techniques can be used for measuring average thicknesses, these techniques have not been satisfactory for determining minimum thicknesses, which 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 geometrics 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 qualifica-tion. However, no present inspection method is completely acceptable for cast stainless steel.

Two other nondestructive examination (NDE) methods available for detecting 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 for inspecting large-diameter (202-mm [8-in.]) pipe. The perpendicular radiographic technique can detect abrupt charges in thickness within 2 percent of the wall thickness. A calibration curve of thickness versus density is required for making accurate measurements.

2. Recordkeeping and Trending

Thickness measurement data can be used to identify sites susceptible to erosion/corrosion.

Monitoring water chemistry--including oxygen level, pH level, and impurities--can assist in estimating erosion/corrosion rates.

An accounting of actual thermal loadings is needed for accurately predicting the residual life of those components. Once the loadings are more accurately defined, fatigue life can be appropriately predicted, and an appropriate inservice inspection program can be implemented with state-of-the-art techniques. A higher degree of 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 discussed below. Research is continuing in this area to assess the effectiveness of these countermeasures.

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. The stress improvement methods introduce compressive stresses at the tip of shallow cracks in the HAZ and effectively inhibit the growth of short cracks that do not exceed 30 percent of the wall thickness. However, more frequent inspections and a larger sample size are required for these welds.

Hydrogen water chemistry has been successfully used at BWRs to suppress IGSCC crack initiation and growth when the mixture 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-mhos per centimeter and the electrochemical potential is kept below -230 mV (standard hydrogen electrode). Testing to verify that online cracking has been arrested is desirable when using this technique for cracks growth control.

In addition, BWRs on hydrogen water chemistry control inject oxygen in the condensate system to maintain 20 to 50 ppb oxygen in the feedwater and condensate systems to control carbon steel corrosion.

Weld overlays introduce compressive stresses in the weldment that inhibit the initiation and growth of IGSCC. Analytical results indicate that weld overlays will inhibit the growth of cracks that do not extend beyond 60 percent of the wall thickness. The major concern with extended use of weld overlays is the difficulty in performing reliable inspections of the weldment under the overlays. However, improved ultrasonic methods are being developed for this purpose. Welds repaired by weld overlays are generally inspected within two refueling cycles following the repair.

Mechanical clamping devices introduce axial and circumferential compressive stresses in the piping and retard crack growth. In addition, such clamp ng devices provide an alternative load path around any degraded weldment to ensure its structural integrity.

When piping shop welds are heat treated with a solution, sensitization in the HAZ is eliminated, thus providing protection against IGSCC. This treatment is applicable to new piping, and about 40 percent of the welds in the recirculation piping currently are solution-heat-treated.

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.

Corrosion-resistant cladding applied 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 crack growth caused by fatigue in both the base metal and the welds of surge and spray lines. These methods potentially can 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 and not as their replacement.

Fatigue crack detection by acoustic emission methods depends on the ability of the instrumentation to detect the acoustic 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 location of the crack is determined by the length of time it takes the signals to arrive 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. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," New York.

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

3. U.S. Nuclear Regulatory Commission, "Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," NUREG-0313, Rev. 2, 1986.

4. Electric Power Research Institute, "BWR Hydrogen Water Chemistry Guidelines," NP-4927-SR, 1987, revision October 1988.

Table C.1.1-1 Summary of degradation processes for PWR reactor coolant system piping

Degradation site	Stressor	Degradation mechanism	Potential failure mode	Inservice inspection surveillance method
Main coolant pipe nozzles*	System operating transients	Fatigue crack initiation and propagation	Through-wall leakage	Volumetric inspection for diameter greater than 4 inches
Thermal end dissimilar metal weld**	System operating transients	Fatigue crack initiation and propagation	Through-wall leakage	Surface inspection for diameter less than 4 inches
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 that are the most heavily loaded are difficult to determine. Reported results are heavily dependent on the type of analysis performed. Actual damage will depend completely on the actual transient usage that occurs in the plant that 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	Stressor	Degradation mechanism	Potential failure mode	Inservice inspection surveillance method
Weld heat-affected zone of furnace-sensitized safe ends	Tensile stress, oxygen environment, sensitized heat-affected zone	Intergranular stress corrosion cracking	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	Stressor	Degradation mechanism	Potential failure mode	Inservice inspection surveillance method
Feedwater nozzles into reactor, thermal sleeves/spargers	Thermal stress, corrosive water, water-hammer, 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/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	Stressor	Degradation mechanism	Potential failure mode	Inservice inspection surveillance method
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/corrosion, mechanical fatigue	Rupture caused by wall thinning, leakage through cracks	Ultrasonic testing, radiography*
Geometric discontinuities on inside surface of piping	Flow velocity, O_2 content and pH level in feedwater, impurities, water-hammer	Erosion/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 inservice inspection requirements.

Table C.1.1-5 Summary of degradation process for PWR pressurizer surge and spray line and nozzles

Degradation site	Stressor	Degradation mechanism	Potential failure mode	Inservice inspection surveillance method
Feedwater and safety injection nozzles	Thermal transient and shock stress loadings, stratified flow stress loadings, flow-induced vibration	High- and low-cycle 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 leaks	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 intervals	Thermal transient stress loadings	Low- and high-cycle 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
	Stratified flow loadings (pipe only), thermal stripping		Thermal sleeve cracking	

Table C.1.1-5 (Cont)

Degradation site	Stressor	Degradation mechanism	Potential failure mode	Inservice inspection surveillance method
	Flow-induced mechanical vibration	Mechanical fatigue	Thermal sleeve cracking, crack initiation in nozzle	

Table C.1.1-6 Summary of countermeasures for managing degradation of reactor coolant pipe

Mechanism	Countermeasure	Mitigation	Repair	Replacement
Intergranular stress corrosion cracking	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			
	Weld overlay	X	X	
	Clamping device	X	X	
Intergranular stress corrosion cracking	Hydrogen water chemistry	X		
	Minimalization cold working in fabrication			X
Thermal embrittlement	Use of less susceptible material			X

Source: Reference 4.

C.1.2 VALVES

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Mechanical Engineering Branch (EMEB)
Materials & Chemical Engineering Branch (EMCB)

I. AREAS OF REVIEW

A. Description

This SRP-LR chapter addresses the valves identified as important to license renewal. The valves typically are those valves required to meet the ASME Code, Section XI, 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. 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.

Typical examples of age-related mechanisms affecting valves, valve components, and operators, which include corrosion, erosion, fatigue, and physical damage are discussed 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. 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.

E. The acceptance criteria for fatigue as described in SRP-LR C.1.1, Item II.D.2, is applicable to all ASME Class 1 valves.

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 performs at least one 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.3, 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

C.1.3 PUMPS

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Mechanical Engineering Branch (EMEB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses those pumps identified as being important to license renewal. These pumps are typically, but not limited to, all pumps required to meet the inservice inspection and testing (ISI/IST) requirements of the ASME Code, Section XI, Classes 1, 2, and 3.

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

The aging concerns and mechanisms listed below have been identified as applicable to the components of reactor coolant and recirculation pumps (Refs. 1-7) and may apply to other pumps covered in the scope of this SRP-LR.

1. Pump Casings

   - thermal embrittlement
   - thermal and mechanical fatigue
   - stress-corrosion cracking
   - high residual stress as a result of no post-weld 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 concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant 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 C.0.1 for Items II.A through II.C.

D. The requirements of ASME Code, Section XI, for ISI/IST of pump components shall be continued throughout the renewal period. The inservice inspection requirements of ASME Code, Section XI, which are currently limited to volumetric examinations, should be supplemented to include visual inspections. Pump shaft inspections done during shutdown should include surface and volumetric examinations.

E. The acceptance criteria for fatigue as described in SRP-LR C.1.1, Item II.D.2, are 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 applicant'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 pump shaft inspections that are performed during outages should include surface and volumetric examinations.

The ISI/IST program should be continued throughout the license renewal period.

The review procedure for fatigue as discussed in SRP-LR C.1.1, Item III.E.3, 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 degradation mechanisms and inspection mechanisms for pump casings, closure studs, and pump shafts are discussed below.

A. Pump Casings

1. Degradation

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 length of time the metal is exposed at that temperature.

The pump casing is subjected to thermal and mechanical fatigue damage caused by the operating transients and pump vibrations. Pump welds made using the electroslag technique are susceptible to fatigue damage because of 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

Pumps within the scope of this SRP-LR will be disassembled for inspection and maintenance at intervals specified in the applicant's 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 generally is used for volumetric examination of welds for a 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 ISI methods. However, advanced ultrasonic testing (UT) methods are being developed that can better detect flaws and determine their size, orientation, and location (Refs. 1 and 2). Those 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 ISI requirements, which are currently limited to volumetric examinations, should be supplemented to include visual examinations. Removing 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. 4). 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 (LWRs) 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. 5 and 6). 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 outages 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

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5. S. D. Leshnoff and P. C. Riccardella, "Reactor Coolant Pump Shaft Crack Investigations at TMI-1," Proceedings of the Conference on Main Coolant Pump Diagnostics, December 1988, EPRI NP-6116, pp. 11-1 through 11-34.

6. S. M. Stoller Corporation, "RCP Shaft Fractured, Cap Screws Cracked, Broken-Fabrication, Thermal and Weld Stresses, IGSCC, Insufficient Preload," p. 522, Nuclear Power Experience, PWR-2, April 1986, p. 52.

7. U.S. Nuclear Regulatory Commission, "RCP Shaft Fractured, Cap Screws Cracked, Major Light Water Reactor Components - Overview," Vol. 1, NUREG/CR-4731 (EGG-2469), June 1987.

C.1.4 HEAT EXCHANGERS

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Mechanical Engineering Branch (EMEB)
Materials & Chemical Engineering Branch (EMCB)

I. AREAS OF REVIEW

A. This SRP-LR chapter 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 overall safety.

Steam generators that are affected by many of the aging issues are also addressed in SRP-LR B.1.2, "Reactor Coolant System." Heat exchangers identified as being important to license renewal potentially include, but are not limited to:

o <u>Heat exchangers</u>

component cooling water
service water
shutdown cooling
regenative
non-regenative
residual heat removal (RHR)

o <u>Coolers</u>

containment cooler/chiller
instrument air cooler
jacket water cooler
lube oil cooler
service air cooler

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

C. 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:

- fouling of heat transfer surfaces
- cracking of shell, tubes, or welds
- distortion of internal parts
- erosion/corrosion of internals
- blockage of flow passages
- fatigue of nozzles, tubes and supports
- microbiologically influenced corrosion (MIC) pitting
- galvanic corrosion

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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 C.0.1 for Items II.A through II.C.

D. Heat exchanger performance characteristics 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 applicant is committed.

E. The applicant shall conduct tests to show that heat exchanger tube wall thickness is no less than the original minimum design value. Alternatively, the applicant may demonstrate that the present wall thickness is adequate by suitable calculations.

III. REVIEW PROCEDURES

See Section III, "Review Procedures," of C.0.1 for Items III.A through III.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 applicant'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 records for most of the heat exchangers of interest. Where the existing 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 maintenance and repair records. These records should address observed degradation including, at the least, leaks, cracks, corrosion (e.g., general, pitting, and galvanic), erosion, fouling and 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 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, "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. Amar, A. S., and V. N. Shah, "Remedies for PWR Recirculating Steam Generator Tube Aging," 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 - License Renewal Project Directorate (LRPD)
Secondary - Materials & Chemical Engineering Branch (EMCB)
Mechanical Engineering Branch (EMEB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses the generic structural and mechanical considerations for tanks and vessels identified as being important to license renewal. Pressure vessel design codes, as applicable and as found in Article 3000 of the ASME Code, Section III (Ref. 1), for Class 2 and 3 vessels, and Section VIII (Ref. 2) 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 applicant 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 SRP-LR chapter does not cover the reactor pressure vessel, pressurizer, or steam generators, which are covered in SRP-LR B.1.1, "Reactor Pressure Vessel," and SRP-LR B.1.2, "Reactor Recirculation System." Concrete tanks are covered in SRP-LR C.4.0, "Civil Structures."

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

C. Erosion/corrosion, embrittlement, fatigue, oxidation, pitting, microbiologically influenced corrosion, and stress-corrosion cracking are typical examples of age-related degradation mechanisms that should be reviewed for tanks and vessels.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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 C.0.1.

III. REVIEW PROCEDURES

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

E. The applicant'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 above (Item I.C) for the license renewal period.

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, Articles 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 applicant's exemptions, as permitted by ASME Code, Section XI, Article IWC-1220, for Class 2 components and systems, that are exempted from the aging mitigation program, should be reviewed for acceptability for the renewal period.

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

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

C.1.6 EQUIPMENT AND COMPONENT SUPPORTS

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Structural & Geosciences Branch (ESGB)
Mechanical Engineering Branch (EMEB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses reactor pressure vessel supports, piping/equipment snubbers, piping/equipment supports, anchor bolts, and reactor shield walls identified as being important to license renewal.

There are five major types of reactor pressure vessel supports: neutron shield tank, column supports, cantilever, bracket-type, and skirt-type.

Snubbers are mechanical or hydraulic devices that allow free thermal movement of piping or equipment during normal operating conditions and that 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 and accommodate thermal displacements and design-basis dynamic loads for the piping and 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 or steel structures that surround portions of the reactor pressure vessel (RPV) to shield instrumentation and electrical equipment. The shield walls also may serve as a structural support for other components such as piping.

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

C. 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 summarized in Table C.1.6-1.

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 such environmental conditions as radiation or temperature. The major aging concern for mechanical snubbers is a loss of free-movement characteristics as a result of such environmental conditions as corrosion, high temperature, and vibration (Ref. 1).

Many plants have experienced failures of piping/equipment supports under normal operating conditions. These failures are the result of such unanticipated or improperly characterized loading phenomena as water hammer or cavitation. Metal fatigue is often the apparent cause of failure, although the root cause is likely to be a failure to properly characterize and design for cyclic loading phenomena.

Piping and equipment supports also are 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-basis accident (DBA).

Material properties of reactor shield walls and doors are subject to degradation associated with radiation exposure (primarily neutrons). Corrosion also may be a significant degradation mechanism.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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 C.0.1. for Items II.A through II.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 applicant is committed.

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 the effects of age-related mechanisms on applicable RPV supports have been adequately addressed, including the mechanisms discussed below.

- Ferritic components of support structures exposed to neutron irradiation will have degraded mechanical properties and should be examined or 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 to replace non-metallic elements of support that have exceeded 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 applicant'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 reviewer should verify that the applicant has visually examined a statistical sample of piping/equipment supports to assess corrosion damage.

4. The reviewer should verify that the applicant has inspected a statistical sample of all types of anchor bolts for torque setting (preload) and corrosion.

5. The reviewer should verify that the applicant has reconciled 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 applicant has performed a visual inspection for corrosion damage.

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 and Service Wear of Hydraulic and Mechanical Snubbers Used on Safety Related Piping and Components of Nuclear Power Plants," Vol. 1, NUREG/CR-4279 (PNL-5479), February 1986.

Table C.1.6-1 Summary of degradation processes for PWR and BWR RPV supports

Degradation site	Stressor	Degradation mechanism	Potential failure mode	Inservice inspection surveillance 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 midplane 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 teh 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	Binding that causes possibly excessive stresses in the primary system during heatup and cooldown	
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	Overload failure on fatigue-cracked component	

C.2.0 ELECTRICAL

C.2.1 CABLING AND WIRING

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Electrical Systems Branch (SELB)
Instrumentation & Control Systems Branch (SICB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses cabling and wiring identified as being important to license renewal. Cables constitute 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 safety both during normal operation and under accident conditions. The material for conductors and shields of most cables is stranded copper, often coated with tin or some other coating to prevent interaction between the copper and the insulating material and to provide good corrosion-resistant connections. The insulating materials of cables are generally polymer-based compounds, except for mineral-insulated cables (Ref. 1). Depending upon circuit requirements, insulation serves 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 I, "Areas of Review," of SRP-LR C.0.1 for Item I.B.

C. A variety of age-related mechanisms can affect the ability of cables to continue to operate reliably. Although 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 cabling and wiring approach the end of their documented qualified life. Typical examples of age-related degradation concerns associated with cabling and wiring are given below:

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.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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 C.0.1. Items II.A through II.C.

D. The applicant has or will implement a program to identify degradation due to aging. This program should include a combination of tests, analyses, and inspections to (1) define the current material condition of cables and (2) establish a replacement interval if appropriate.

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 the current material condition of cabling 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," of SRP-LR C.0.1.

VI. GENERAL INFORMATION

Studies performed as a part of the NRC Nuclear Plant Aging Research (NPAR) Program show 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 cabling.

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 normally can not be seen 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 cabling systems in a way that relates electrical measurements to the pre-aging 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 cabling. None of those test methods have been used in evaluating cabling 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 cabling; such techniques must have the potential for relating to qualification pre-aging programs. Several new or evolving aging assessment test methodologies for monitoring cabling systems include the following:

1. A mechanical cabling indenter is being developed that may be effective for surveying overall cabling environmental aging severity in a containment and for tracking the aging of cabling jackets or exposed insulations with reference to a preaged condition in qualification.

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 cabling. New concepts of these techniques that are applicable to nuclear plants are being investigated by the National Institute of Standards and Technology (NIST) and the Electric Power Research Institute (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 test which of the many previous electrical and mechanical cable monitoring tests produced data that correlate with cabling performance under LOCA/MSLB stressor conditions (Ref. 2).

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 cabling.

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, (EGG-2469) November 1989.

2. 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 PV), (Draft), March 1990.

*Formerly the National Bureau of Standards.

Table C.2.1-1 Summary of potentially important failure modes and degradation factors for LWR containment cables[a]

Failure mode	Age degradation mechanism	Dominant aging stressor	Component	Degradation site
Circuit ground or short when subject to condensing steam, spray or water common cause (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 open circuits in, total loss of, or multiple grounds on shields (CCF, random failure (RF).	Jacket cracking or moisture diffuses through jacket and condenses.	Moisture, high temperature.	Shielded coaxial or multiconductor 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, 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.	Thermal and radiation hot spots, connections where cable is moved.

See footnotes at end of table.

Table C.2.1-1 (continued)

Failure mode	Age degradation mechanism	Dominant aging stressor	Component	Degradation site
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.	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.	Nonhermetic 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, these are the problems that should be considered if qualification practices were not complete or rigorously applied or in considering the extension of the original life of components.

b. Notations in parentheses indicate potential for common cause failures (CCFs) after a design-basis accident (DBA) or submersion and for random failures (RFs) during normal or abnormal service.

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 and Environmental Interactions)

Component	Subcomponent	Typical material	Aging concern
Cable	Insulation	Crosslinked polyethylene	Thermal and radiation embrittlement, oxidation, cracking, and moisture intrusion.
	Jacket	Chlorosulfonated polyethyline	Mechanical stress, thermal and radiation embrittlement, cracking, and moisture intrusion.
	Conductor and 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	Fluorosilicone 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	Polyolefin	Cracking
Junction box	Seals	Elastomer	Embrittlement

Table C.2.1-2 (continued)

Managing Aging

Inspection and Monitoring: Preservice	Inservice	Maintenance
<u>10 CFR 50.49</u>	No requirement for inservice inspection	Performed when system performance has degraded
Calls for artificial or natural aging before environmental qualification testing	Inspection of connections following maintenance	Performed when testing identifies specific problems
<u>Regulatory Guide 1.89</u>	Monitor redundant channels for discrepancies	Components replaced at end of qualified life
Qualified life may be demonstrated, based on Arrhenius theory and a surveillance and maintenance program	End-to-end system tests during refueling outage	
	<u>Recommendations</u>	<u>Recommendations</u>
<u>IEEE-383</u>		
Provides industry guidance for qualifying Class 1E cables and connections	Develop inservice surveillance criteria	Develop advanced remote monitoring for detecting deterioration
<u>IEEE-317</u>	Perform periodic inspections	
Covers design, installation, and testing of electrical penetrations in containment structures	Perform temperature and radiation mapping in containment cable locations	Develop criteria for replacing cable in a severe environment

C.2.2 JUNCTIONS

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Plant Systems Branch (SPLB)
Electrical Systems Branch (SELB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses junctions identified as being important to license renewal. The most common types of junctions and 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 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, "Areas of Review," of SRP-LR C.0.1 for Item I.B.

C. Typical examples of age-related degradation concerns associated with junctions are provided below. However, 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 resistant to aging degradation. However contaminants may cause excessive leakage in instrumentation circuits. Butt and bolt splices may have insulation usually nylon or Kynar, that could be vulnerable to aging degradation, Raychem heat shrink tubing and 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 its mechanical separation 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 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.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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 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

Most NRC information that has been disseminated regarding connections discussed design, selection, installation, and quality assurance inadequacies, not the effects of aging.

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 reason for connector failure 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 even looser.

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. REFERENCE

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.

C.2.3 ELECTRICAL PENETRATIONS

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Plant Systems Branch (SPLB)
Electrical Systems Branch (SELB)

I. AREAS OF REVIEW

A. This SRP-LR addresses electrical penetrations identified as being important to license renewal. The electrical penetration assemblies (EPAs) covered by this SRP-LR chapter perform 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 O-rings, gaskets, and such materials as metals, plated metals, polymer-based rubbers, ceramic materials, high-strength and high-temperature glass (Ref. 1).

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

C. 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 concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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 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," M. J. Jacobs, NUREG/CR-5461 (SAND-2369), July 1990.

C.2.4 RELAYS, CIRCUIT BREAKERS, AND SWITCHGEAR

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Plant Systems Branch (SPLB)
Electrical Systems Branch (SELB)

I. AREAS OF REVIEW

A. This SRP-LR chapter addresses circuit breakers, load centers, motor control centers, power panel boards, and relays, regardless of nomenclature, identified as being important to license renewal. 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, "Areas of Review," of SRP-LR C.0.1 for Item I.B.

C. This review is restricted in scope to 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 such root causes, 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 (NUREG-0800) 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 result in 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 such operation is 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 insulation (if any) 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.

Table C.2.4-1 summarizes the stresses associated with relays and their effect on operation of the relay.

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 occur. 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.

Table C.2.4-2 summarizes aging degradation concerns associated with circuit breakers.

3. Switchgears

Switchgears 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, conductors, electrical interconnections, and accessories. The component aging mechanisms discussed above are applicable to switchgears that contain these components.

The following conditions can cause switchgears 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 switchgears 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 switchgears 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 unintended intermittent operation. 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. Excess component heating, enclosure temperature rises, and elevated ambient temperature accelerate 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 switchgears, 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.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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 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 is important in the switchgear cubicle. Dry-type transformers used in load centers are addressed in SRP-LR C.2.6.

VII. REFERENCE

1. U.S. Nuclear Regulatory Commission, "Nuclear Plant Aging Research: The 1E Power System," L. C. Meyer and J. L. Edson, NUREG/CR-5181 (EGG-2545), May 1990.

Table C.2.4-1 Stresses and effects on relays

Stress	Effect of stress on component	Effect on operation
ELECTRICAL		
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		
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 C.2.4-1 (continued)

Stress	Effect of stress on component	Effect on operation
THERMAL		
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

Table C.2.4-2 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
		Degraded insulation	
	Elevated humidity	Oxidation	Shorting and arcing
	Dirt	Hardening of lubricant	Improper operation
	Chemicals		
	Rust	Embrittlement of materials	

C.2.5 TRANSFORMERS

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Electrical Systems Branch (SELB)

I. AREAS OF REVIEW

A. Description

This SRP-LR chapter addresses transformers identified as being important to license renewal. Transformers are used extensively in nuclear power stations to transfer power by electromagnetic induction between circuits of differing 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 SRP-LR chapter. 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, "Areas 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 given below:

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 a 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.

The areas of aging concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing potential aging mechanisms. Site-specific conditions and experience are documented integrated plant assessment 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

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.

C.2.6 SOLENOID-OPERATED VALVES

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary- Electrical Systems Branch (SELB)

I. AREAS OF REVIEW

A. Description

This SRP-LR chapter addresses solenoid-operated valves identified as being important to license renewal. 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, "Areas of Review," of SRP-LR C.0.1 for Item I.B.

C. Aging Concerns and Mechanisms

This review is restricted in scope to 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.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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 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, "Nuclear Plant Aging Research: The 1E Power System," L.C. Meyer and J.L. Edson, NUREG/CR-5181 (EGG-2545), May 1990.

C.2.7 ELECTRIC MOTORS

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Electrical Systems Branch (SELB)

I. AREAS OF REVIEW

A. Description

This SRP-LR chapter addresses electric motors identified as being important to license renewal. These are used in systems needed for safe operation 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 I, "Areas 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 lubricating 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 concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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 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.3.0 INSTRUMENTATION

C.3.1 SENSORS

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Instrumentation & Control System Branch (SICB)

I. AREAS OF REVIEW

A. Description

This SRP-LR chapter addresses electronic sensors identified as being important to license renewal. These sensors 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, "Areas of Review," of SRP-LR C.0.1 for Item I.B.

C. 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 degrade with age or performance through broken connectors, lead-in wire damage, resistance changes, and electronic device failure.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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 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 (see also SRP-LR C.3.2).

VII. REFERENCE

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 - License Renewal Project Directorate (LRPD)
Secondary - Instrumentation & Control Systems Branch (SICB)

I. AREAS OF REVIEW

A. Description

This SRP-LR chapter addresses electronic components identified as being important to license renewal. They are the items from which 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, "Areas of Review," of SRP-LR C.0.1 for Item I.B.

C. 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. The main stressors may be electrical and/or thermal. Typical examples of age-related degradation concerns associated with electronic components are given below(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 high-humidity conditions.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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 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 or thermal stress or both. Other 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. Electric Power Research Institute, "A Review of Equipment Aging Theory and Technology," NP-1558, EPRI Project 890-1, September 1980.

C.3.3 ELECTRONIC DEVICES

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Instrumentation & Control Systems Branch (SICB)

I. AREAS OF REVIEW

A. Description

This SRP-LR chapter addresses the electronic devices used in systems identified as being important to license renewal. They 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 the operation of the safety-related circuit connected to the device's Class 1E side below an acceptable level. Isolation devices are used to separate-related safety circuits from non-safety-related 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, "Areas of Review," of SRP-LR C.0.1 for Item I.B.

C. 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, "Electronic Components").

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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 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 electrical stress or thermal stress or both. 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 STRUCTURES

REVIEW RESPONSIBILITIES

Primary - License Renewal Project Directorate (LRPD)
Secondary - Structural & Geosciences Branch (ESGB)

I. AREAS OF REVIEW

A. Description

This SRP-LR chapter addresses the aging factors and environmental stressors that can degrade structures identified as being important 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 buildings, and diesel generator buildings; primary containments (addressed in SRP-LR B.3.1); major load carry features such as the reactor pressure vessel (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 structures required to withstand such a design-basis event as an earthquake or large-scale loss-of-coolant accident (LOCA). For specific applications at certain plants, masonry block walls must also be considered.

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

C. Potential age-related degradation mechanisms for the various structural components include (Refs. 1 and 2):

1. Concrete

- freeze-thaw
- leaching of calcium hydroxide
- aggressive chemicals
- reactions with aggregates
- corrosion of embedded steel

Elevated temperature

- irradiation
- creep
- shrinkage
- abrasion and cavitation
- cracking of masonry block walls

3. Reinforcing steel

   - corrosion
   - elevated temperature
   - irradiation

4. Piles

   - corrosion

5. Structural steel

   - corrosion
   - elevated temperature
   - irradiation

6. Stainless steel liner plate

   - corrosion, intergranular stress corrosion cracking (IGSCC)
   - elevated temperature
   - irradiation

7. Miscellaneous degradation issues

   - fatigue
   - cathodic protection effects on bond strength
   - settlement

Although the majority of these potential degradation mechanisms are not significant at all plants, each should be assessed on a plant-by-plant basis before dismissing it.

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 and thermal expansion. Metallic materials degradation 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.

The areas of concern regarding a facility's aging should be reviewed to evaluate the acceptability of the applicant's program for managing 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 C.0.1 for Items II.A through II.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 applicant is committed.

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 confirm that the applicant 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 evidence 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, boiling-water reactor (BWR) suppression pools, and other water pools. Assess effects of observed and projected corrosion/ 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 applicant's IPA should address all of the potential age-related degradation mechanisms discussed in Item I.C and Section VI of this SRP-LR and any potential age-related degradation mechanisms experienced during the current license period for civil structures important to license renewal. Results of this one-time inspection shall be addressed and ongoing monitoring programs, analyses and proposed corrective actions shall be defined to provide assurance that these structures will perform their required functions through the license renewal period. The applicant shall provide an adequate technical justification for those potential degradation mechanisms that the applicant proposes to eliminate from consideration.

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.

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Prepared with Draft Regulatory Guide 1009 to support Part 54 rulemaking.

11. ABSTRACT (200 words or less)

The SRP-LR is to be used by the NRC staff when performing safety reviews of applications for the renewal of power reactor licenses. The use of the SRP-LR when reviewing license renewal applications provides a framework for the staff to determine whether or not (1) the application is sufficient to allow the timely renewal provisions of 10 CFR 2.109 to apply, (2) systems, structures, and components important to license renewal have been identified, (3) significant age-related degradation has been identified and its effects evaluated, and (4) programs for age-related degradation management have been or will be implemented such that the current licensing basis will be maintained during the renewal term. The draft SRP-LR has been developed to enable the staff to identify areas and issues requiring review, and provides acceptance criteria to assist the reviewers.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

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