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DRAFT REGULATORY GUIDE DG-1009

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

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This regulatory guide is being issued in draft form to involve the public in the early stages of the development of a regulatory position in this area. It has not received complete staff review and does not represent an official NRC staff position.

Public comments are being solicited on the draft guide (including any implementation schedule) and its associated regulatory analysis or value/impact statement. Comments should be accompanied by appropriate supporting data. Written comments may be submitted to the Regulatory Publications Branch, DFIPS, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Copies of comments received may be examined at the NRC Public Document Room, 2120 L Street NW., Washington, DC. Comments will be most helpful if received by March 8, 1991.

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A. INTRODUCTION

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

Purpose

The purpose of this regulatory guide is to establish a uniform format and content acceptable to the NRC staff for structuring and presenting the technical information to be compiled by an applicant for a renewed nuclear power plant operating license. The guide also establishes a uniform format for structuring and presenting the technical information to be submitted by the applicant as part of an application for a renewed license. This regulatory guide identifies the content of and provides technical criteria for the compiled technical information. Use of this format will help to ensure the completeness of the information compiled or provided, will assist the NRC staff and others in locating the information, and will aid in shortening the time needed for the process of reviewing the license renewal application.

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

Scope

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

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

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

Applicability

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

Any information collection activities mentioned in this draft regulatory guide are contained as requirements in those sections of 10 CFR Part 54 that provide the regulatory basis for this guide. The proposed additions to 10 CFR

Part 54 have been submitted to the Office of Management and Budget for clearance, as appropriate, under the Paperwork Reduction Act. Such clearance, if obtained, would also apply to any information collection activities identified in this guide.

B. DISCUSSION

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

The technical information will be reviewed by the NRC staff to assess the effectiveness of an applicant's established or proposed programs for understanding and managing age-related degradation of SSCs important to license renewal during a renewed license term and to evaluate acceptability of the application for a renewed license.

Use of Standard Format

Conformance with the standard format described in the Regulatory Position is encouraged but not required. License renewal applications with different formats will be acceptable to the staff if they provide an adequate basis for the findings requisite to the issuance of a renewed license. However, because it may be more difficult to locate needed information, the staff review time

for such applications may be longer, and there is a greater likelihood that the staff may regard the application as incomplete.

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

C. REGULATORY POSITION

The methods described in this section are acceptable to the NRC staff for satisfying the requirements proposed in 10 CFR 54 and 10 CFR 2.109 pertaining to the technical information to be compiled or submitted in support of an application for a nuclear power plant operating license renewal.

1. FORMAT FOR TECHNICAL INFORMATION

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

The license renewal application is to contain two separate parts: a formal application and a supplement to the current FSAR. Regulatory Position 1.1 describes the basic information that should be included in the formal application. Regulatory Position 1.2 describes the information that should be included

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

1.1 Formal Application

The formal application is to contain the following elements:

1. A table of contents.

2. An introduction providing general information concerning the application. This should include:

   a. The information specified in 10 CFR 50.33(a) through (e), 50.33(h), and 50.33(i),

   b. The earliest effective date and length of the renewal term,

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

   d. A description of the scope and organization of the remaining sections of the application,

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

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

3. A characterization and summation of the licensee's findings.

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

4. An implementation plan that includes the following elements:

a. Summary of Commitments

List the commitments described in the license renewal application.

b. Description of Administrative Controls

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

c. Tasks and Schedule

Detail the commitments pertaining to age-related degradation that will be completed following renewal of the operating license. These commitments may include system design changes, one-time replacements, program enhancements, and new programs.

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

5. Submittals of any Technical Specification and program changes and additions identified as necessary to manage age-related degradation during the renewal term.

6. A list of changes in calculations or analytical approaches resulting from licensing renewal activities, including justification for the changes.

1.2 FSAR Supplement

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

The supplemental FSAR chapter should contain sections that describe the licensee's methodology and results for satisfying each element of 10 CFR 54.21. The NRC staff will review the licensee's application on a systems basis according to review procedures set forth in a new Standard Review Plan chapter that deals with the review of applications for renewed licenses, Draft NUREG-1299, "Standard Review Plan for License Renewal" (SRP-LR).*

The technical information compiled or submitted by a licensee in compliance with 10 CFR 54.21 should conform in format and content to this regulatory guide. This expectation is based on the requirements in the proposed 10 CFR 54.21 to ensure that a facility's licensing basis will be maintained throughout the term of the renewed license and on the definition of current licensing basis as stated in 10 CFR 54.3(a). Maintenance of the current licensing basis, as defined in 10 CFR 54.3(a), requires a licensee to comply with 10 CFR 50 among other things. This includes the requirement cited in 10 CFR 50.34(g) that nuclear power plant operating licenses docketed after May 17, 1982, include an evaluation of the facility against the SRP revision in effect six months prior to the docket date of the application.

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

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*Single copies of Draft NUREG-1299 are available from the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Director, Division of Information Support Services, while supplies last.

Part A: General Information and Discussion

Information of an introductory or general nature such as purpose, scope, definitions, organization, and relationship to 10 CFR 54; conformance to regulatory guides; citations for referenced information; general technical information required by 10 CFR 54 and described in Regulatory Position 1.2.2; and a description of any deviations from the acceptance criteria contained in the SRP-LR.

Part B: Nuclear Power Plant Systems

Information specific to the principal systems and subsystems of the plant that contain structures or components important to license renewal.

Part C: Generic Components

Information related to structures and components important to license renewal for which age-related degradation may be generically addressed.

Not all the items included in the lists of SSCs in Table I are germane to all plants, and the lists may not include all SSCs important to license renewal for any particular plant.

1.2.1 Review Objectives

The FSAR supplement should allow the NRC staff to reach the following conclusions:

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

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

3. The licensee's methodologies for identifying established effective programs [defined in 10 CFR 54.3(a)] for structures and components requiring evaluation of age-related degradation are acceptable.

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

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

1.2.2 Supporting Documentation

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

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

2. Information specific to the total facility as required by 10 CFR 54.21, specifically:

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

   b. A list of all structures and components that are constituent elements of the SSCs listed in 2.a.

   c. A list of all structures and components from 2.b that require evaluation of age-related degradation.

   d. Justification for conclusions that any structures and components listed in 2.b but not listed in 2.c do not contribute to the performance of a safety function of an SSC important to license renewal or that their failure would not prevent an SSC important to license renewal from performing its intended safety function.

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

f. A list of all structures and components that are subject to an established effective program as defined in 10 CFR 54.3 (a), the associated effective programs, and the basis for continuing these programs through the renewed license period.

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

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

i. A list and description of any proposed modifications related to age-related degradation that have been or will be made to the facility or its administrative control procedures resulting from the integrated plant assessment [10 CFR 54.21(a)] or exemptions and reliefs described in 2.h [10 CFR 54.21(b)].

3. Information specific to the systems listed in Part B of Table I and the generic structures and components listed in Part C of Table I. Information related to components that can be grouped in terms of component type and expected age-related degradation may be cited once in generic form, and

that citation may be referenced for subsequent components that fit the appropriate grouping. For each system (Part B of Table I) or generic structure or component (Part C of Table I) applicable to the facility, the following information should be presented:

a. An SSC-specific list of constituent structures and components that are important to license renewal. A reference to the lists provided pursuant to 2.a and 2.b will suffice, provided these lists clearly identify all structures and components associated with that particular SSC.

b. Identification of age-related degradation sites, site-specific aging mechanisms, and root causes, when practicable, for structures and components listed pursuant to 2.c. Appendix A summarizes the age-related degradation processes that should be discussed.

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

d. An identification of the structures and components listed pursuant to 3.a that are subject to established effective programs, including a technical justification of how these programs effectively manage

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

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

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

and reliefs associated with each system. Justification for continuation may be supplied either in each system section or with the list in 2.h.

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

h. For existing and new programs identified as necessary for managing age-related degradation, a description of how these programs are or will be implemented and controlled to ensure that their effectiveness is maintained throughout the renewal term.

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

Additions or other changes to the Technical Specifications pertaining to age-related degradation may be necessary to account for modifications in the plant design or limitations on plant operations during the renewal term. If such changes are deemed necessary, the license renewal application (described in 1.1 Formal Application) should specifically request such changes and should provide the technical justification for the changes. Such changes should be limited to those necessary to address effects of age-related degradation.

Items 3.a through 3.i above, specify information that should be included in the FSAR supplement for each SSC important to license renewal. Additional information that should be submitted as part of a license renewal application

or compiled and retained by the licensee is described in Table II. Some of the information items listed in Table II should be included in the FSAR supplement (information described in Regulatory Position 1.2.2). The remainder of the information specified in Table II should be submitted as part of the application but independent of the FSAR supplement or should be retained by the licensee as appropriate.

2. TECHNICAL INFORMATION CONTENT

As required by 10 CFR 54.21, "The FSAR supplement (that presents the technical information requirement for license renewal) must include an evaluation of the aging mechanisms that are present and that result in degradation of the plant's systems, structures, and components, and a demonstration that the effects of such degradation will be effectively managed throughout the renewal term." To meet these requirements, the evaluation must cover those SSCs that are important to license renewal and must be based on principles of understanding and managing age-related degradation. This Regulatory Position 2 provides guidance on selection criteria for SSCs important to license renewal and on programs and practices for understanding and managing aging that are acceptable to the NRC staff as the bases for the "evaluation" and "demonstration" required by 10 CFR 54.21 as noted above.

2.1 Types of SSCs for Which Aging Should Be Considered for License Renewal

The process for identifying SSCs important to license renewal and their constituent structures and components requiring evaluation of age-related degradation is illustrated in Figure 1A. Figures 1A and 1B also summarize some of the information to be collected by the licensee and either submitted as part of a license renewal application or retained in auditable, retrievable form for the term of the renewed operating license. Figure 1B delineates approaches to information collection and evaluation for a licensee to demonstrate an acceptable level of understanding of age-related degradation in those structures and components identified through the selection process illustrated in Figure 1A. Acceptable management of age-related degradation for structures and components that are important to license renewal requires evaluations of the effectiveness of existing programs and practices for timely mitigation of the age-related degradation.

Figure 1B also delineates a process for identifying deficiencies in existing programs for addressing age-related degradation in specific structures and components and for specifying actions to be pursued in addressing these deficiencies in support of a license renewal application.

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

2.2 Integrated Plant Assessment - Programs for Addressing Age-Related Degradation*

Elements of the integrated plant assessment for the evaluation of age-related degradation should be based on sound principles and practices for understanding and managing aging. SSC-specific understanding of aging mechanisms and the degradation sites at which they operate is useful to evaluate the effectiveness of existing programs and replacement/refurbishment practices for managing aging or to develop acceptable new programs or practices. An established effective program should include, but is not limited to, inspection, surveillance, maintenance, trending, recordkeeping, replacement/refurbishment, and the assessment of operational life for the purpose of timely mitigation of the effects of age-related degradation.

The prime criteria for such a program are that it be documented and that it ensures that SSCs important to license renewal will continue to perform

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*For expanded discussions, see Appendix A to this regulatory guide.

adequately, thus ensuring that the current licensing basis will be maintained during the renewal period. In addition, an established effective program will (1) be clearly defined and documented in the FSAR, (2) be approved by onsite review committees, (3) be implemented by the facility operating procedures, (4) establish documented acceptance criteria against which the need for corrective action is evaluated to ensure that age-related degradation will not directly or indirectly prevent SSCs from performing their intended functions, and (5) ensure that corrective action is taken when applicable acceptance criteria are not met. Programs for understanding and managing aging should be implemented and maintained through a system of specific administrative controls that ensures continuing program effectiveness throughout the term of a renewed license.

Requirements for a licensee to demonstrate the adequacy of the plant program for addressing age-related degradation in structures and components important to license renewal are specified in 10 CFR 54.21. These are (1) by substantiating that established programs (ongoing programs that are currently in place) are effective in ensuring the capability of SSCs important to license renewal to perform their safety functions throughout the renewal period, (2) by taking actions or committing to actions to manage age-related degradation not adequately addressed by established programs [10 CFR 54.21(a)(4)], or (3) by demonstrating that age-related degradation is not significant with respect to the current licensing basis [10 CFR 54.21(a)(4)]. Adequacy of the program for addressing age-related degradation must be demonstrated for each structure, component, or group of similar components important to license renewal using one of the aforementioned methods. Both existing programs and new actions taken or to be taken to manage age-related degradation are subject to the same effectiveness criteria. Criteria that relate to program structure and administration are cited in the preceding paragraph. Criteria that relate to established effective programs and to new actions taken or to be taken to manage aging are discussed in Regulatory Positions 2.2.1 and 2.2.2, respectively. Regulatory Position 2.2.3 deals with the exclusion of structures and components not subject to significant age-related degradation during the renewed licensing term. Technical criteria for practices employed in understanding and managing aging are described in Regulatory Positions 2.2.4 and 2.2.5, respectively.

2.2.1 Established Effective Programs

Established effective programs for managing age-related degradation are defined in 10 CFR 54.3(a). The structural and administrative criteria provided in that definition plus the technical criteria in Regulatory Positions 2.2.4 and 2.2.5 should be applied by a licensee in evaluating the current plant program for managing aging in structures and components important to license renewal. The methodology employed in performing these evaluations should be described, and results of the evaluations should be provided for all important-to-license-renewal structures and components that are subject to established programs. Two essential products of these evaluations will be identification of (1) those important-to-license-renewal structures and components that are regularly inspected and routinely replaced/refurbished at defined intervals and (2) structures and components for which established programs do not effectively address age-related degradation.

2.2.2 Actions To Be Taken To Manage Age-Related Degradation

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

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

For some structures and components important to license renewal, age-related degradation may not lead to significant reduction in the capacity of the structure or component to perform its safety functions, i.e., functions defined as important to license renewal. If a licensee explicitly demonstrates that age-related degradation of an important-to-license-renewal structure or component

will not compromise the current licensing basis during the license renewal term, that structure or component may be excluded from further consideration of age-related degradation.

2.2.4 Summary of Acceptable Programs and Practices for Understanding Aging

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

The process of assessing age-related degradation involves integrating the relevant materials science concepts that describe degradation processes with plant-, SSC-, and structure-or-component-specific design and operational information to understand aging mechanisms, degradation sites and rates, and the consequences of degradation with respect to plant safety. The individual and interactive influences of structure and component design, constituent materials, and both normal and abnormal stressors and environments establish the feasibility of the mechanisms that can degrade SSCs important to license renewal and determine the rates at which degradation progresses. An effective program to understand

aging will selectively integrate a sound understanding of these basic principles with plant-specific design, operational experience, manufacturers' and design information, research and test data, applicable regulatory instruments and requirements, and qualified technical judgment to characterize age-related degradation processes that are operative in important-to-license-renewal structures and components. These characterizations should be expressed in terms of specific degradation sites, site-specific aging mechanisms, root causes for degradation when practicable, and projected effects of degradation on SSC functions.

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

Acceptable practices for managing aging in structures and components for which no defined replacement/refurbishment programs exist may employ combined mechanistic and empirical approaches for understanding aging mechanisms and identifying degradation sites, and, where practicable, root causes. This information may form the basis for inspection, surveillance, condition monitoring, test methods and frequencies, and residual lifetime evaluations that will dictate timely and effective preventive and corrective maintenance methods and frequencies, as well as associated recordkeeping needs during the renewal term.

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

Practices for managing aging in structures and components that are regularly inspected and routinely replaced or refurbished at defined intervals should be evaluated for the adequacy of the inspection and replacement or refurbishment programs to ensure timely mitigation of age-related degradation during the renewal term. The evaluation process should include reviews of the operational experience and, as appropriate, design and manufacturers' information, known aging mechanisms, and other available information.

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

The maintenance program is the principal vehicle through which age-related degradation is managed. Operational and maintenance records and input from monitoring programs should be employed in the maintenance program for scoping and scheduling both preventive and corrective maintenance activities intended to manage age-related degradation. These activities should be carried out by experienced, qualified maintenance personnel and should lead to needed servicing, repair, refurbishment, or replacement with a frequency sufficient to ensure the capability of SSCs important to license renewal to perform their safety functions during the renewal term. In evaluating the effectiveness of maintenance in managing aging, the maintenance and surveillance intervals should be considered along with (1) the probability of defect detection and diagnosis and (2) the probability of effective defect correction given defect detection and diagnosis.

A critical first step in implementing an aging management program is to define those surveillance, maintenance, or other mitigative program elements to be implemented to deal with the degradation processes revealed by programmatic efforts to understand aging (Regulatory Position 2.2.4). These program elements may include inspections, tests using a wide variety of nondestructive examination (NDE) and other methods, general surveillances, condition monitoring, diagnostic and root-cause analysis, preventive maintenance, corrective maintenance, predictive maintenance, and reliability-centered maintenance. The primary goal should be to develop effective aging management practices implemented through replacement, refurbishment, and repair programs that accurately reflect residual lifetimes for specific structures and components and the safety significance of anticipated degradation. The effectiveness of aging management programs should be evaluated for specific structures and components using guidance such as that contained in relevant codes and standards and approved industry technical reports. The process for evaluating effectiveness should reveal those deficiencies that require correction through improved or new programs for managing aging. This process is outlined schematically in Figure 1B. An accurate assessment of current program effectiveness, based on the principles of understanding aging summarized in Regulatory Position 2.2.4 and in Appendix A, and implementation

of program enhancements to address revealed deficiencies in aging management practices are prerequisites to operating license renewal in nuclear power plants.

An important aspect of plant surveillance and maintenance programs is retention of essential data in complete, auditable, easily retrievable records. The records system and its contents should conform to good maintenance practices as well as the requirements of Appendix B to 10 CFR Part 50 and the plant quality assurance program to the extent that these documents apply to SSCs important to license renewal. A record of the documentation required by or necessary to document compliance with the provisions of 10 CFR Part 54 and a record of the administrative processes for controlling changes to such documents should be retained by the licensee in an auditable and retrievable form for the renewal term. This record should include a listing of and the justification for those structures, systems, and components important to license renewal and included in established effective programs as defined in 10 CFR 54.3(a) or subject to actions taken or to be taken. Records and related data should be employed to the extent practicable in trending degradation processes, thereby providing assurance of controlled, timely maintenance. Trends that are based on data contained in the maintenance records and that account for both normal and off-normal operating conditions should be used to monitor the effectiveness of aging management for selected structures and components.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants regarding the NRC staff's plans for using this regulatory guide.

This draft guide has been released to encourage public participation in its development. Except in those cases in which an applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method to be described in the active guide reflecting public comments will be used by the NRC staff in evaluating applications for renewal of operating licenses for commercial nuclear power plants.

FIGURE 1

PROCESS FOR SELECTING SYSTEMS, STRUCTURES, AND COMPONENTS IMPORTANT TO LICENSE RENEWAL AND FOR UNDERSTANDING AND MANAGING AGE-RELATED DEGRADATION

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

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

Two methods are presented as guidance for determining the scope and depth of analysis necessary to define age-related degradation mechanisms and to evaluate the adequacy of aging management programs. The first method may be applicable when evaluating structures and components that are regularly inspected and replaced or refurbished routinely at defined intervals. These structures and components include items such as batteries, relays, and selected snubbers. Programs that may be considered to provide for management of aging in these structures and components include inspection, surveillance, testing, condition monitoring, residual lifetime evaluations, predictive maintenance, preventive maintenance, reliability centered maintenance, and other maintenance or similar programs that provide timely mitigation of age-related degradation during the license renewal term. The second method may be applicable when evaluating structures and components that are not regularly inspected and routinely replaced or refurbished. Such structures and components typically were designed for and expected to be in place for the original 40 years of plant operation. These "long-lived" structures and components include items such as the reactor coolant system, large-diameter piping, the reactor pressure vessel, steam generators, and cables.

Structures and components that are regularly inspected and routinely replaced or refurbished are not automatically considered as subject to "established effective programs." The effectiveness of aging management programs for these structures and components should be evaluated based on a review of the adequacy of the inspection and replacement or refurbishment programs for timely mitigation of age-related degradation during the license renewal term. An acceptable approach would be to evaluate operational experience, replacement or refurbishment intervals, and, as appropriate, relevant design and manufacturers' information, known aging mechanisms, and other available information. Based on this review and a conclusion that the structure or component will remain functional during the defined interval between replacement or refurbishment, an applicant may establish that the current program is adequate. When the ongoing

replacement program is demonstrated to be adequate for timely mitigation of age-related degradation and if it is in the FSAR, approved by the onsite review committee, and implemented by the facility operating procedure, it is considered to be an "established effective program."

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

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

Throughout Figures 1A and 1B, individual activities and decision points have been referenced as appropriate to the specific parts and subparts of 10 CFR Part 54 that provide their justification. These references to 10 CFR Part 54 are denoted within appropriate nodes in the figures.

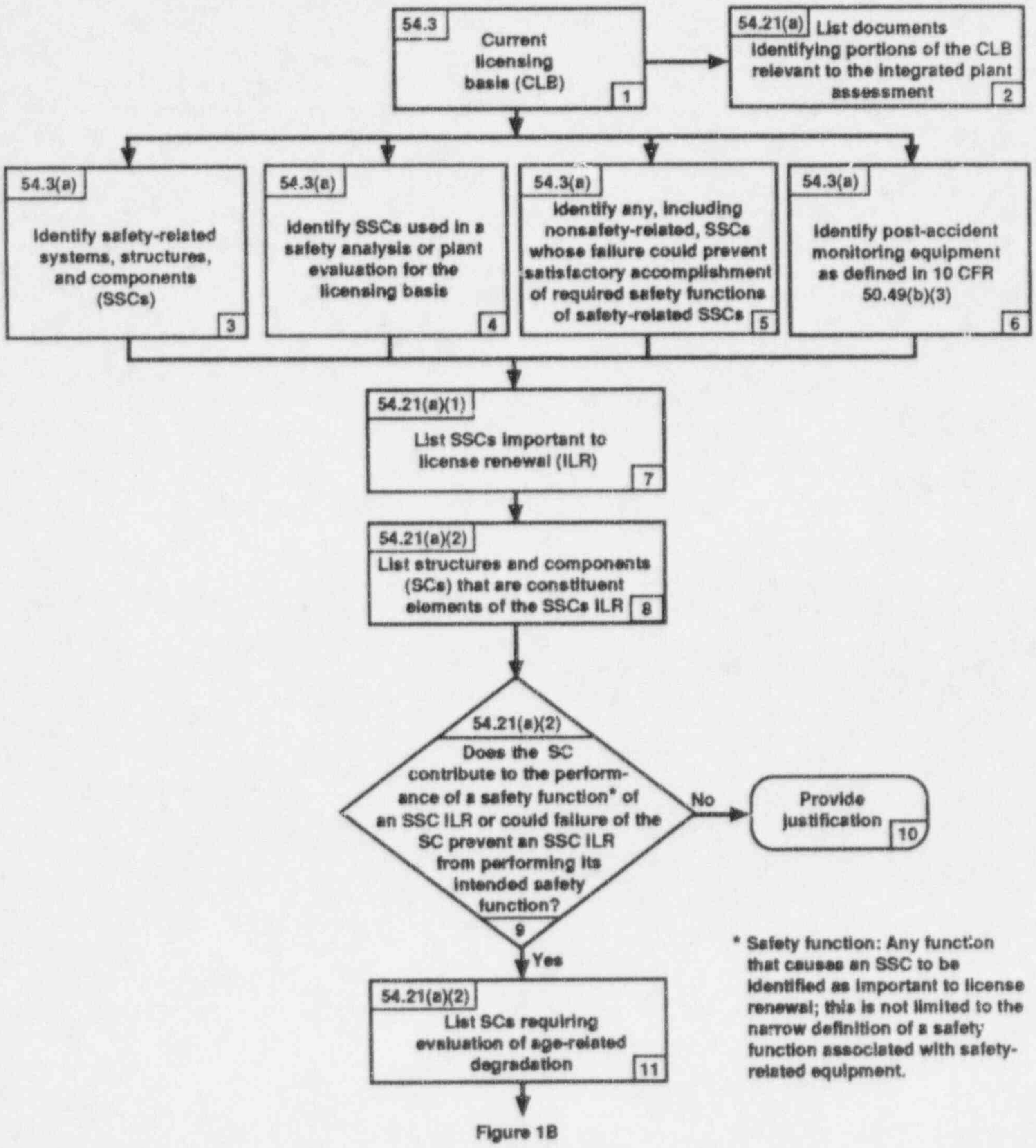
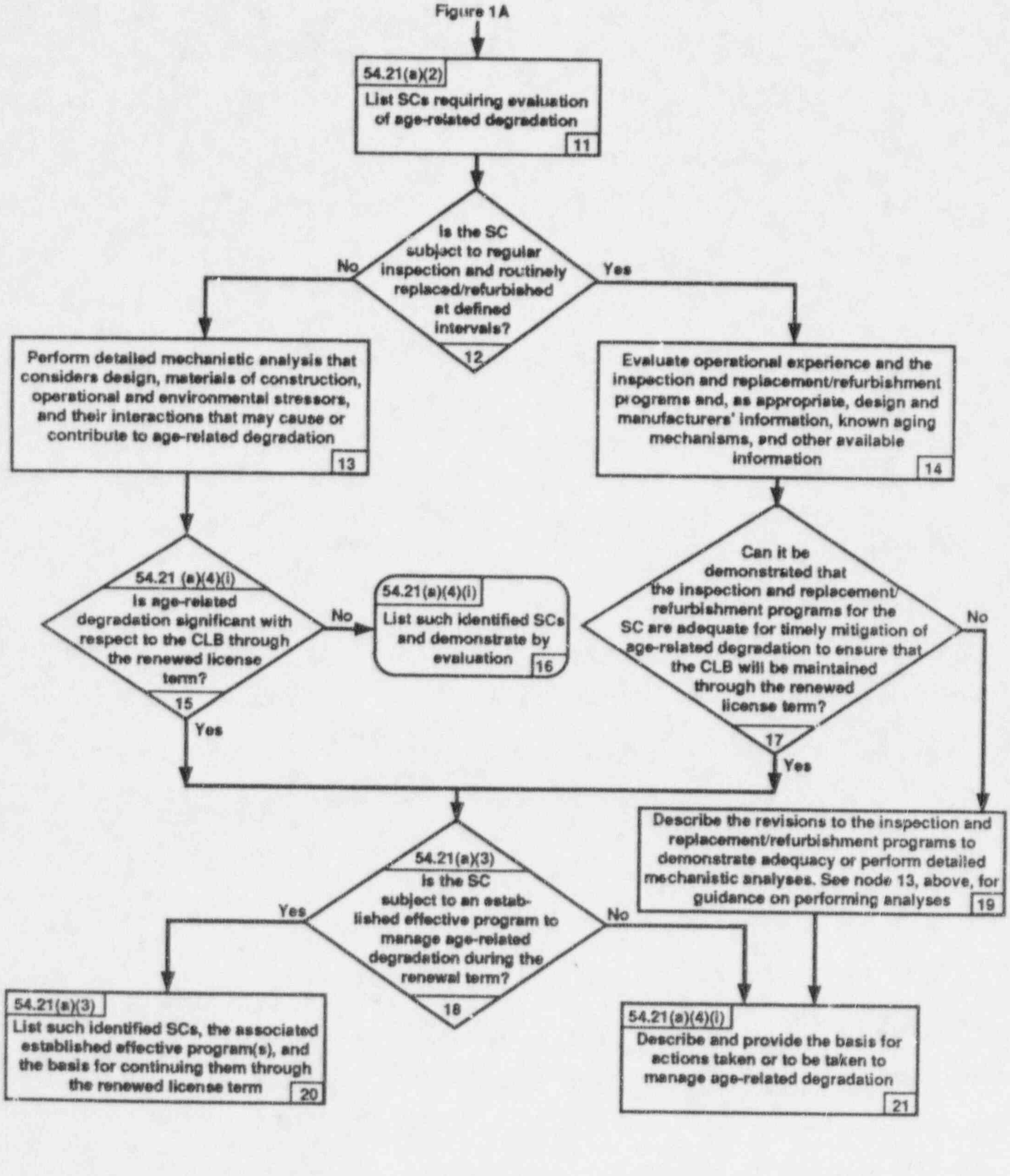


FIGURE 1A Integrated Plant Assessment--Identification of Important-To-License-Renewal SSCs and SCs Requiring Evaluation of Age-Related Degradation



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

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

1 Compile and maintain all documents describing the current licensing basis (CLB) in an auditable and retrievable form as per 10 CFR 54.21(a).

2 Submit a list of documents identifying portions of the CLB relevant to the integrated plant assessment (IPA) as per 10 CFR 54.21(a).

3, 4, 5, 6 These four categories are defined by 10 CFR 54.3(a) as being important to license renewal. Assignment of systems, structures, and components (SSCs) to the appropriate categories should be accomplished by starting with all plant SSCs and distributing these among the four categories by applying guidelines such as those contained in the sources referenced in the following four notes. SSCs that fit none of the four categories are not important to license renewal and require no further consideration for license renewal purposes.

3 Q List, Final Safety Analysis Report (FSAR), Quality Assurance Program and all other elements of the Current Licensing Basis as defined in 10 CFR 54. SSCs used in a safety analysis or plant evaluation for the Licensing Basis include, but are not limited to, SSCs identified in the FSAR, the Technical Specifications, Plant Operations Manuals, Piping and Instrumentation Diagrams (P&IDs),Table III of this regulatory guide (Generic Functional SSCs Important to License Renewal), and evaluations submitted to show compliance with the Commission's regulations such as Anticipated Transients Without Scram (ATWS), Station Blackout, Fire Protection, Pressured Thermal Shock (PTS), and Environmental Qualification.

4 Current Licensing Basis.

5 FSAR, P&IDs, as-built current drawings, plant modification records, master equipment list, plant configuration control system data.

6 10 CFR 50.49 and associated regulatory guides (e.g. R.G.1.89, R.G.1.97), FSAR.

7 Describe methodology for identifying SSCs important to license renewal, as defined in 10CFR54.3(a), and submit a list of the identified SSCs as part of the Integrated Plant Assessment (IPA).

8, 9, 10, 11 Describe methodology for identifying those structures and components (SCs) that are constituent elements of the SSCs important to license renewal and that require evaluation of age-related degradation.

11 Probabilistic risk assessment (PRA) techniques may also be used to supplement the deterministic approach shown in Figure 1A by adding additional SCs to the list of those requiring evaluation of age-related degradation.

12 Plant records and CLB.

13, 14 Input from the Nuclear Plant Aging Research (NPAR) Program; codes and standards; nondestructive examination (NDE); industry studies, and other programs related to effects of material variables, environment, and stressors on age-related degradation.

<u>Materials</u> characterization includes initial design, construction, installation plus changes introduced by modification, maintenance, or replacement.

<u>Stressors</u> derive from both environmental and service conditions

- Environmental conditions include radiation, temperature, atmosphere, humidity, and chemical environment for all design basis events
- Service conditions include steady-state, cyclic, or other transient loadings imposed during normal operation, testing, or offnormal events. Mechanical and electrical loadings predominate.

15, 16 For those structures and components (SCs) for which it can be demonstrated that age-related degradation is not significant with respect to the CLB through the renewed license period, no further actions need be taken to manage their age-related degradation.

17, 18 Decisions to be based upon criteria developed by the licensee and subject to review by the NRC. Need for in-depth evaluation will reflect such considerations as safety significance, failure consequence, system and degradation process complexity, intensity or conservatism of current mitigation program, and other factors deemed important by the licensee and NRC for specific cases. Supporting information for these decisions will derive from various plant-specific sources including records of operations and maintenance, technical specifications, maintenance/surveillance procedures, inservice inspection program (ASME Section XI), PTS analyses, preventive-predictive maintenance programs.

19, 20, 21 Programs for managing age-related degradation include inspection, testing, surveillance, condition monitoring, root cause analysis, predictive/preventive/corrective/risk-based/reliability centered maintenance, record keeping and trending, other replacement/refurbishment/repair activities, residual life assessment, and responses to changes in operating and design parameters and environments. Relevant information can be found in records of operations and maintenance, technical specifications, maintenance/surveillance procedures, inservice inspection program (ASME Section XI), pump and valve testing programs, environmental qualification programs, preventive-predictive maintenance program records, and the current body of regulatory requirements.

Note A: Results of activities 2, 7, 8, 10, 11, 16, 19, 20, and 21 should be submitted with the license renewal application. These results plus results of activities 1, 3, 4, 5, 6, 13, 14, and 17 should be documented and retained by the licensee in auditable, retrievable form.

TABLE I

INFORMATION TO BE INCLUDED IN THE FSAR SUPPLEMENT -- GENERAL AND SSC SPECIFIC

Part A: General Information and Discussion*

(Part A should include, but is not limited to, the following types of information.)

1. Introduction

    1.1 Purpose of the FSAR Supplement

    1.2 Scope of the FSAR Supplement

    1.3 Definitions of Terms

    1.4 Organization of the FSAR Supplement

    1.5 Relationship of the FSAR Supplement to 10 CFR 54

2. Conformance to Applicable Regulatory Guides

3. Listing and Summary of Material Incorporated by Reference

4. Description of Integrated Plant Assessment (see Regulatory Position 1.2.2 of this regulatory guide)

5. Information Required by 10 CFR 54.21 (see Regulatory Position 1.2.2 of this regulatory guide)

6. Deviations from SRP-LR Acceptance Criteria*

Part B: NPP Systems

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

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

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

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*An applicant for a renewed license will find it useful to consult NUREG-1299, "Standard Review Plan for License Renewal," the SRP-LR, for descriptions of criteria and review procedures to be applied by the NRC to applications for license renewal.

TABLE I (contd)

1.2 Reactor Coolant System (includes piping, reactor coolant pumps, and steam generators) [5,3]

1.3 Reactor Control System [3]

1.4 Control Rod Drive System [3]

1.5 Reactor Protection System [7,3]

1.6 Nuclear Monitoring/Nuclear Instrumentation System [7,3]

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

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

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

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

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

2.2 Safety Injection Systems [3,5,6]

2.2.1 Reactor Core Isolation Cooling (BWR)

2.2.2 High Pressure and Intermediate Pressure Safety Injection System (PWR)

2.2.3 Core Flood System (PWR)

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

2.2.5 Core Spray Systems (BWR)

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

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

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

2.5 Remote Shutdown System/Safe Shutdown Systems [7]

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

3.1 Primary Containment Structure [3,6]

TABLE I (contd)

3.2 Secondary Containment [3,6]

3.3 Containment Heat Removal System [6]

3.4 Containment Isolation System [6]

3.5 Containment Purge System [6]

3.6 Standby Gas Treatment System (BWR) [6]

3.7 Containment Combustible Gas Control System [6]

3.8 Containment Spray System [6]

3.9 Containment Ventilation System [9]

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

4.1 Main Power [8]

4.1.1 Protective Relaying and Controls

4.2 Plant AC Distribution System [8]

4.2.1 Essential Power System

4.2.2 Nonessential Power System

4.2.3 HPCS Power System (BWR)

4.3 Instrument and Control Power Systems [8]

4.3.1 DC Power System

4.3.2 Instrument AC Power System

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

4.4.1 EDG Instrumentation and Control Subsystem [8]

4.4.2 EDG Starting Subsystem [9]

4.4.3 EDG Cooling Subsystem [9]

4.4.4 EDG Fuel Oil Subsystem [9]

4.4.5 EDG Lubricating Oil Subsystem [9]

4.5 Plant Essential Lighting System [8,9]

TABLE I (contd)

4.6 Plant Computer [7]

4.7 Switchyard [8]

4.7.1 DC Control Power System

4.8 Information Systems Important to Safety [7]

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

5.1 Offgas System (BWR) [11]

5.2 Radiation Monitoring System [12]

5.3 Component Cooling Water System [9]

5.4 Service Water System [9]

5.5 Ultimate Heat Sink [9]

5.6 Refueling System [9]

5.7 Spent Fuel Storage [9]

5.8 Compressed Air System [9]

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

6.1 Fire Protection System [9]

6.2 Communications [9]

6.3 Control Room Habitability System [6]

6.4 Auxiliary HVAC Systems [6]

Part C: Generic Components

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

1. Mechanical

1.1 Piping

TABLE I (contd)

1.2 Valves

1.3 Pumps

1.4 Heat Exchangers

1.5 Tanks and Vessels

1.6 Equipment and Component Supports

2. Electrical

2.1 Cable and Wiring

2.2 Junctions

2.3 Electrical Penetrations

2.4 Relays, Circuit Breakers, and Switchgear

2.5 Transformers

2.6 Solenoid Operated Valves

2.7 Electric Motors

3. Instrumentation

3.1 Sensors

3.2 Electronic Components

3.3 Electronic Devices

4. Civil Structures

TABLE II

TECHNICAL INFORMATION NEEDED FOR LICENSE RENEWAL (LR)

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

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

TABLE II. (contd)

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

TABLE II. (contd)

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

TABLE II. (contd)

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

TABLE III

GENERIC FUNCTIONAL NUCLEAR POWER PLANT SSCs IMPORTANT TO LICENSE RENEWAL*

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

A. All components that constitute the reactor coolant pressure boundary.

B. The reactor core and reactor vessel internals.

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

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

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

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

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

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

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

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

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*This table provides supplemental guidance for developing plant-specific lists of SSCs important to license renewal. This guidance is derived from Regulatory Guide 1.29, "Seismic Design Classification."

TABLE III (contd)

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

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

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

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

O. Primary and secondary reactor containment.

P. Systems, other than radioactive waste management systems, not covered by items (A) through (O) above that contain or may contain radioactive material and whose postulated failure would result in conservatively calculated potential offsite doses (using meteorology as recommended in Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," and Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors") that are more than 0.5 rem to the whole body or its equivalent to any part of the body.

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

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

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

APPENDIX A

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

This appendix discusses the significant mechanisms that cause age-related degradation in nuclear power plants and the principles involved in understanding and mitigating this degradation. Methods for selecting systems, structures, and components (SSCs) in which aging is a license renewal concern are also described. The information that follows is of a summary nature and is not intended to characterize in detail the age-related degradation in nuclear power plants. As research continues, additional information concerning age-related degradation will be forthcoming.

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

The following sections provide information that relates to (1) selecting SSCs important to license renewal, (2) understanding age-related degradation in

structures and components important to license renewal, and (3) managing aging in structures and components important to license renewal.

A.1 SELECTION OF SSCs IMPORTANT TO LICENSE RENEWAL

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

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

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

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

An effective program for addressing age-related degradation will provide for both understanding and managing the aging that occurs in nuclear power plants. Aging mechanisms and their effects should be understood with sufficient accuracy and detail to provide the basis for developing and implementing aging management strategies that address, in a prioritized and timely fashion, actual or potential root causes of SSC failure.

A.2.1 Understanding Age-Related Degradation

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

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

A.2.1.1 Materials

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

A.2.1.2 Aging Stressors

Of the factors that can affect the age-related degradation of nuclear power plant SSCs, the stressors associated with environmental and service conditions are generally the most difficult to understand. Stressors caused by service

conditions assume various forms (e.g., mechanical, electrical) and can originate or are intensified during component fabrication, assembly, transportation, installation, operation, testing, and maintenance. Those who design, fabricate, operate, and maintain structures and components should understand how stressors can degrade their operational capabilities.

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

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

1. Mechanical loads are generally associated with physical movements, pressure differentials, and dimensional changes. The operation of SSCs either during normal operation (including testing) or under accident conditions usually induces time-dependent mechanical stresses. These stresses are caused by dynamic loads, internal or external pressure changes, impact, vibration loads, temperature changes, component test loads, and seismically induced motions. The operational motions of active SSCs (e.g., valve operation and pump rotations) produce time-dependent distortions and inertial stresses as well as wear. The effects of these loads in degrading SSCs are generally understood, but degradation rates are usually only estimates obtained from the analysis of inservice monitoring data, inspection reports, and maintenance information. Proper maintenance can mitigate much or all of the degradation caused by mechanical loads.

Internal and external pressure loads approaching operational or accident design limits also can produce high stresses that can cause distortions and, after sufficient cycles, can result in strain hardening and fatigue damage to SSC materials. If these stresses are combined with vibration and thermal stresses, measurable degradation can occur in a period that is short relative to the anticipated operational life of the SSCs.

Seismic events or similar but more localized events, e.g., water hammer, can inflict immediate damage to SSCs at any point during their operational life. Even though the SSCs may not fail during the impact, their functional capability may be degraded such that the operational life is shortened. The extent of the damage to SSCs resulting from external sources must be understood to anticipate any associated reduction in lifetime.

Vibrational loads can cause fatigue damage. Methods of analyzing vibrational fatigue damage are available; however, the results often include large uncertainties. These uncertainties are associated with material fatigue properties and the distribution and magnitude of the induced dynamic stresses. Vibrational stresses may be induced by plant operational modes, during transportation if a component is not properly isolated, and by ground or seismic vibrations. The source of vibrational loads that develop during the operational life of SSCs, the distribution of the associated stresses, and the endurance limits of the materials must be known for lifetime prediction.

Thermal stresses develop in SSCs because of temperature-gradient-induced differences in thermal expansion and the fact that different materials expand at different rates when heated. Differential expansions may be resisted internally or by interference with adjacent component surfaces. This resistance results in time-dependent thermal stresses that can cause age-related degradation, either separately or when combined with the effects of other stressors. Typical of such degradation are the thermal fatigue cracks that have appeared in high-temperature coolant water piping and nozzles and embrittlement of insulating materials.

2. Electrical stresses are induced in the insulating materials used in the fabrication of electrical and electromechanical parts and components. Both

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

A.2.1.3 Aging Mechanisms

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

1. Corrosion

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

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

Pitting is a localized form of corrosion that results in small craters or holes in the metal. Pitting is potentially one of the most insidious forms of corrosion because it can lead to component failure by perforation while producing only a small loss of metal. Because of their small size and because the pits are often covered with corrosion products, they can be difficult to detect. Pitting occurs when one area of a metal surface becomes anodic with respect to the rest of the surface or when highly localized changes in the environment in contact with the surface cause accelerated attack. Causes of pitting include local inhomogeneities on or beneath the metal surface, local loss of passivity, mechanical or chemical rupture of the protective oxide surface film, galvanic corrosion from a relatively distant cathode, and the formation of a metal ion or oxygen concentration cell under a solid deposit (crevice corrosion). The rate of penetration into the metal by pitting may be 10 to 100 times greater than for general corrosion. The most common causes of pitting in steels are surface deposits that set up local concentration cells and dissolved halides that produce local anodes by rupture of the protective surface scale. With corrosion-resistant alloys such as stainless steels, the most common cause of pitting is the highly localized destruction of passivity through contact with a halide-containing environment.

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

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

Stress corrosion cracking (SCC) is an aging mechanism that occurs in engineering materials by the combined and synergistic interaction of a chemically aggressive environment, a susceptible material, and a tensile stress or radiation field. The material fails by slow, environmentally induced crack growth that occurs with little or no attendant macroscopic plastic deformation. Although a tensile stress is not necessary for irradiation-assisted SCC, it can aggravate the phenomenon. The stresses required to cause SCC are usually below the yield strength and are tensile in nature. These stresses can be either applied or residual and may result from the fabrication process or inservice loading of the component or structure. Common sources of stress include thermal processing and stress risers created during surface finishing, fabrication, or assembly. The length of time required to produce SCC decreases for increasing stress level. The minimum stress at which cracking will occur depends on the temperature, the composition and microstructure of the alloy, and the environment. SCC may initiate at pre-existing mechanical cracks or other surface discontinuities such as pits produced by chemical attack.

Microbiologically influenced corrosion (MIC) occurs when biological organisms affect corrosion processes on metals by directly influencing the anodic and cathodic reactions, by affecting the protective surface scales on metals, by producing corrosive substances, or by creating solid deposits. These organisms include microscopic forms such as bacteria and macroscopic types such as algae and barnacles. Microscopic and macroscopic organisms have been observed to live and reproduce under broad ranges of pressure, temperature, humidity, and pH; thus, biological organisms may influence

corrosion in a variety of environments. MIC effects on carbon steel may result in random pitting, general corrosion, or severe hydraulic effects caused by formation of tubercles and massive corrosion product deposits. MIC attack on stainless steel is characterized by pitting, most commonly at weldments.

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

2. Erosion

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

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

3. Embrittlement

Structural or chemical changes induced by radiation, elevated temperature, or atmospheric contaminants cause embrittlement of metals and polymers used as electrically insulating barriers that can lead to fragility and failure under dynamic loading. Metallic components are most susceptible to embrittlement from neutron radiation; thus, components in proximity to the reactor core are most affected. Embrittlement with loss in toughness for critical components such as pressure vessels and supports represents the most significant contribution of radiation to aging. Organic and electronic materials are particularly susceptible to radiation damage from gamma rays. Thermal embrittlement is associated with chemical or metallurgical changes and results from such processes as thermal aging leading to reduced toughness of ferrous alloys, high temperature sensitization to intergranular stress corrosion cracking in austenitic stainless steels, and oxidation or cross linking of polymers with a resultant loss in toughness and dielectric strength. Hydrogen absorption by metallic alloys can also lead to loss of toughness and brittle fracture.

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

Thermal embrittlement can occur in cast austenitic-ferritic (duplex) stainless steel piping. The embrittlement is associated with the formation of precipitates in the ferritic phase, leading to cleavage of the ferrite or separation of the ferrite/austenite phase boundaries. The degree of aging is related to the volume fraction of ferrite in the material. In

addition, the precipitation and growth of phase-boundary carbides or nitrides can lead to brittle fracture. In general, low carbon grades of cast stainless steel are the most resistant, and molybdenum-containing high carbon grades are the most susceptible to thermal embrittlement.

Hydrogen damage is an environmentally assisted degradation process that usually results from the combined action of hydrogen and residual or applied tensile stresses. Hydrogen damage occurs in several ways, such as hydrogen embrittlement, blistering, and cracking from hydride formation. Hydrogen embrittlement is usually associated with loss of tensile ductility in carbon steels and high-strength alloys and is a function of the stress level and time. Steel can be embrittled by only a few parts per million hydrogen, which can originate from the fabrication process or inservice corrosion reactions. A similar effect may occur in austenitic stainless steels, but required hydrogen levels are many times the levels for carbon steels. Above about 400°F, hydrogen diffuses rapidly in steel and is eliminated by offgasing. Unless the steel has an impermeable coating or is used in a high hydrogen pressure environment, embrittlement should not pose a problem at these temperatures. At low temperatures, hydrogen embrittlement can occur in high strength, low alloy steel.

4. Mechanical Degradation Mechanisms

Fatigue is a common degradation process that occurs in rotating or reciprocating equipment or under other service conditions that place periodic or cyclic loads on SSCs. Fatigue damage results in progressive, localized structural change in materials subjected to fluctuating stresses and strains. Associated failures may occur at either high or low cycles in response to various kinds of loads, e.g., mechanical or vibrational loads, thermal cycles, or pressure cycles. The process of fatigue consists of three stages: (1) initial fatigue damage leading to crack initiation, (2) crack propagation, and (3) sudden fracture of the remaining ligament. Fatigue cracks initiate and propagate in regions of stress concentration that intensify strain, e.g., structural defects. The fatigue life of any SSC is the number of stress or strain cycles required to cause failure. This number is a function of several variables such as stress level, stress state, cyclic waveform, fatigue environment, and the metallurgical condition

of the material. Stress cycles can be generated by the direct application of mechanical loads, differential thermal expansion of mechanically constrained components, or temperature fluctuations. Although the loading conditions are different, the resultant fatigue is considered to be additive. Fatigue cracks form at the point of maximum local stress and minimum local strength. The local stress pattern is governed by the geometry of the SSC, including local features such as surface and metallurgical imperfections that concentrate stress, and by the type and amplitude of the loading. Surface imperfections such as scratches, mars, burrs, and other fabrication flaws are locations where fatigue cracks may start. Inclusions, hard precipitates, and crystal discontinuities such as grain boundaries are examples of microscopic stress concentrators. Pitting corrosion, stress corrosion cracking, and other effects of a hostile environment may also be important. For example, many fatigue failures originate in fretted areas. In many large structural components, the existence of a crack does not necessarily imply imminent failure of the component. Significant structural life may remain before the crack grows to a size at which failure occurs. The growth of a fatigue crack under cyclic loading is principally controlled by the maximum load and the ratio of maximum to minimum load.

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

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

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

A.2.1.4 Degradation Resulting from Operational Environment

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

transients did not contribute to limiting the expected lifetime of the affected SSC. Normal operating, testing, and environmental stressors (including those caused by electrical, mechanical, and thermal loadings), also contribute to age-related degradation that should be evaluated prior to extended life. Original equipment designers and manufacturers should be consulted for identification of aging mechanisms specific to particular SSCs.

A.2.1.5 Degradation Sites

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

A.2.2 Managing Aging Degradation

When the interactive effects of materials, designs, and stressors from environmental and service conditions are understood, the root causes of age-related degradation can be identified and programs to ensure that SSCs will adequately perform their intended functions can be implemented. Inspections and surveillance to monitor degradation in SCCS important to license renewal should be regularly performed. Selectively applied condition monitoring and trending can also be useful in this respect. Effective management of aging

will permit timely repair, replacement, or servicing through preventive or corrective maintenance.

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

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

1. Identify monitorable indicators that account for both normal and off-normal conditions that can be trended to show aging effects on the performance or reliability of SSCs important to license renewal.

2. Develop and implement methods for monitoring the indicators identified in number 1 above that will provide the total life history for the specific SSCs.

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

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

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

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

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

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

A.2.2.1 Root-Cause Determination

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

A.2.2.2 Monitoring Aging Degradation

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

degradation through root-cause analysis and corrective action. Monitoring programs should provide total life histories for SSCs of concern. Approaches to monitoring degradation include the following.

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

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

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

A.2.2.2.4 Residual Life Assessment. For monitored trends in age-related degradation to have meaning in terms of frequency of service, replacement, or refurbishment, it is necessary to correlate the level of monitored parameters

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

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

A.2.2.3 Mitigating Aging

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

Maintenance activities range from simple straightforward tasks to complex activities that require extensive coordination, training, and technical expertise. The level of oversight and resources devoted to these activities should reflect their complexity and importance to plant safety and reliability. A maintenance program has many important elements. Those considered here as being particularly relevant to age-related degradation include preventive maintenance,

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

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

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

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

- A list of SSCs whose failure or loss of function could have significant safety consequences. These SSCs require scheduled preventive maintenance that may have further priorities based upon risk, operating experience, and expert opinion.

- A list of SSCs whose failure or loss of function would not be self-evident. These SSCs should also be subject to scheduled oversight and maintenance.

- All other SSCs. Failure or loss of function for these will have economic consequences only.

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

A.2.2.4 Recordkeeping and Trending

Recordkeeping and trending are essential elements of both monitoring and maintenance programs. The sole product of monitoring programs is information.

In order to be useful, this information must be translated into effective maintenance practices. Therefore, (1) the information obtained by monitoring activities must be recorded in adequate, unambiguous detail in a form that allows ready retrievability, and (2) the information must be reliably relatable to specific maintenance practices that effectively address the age-related degradation that is actually occurring. These records are needed to set priorities for maintenance resources and to correlate actual operating environments and stressors with design assumptions and computed lifetimes so that SSC lifetimes and maintenance intervals can be realistically anticipated.

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

Recordkeeping can be supplemented or requirements offset by conservative maintenance practices based on equipment history or conservative condition assessments for selected SSCs; however, detailed, usable, and retrievable records of such practices and condition assessments and supplementary raw data should be maintained. This is a task that, in principle, can be simplified greatly by modern computer technology, which has enhanced the technical and economic feasibility of maintaining high quality records. Trending of information obtained by monitoring activities may be a straightforward process that leads directly to maintenance recommendations. More often, however, trending intended to lead to improved oversight and control requires considerable initial development of the basic trending program and qualification of the measures to be trended if results are to be meaningful. It is particularly important that trending methods take into account off-normal as well as normal operating conditions. Records of component failure data can be trended and monitored to assess maintenance program effectiveness. Process indicators, such as post-maintenance test results, surveillance test results, ratio of preventive to corrective maintenance, maintenance backlog, and rework frequency, should also be trended to provide indications of overall maintenance effectiveness and areas requiring improvement.

APPENDIX B

REPRESENTATIVE SYSTEMS, STRUCTURES, AND COMPONENTS POTENTIALLY IMPORTANT TO LICENSE RENEWAL*

I. PRESSURIZED WATER REACTORS

	Standard Review Plan, NUREG-0800	Generic Functional Table III	Standard Technical Specifications
A. RELIED UPON FOR PRESSURE BOUNDARY INTEGRITY, SHUTDOWN AND ACCIDENT MITIGATION			
1. Reactor Coolant Pressure Boundary			
Reactor Vessel	5.3.3	A	3/4.2.1, 4.9.1, 4.10, 3/4.9.10
Steam Generator	5.4.1, 5.4.2.2	A	3/4.4.5, 3/4.4.6, 3/4.4.10
Reactor Coolant Pump	5.4.1	A	3/4.4.1, 3/4.4.10
Piping	5.4.3	A	3/4.4.1.1, 3/4.4.10,
Pressurizer	5.4.10	A	3/4.4.3, 3/4.4.10, 3/4.4.10
Instrumentation	7.1	A	3/4.3
Valves	5.4.12	A	3/4.4.2, 3/4.4.4
2. Power Operated Relief Valves, Block Valves, and Interconnected Piping			
Pressurizer PORV	5.4.13	A	3/4.4.4, 3/4.4.9.3, 3/4.4.10
Pressurizer Block Valves	5.4.13	A	3/4.4.4, 3/4.4.10
Pressurizer Piping	5.4.3	A	3/4.4.10
Safety Valves			3/4.4.2, 3/4.4.10
3. Reactor Protection System		J	
Detector	7.2	J	3/4.3.1, 3/4.2, 2.2.1
Signal Comparator	7.2	J	3/4.3.1, 3/4.2, 2.2.1
Logic Circuit	7.2	J	3/4.3.1, 3/4.2, 2.2.1
Master Relay	7.2	J	3/4.3.1, 3/4.2, 2.2.1
Slave Relay	7.2	J	3/4.3.1, 3/4.2, 2.2.1
Connecting Wire/Cable	7.2	J	3/4.3.1, 3/4.2, 2.2.1
4. Engineered Safety Features Actuation System		K	
Detector	7.3	K	3/4.3.2, 2.2.1
Signal Comparator	7.3	K	3/4.3.2, 2.2.1
Logic Circuit	7.3	K	3/4.3.2, 2.2.1
Master Relay	7.3	K	3/4.3.2, 2.2.1
Slave Relay	7.3	K	3/4.3.2, 2.2.1
Connecting Wire/Cable	7.3	K	3/4.3.2, 2.2.1

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

	Standard Review Plan, NUREG-0800	Generic Functional Table III	Standard Technical Specifications
A. RELIED UPON FOR PRESSURE BOUNDARY INTEGRITY, SHUTDOWN AND ACCIDENT MITIGATION (contd)			
5. Control Room and Auxiliary Shutdown	7.4	N	
Cable	7.4	N	3/4.3.3.5, 2.2.1
Instrumentation	7.4	N	3/4.3.3.5, 2.2.1
6. Nuclear Instrumentation		J,K	
Source Range Detectors	7.2	J,K	3/4.3.1
Intermediate Range Detectors	7.2	J,K	3/4.3.1
Power Range Detectors	7.2	J,K	3/4.3.1
Connecting Cable		J,K	3/4.3.1
7. Non-nuclear Instrumentation		J,K	
Temperature, RCS	7.2	J,K	3/4.3.1
Pressure, RCS	7.2, 7.3	J,K	3/4.3.2, 3/4.3.1
Pressurizer Level	7.2	J,K	3/4.3.1
Flow, RCS	7.2	J,K	3/4.3.1
Reactor Vessel Level Instrumentation	7.5	--	3/4.3.3.6
Sub-cooling	7.5	J,K	3/4.3.3.6
Pressurizer Pressure	7.2, 7.3, 7.4,	J,K	3/4.3.1, 3/4.3.3, 3/4.3.2, 3/4.3.3
Steam Generator Level	7.2, 7.4, 7.5	J,K	3/4.3.1, 3/4.3.3.6
Impulse Pressure	7.7	J,K	
Steam Flow	7.7, 7.2	--	3/4.2
Feedwater Flow	7.7, 7.2	--	3/4.3.3.5 (AF)
Steam Pressure	7.7, 7.2	J,K	3/4.3.3.5
Feedwater Pressure		--	
8. In-Core Instrumentation		J,K	
Flux Detector	7.2	--	3/4.3.3.2
Thermocouple	7.5	--	3/4.3.3.6
Drive Assembly	7.2	--	3/4.3.3.2
Transfer Device	7.2	--	3/4.3.3.2
Connecting Tubing	7.2	--	
Drive Cable		--	3/4.3.3.2
Readout/Control Equipment		--	3/4.3.3.2
Gas Purge System		--	
Leak Detection System		--	
9. Seismic Category I Piping, Raceways, Cables, Hangers, Structures		A,C	
Piping	5.2.4, 5.2.3	A,C	3/4.4.10, 3/4.1.2, 3/4.4.10
Raceways	5.2.4	C	
Cables		C	
Hangers	5.2.4	C	3/4.7.9
Structures		C	

	Standard Review Plan, NUREG-0800	Generic Functional Tab. II	Standard Technical Specifications
A. RELIED UPON FOR PRESSURE BOUNDARY INTEGRITY, SHUTDOWN AND ACCIDENT MITIGATION (contd)			
10. Auxiliary Feedwater System	7.4/10.49	G	
Pumps		G	3/4.4.10, 3/4.7.1.2
Motor		G	3/4.7.1.2
Turbine		G	3/4.7.1.2
Valves		G	3/4.4.10, 3/4.7.1.2
Piping		G	3/4.4.10, 3/4.7.1.2
Pipe Supports		G	3/4.7.9
Pipe Restraints		G	3/4.7.9
Condensate Storage Tank		G (not always)	3/4.7.1.3, 3/4.4.10
Automatic Steam Generator Overfill Protection		M	3/4.3.1
Control Air			
11. Emergency Diesel Generators	8.3.1	Q	
Diesel Engine		Q	3/4.8.1
Alternator		Q	3/4.8.1
Starting Air Compressor	9.5.6	Q	3/4.8.1
Aftercooler		Q	
Air Dryer		Q	
Air Receiver		Q	
Filters		Q	
Valves		Q	
Piping		Q	
Pipe Supports		Q	
Pipe Restraints		Q	3/4.7.9
Intake Air Filter	9.5.8	Q	3/4.8.1
Silencers		Q	
Intercoolers		Q	
Ducting		Q	
Turbocharger		Q	
Exhaust Air Silencer	9.5.8	Q	3/4.8.1
Fuel Oil Storage Tank	9.5.4	Q	3/4.8.1
Day Tank		Q	3/4.8.1
Transfer Pumps	9.5.4	Q	3/4.8.1
Filters		Q	
Strainers		Q	
Piping		Q	
Valves		Q	
Injector Pumps		Q	3/4.8.1
Drain Tank		Q	3/4.8.1
Drain Tank Pump		Q	3/4.8.1
Intercooler Heat Exchanger		Q	3/4.8.1
Jacket Water Heat Exchangers		Q	3/4.8.1
Jacket Water Pumps		Q	3/4.8.1
Jacket Water Auxiliary Pump		Q	3/4.8.1
Lube Oil Cooler		Q	3/4.8.1
Valves		Q	3/4.8.1
Jacket Water Heaters		Q	3/4.8.1
Expansion Tank		Q	3/4.8.1
Piping		Q	3/4.8.1
Instrumentation		Q	3/4.8.1

	Standard Review Plan, NUREG-0800	Generic Functional Table III	Standard Technical Specifications
A. RELIED UPON FOR PRESSURE BOUNDARY INTEGRITY, SHUTDOWN AND ACCIDENT MITIGATION (contd)			
11. Emergency Diesel Generators (contd)			
Lube Oil Pumps	9.5.7	Q	3/4.8.1
Auxiliary Lube Oil Pump		Q	3/4.8.1
Motor		Q	3/4.8.1
Electric Heater		Q	3/4.8.1
Filter		Q	3/4.8.1
Strainers		Q	3/4.8.1
Valves		Q	3/4.8.1
Heat Exchangers		Q	3/4.8.1
Auxiliary Tank		Q	3/4.8.1
Rocker Lube Oil Pump		Q	3/4.8.1
Pre-Lube Pump		Q	3/4.8.1
Motor		Q	3/4.8.1
Reservoir		Q	3/4.8.1
Gas Ejector		Q	3/4.8.1
Separator		Q	3/4.8.1
Sump		Q	3/4.8.1
Tubing		Q	3/4.8.1
Instrumentation		Q	3/4.8.1
12. Station Batteries and Vital Power Supplies)	8.3.2	Q	
Battery		Q	3/4.8.2
Battery Charger		Q	3/4.8.2
Cable		Q	3/4.8.2
Breakers		Q	
13. Electrical Distribution, Safety Related	8.3.1	Q	3/4.8.3
All Components with Safety Function		Q	
14. Containment Building		0	
Containment Lines	3.8.1-3	0	3/4.6.1, 3/4.6.1.7
Shield Building		0	3/4.6.1
Primary Shield Wall		0	
Missile Shield		0	
Refueling Cavity		0	
Recirculation Sump		0	
Base Mat		0	3/4.6.1
Relief Valves			3/4.6.7
Tendons		0	3/4.6.1.7
Isolation Valve		0	3/4.6.4, 3/4.6.1.2, 3/4.9.9
Air Locks		0	3/4.6.1.3, 3/4.9.4
15. Containment Isolation System	6.2.4	0	3/4.6.1.4
Cable		0	
Instrumentation		0	

	Standard Review Plan, NUREG-0800	Generic Functional Table III	Standard Technical Specifications
A. RELIED UPON FOR PRESSURE BOUNDARY INTEGRITY, SHUTDOWN AND ACCIDENT MITIGATION (contd)			
16. Containment Spray System	6.5.2	C	
Containment Spray Pumps		C	3/4.6.2.1, 3/4.4.10
Spray Additive Tank		C	3/4.6.2.2, 3/4.4.10
Piping		C	3/4.4.10
Nozzles		C	3/4.6.2.1
Instrumentation		C	
Valves		C	3/4.4.10
17. Containment Air Cooling System	6.2.2	C	
Fans		C	3/4.6.2.3
Motors		C	
Coolers		C	3/4.6.2.3
Roughing Filters		C	3/4.6.1.9
HEPA Filters		C	3/4.6.1.9
Dampers		C	
Ductwork		C	
Instrumentation		C	
Moisture Separator		C	
Relief Devices		C	
Charcoal Filters		C	3/4.6.4, 3/4/6.1.9
18. Component Cooling Water System	9.2.1	G	
Pumps		G	3/4.7.3, 3/4.4.10
Heat Exchangers		G	3/4/7.3, 3/4.4.10
Surge Tanks		G	3/4.4.10
Valves		G	3/4.4.10, 3/4.7.3
Piping		G	3/4.4.10, 3/4.7.3
Instrumentation		G	
19. Service Water System, Safety-Related	9.2.1	G	
Pumps		G	3/4.7.4, 3/4.4.10
Strainers		G	
Piping		G	3/4.4.10, 3/4.7.4
Valves		G	3/4.4.10, 3/4.7.4
Instrumentation		G	
Cooling Towers		G	3/4.7.5
20. Emergency Core Cooling System	6.3	C	
Accumulators		C	3/4.5.1, 3/4.4.10
Boron Injection Tank		C,M	3/4.5.4.1, 3/4.4.10
Refueling Water Storage Tank		C	3/4.5.5, 3/4.4.10
Intermediate Head Injection System		C	3/4.5.2, 3/4.4.10
Low Head Injection System		C	3/4.5.2, 3/4.4.10
High Head Injection System		C	3/4.1.2.2, 3/4.1.2.4, 3/4.5.2, 3/4.1.2.1, 3/4.4.10

	Standard Review Plan, NUREG-0800	Generic Functional Table III	Standard Technical Specifications
A. RELIED UPON FOR PRESSURE BOUNDARY INTEGRITY, SHUTDOWN AND ACCIDENT MITIGATION (contd)			
20. Emergency Core Cooling System (contd)			
Containment Recirculation Sump		C	3/4.5.2
Valves		C	3/4.5.2, 3/4.4.10
Piping		C	3/4.5.2, 3/4.4.10
21. Residual Heat Removal System	5.4.7	D	
Pumps		D	3/4.5.2, 3/4.4.10, 3/4.9.8
Heat Exchangers		D	3/4.5.2, 3/4.4.10, 3/4.9.8
Valves		D	3/4.5.2, 3/4.4.10, 3/4.9.8
Piping		D	3/4.5.2, 3/4.4.10, 3/4.9.8
Instrumentation		D	3/4.3.3.5, 3/4.9.8
22. Chemical and Volume Control System	9.3.4	D,M	
Regenerative Heat Exchanger		--	3/4.4.10
Letdown Heat Exchanger		--	3/4.4.10
Ion Exchangers		--	3/4.4.10
Volume Control Tank		--	3/4.1.2, 3/4.4.10
Primary Water Storage Tank		--	3/4.1.2, 3/4.4.10
Boric Acid Tanks		D,M	3/4.1.2, 3/4.4.10
Boric Acid Batch Tank		D,M	3/4.1.2, 3/4.4.10
Boric Acid Transfer Pumps		D,M	3/4.1.2, 3/4.4.10
Filter		D,M	
Blender		D,M	
Excess Letdown Heat Exchanger		--	3/4.4.10
Valves		D,M	3/4.1.2, 3/4.4.10
Piping		D,M	3/4.1.2, 3/4.4.10
Instrumentation		--	
Positive Displacement Pump		--	3/4.4.10
23. Combustible Gas Control	6.2.5	C	
Post-Accident Hydrogen Venting System		C	3/4.6.5.3
Post-Accident Hydrogen Sampling System		C	3/4.6.5.1
Post-Accident Hydrogen Mixing System		C	3/4.6.5.4
Internal Hydrogen Recombiners		C	3/4.6.5.2
External Hydrogen Recombiners		C	3/4.6.5.2
24. HVAC, Control Room and ESF		C	
Purge and Exhaust System	6.4	C	3/4.7.7
Reactor Containment Fan Cooler System		C	3/4.6.2.3

	Standard Review Plan, NUREG-0800	Generic Functional Table III	Standard Technical Specifications
A. RELIED UPON FOR PRESSURE BOUNDARY INTEGRITY, SHUTDOWN AND ACCIDENT MITIGATION (contd)			
24. HVAC, Control Room and ESF (contd)			
Containment Activated Charcoal Filter Units System		C O	3/4.6.1.9, 3/4.6.4
Reactor Cavity and Excore Instrumentation Ventilation System		--	
Control Rod Drive Mechanism Ventilation System			--
Manipulator Crane Ventilation System		--	3/4.9.12
Pressure Vacuum and Relief System		--	3/4.6.7
Control Room Ventilation System		N	3/4.7.7
25. Instrument Air System	9.3.1	R	
Compressors		R	
After Cooler		R	
Receiver		R	
Dryer/Filter Train		R	
Accumulators		R	
Instrumentation		R	
26. Fuel Pool Structure and Cooling System	9.1.3	D	
Pumps		D	3/4.9.12, 3/4.4.10
Heat Exchanger		D	3/4.9.12, 3/4.4.10
Purification Pumps		--	3/4.9.12, 3/4.4.10
Demineralizer		--	3/4.4.10
Piping		D	3/4.9.12, 3/4.4.10
Strainers, Filters		--	3/4.9.12
Valves		D	3/4.9.12, 3/4.4.10
27. Fire Protection	9.5.1	R	
Pumps		--	3/4.7.11.1, 3/4.7.11.2
Valves		--	3/4.7.11.1, 3/4.7.11.2
Piping		--	3/4.7.11.1, 3/4.7.11.2
Tanks		--	3/4.7.11.1, 3/4.7.11.2
Instrumentation		--	3/4.3.3.8
Halon		--	3/4.7.11.4
CO_2		--	3/4.7.11.3.3
28. Ultimate Heat Sink	9.2.5	--	3/4.7.5
NA			
B. FAILURE CAN AFFECT FUNCTIONING OF CATEGORY A SSC			
1. Condensate/Feedwater System, Including Reheat	10.4.7, 10.3.6	F,R	
Main Condenser	10.4.1	--	3/4.4.10
Condensate Pumps		--	3/4.4.10

	Standard Review Plan, NUREG-0800	Generic Functional Table III	Standard Technical Specifications
B. FAILURE CAN AFFECT FUNCTIONING OF CATEGORY A SSC (contd)			
1. Condensate/Feedwater System, Including Reheat (contd)	10.4.7 10.3.6		
Demineralizers		--	3/4.4.10
LP Feedwater Heaters		--	3/4.4.10
Piping		--	3/4.4.10
Valves		--	3/4.4.10, 3/4.7.1
Main Feed Pumps		--	3/4.4.10
HP Feedwater Heaters		--	3/4.4.10
Startup Feedwater System		--	3/4.4.10
Heater Drain System		--	3/4.4.10
Condensate Storage and Transfer System		--	3/4.4.10, 3/4.7.1.3
2. Turbine, Main Generator, Controls	10.2	F,R,S	
HP Turbine		--	
LP Turbines		--	
Valves		--	
Piping		--	
Gland Steam Condenser		--	
Condenser Exhausters		--	
Regulators		--	
Piping		--	
Valves		--	
Oil Pumps		--	
Oil Reservoir		--	
Oil Coolers		--	
Turning Gear		--	
Ejector		--	
Moisture Separator Reheater		--	
Main Generator		--	
Excitation System		--	
Instrumentation		--	3/4.3.4
3. Main Steam System	10.3	A,F,R,S	
Steam Generator			3/4.7.2, 3/4.4.5, 3/4.4.10
Piping			3/4.4.10
Valves			3/4.4.10, 3/4.7.1
4. Reactor Control System			
Control Rod Drive Mechanism	3.9.4	D,M	3/4.1.3, 3/4.3.3
Logic Cabinet			
Power Cabinet			
Instrumentation			

	Standard Review Plan, NUREG-0800	[illegible]ic Functional Table III	Standard Technical Specifications
B. FAILURE CAN AFFECT FUNCTIONING OF CATEGORY A SSC (contd)			
5. Condenser Cooling System	10.4.5	F,R	
Circulating Water Pumps		--	
Valves		--	
Piping		--	
Condenser		--	
Cooling Towers		--	
6. Instrument/Service Air	9.3.1	R	
Compressors		--	
After Coolers		--	
Air Receivers		--	
Dryer/Filter Train		--	
Instrumentation		--	
C. OTHER SSCs IMPORTANT TO LICENSE RENEWAL			
1. Reactor Post-Accident Monitoring System	7.5	J,K	3/4.3.3.6
Instrumentation			
2. Safety Parameter Display System		K	
Computer	7	--	
Instrumentation		--	
3. Waste Systems: Liquid, Gas, Solid	11.4, 11.2 11.3	P	
Liquid Subsystems			
Solid Subsystems			
Gaseous Subsystems			
4. Fuel Handling Systems	L,P	3/4.9	
New Fuel Storage Area			
Spent Fuel Storage Pool			3/4.4.10, 3/4.9.11
Fuel Storage Building Crane			
Spent Fuel Bridge Crane			
New Fuel Elevator			
New Fuel Handling Tool		--	
Spent Fuel Handling Tool		--	
Refueling Cavity			
Transfer Canal			3/4.4.10
Polar Crane			3/4.9.7
Manipulator Crane			3/4.9.6
Red Cluster Assembly Change Fixture		--	
Reactor Vessel Head Lifting Device		--	
Reactor Internals Lifting Device			--
Stud Tensioner		--	
Refueling Tools		--	
Conveyer Car Assembly			

	Standard Review Plan, NUREG-0800	Gener. Functional Table III	Standard Technical Specifications
C. OTHER SSCs IMPORTANT TO LICENSE RENEWAL (contd)			
Drive Frame Assembly			
Lifting Mechanism		--	
Valve			3/4.3.3.1, 3/4.9.2
Instrumentation			
Controls			
5. Radiation and Environmental Monitoring		K	
Containment Air Particulate Detector	11.3	--	3/4.3.3.1
Containment Noble Gas Monitor	11.3	--	3/4.3.3.1
Containment Purge Exhaust Monitor	11.3	--	3/4.3.3.1
Auxiliary Building Ventilation System Monitor	11.3	--	3/4.3.3.1
Plant Vent Stack Monitor	11.3	--	3/4.3.3.1
Control Room Air Intake Monitor	11.3, 9.4.1	--	3/4.3.3.1
Condenser Air Ejection Gas Monitor	11.3	--	3/4.3.3.1
Steam Generator Blowdown Liquid Monitor	11.2	--	
Component Cooling Water System Monitor		--	
Service Water Effluent Discharge Monitor			--
Waste Disposal System Liquid Effluent Monitor		--	
Gas Decay Tank Effluent Gas Monitor		--	
6. Communications Equipment	9.5.2	K	3/4.9.5
Telephone System		--	
Radio System		--	
Page System		--	
7. Intrusion Detection		--	
Motion Detection System		--	
Sound Monitoring System		--	
Television System		--	
RF Field System		--	
E-Field System		--	
8. Access Control		K	
Door Control System		--	
Badging/ID System		--	

	Standard Review Plan, NUREG-0800	Generic Functional Table III	Standard Technical Specifications
C. OTHER SSCs IMPORTANT TO LICENSE RENEWAL (contd)			
9. Guard Response Support			
Weapons Systems		--	
Communications Systems		--	
10. Alarm Station Operation		J	
Instrumentation		--	
11. Area Radiation Monitors		J,K	3/4.3.3.1
Area Radiation Monitoring System			--
12. Radiation Survey Instruments			
Radiation Monitoring Systems		--	
13. Personnel Monitoring Devices			
Radiation Detectors	12.3/4	--	3/4.6
14. Personnel Protection Barriers			
Machinery		--	
Structural		--	

II. BOILING WATER REACT(

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

	Standard Review Plan, NUREG-0800	Ger Fun(,al Table ıII	Standard Technical Specifications
A. RELIED UPON FOR PRESSURE BOUNDARY INTEGRITY, SHUTDOWN AND ACCIDENT MITIGATION (contd)			
7. Seismic Category 1 Piping, Raceways, Cables, Hangers, and Structures to Support Dynamic Loads		A	
Reactor Vessel, System	3.0, 3.10	A,B	3/4, 4.6
Recirculation System	3.0, 3.9, 6, 3.10	A	3/4, 4.6
Main Steam System	3.0, 3.9	A,C	3/4, 4.7
Condensate and Feedwater System	3.0, 3.7	A,G	3/4, 4.4
Automatic Reactor Vessel Overfill Protection	3.0, 3.7	M	3/4.3.1
Reactor Core Isolation Cooling System	5.46	A,G	3/4, 7.3
Reactor Water Cleanup System (Category 1 piping and valves)	5.4.8	A,P	3/4, 4.4
8. Primary Containment		--	
Reactor Building Foundation	3.2.1	O,S	3/4, 6.1, 6.2, 6.3, 6.4, 6.5
Drywell	3.2.1	O	3/4, 6.1, 6.2, 6.3, 6.4, 6.5
Drywell Access Penetrations	3.2.1	O,S	3/4, 6.1, 6.2, 6.3, 6.4, 6.5
Drywell Electrical Penetrations	3.0	O,S	3/4, 6.1, 6.2, 6.3, 6.4, 6.5
Drywell Pipe Penetrations	3.0	O,S	3/4, 6.1, 6.2, 6.3, 6.4, 6.5
Horizontal Vents and Weir Wall	3.0	O	3/4, 6.1, 6.2, 6.3, 6.4, 6.5
Containment	3.0	O	3/4, 6.1, 6.2, 6.3, 6.4, 6.5
Fuel Transfer Tube	9.1	O	3/4, 6.1, 6.2, 6.3, 6.4, 6.5
Suppression Pool	9.0, 3.0	O,G	3/4, 6.1, 6.2, 6.3, 6.4, 6.5
Containment Upper Pool	9.1	O	3/4, 6.1, 6.2, 6.3, 6.4, 6.5
Primary Containment HVAC System	9.4	O	3/4, 6.1, 6.2, 6.3, 6.4, 6.5
Primary Containment Auxiliary System	9.4	O	3/4, 6.1, 6.2, 6.3, 6.4, 6.5
Containment Spray	9.4, 6, 5.2	O,G	3/4, 6.1, 6.2, 6.3, 6.4, 6.5
9. Containment Air Cooling		C	
Drywell Recirculation System	6.2.2	C	3/4, 6.7
Drywell Purge Ventilation System	6.2.2	C	3/4, 6.7

A. RELIED UPON FOR PRESSURE BOUNDARY INTEGRITY, SHUTDOWN AND ACCIDENT MITIGATION (contd)

	Standard Review Plan, NUREG-0800	General Functional Table III	Standard Technical Specifications
Containment Normal Ventilation System	6.2.2	C	3/4, 6.7
Containment High Flow Purge System	6.2.2	C	3/4, 6.7
Containment Recirculation System	6.2.2	C	3/4, 6.7
10. Hydrogen Control System		C	
Containment Combustible Gas Control System	6.2.5	C	3/4, 6.7
Distributed Igniter System	6.2.5	C	3/4, 6.7
Containment Atmospheric Monitoring System	6.2.5	C	3/4, 6.7
11. Station Batteries and Vital Power Supplies		Q	
4.16 KV Switchgear	8.3.1	Q	3/4, 8.1, 8.2, 8.3, 8.4
Division 1&2 Diesel Generators	9.5	Q	3/4, 8.1, 8.2, 8.3, 8.4
Division 3 Diesel Generators	9.5	Q	3/4, 8.1, 8.2, 8.3, 8.4
480 V Switchgear	8.3.1	Q	3/4, 8.1, 8.2, 8.3, 8.4
Essential AC Power Supplies	8.3.1	Q	3/4, 8.1, 8.2, 8.3, 8.4
Batteries 125, 250, VDC	8.3.2	Q	3/4, 8.1, 8.2, 8.3, 8.4
Battery Chargers	8.3.2	Q	3/4, 8.1, 8.2, 8.3, 8.4
Nuclear System Protection Separate Divisional Power Supplies	8.1	K	3/4, 8.1, 8.2, 8.3, 8.4
12. EDG (Including Air Storage, Fuel Storage and Transmission and Cooling)		I	
Cooling Water System	9.5.5	C	3/4, 5.1
Lube Oil System	9.5.7	C	3/4, 5.1
Air Compressors	9.5.6	C,I	3/4, 5.1
Air Storage Tanks	9.5.6	C,I	3/4, 5.1
Diesel Engine	9.5.8		3/4, 8.1-8.4
Generator	9.5.8.1		3/4, 8.1-8.4
Intake & Exhaust	9.5.8		3/4, 8.1-8.4
Fuel Oil System	9.5.4		3/4, 8.1-8.4
Instrumentation & Control	9.5		3/4, 8.1-8.4

	Standard Review Plan, NUREG-0800	Generic Functional Table III	Standard Technical Specifications
A. RELIED UPON FOR PRESSURE BOUNDARY INTEGRITY, SHUTDOWN AND ACCIDENT MITIGATION (contd)			
13. Electrical Distribution - Safety Related		Q	
Divisional Power 1,2,3,4,5	8.1	Q	3/4, 8.1-8.4
Static Bypass Switch	8.1	Q	3/4, 8.1-8.4
Inverter	8.1	Q	3/4, 8.1-8.4
14. Reactor Core Isolation Cooling System (Including Isolation Condenser)		A,C,E,G	
Steam Isolation Valves	5.4.6	A,C,E	3/4, 7.3
Steam Flow Elements	5.4.6	A,C,E	3/4, 7.3
Turbine Trip Throttle Valve	5.4.6	A,C,E	3/4, 7.3
Turbine Governor Valve	5.4.6	A,C	3/4, 7.3
Turbine	5.4.6	A,C	3/4, 7.3
Turbine Oil System	5.4.6	C	3/4, 7.3
Gland Seal Elements	5.4.6	G,C	3/4, 7.3
Exhaust Piping	5.4.6	G,C	3/4, 7.3
Suction Strainer	5.4.6	G,C	3/4, 7.3
Suction Valves	5.4.6	G,C	3/4, 7.3
Water Log Pump	5.4.6	G,C	3/4, 7.3
RCIC Pump	5.4.6	G,C	3/4, 7.3
Auxiliary Equipment Cooling	5.4.6	G,C	3/4, 7.3
Minimum Flow Bypass Line	5.4.6	G,C	3/4, 7.3
Test Recirculator Line	5.4.6	G,C	3/4, 7.3
Testable Check Valve	5.4.6	G,C	3/4, 7.3
Flow Controller	5.4.6	G,C	3/4, 7.3
15. High Pressure Cooling Injection System		G,C	
Suction Path	6.3	G,C	3/4, 5.1
HPCS(I) Pump	6.3	G,C	3/4, 5.1
Discharge Path	6.3	G,C	3/4, 5.1
HPCS(I) Water Leg Pump	6.3	G,C	3/4, 5.1
Leak Detection System	6.3	G,C	3/4, 5.1
Valve Interlock	6.3	G,C	3/4, 5.1
16. Automatic Depressurization System		A	
Safety/Relief Valves	6.0	A	3/4, 5.1
Air Supply	6.0	--	3/4, 5.1
Vacuum Breakers	6.0	--	3/4, 5.1

	Standard Review Plan, NUREG-0800	Gene Functional Table III	Standard Technical Specifications
A. RELIED UPON FOR PRESSURE BOUNDARY INTEGRITY, SHUTDOWN AND ACCIDENT MITIGATION (contd)			
17. Core Spray System or Low Pressure Injection System		--	
Suction Path	5.4.7	C	3/4, 5.1
LPCS Pump	5.4.7	C	3/4, 5.1
Discharge Path	5.4.7	C	3/4, 5.1
LPCS Water Leg Pump	5.4.7	C	3/4, 5.1
LPCI System	5.4.7	C	3/4, 5.1
18. Reactor Circulation System		A,H	
Recirculation Loop Suction Piping	5.2.3	A.H	3/4, 4.1
Suction Isolation Valve	5.2.3	A,H	3/4, 4.1
Recirculation Pumps	5.4	A,H	3/4, 4.1
Recirculation Pump Shaft Seals	5.4	A,H	3/4, 4.1
Recirculation Pump Discharge Piping	5.4	A,H	3/4, 4.1
Flow Control Valve	5.4	A,H	3/4, 4.1
Discharge Isolation Valve	5.4	A,H	3/4, 4.1
Reactor Water Sample Connnection	5.4	A.H	3/4, 4.1
Discharge Manifold and Risers	4.5.2	A.H	3/4, 4.1
Jet Pumps	4.5.2	A,H	3/4, 4.1
Recirculation Pump Motors	5.4	A,H	3/4, 4.1
Low Frequency Motor Generator Sets	5.4	H	3/4, 4.1
19. Residual Heat Removal System (Including Drywell Spray)		C	
Suction Strainers	5.4.7	C,D,G	3/4, 4.9, 4.1
RHR Water Leg Pump	5.4.7	C,D,G	3/4, 4.9, 4.1
RHR Pumps	5.4.7	C,D,G	3/4, 4.9, 4.1
RHR Heat Exchangers	5.4.7	C,D,G	3/4, 4.9, 4.1
Motor Operated Valves	5.4.7	C,D,G	3/4, 4.9, 4.1
Testable Check Valves	5.4.7	C,D,G	3/4, 4.9, 4.1
Containment Spray Spargers	5.4.7	C,D,G	3/4, 4.9, 4.1
Air Operated Control Valves	5.4.7	D	3/4, 4.9, 4.1
Electro Pneumatic Controllers	5.4.7	D	3/4, 4.9, 4.1
20. RHR/Shutdown Service Water System		--	
Ultimate Heat Sink Basin and Towers	9.2.1, 9.2.5	G	3/4, 5.1, 7.1
Standby Service Water Pumps	9.2.1	G	3/4, 5.1, 7.1
Heat Exchangers	9.2.5	G	3/4, 5.1, 7.1

A. RELIED UPON FOR PRESSURE BOUNDARY INTEGRITY, SHUTDOWN AND ACCIDENT MITIGATION (contd)

	Standard Review Plan, NUREG-0800	Generic Functional Table II	Standard Technical Specifications
21. Emergency Equipment Cooling		C	
Heat Exchangers	9.2.1	C	3/4, 7.1
Closed Cooling Water System	9.2.1	C	3/4, 7.1
22. HVAC-Control Room and ESF		N,R	
Supply Air Handling Units	6.4	N,R	3/4, 7.2
Recirculation Fans	6.4	N,R	3/4, 7.2
Makeup Air Cleaning Units	6.4	N,R	3/4, 7.2
23. Instrument Air--Important to Safety		N,R	
Service and Instrument Air Compressors	9.3.1	N,R	N/A
Air Receiver	9.3.1	N,R	N/A
Refrigeration Air Dryers and After Filters	9.3.1	N,R	N/A
Dessicant Air Dryers	9.3.1	N,R	N/A
Booster Instrument Air Compressors	9.3.1	N,R	N/A
24. Fuel Pool Structure and Cooling System		D,L	
Skimmers Weirs and Scuppers	9.1	D,L	3/4, 9.1-9.12
FPCC Drain Tank	9.1	D,L	3/4, 9.1-9.12
FPCC Pumps	9.1.2	D,L	3/4, 9.1-9.12
FPCC Heat Exchangers	9.1.2	D,L	3/4, 9.1-9.12
Filter/Demineralizers	9.1.2	D,L	3/4, 9.1-9.12
Diffusers	9.1.2	D,L	3/4, 9.1-9.12
Spent Fuel Pool	9.1.3	D,L	3/4, 9.1-9.12
Transfer Pool	9.1.3	D,L	3/4, 9.1-9.12
25. Fire Protection (Including Suppression)		R	
Fresh Water Supplies	9.5.1	R	3/4, APPENDIX "R"
Fire Water Supplies	9.5.1	R	3/4, APPENDIX "R"
Fire Jockey Pumps	9.5.1	R	3/4, APPENDIX "R"
Fire Main	9.5.1	R	3/4, APPENDIX "R"
Manual Hose Station	9.5.1	R	3/4, APPENDIX "R"
Preaction Type Sprinkler System	9.5.1	R	3/4, APPENDIX "R"
Deluge Type Sprinkler System	9.5.1	R	3/4, APPENDIX "R"
Wet Pipe Type System	9.5.1	R	3/4, APPENDIX "R"
High Pressure CO_2	9.5.1	R	3/4, APPENDIX "R"
Low Pressure CO_2	9.5.1	R	3/4, APPENDIX "R"
Halon System	9.5.1	R	3/4, APPENDIX "R"
Heat Detection Systems	9.5.1	R	3/4, APPENDIX "R"
Smoke Detectors	9.5.1	R	3/4, APPENDIX "R"

	Standard Review Plan, NUREG-0800	Gener Functi Table III	Standard Technical Specifications
A. RELIED UPON FOR PRESSURE BOUNDARY INTEGRITY, SHUTDOWN AND ACCIDENT MITIGATION (contd)			
25. Fire Protection (Including Suppression) (contd)			
Flame Detectors	9.5.1	R	3/4, APPENDIX "R"
26. Ultimate Heat Sink		R	
Circulating Water Pumps	9.2.5	R	3/4, 7.1
Cooling Towers	9.2.5	R	3/4, 7.1
Main Condenser	9.2.5	R	3/4, 7.1
Major Valves	9.2.5	R	3/4, 7.1
Major Piping	9.2.5	R	3/4, 7.1
B. FAILURE CAN AFFECT FUNCTIONING OF CATEGORY A SSC			
1. Condensate/Feedwater System Including Reheat	10.4.7	R	4.4
2. Turbine-Generator and Controls	10.2	R,S	3/4, 3.8
3. Main Steam System	10.3	R,S	3/4, 4.7
4. Reactor Control System	7.1	D,M	3/4, 4.1
5. Condenser Cooling System (Circulation Water System)	10.4.5	R	N/A
6. Instrument Air/Service Air, Not S.R.	9.3.1	R	N/A
7. Switchyard	8.2	R	3/4, 8.1-8.4
C. OTHER SSCs IMPORTANT TO LICENSE RENEWAL			
1. Reactor Post-Accident Monitoring System	9.3.2	J,K	3/4, 3.1-3.9
Instrumentation	9.3.2	K	3/4, 3.1-3.9
2. Safety Parameter Display System		K	3/4, 3.1-3.9
Computer	7.1	--	3/4, 3.1-3.9
Instrumentation	7.1	--	3/4, 3.1-3.9
3. Waste Systems: Liquid, Gas, Solid	11.0	P	3/4, 11.1-11.4
Liquid Subsystems	11.0	--	3/4, 11.1-11.4
Solid Subsystems	11.0	--	3/4, 11.1-11.4
Gaseous Subsystems	11.0	--	3/4, 11.1-11.4

A. RELIED UPON FOR PRESSURE BOUNDARY INTEGRITY, SHUTDOWN AND ACCIDENT MITIGATION (contd)

	Standard Review Plan, NUREG-0800	Generic Functional Table II	Standard Technical Specifications
4. Fuel Handling Systems	9.0	L,P	3/4, 9.1-9.12
New Fuel Storage Area	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
Spent Fuel Storage Pool	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
Fuel Storage Building Crane	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
Spent Fuel Bridge Crane	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
New Fuel Elevator	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
New Fuel Handling Tool	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
Spent Fuel Handling Tool	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
Refueling Cavity	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
Transfer Canal	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
Polar Crane	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
Manipulator Crane	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
Red Cluster Assembly Change	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
Fixture	9.1.3	--	3/4, 9.1-9.12
Reactor Vessel Head Lifting	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
Device	9.1.3	--	3/4, 9.1-9.12
Reactor Internals Lifting	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
Device	9.1.3	--	3/4, 9.1-9.12
Stud Tensioner	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
Refueling Tools	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
Conveyer Car Assembly	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
Drive Frame Assembly	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
Lifting Mechanism	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
Valve	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
Instrumentation	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12
Controls	9.1.1, 9.1.2,	--	3/4, 9.1-9.12
	9.1.3	--	3/4, 9.1-9.12

	Standard Review Plan, NUREG-0800	Gener Functio... Table III	Standard Technical Specifications
A. RELIED UPON FOR PRESSURE BOUNDARY INTEGRITY, SHUTDOWN AND ACCIDENT MITIGATION (contd)			
5. Radiation and Environmental Monitoring		K	
Containment Air	12.1-12.5		3/4, 3.7
Particulate Detector	12.1-12.5	--	3/4, 3.7
Containment Noble Gas Monitor	12.1-12.5	--	3/4, 3.7
Containment Purge Exhaust Monitor	12.1-12.5	--	3/4, 3.7
Auxiliary Building Ventilation System Monitor	12.1-12.5	--	3/4, 3.7
Plant Vent Stack Monitor	12.1-12.5	--	3/4, 3.7
Control Room Air Intake Monitor	12.1-12.5	--	3/4, 3.7
Condenser Air Ejection Gas Monitor	12.1-12.5	--	3/4, 3.7
Steam Generator Blowdown Liquid Monitor	12.1-12.5	--	3/4, 3.7
Component Cooling Water System Monitor	12.1-12.5	--	3/4, 3.7
Service Water Effluent Discharge Monitor	12.1-12.5	--	3/4, 3.7
Waste Disposal System Liquid Effluent Monitor	12.1-12.5	--	3/4, 3.7
Gas Decay Tank Effluent Gas Monitor	12.1-12.5	--	3/4, 3.7
6. Communications Equipment	9.5.2	K	N/A
Telephone System	9.5.2	--	N/A
Radio System	9.5.2	--	N/A
Page System	9.5.2	--	N/A
7. Intrusion Detection	13.6	--	N/A
Motion Detection System	13.6	--	N/A
Sound Monitoring System	13.6	--	N/A
Television System	13.6	--	N/A
RF Field System	13.6	--	N/A
E-Field System	13.6	--	N/A
8. Access Control	13.6	K	N/A
Door Control System	13.6	--	N/A
Badging/ID System	13.6	--	N/A
9. Guard Response Support			
Weapons Systems	13.6	--	N/A
Communications Systems	13.6	--	N/A
10. Alarm Station Operation		J	
Instrumentation	13.6	--	N/A

	Standard Review Plan, NUREG-0800	Generic Functional Table III	Standard Technical Specifications
A. RELIED UPON FOR PRESSURE BOUNDARY INTEGRITY, SHUTDOWN AND ACCIDENT MITIGATION (contd)			
11. Area Radiation Monitors		J,K	
Area Radiation Monitoring System	12.0	--	3/4, 3.7
12. Radiation Survey Instruments			
Radiation Monitoring Systems	12.0	--	3/4, 3.7
13. Personnel Monitoring Devices			
Radiation Detectors	12.0	--	3/4, 3.7
14. Personnel Protection Barriers			
Machinery	13.6	--	N/A
Structural	13.6	--	N/A

REGULATORY ANALYSIS

A separate regulatory analysis was not prepared for this draft regulatory guide. The regulatory analysis prepared for the proposed 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," NUREG-1362, "Regulatory Analysis for Proposed Rule on Nuclear Power Plant License Renewal," July 1990, provides the regulatory basis for this guide and examines the costs and benefits of the rule as implemented by the guide. A copy of the regulatory analysis is available for inspection and copying for a fee at the NRC Public Document Room, 2120 L Street NW., Washington, DC.

BACKFIT ANALYSIS

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

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