RECEIVED

OCT 23 1995

OSTI

**Regulatory Instrument Review:
Aging Management of LWR Cables,
Containment and Basemat,
Reactor Coolant Pumps, and
Motor-Operated Valves**

**E. V. Werry
S. Somasundaram**

September 1995

**Prepared for
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
under Contract DE-AC06-76RLO 1830
NRC FIN B2865**

**Pacific Northwest Laboratory
Operated for the U.S. Department of Energy
by Battelle Memorial Institute**

PNL-8475

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor Battelle Memorial Institute, nor any of their employees, makes any warranty, expressed or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof, or Battelle Memorial Institute. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

PACIFIC NORTHWEST LABORATORY
operated by
BATTELLE MEMORIAL INSTITUTE
for the
UNITED STATES DEPARTMENT OF ENERGY
under Contract DE-AC06-76RLO 1830

Printed in the United States of America

Available to DOE and DOE contractors from the
Office of Scientific and Technical Information, P.O. Box 62, Oak Ridge, TN 37831;
prices available from (615) 576-8401. FTS 626-8401.

Available to the public from the National Technical Information Service,
U.S. Department of Commerce, 5285 Port Royal Rd., Springfield, VA 22161.

The contents of this report were printed on recycled paper

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

**Regulatory Instrument Review:
Aging Management of LWR Cables,
Containment and Basemat,
Reactor Coolant Pumps, and
Motor-Operated Valves**

E. V. Werry
S. Somasundaram

September 1995

Prepared for
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
under Contract DE-AC06-76RLO 1830
NRC FIN B2865

Pacific Northwest Laboratory
Richland, Washington 99352

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED
DISTRIBUTION OF UNLIMITED

MASTER

30

Preface

This report was prepared by Pacific Northwest Laboratory to provide a preliminary assessment regarding the extent to which regulatory instruments contain information pertinent to managing aging. Assessments of the applicable regulatory instruments regarding aging management are complex and subject to differences in interpretation. Therefore, the perspectives in this report should be considered

preliminary. These perspectives are not established needs or views and do not reflect regulatory positions or requirements.

The material provided in this report was prepared and reviewed in 1991/1992. However, the report was not published until 1995. Therefore, any changes in the regulatory instruments during the interim period are not reflected in this report.

Abstract

The results of Stage 2 of the Regulatory Instrument Review are presented in this volume. Selected regulatory instruments, such as the Code of Federal Regulations (CFR), U.S. Nuclear Regulatory Commission (NRC), Regulatory Guides, and ASME Codes, were investigated to determine the extent to which these regulations apply aging management to selected safety-related components in nuclear power plants. The Regulatory Instrument Review was funded by the NRC under the Nuclear Plant Aging Research (NPAR) program. Stage 2 of the review focused on four safety-related structures and components; namely, cables, containment and base-mat, reactor coolant pumps, and motor-operated valves. The review suggests that the primary emphasis of the regulatory instruments was on the

design, construction, start-up, and operation of a nuclear power plant, and that aging issues were primarily addressed after an aging-related problem was recognized. This Stage 2 review confirms the results of the prior review; (see Regulatory Instrument Review: Management of Aging of LWR Major Safety-Related Components, NUREG/CR-5490. The observations indicate that the regulations generally address management of age-related degradation indirectly. Specific age-related degradation phenomena frequently are dealt with in bulletins and notices or through generic issues, letters, etc. The major recommendation of this report, therefore, is that the regulatory instruments should more directly and explicitly address the aging phenomenon and the management of the age-related degradation process.

Summary

A review of regulatory documents, referred to here as regulatory instruments, has been conducted to determine the extent to which aging management is specifically applied to selected major light-water reactor (LWR) safety-related components. The review initially addressed the reactor pressure vessel, steam generator, primary piping, pressurizer vessel, and emergency diesel generators. The results were published as Regulatory Instrument Review: Management of Aging of LWR Major Safety-Related Components, NUREG/CR-5490 (Werry 1990).

This report presents the results of Stage 2 of the Regulatory Instrument Review and includes cables, containment and basemat, reactor coolant pumps, and motor-operated valves as the components. The review, which was conducted for the U.S. Nuclear Regulatory Commission (NRC) under the Nuclear Plant Aging Research (NPAR) Program at Pacific Northwest Laboratory (PNL), has continued since 1988 and concludes with this report. The scope of this review covers the regulatory instruments relating to cables, containment and basemat, reactor coolant pumps, and motor-operated valves.

It was concluded that these regulatory instruments do contain aging guidance; however, the regulatory instruments are neither primarily designed to account for aging of components nor to manage the consequences of aging over time. For example, the regulatory instruments typically emphasize design, construction, start-up, and operation. Actions to recognize and mitigate age-related degradation are often not explicitly mentioned in the regulatory instruments. However, such actions are addressed through bulletins, general safety issues, information notices, etc., in response to problems related to aging. An exception to this general regulatory instrument emphasis is related to the component cables because of the influence of the IEEE Standards that specify component/equipment qualification and testing.

It is recommended that the recognition and mitigation of significant age-related degradation be made more explicit in the applicable regulatory instruments. In general, the principal observations of this report, such as the need to address replacement

methods and the need for improved non-destructive evaluation methods, agree with the observations reported in NUREG/CR-5490. The findings related to the containment and basemat, however, focus on the generic issues associated with the areas of the containment and basemat that are inaccessible to inservice inspection.

This report provides a brief background of the Stage 1 review and the scope for the Stage 2 review. The Stage 2 work includes a summary of regulatory instruments studied, the review methodology, a discussion of the aging issues and observations, and recommendations and conclusions. The results of the review are presented in tabular form in APPENDICES I, II, III, and IV.^(a)

The regulatory instruments reviewed are those historically associated with nuclear power plant design, construction, and operation. They include regulations, guides, codes and standards, and technical specifications, such as the Code of Federal Regulations, the American Society of Mechanical Engineers-Boiler and Pressure Vessel Code, and the U.S. NRC Regulatory Guides (see APPENDIX V). The review method consists of an evaluation of the requirements found in the regulatory instruments to determine whether they relate to a specific component aging issue. Finally, a brief description of aging issues is provided in the APPENDICES.

A discussion is provided for each component's aging issues and the rationale for choosing those particular aging issues. The aging issues were primarily chosen from NPAR reports and from experience and judgment of the PNL staff.

Selected examples of interpretations of aging management provided by the regulatory instruments for the different components are also included.

(a) The regulatory instruments, e.g., Code of Federal Regulations and American Society of Mechanical Engineers Code, frequently make use of Appendix in their text; therefore, the Appendices attached to this report will be identified in CAPITAL LETTERS: e.g., APPENDIX I, II, III, etc., to distinguish them from those found in the regulatory instruments.

Acknowledgments

The authors of this report acknowledge the support and technical guidance provided by J. P. Vora of the U.S. Nuclear Regulatory Commission during the course of the regulatory instrument review under the Nuclear Plant Aging Program. Our thanks also go to V. N. Shah, Idaho National Engineering Laboratory, and to H. D. Haynes, Oak Ridge National Laboratory, for their contribution to the review by providing the "Understanding and Managing Aging Tables and Figures," for reactor coolant pumps, boiling-water

reactor containments, and motor-operated valves.

The authors also wish to thank the following Pacific Northwest Laboratory staff for their contributions to this report: A. B. Johnson, Jr., for Project Management support and technical guidance; and P. M. Daling, S. H. Bush, and J. C. Spanner for their expertise, review of the regulatory instruments, and input to the review of the regulatory instruments. A special thanks goes to R. C. Pedersen for his timely editing of the report.

Acronyms/Abbreviations

ACI	American Concrete Institute
ANS	American Nuclear Society
ANSI	American National Standards Institute, Inc.
ASME	American Society of Mechanical Engineers
ASTM	American Society of Testing and Materials
BOP	balance-of-plant
BPVC	Boiler and Pressure Vessel Code
BWR	boiling-water reactor
CFR	Code of Federal Regulations
DBA	design-basis accident
ECCS	emergency core cooling system
EDG	emergency diesel generator
EPAs	electrical penetration assemblies
FSAR	final safety analysis report
GDC	general design criterion/criteria
GSI	generic safety issues
IEEE	Institute of Electrical and Electronic Engineers
IGSCC	intergranular stress corrosion cracking
ISI	inservice inspection
IST	inservice testing
LER	licensee event report
LOCA	loss-of-coolant accident
LR	license renewal
LWR	light-water reactor
MOV	motor-operated valve
NDE	nondestructive examination
NPAR	Nuclear Plant Aging Research
NPEC	IEEE Nuclear Power Engineering Committee
NPP	nuclear power plant
NRC	U.S. Nuclear Regulatory Commission
NUREG	Nuclear Regulatory Commission Report
ORNL	Oak Ridge National Laboratory

PCCV	prestressed concrete containment vessel
PLEX	Plant Life Extension
PNL	Pacific Northwest Laboratory
PVRC	Pressure Vessel Research Committee
PWR	pressurized-water reactor
RCCV	reinforced concrete containment vessel
RCP	reactor coolant pump
RG	(NRC) regulatory guides
RPV	reactor pressure vessel
SAR	safety analysis report
SCC	stress corrosion cracking
SRP	Standard Review Plan
SS	stainless steel
SSC	systems, structures, and components
SSE	safe shutdown earthquake
TS	technical specification

Acronyms/Abbreviations Used Only in the Appendices

BTP	branch technical position
CC	concrete containment
EAF	environmental assisted fatigue
EPRI	Electric Power Research Institute
EQ	equipment qualification
ESF	engineered safety feature
LCO	limiting conditions for operation
LE	life extension
MSLB	main steam line break
NDT	nil-ductility transition temperature
NFPA	National Fire Protection Association
NUMARC	Nuclear Management and Resources Council
PSAR	Preliminary Safety Analysis Reports
RCS	reactor coolant system
RT _(NDT)	reference nil-ductility temperature

SAR	safety analysis report
TMI	Three Mile Island
UT	ultrasonic testing

Contents

Preface	iii
Abstract	v
Summary	vii
Acknowledgments	ix
Acronyms/Abbreviations	xi
1 Introduction	1.1
1.1 Purpose of the Regulatory Instrument Review	1.1
1.2 Background of Previous Reviews	1.1
1.2.1 Stage 1 Review, Before FY 1990	1.1
1.2.2 Stage 2 Review, FY 1990 and FY 1991	1.1
1.3 Scope of Regulatory Instruments Reviewed	1.1
2 Review Methodology	2.1
3 Discussion of Component Aging Issues and Aging Management	3.1
3.1 Cables	3.1
3.1.1 Aging Issues -- Cables	3.1
3.1.2 Applicable Aging Management	3.2
3.1.2.1 Aging Management Features Found in the Regulatory Instruments for Cables	3.2
3.2 Containment and Basemat	3.3
3.2.1 Aging Issues -- Containment and Basemat	3.4
3.2.1.1 BWR Stressors and Aging Issues	3.4
3.2.1.2 PWR Stressors and Aging Issues	3.5
3.2.2 Applicable Aging Management	3.6
3.2.2.1 BWR Containment and Basemat Aging Management Features	3.6
3.2.2.2 PWR Aging Management Features	3.7
3.3 Reactor Coolant (or Recirculating) Pumps	3.8
3.3.1 Aging Issues -- Reactor Coolant Pumps	3.8
3.3.2 Applicable Aging Management	3.10
3.4 Motor-Operated Valves	3.11
3.4.1 Aging Issues -- Valves	3.13
3.4.2 Applicable Aging Management	3.14
4 Observations, Recommendations, and Conclusions	4.1

4.1	Observations	4.1
4.1.1	General Observations	4.1
4.1.2	Specific Observations for Containment and Basemat	4.1
4.2	Recommendations and Conclusions	4.2
5	References	5.1
Appendix I:	Regulatory Instrument Review for Cables	I.1
Appendix II:	Regulatory Instrument Review for Containment and Basemat	II.1
Appendix III:	Regulatory Instrument Review for Reactor Coolant (or Recirculating) Pumps	III.1
Appendix IV:	Regulatory Instrument Review for Valves	IV.1
Appendix V:	Glossary of Regulatory Instrument Descriptions	V.1
Appendix VI:	Glossary of Aging Issues	VI.1

1 Introduction

1.1 Purpose of the Regulatory Instrument Review

The purpose of the Regulatory Instrument Review is to evaluate the existing regulatory guides, standards, specifications, and codes for applicable instructions and/or guidance on the management of aging of selected safety-related components. The review is an element of the U.S. Nuclear Regulatory Commission's (NRC) Nuclear Plant Aging Research (NPAR) Program at Pacific Northwest Laboratory (PNL)^(a). Documented in this report are the results of PNL's Stage 2 review, which has been limited to cables, containment and basemat, reactor coolant pumps (RCPs), and motor-operated valves (MOVs). These components were selected from the NPAR program plan, NUREG-1144 (NRC 1987), which identified a list of components with higher priorities related to the management of aging. The list included safety-related components that have been subject to aging problems. Ultimately, at the completion of all component reviews, specific recommendations will be made for changes, revisions, and research, if required, to improve the guidance of the regulatory instruments to more effectively manage aging of safety-related components and to assist in life extension of the components. Age-related issues that are specific to license renewal (LR) need to be considered and may need to be further addressed in the regulations. While there may be some value in a near-term review specific to LR, the optimum timing for a comprehensive review would seem to be

- a) after the Industry Reports have been issued, and
- b) after the LR rule and Standard Review Plan (SRP) for LR have been tested by reviews of one or two plant applications.

1.2 Background of Previous Reviews

1.2.1 Stage 1 Review, Before FY 1990

The regulatory instrument review in FY 1988 and 1989 focused on five light-water reactor (LWR) major safety-related components: reactor pressure vessels (RPVs), steam generators, primary piping (reactor coolant piping), pressurizer vessel, and emergency diesel generators (EDGs). A report covering this Stage 1 work, entitled Regulatory Instrument Review: Management of Aging of LWR Major Safety-Related Components, NUREG/CR-5490, was published in October 1990 (Werry 1990).

1.2.2 Stage 2 Review, FY 1990 and FY 1991

The regulatory instrument review continued during FY 1990 focusing on cables and the containment and basemat; a draft letter report provided these results. The FY 1991 review was extended to RCPs and selected valves. The results of the Stage 2 review for the four components are presented as follows: APPENDIX I - Cables; APPENDIX II - Containment and Basemat; APPENDIX III - Reactor Coolant Pumps; APPENDIX IV - Valves.

1.3 Scope of Regulatory Instruments Reviewed

The terms **regulatory instruments** and **instruments** are used throughout this report. The term "instrument" refers to a type of published document, such as the Code of Federal Regulations (CFR).^(b) Historically, the regulatory instruments applicable to

(a) Pacific Northwest Laboratory is operated for the U.S. Department of Energy by Battelle Memorial Institute under contract DE-AC06-76RLO 1830.

(b) These "instruments" are not to be confused with instruments of measurement such as pressure gauges or flow meters.

this review have served as the basis for the design, construction, start-up, and operation of nuclear power plants (NPPs).^(a)

Nine types of regulatory instruments were selected for the review:

- Code of Federal Regulations (CFR)
- Technical Specifications (TS)
- Standard Review Plan (SRP)
- NRC Regulatory Guides (RG)
- American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Sections III and XI
- Generic Safety Issues (GSI)
- American Nuclear Society (ANS) Standards

- Institute of Electrical and Electronics Engineers, Inc. (IEEE) Standards

- American Concrete Institute (ACI) Standards

Except for the ACI standards these are the same regulatory instruments used in the Stage 1 review. The ACI standards have been added to this review because of the concrete structures in the containment and basemat. Generally, only those regulatory instruments in the list above that have direct application to a component will be reviewed for aging management content; for example, IEEE standards were reviewed for references to the aging of cables and instrumentation, whereas those same standards were not reviewed for references to aging of concrete structures. We acknowledge that certain of the codes and standards are not endorsed by the NRC. A brief summary of the contents of each regulatory instrument can be found in APPENDIX V. APPENDIX VI provides a glossary of terms related to aging issues.

(a) No attempt was made to take into account that approximately 50% of the NPPs now operating were licensed before 1975. Therefore, not all of the plants were licensed under the same set or edition of instruments noted in this report.

2 Review Methodology^(a)

Stage 2 of the Regulatory Instrument Review was based on the technical concerns for aging phenomena found in the following safety-related components: cables, containment and basemat, RCPs, and MOVs. The review method consists of an evaluation of the requirements found in the regulatory instruments to determine whether or not they relate to a specific component aging issue. For example, how and to what extent does the CFR address the aging issue of fatigue and vibration of NPP cables? From this initial question, a judgment was made about whether the regulatory instrument specifically addresses aging and life extension issues.

For the issue of cable corrosion, for example, each applicable regulatory instrument, such as the IEEE Standards, was reviewed to identify the extent of the guidance that would provide the basis to mitigate and manage age-related corrosion. In addition, relevant life extension considerations were identified to determine whether they addressed the effects of corrosion. Further, if determinable, suggestions were made on the aging needs and life extension needs of the regulatory instruments.

Current initiatives are noted and explanatory and additional technical information is provided in the comment columns of the APPENDICES. Results and specifics of the review are provided in tabular form in APPENDICES I, II, III, and IV. The APPENDIX Columns 4 through 8 discuss the following review topics:

Aging Features: What parts of the regulatory instrument, if any, identify the management of

component aging? Are aging issues addressed by the regulatory instrument? Is the information implied or explicitly expressed?

Life Extension Features: Does the regulatory instrument address life extension or component replacement needs?

Current Initiatives: Is work currently underway to update or change the regulatory instrument to include aging features or life extension features? What, if any, is the status of special working groups, of technical committees, or of on-going research that could lead to the revision of the regulatory instruments to include these features?

Aging Needs: What changes or revisions generally are needed in the regulatory instrument to address aging issues?

Life Extension Needs: Does the regulatory instrument require further investigation/research of the aging issue to accommodate life extension? How could the regulatory instrument be used or modified to meet life extension needs?

Addressing the subjects in the columns, such as aging features and aging needs, shows how the aging management status of the selected regulatory instruments relates to the various selected component and the aging issues.

(a) This review methodology, which is similar to that used in the Stage 1 review, is described in more detail in NUREG/CR-5490.

3 Discussion of Component Aging Issues and Aging Management

The regulatory instruments frequently provide guidance relevant to aging management. However, such guidance is generally implied and not expressed explicitly. Also, the regulatory instruments typically focus on the safety considerations of design, construction, and initial start-up rather than the effects and processes of long-term aging. However, aging management and safety management have similar goals and require similar engineering principles for successful NPP operation and maintenance. Extensive analytical and experimental studies, such as those coordinated by the NPAR program, have been directed toward characterizing the aging process of selected systems, structures, and components (SSCs) (Copeland 1988; Hoopingarner and Zaloudek 1989; Johnson et al. 1991). These efforts have identified practices and mechanisms that promote age-related degradation, such as harsh testing and environmentally induced vibration. The relationships among these aging mechanisms vary widely; a very close similarity therefore exists among some mechanisms, whereas little or no relationship exists among other mechanisms except that they are age-related. Additionally, the mechanisms vary between being component-specific, such as RPV embrittlement, to one that can affect virtually all components, such as corrosion. In general, controlling the effects of environmental factors is a predominant consideration for the components and the aging issues chosen for Stage 2 of the review.

3.1 Cables

Cables have been identified (NRC 1987) as a major safety-related component. Cables are vitally important to plant safety instrumentation and to safety-related components because they provide the power supply and sensory carriers for these devices. They are used extensively in power plants and are found in every electric circuit in all areas of the plant and its systems. They have also been identified

by the IEEE's Nuclear Power Engineering Committee^(a) (NPEC) as sample Class IE equipment for study of plant life extension (PLEX), especially as related to the adequacy of the existing IEEE standards. For the purpose of this review, the cables referred to are those used in power, control, or regulatory instrumentation systems. Connectors were not included in the review unless the connection was shown to be involved with cable degradation. A recently published NPAR report notes the following:

"Cables, connections, and electrical penetration assemblies (EPAs) are highly reliable devices under normal plant operating conditions, with no evidence of significant failure rate increases with aging. Consequently, they receive little or no preventative maintenance. Under accident conditions, however, the reliability of these components is relatively unknown" (Jacobus 1990).

This report also states that "...the most safety significant aging effects are those that have potential to lead to common cause failures during accident conditions." The report concluded that "The major causes of cable functional failures under accident conditions result from high temperature and/or moisture penetration."

Cables are believed to age as the result of exposure to elevated temperatures and low-level radiation. The effects of environmental conditions are, therefore, a predominant factor in cable aging issues addressed in this report. The issues related to cables and the applicable aging management found in this review are discussed below.

3.1.1 Aging Issues -- Cables^(b)

Cases of premature cable aging have occurred in selected plants where abnormally high temperatures

(a) Working Group 3.4 of IEEE Subcommittee 3, *Interim Report on Plant Life Extension*, for submission to IEEE/NPEC, September 1988.

(b) A list and definition of the aging issues is found in APPENDIX VI.

and/or radiation levels have resulted in a need for cable replacement. The aging issues chosen for this report and their categorization mainly come from two sources: 1) the review of available aging studies on the components (NRC 1987; Jacobus 1990; Meyer and Edson 1989; Shah, Garner, and Conley 1988; Copeland 1988; Gillien and Clough 1988) and 2) PNL staff experience. In this report, inspection and testing is not specifically categorized as an aging issue; however, testing can lead to age-related degradation, especially if frequent and severe (Hoopingarner and Zaloudek 1989).

Eight cable aging issues were selected. The issues are listed below:

- environmental effects (biological, temperature, humidity, radiation, etc.)
- creep and shrinkage
- inspection and testing (excessively harsh testing)
- operational hazards and electrical and mechanical loads
- wear
- fatigue and vibration
- embrittlement
- corrosion.

3.1.2 Applicable Aging Management

Although aging management relative to cables is influenced by IEEE standards, the results of the Stage 1 and Stage 2 reviews are similar. For example, the IEEE Nuclear Power Engineering committee noted that "...although the existing (IEEE) NPEC standards address all aspects, they address the design and construction aspects greater than they do operation and maintenance aspects of a nuclear power plant."^(a) Furthermore, the working group noted that the IEEE standards "...do not have the desired coverage of the PLEX (plant life extension) issues as

sought by industry." Yet the IEEE standards, as opposed to the majority of the other regulatory instruments, do emphasize aging by defining aging within the standards and by their provisions for qualification testing requirements for NPP electrical equipment and gear. Of prime importance within the IEEE standards is qualification testing for Class IE equipment that determines the "qualified life" for a specific component. Qualified life is defined by IEEE as the period before the start of a design-basis event for which the equipment was demonstrated to meet the design requirements for the specified service conditions. None of the other regulatory instruments, such as the ASME BPVC or the SRP, allocate as much attention to qualification, qualified life, and replacement of equipment as do the IEEE standards.

3.1.2.1 Aging Management Features Found in the Regulatory Instruments for Cables

The following are ten typical applications of indirect or direct aging-related features as listed in APPENDIX I, Regulatory Instrument Review for Cables, of this report:

• All Aging Issues

1. As addressed in the CFRs

APPENDIX I, Item 4 - 10 CFR 50 § 50.73 requires that a Licensee Event Report (LER) be prepared for "its principal safety barriers" that are subject to serious degradation. These events may or may not be aging related. The comments of Column 9 acknowledge that this is an implied aging scenario and that the LER does not have to specifically address the status or effect of aging or its contribution to aging.

• Environmental Effects

2. As addressed in the CFRs

APPENDIX I, Item 18 - 10 CFR 50 § 50.49 stipulates the environmental conditions that must be addressed for the qualification of

(a) Working Group 3.4 of IEEE Subcommittee 3, *Interim Report on Plant Life Extension*, for submission to IEEE/NPEC, September 1988.

electrical equipment, including operating temperatures, pressures, humidity, chemical effects, radiation, and aging.

3. As addressed in the SRP

APPENDIX I, Item 19 - SRP 8.3.1 specifies that NPP power systems shall be capable of withstanding the environmental conditions of both normal operations and postulated accidents.

4. As addressed in the RGs

APPENDIX I, Item 22 - RG 1.131 notes that the users of IEEE 323 are expected to require service limits compatible for the service conditions of the specific NPP. The limits of IEEE 323 do not necessarily represent acceptable limits for all NPPs. The comments of Column 9 discuss the NRC staff exceptions and acceptability of the existing standard, IEEE 323, as set forth in RG 1.131.

5. As addressed in the IEEE standards

APPENDIX I, Item 28 - IEEE standard 384 stipulates that equipment must be protected from operational (environmental) hazards, e.g., pipe whip, steam releases, radiation, and from design-basis events.

• **Wear**

6. As addressed in the CFRs

APPENDIX I, Item 74 - 10 CFR 50, Appendix A, Criterion 4, requires that SSCs be designed to accommodate normal operations, testing, maintenance and off-normal conditions. The comments of Column 9 state that wear is not specifically mentioned, yet it is implied to be included in normal and off-normal operations.

7. As addressed in the IEEE standards

APPENDIX I, Item 76 - IEEE standard 383 requires that qualifying tests should include mechanical stresses for normal operation and design-basis events for cables. Column 9, Comments, notes that although wear is not

specifically mentioned, the factors that enhance wear are described, e.g., bending or flexing, vibration, and sidewall pressures. Thus, attention to wear is strongly implied but not explicitly mentioned.

• **Embrittlement (thermal and radiation)**

8. As addressed in the CFRs

APPENDIX I, Item 86 - 10 CFR 50, Appendix A, Criterion 23, requires that the protection system failure modes shall be designed to fail in a safe mode in extreme environmental conditions, including heat, cold, fire, steam, and radiation. This is an implied aging management scenario because the CFR, Criterion 23, does not discuss how the design of protection systems would or should compensate for component aging over time.

9. As addressed in the IEEE standards

APPENDIX I, Item 91 - IEEE standard 627 provides basic principles for design qualification of safety equipment used in NPPs. All local (environmental) service conditions shall be addressed, including radiation and temperature. Column 9, Comments, notes that the standard recognizes the need to consider the effects of significant aging as part of the qualifying process. "Significant" aging is defined in the standard under paragraph 4.4.1-Aging.

3.2 Containment and Basemat

The function of a containment and basemat structure is to house the nuclear steam supply system and restrict the release of radioactive material or gases to the environment during normal plant operation, and in the event of an accident to protect the public from fission products. In particular, the containment and basemat provide the shielding and barriers needed against radiation in the event of an accident. The containment provides protection to the reactor system from the effects of severe and extreme environmental conditions, such as tornadoes, floods, external accidents, and penetration by moving objects, such as aircraft. Containment structures (including the

basemat) are designed to withstand design earthquakes, loss-of-coolant accident (LOCA) pressures and temperatures, and normal operating loads and temperatures. In the rest of the discussion, basemats have been addressed as an integral part of the containment structure and foundation.

Containment structures for commercial U.S. LWRs are provided principally for two reactor designs, boiling-water reactors (BWRs) and pressurized-water reactors (PWRs). The general configurations of the containments for these reactor designs are discussed below:

- **Boiling-Water Reactors**

All operating BWRs are housed in "pressure suppression" type containments. The term "pressure suppression" comes from the steam that would be generated as a result of a LOCA; the steam is channeled to a suppression pool, where it would be condensed. This would reduce the pressure that otherwise would be manifest in the primary containment; i.e., reduce it to a moderate pressure compared with the pressures that may occur in PWR containments during similar accidents. All pressure-suppression-type primary containments consist of a drywell that contains the RPV, recirculating piping, and other associated piping, and a wetwell or suppression chamber that contains a large volume of water. In addition to helping condense the steam during a LOCA, the suppression pool provides a primary source of water for the emergency core cooling systems (ECCSs).

There are three designs, or generations, of pressure suppression type containments: Mark I, Mark II, and Mark III. Each type can be a metal, reinforced concrete, or prestressed concrete containment, depending on whether the containment is a free-standing metal vessel, or a reinforced or prestressed concrete vessel with a metal liner. Twenty-six (26) domestic BWR plants have metal containments and 10 have concrete containments.

- **Pressurized-Water Reactors**

The majority of the PWR containments in the United States are prestressed concrete containment vessels (PCCVs) with steel liners covering the inside surfaces. The major components of a PCCV are a flat reinforced concrete slab, an

upright prestressed concrete cylinder, and a prestressed concrete dome; all the internal surfaces are lined with steel plates anchored to the concrete. Reinforced concrete containment vessels (RCCVs) with steel liners are also commonly used in the PWRs. The only difference in the design of these and the PCCVs is that the cylinder and dome are not prestressed, but rely completely on the reinforcing steel and concrete for load resistance. Containments constructed as steel cylinders or steel spheres have also been used for some PWRs. The freestanding cylinder and dome (or a freestanding sphere), the bottom of which is anchored to a concrete foundation, are enclosed inside a reinforced concrete reactor building that forms the shield around the containment.

Another type of pressure suppression system for PWRs is the use of an ice-condenser containment system. This concept is based on the use of a suitable quantity of ice as a low-temperature, passive heat sink to condense the steam and thus absorb the rupture in the reactor coolant system. Only 10 reactors presently use this type of a pressure suppression system.

3.2.1 Aging Issues – Containment and Basemat

While aging issues for PWR and BWR containment designs have many similarities, each has aging issues specific to its design; therefore, aging issues related to each design will be addressed separately in the following subsections.

3.2.1.1 BWR Stressors and Aging Issues

The internal and external environments are the major stressors acting on BWR containments. Additional stressors during testing and operation include cyclic loads on the containment vessels. The internal environments in both metal and concrete containments consist of an operating pressure in the range of -1 to +2 psig and operating temperatures for the drywell and suppression pool in the range of 54-66°C (130-150°F) and 10-66°C (50-150°F), respectively. The relative humidity of 40-60% is higher than in PWR containments because of water in the suppression pool, which can increase corrosion, especially on submerged portions of the metal surfaces. Therefore, limits are set on the water quality in the pool. Additionally, Mark-I and

Mark-II containments are included and have low oxygen contents while the Mark-III design is air-filled (a more corrosive environment). The external environment comprises exposure to atmosphere involving a range of potentially corrosive conditions. The likelihood of corrosion increases, especially for Mark I and II drywells, if there is any uncoated surface and/or the compressible fill material is not removed from the gap. The fill material between the shield wall and drywell can absorb moisture and produce corrosive products; e.g., at McGuire Unit 1. Additionally, if degraded, the material can ignite easily and burn vigorously, producing toxic gases. The Mark-III containment is less likely to corrode because it is accessible to recoat if needed, and no compressible fill material is used. The major degradation mechanisms for concrete are very similar to those for PWR containments, which are discussed in Section 3.2.1.2.

The stainless steel (SS) bellows welded to the carbon steel piping and located in the high-energy piping penetrations are susceptible to galvanic corrosion. The bellows are cold rolled from seamless tubing and the heat affected zones are sensitized during welding, which may cause intergranular stress corrosion cracking (IGSCC). The galvanic corrosion can be prevented by a zinc-rich (galvanized) coating, because zinc is anodic to both carbon steel and SS and would corrode preferentially, protecting the steel. Other degradation mechanisms exclusive to BWR metal containments are corrosion caused by differential aeration (caused by the difference in the amount of dissolved oxygen near a water line) and aqueous corrosion (caused by acidic conditions).

Nine principal aging issues were examined for the BWR containment and basemat (Shah, Garner, and Conley 1988):

- corrosion - uniform, galvanic, microbiologically influenced, crevice, pitting, and corrosion caused by differential aeration
- fatigue - thermal, mechanical, environmentally assisted

- irradiation embrittlement
- mechanical wear and erosion
- concrete cracking and spalling (loss of free water)
- creep or shrinkage of concrete
- relaxation of tendons
- hydrogen embrittlement
- inspection and testing.

3.2.1.2 PWR Stressors and Aging Issues

The most likely degradation site for PCCVs is the prestressing system, which consists of tendons and anchors. The potential degradation mechanisms are time-dependent relaxation of tendon material, hydrogen embrittlement, and stress corrosion cracking (SCC) of the heat-treated anchor heads. Problems may develop with the metallic liner in PCCVs and RCCVs during a design-basis accident (DBA) and severe accidents, because liner-concrete interactions may take place at locations of major concrete cracks and distortions and introduce highly localized strains in the liner. The relevant degradation modes affecting the concrete material properties are deterioration^(a) resulting from aggressive environments and internal chemical reactions.

Effects of radiation and nuclear (thermal) heating on the physical properties of concrete are phenomena applicable to structures that house nuclear reactors. However, in the initial selection of the concrete constituents, shielding design considerations (more water as an effective neutron moderator and attenuator) must be balanced with structural considerations (minimum water content). The most significant degradation of concrete results from the heat of reactor operations, which causes evaporation of the free water in the concrete. However, the amount of water loss is not expected to be substantial, even during a 40-year operating life (Hungerford 1975).

(a) ASME BVPC, Division 2, Subsections CB, concrete reactor vessels and Subsection CC, concrete containments, both state in Articles CB-2140 and CC-2140 that consideration of deterioration of material caused by service is outside the scope of the code.

In comparison, direct radiation damage to concrete is an even less serious concern.

Concrete and steel can both fail by fatigue. The effects of fatigue on concrete are more pronounced if the stress reversals alternately place the concrete in tension and compression. Excessive vibration can lead to cracking of the concrete.

Unlike concrete, the major degradation mechanism for steel is corrosion. Elevated temperatures and radiation can also degrade steel. Corrosion in prestressed system components may be either localized or uniform. The most typical incidence of corrosion is pitting on the tendon wires or strands. Loss of prestressing force in PCCs due to creep and shrinkage of the concrete and relaxation of the tendon material may become more important with increasing age.

Five principal aging issues were chosen for PWR containments (Shah, Garner, and Conley 1988):

- hydrogen embrittlement
- corrosion - uniform crevice, pitting, and microbiological influences
- concrete cracking and spalling
- fatigue - thermal, mechanical, environmentally assisted
- inspection and testing.

3.2.2 Applicable Aging Management

The following subsections discuss a few examples of the aging management features, implied and explicit, as found during the regulatory instrument review. The CFR and ASME BPVC are the principal regulatory instruments that impact the design and operation of reactor containments. These examples represent some typical findings of the review, the details of which are given in APPENDIX II.

3.2.2.1 BWR Containment and Basemat Aging Management Features

The BWR instrument review, as presented in APPENDIX II, is further delineated to address

1) BWR containment *concrete structural structure* components and 2) BWR containment *steel structure* components. The following six regulatory instruments provide direct and indirect guidance for managing the aging of concrete and steel structures:

• Corrosion of BWR Concrete Containments

1. As addressed in the CFRs

APPENDIX II, Item 1 - 10 CFR 50 (Appendix A) Criterion 50, 51, 52, 53, 54 and 55 (Section V, Reactor Containment) provides the general design criteria (GDC) for reactor containment. Criterion 53 requires that the design shall provide for appropriate provisions for periodic inspection and testing for all areas of the containment. Corrosion is not specifically mentioned as an aging problem; however, it is implied that inspections would include monitoring for corrosion and other aging problems. Column 7 indicates that a method for corrosion inspection of concrete reinforcing steel is needed.

2. As addressed in the ASME BPVC

APPENDIX II, Item 9 - ASME BPVC Section III, Division 2 provides material, design, construction, and examination requirements for concrete containment structures.

Application to corrosion is implied. Column 9 indicates that the Code does not cover long-term deterioration. Article CC-2140 for Concrete Containments indicates that deterioration of material *caused by the service conditions* is outside the scope of subsections CB and CC of Division 2. Furthermore, the article indicates that the *owner* is responsible for selecting materials suitable for the conditions stated in the Design Specifications and considering the effects of the service conditions on the *properties of the material*.

3. As addressed in the GSIs

APPENDIX II, Item 13 - GSI B-49 addresses the design criteria for corrosion prevention and inservice testing (IST). Column 9 indicates that the issue deals with inservice

inspection (ISI) for all types of containment and is a "licensing issue." The aging needs, Column 7, are identified as a need to evaluate long-term corrosion of reinforcing steel and steel liner and steel/water contact surfaces. The purpose of the GSI is an explicit example of aging management, especially if a solution is found to the GSI, such as improved evaluation and inspection methods or testing methods for long-term corrosion of containments.

- **Hydrogen Embrittlement of BWR Concrete Containments**

4. As addressed in the ACI standards

APPENDIX II, Item 24 - Regulatory Guide 1.35 requires that a surveillance program and visual inspection of the tendon be established. Filler grease analysis is also required. Analysis of mixing water is required in ACI 359-89.

- **Corrosion of BWR Metal Containments**

5. As addressed in the CFRs

APPENDIX II, Item 30 - 10 CFR 50 (Appendix J) describes the leakage testing requirements for containment including leakage tests A, B and C. As noted in Column 9, Comments, aging management is implied. Corrosion is not specifically mentioned, although accessible interior and exterior inspections are required before initial or periodic Type A testing is conducted to search "...for evidence of structural deterioration." Furthermore, if there is evidence of deterioration, it shall be corrected before the testing is performed. Thus, Appendix J specifies corrective measures for containment structures that could include repairs to structures damaged by corrosion.

- **Fatigue of BWR Metal Containments**

6. As addressed in the CFRs

APPENDIX II, Item 34 - 10 CFR 50 (Appendix A, Criterion 51) requires that the reactor containment design shall consider containment boundary materials for all service conditions.

The criterion specifically mentions the uncertainties of determining *material properties* and transient stresses. Thus, attention given to fatigue life and fatigue of materials is implied but not explicitly required. The life extension needs, Column 8, note that the fatigue-design basis for the containment needs to account for the significance of continued cyclic loading beyond established life.

3.2.2.2 PWR Aging Management Features

The PWR aging management features given in APPENDIX II, similar to those for BWRs, are categorized for 1) *concrete structure* components and 2) *steel structure* components. The following four regulatory instruments provide direct and indirect guidance for managing aging of PWR containment and basemat components:

- **Corrosion of PWR Concrete Containments**

1. As addressed in the CFRs

APPENDIX II, Item 57 - 10 CFR 50 (Appendix A, GDC) addresses fracture prevention, testing and inspection, and containment leak testing. The inspection and testing requirement, while not specific to all aging concerns, provides implied aging management. One of the principal concerns is the reinforcing steel in the concrete base slab, walls, and dome; Column 8 notes the need for NDE for extent of corrosion of reinforcing steel.

- **Cracking and Spalling of PWR Concrete Containments**

2. As addressed in the ACI Standards

APPENDIX II, Item 65 - ACI Standard 359-89 (CC-3535) addresses the issue of concrete crack control, focusing on reinforcement of areas to control surface and membrane cracking.

- **Corrosion of PWR Metal Containments**

3. As addressed in the GSIs

APPENDIX II, Item 83 - GSI B-49 provides design criteria for containment corrosion

prevention and IST. (This GSI is applicable to all types of containment.) The GSI deals with the long-term evaluation of reinforcements and steel liner and steel surface corrosion in contact with water. The containment ISI requirements are addressed as a "licensing issue."

- **Inspection and Testing of PWR Metal Containments**

4. APPENDIX II, Item 92 - ASME BPVC (Section XI, Subsection IWE) provides for visual inspection of accessible containment shell, vents, flanges, and penetrations. These inspections provide aging management through the detection of age-related degradation. Subsection IWE is currently under consideration by the NRC to require use of the IWE requirements in the CFRs.

3.3 Reactor Coolant (or Recirculating) Pumps

The RCPs perform the important function of circulating coolant through the reactor core and subsequently to a turbine or intermediate heat exchanger. The RCPs are the only rotating heavy machines on the nuclear side of a plant and are critical pressure boundary components. Because the RCP is part of the pressure boundary, the pump casing has been designated as a key component for LR. Other pump parts (impeller, motor, flywheel, closure fasteners, shafts, etc.) are reviewed from a maintenance and logistics viewpoint because they affect the operability of the pump. However, these parts are not of critical concern for the pressure boundary.

In general, the RCPs used in NPPs combine metallic and nonmetallic components such as gaskets, seals, bearings, lubricants, and diaphragms. These components are required to maintain either an active function, such as rotation, or a passive function, such as prevention of pressure boundary leakage. Difficulties encountered in procurement time and replacement of a pump body in a reactor coolant system are also important LR factors. The major pump body degradation issues are fatigue and thermal aging. Fatigue damage is small, and the actual service life

of an RCP body can most likely be extended well beyond the 40-year license life without replacement of the pump body. However, each individual utility must assess the quality, design, operating, and maintenance records to ensure that their pumps can operate effectively throughout PLEX. Similarly, the concern for thermal aging appears to be small, but consideration might be given to monitoring the degree of thermal aging embrittlement of SS casings.

The most likely failure mode for a pump casing would be through-wall leakage of primary coolant water. If thermal embrittlement (long-term aging) becomes a problem, unstable ductile tearing of the pump body during design operational transients would be a potential failure mode. The RCP body fatigue life is usually very conservative and is not considered to be a limiting factor for LR.

Failures of RCP shafts have been observed due to both high-cycle mechanical and thermal fatigue damage (Shaw and McDonald 1993). However, failure of an RCP shaft, while resulting in shutdown of the RCP and a forced outage, will not compromise the integrity of the pressure boundary. It is recommended that inspections should include surface and volumetric examinations of the shaft; however, ASME Section XI does not require these examinations. Results from using conventional ultrasonic inspection techniques have been inconclusive and potentially misleading.

3.3.1 Aging Issues -- Reactor Coolant Pumps

Extensive analytical and experimental efforts are being conducted to characterize the aging processes of different components within NPPs. These efforts have identified many aging mechanisms. The relationship among these mechanisms varies widely; among some mechanisms a close similarity exists while for others little or no relationship exists other than that they are age-related. To further complicate matters, the mechanisms vary between being component-specific to mechanisms that can affect virtually all components.

Vibration

The aging mechanisms grouped within this category apply to the fatigue category and exhibit the

same failure mode. The differences are associated with the initiating event. The following is a brief description of the mechanisms:

High and Low Cycles Vibrational loads imparted to rotating equipment (pumps), supports, and other components.

Pressure Cycles Loads imparted due to periodic pressure excursions (both high and low) during normal operation.

Excessive and unexpected vibrational loads beyond those for which the components were designed were considered in this review. The techniques of IST and inspection are adequate for record-keeping and maintenance in the short-term, but the long-term implications of some of the routine measures is extremely difficult to quantify.

Fatigue

Fatigue can occur whenever there is rotating equipment or a cyclic occurrence of service conditions that can put periodic loads on either the system or component of interest. The following mechanisms are generally grouped within this category:

Thermal Cycles Loads imparted to components due to start-up and shutdown events.

Pressure Cycles Loads imparted due to periodic pressure excursions (both high and low) during normal operation.

A variation of temperature without any change in pressure may also promote fatigue. For example, in addition to the primary system, ECCSs enter this category because the testing requirement introduces a cyclic pressure stress on the major components.

Visual inspections alone, or included with non-destructive examination (NDE), are of limited value in detecting fatigue damage. Fatigue progresses in a component with little or no outward evidence. It is

only when the process is at or near the terminal phase that physical conditions, such as minor cracking, become evident. Therefore, a fatigue analysis should be conducted when fatigue-sensitive components are evaluated. However, there is one possible external monitoring technique that could be an alternative to analysis and may provide some insight into the fatigue susceptibility of specific components. Vibration sensors placed on rotating components, such as pumps and motors, may provide signatures that can detect very early changes or damage in components due to fatigue.^(a)

Corrosion

The aging mechanisms considered here are those that alter the internal surfaces of flowing systems by the buildup of deposits and scale. Specific mechanisms are:

General Corrosion/Deposition Locations for deposition generally occur in low flow regions. (General corrosion occurs over a wide range of flows.)

Crevice Corrosion Can occur under both high and low temperature conditions. Normally most evident in low velocity regions, weld areas, sleeved regions, and welds with backing rings.

Corrosion is quite similar to erosion with respect to detection, and the problems discussed in the following sections are equally applicable. However, there are additional considerations that are unique to corrosion. Monitoring of the key parameter believed to be the principal cause of corrosion may be an important equivalent to an internal inspection. For example, periodic water chemistry analysis (for oxygen concentration, acidity, presence of chlorides, sulfates, bacteria, etc.) and control may be effective in reducing the rate of corrosion buildup and therefore may reduce the frequency of needed internal inspections.

(a) Vibration monitoring is also useful to predict other phenomena such as fretting and wear.

Erosion/Wear

The aging mechanisms associated with erosion and wear are obviously significant in light of the serious failures that occurred in the past and are anticipated in the future. The mechanisms identified to date are:

<i>Water Erosion</i>	High velocity water impinging on internal pump parts causing local wear and thinning of surfaces.
<i>Erosion/Corrosion</i>	Attack on a metal surface resulting from the combined effects of erosion and corrosion.

Erosion and wear are of special concern in rotating surfaces where tolerances can affect the performance of the component. Because erosion and wear are both internal mechanisms, internal inspections are the best approach to measure their degree of damage. Also, external surface inspections would be considered effective only if the results of erosion (and wear) have become sufficiently severe to develop external leakage or cause substantial degradation of performance.

Performance testing of the pump may be effective in monitoring the effects of severe erosion and wear. However, the effectiveness of the particular test would have to be evaluated.

Surveillance

The specific aging effects that are considered in this category include equipment degradation that may occur during ISI, IST, and other regularly prescribed operation and maintenance procedures. As stated earlier, there are wide variations in aging impacts for this category, but long-term effects can be developed with repeated testing, particularly if testing is frequent and severe; e.g., fast-start testing of emergency diesel generators (EDGs) (Hoopingarner and Zaloudek 1989).

3.3.2 Applicable Aging Management

The following subsections discuss a few examples of the applicable aging management features,

principally implied, that emerged from the regulatory instrument review. These examples only represent some typical findings of the review, the details of which are given in APPENDIX III. The CFR and the ASME codes provide the primary impact on the design and operation of RCPs. The review was conducted with emphasis on the aging issues related to the coolant loop pressure boundary.

Aging Features Addressed in the Regulatory Instruments for RCPs

The following are typical examples of indirect or direct aging-related features reviewed in APPENDIX III, the Regulatory Instrument Review for RCPs:

• All Aging Issues

1. As addressed in the RGs

APPENDIX III, Item 7 RG 1.29 specifically mentions that cooling water and seal water systems required for reactor coolant system components, such as the RCPs, shall be designed to withstand the effects of a safe shutdown earthquake (SSE). The comments in Column 9, however, acknowledge that there is no specific mention of aging in this RG but only an implied relationship to the long-term reliability of NPP components.

Items (12) and (13) of APPENDIX III indicate that RGs 1.84 and 1.85 provide a list of ASME Code cases applicable to design and materials and testing acceptable to NRC staff. The comments in Column 9 acknowledge that most Code cases are eventually superseded by revisions to the Code and then annulled by action of the ASME. The Code case procedure provides aging management by offering alternative methods of design and repair not found in current codes.

• Vibration

2. As addressed in the SRP

APPENDIX III, Item 3 - SRP 3.9.3 specifies that the component design must meet 10 CFR 50 and the GDC as it relates to quality standards and effects of normal or

accident conditions. The comments in Column 9 further indicate that components that are part of the reactor coolant pressure boundary must meet the requirements for Class 1 components in ASME Section III.

- **Fatigue**

3. As addressed in the CFRs

APPENDIX III, Item 27 - Although 10 CFR 50 and the associated GDC do not specifically address the aging issue, the design margin provided by them may support PLEX beyond the normal 40-year interval. The aging and life extension needs are to address component deterioration over time.

4. As addressed in the GSIs

APPENDIX III, Item 33 - Monitoring of fatigue transient limits in reactor coolant systems is still under consideration.

- **Corrosion**

5. As addressed in the ASME BPVC

APPENDIX III, Item 24 - The subsections NB and NC provide for design loads affecting the strength and integrity of the pressure boundary. Under Initiatives (Column 6), ASME through the Pressure Vessel Research Committee (PVRC) is assessing the changes needed in Sections III and XI for the treatment of corrosion.

6. As addressed in the ANS standards

APPENDIX III, Item 25 - ANS standard 59.1 provides for sizing and installation criteria affecting the strength and integrity of the pressure boundary. The comments in Column 9 question if the existing design meets the currently acceptable aging needs of the plant and the performance specifications for PLEX.

- **Wear and Erosion**

7. As addressed in the RGs

APPENDIX III, Item 35 - The draft Design Guide 1008 discusses the NRC regulatory position on the treatment of RCP seal failures. The comments in Column 9 indicate that a seal failure may be classified as a LOCA when the seal leakage exceeds the capacity of normal makeup systems, and that this could present serious consequences in plant operations.

- **Surveillance**

8. As addressed in the ASME Codes

APPENDIX III, Item 42 - ASME OM Code-1990 requires detection of changes in the hydraulic and mechanical condition of a pump relative to a reference set of parameters established previously. Also in the Life Extension Features, Column 5, the Code requires that when a reference or set of values may have been affected by repair, replacement, or routine servicing of a pump, a new reference value or set of values shall be determined or the previous value reconfirmed by an IST run before declaring the pump operable. The comments in Column 9 contain the details of the IST requirements for LWR pumps.

3.4 Motor-Operated Valves

MOVs are used extensively within PWR and BWR NPPs for service in safety-related and BOP systems. Sizes of MOVs vary depending on service requirements and range between 5.1 cm (2 in.) and 1.5 m (60 in.) (nominal pipe diameter). The most commonly used types are gate, globe, and butterfly valves. The valve operators can be either ac or dc motor-powered, geared-type valve actuation devices. The majority of MOVs in domestic NPPs utilize operators manufactured by Limitorque Corporation

(Greenstreet, Murphy, and Gissenberg 1985). The valve motor operator is defined as all components within the motor-operator housing -- the gearbox assembly, motor assembly, and switch assembly. The valve itself, on the other hand, is defined as the body assembly, internals, and seals. MOVs used in nuclear plant safety systems are located both inside and outside containment. Under normal conditions, valves inside containment will be exposed to the same or slightly more severe external stressors than those outside containment. Under accident conditions, however, the external environmental stressors inside containment will (depending on location and type of accident) be much more severe.

Failures of valve components that are not caused by the motor operator are categorized as valve failures. Failures of components outside the equipment boundaries are not considered to be MOV failures even though the valve function may have been impaired. Specific information needed for MOV failure characterization includes: 1) failure modes, failure causes, and mechanisms; 2) frequencies of failures; 3) methods of failure detection -- incipient, degraded, and catastrophic; 4) maintenance actions; and 5) modifications resulting from failures. Each of these items serves to build a failure "signature," which when taken in total can provide a comprehensive assessment of the component failure.

The primary reported failure mode of MOVs is failure to change position when commanded. The predominant cause of these failures is generally torque- or limit-switch misadjustment or failure, with motor failures as the second most common reason. An exception to these failure modes was noted in a 1989 service water system NPAR study (Jarrell et al. 1989). This study concluded that although the available literature indicated valve torque switches were a prime suspect in failures, more detailed investigation contradicted this conclusion. Utility personnel accounts and maintenance records indicated that biologic and inorganic accumulations and corrosive attack on service water component surfaces were, in fact, the primary degradation mechanisms. Nevertheless, the root causes of premature aging failures are complex and interrelated -- improper maintenance and troubleshooting, misapplication, inadequate engineering data, improper packing adjustment, too frequent operation, improper manual operation, or

combinations of all of these. No reported cases of an operator failing from only age-related degradation were reported.

Valve failures attributable to aging effects were compiled in a report from Oak Ridge National Laboratory (ORNL) (Murphy et al. 1984). On a component basis, valves make up 20% of the total reported failed components, and over one-third were containment isolation valves. Most of the reported events for these types of valves resulted from failures to pass the 10 CFR 50, Appendix J leakage tests required during periodic surveillance testing. In most cases, foreign material or wear on the valve seat was identified as the cause of the leakage. Other types of valves, such as check, control, and drain valves, account for slightly over half of all failed valves. Most failures involved internal leakage and packing leakage.

Further NPAR aging studies by ORNL (Greenstreet, Murphy, and Eissenberg 1985) identified six main MOV failure modes, namely, 1) failure to open, 2) failure to close, 3) failure to operate as required, 4) plugging (failure to remain open), 5) internal leakage, and 6) external leakage. Several MOV sites, including the gearbox assembly, electric motor assembly, switches, and the valve, were also identified as being susceptible to age-related degradation. MOV operations can be adversely affected not only by aging but by improper maintenance and misapplication as well. However, MOV degradation, whether a result of normal aging phenomena or of improper maintenance, may lead to MOV failure if not detected and remedied. Therefore, effective surveillance and monitoring actions should reduce the total number of MOV failures from all causes and consequently improve plant operational safety (Haynes 1989).

The purpose of a motor operator is to permit remote positioning of a valve; failure detection should occur whenever 1) the valve fails to respond to an open/close command during normal plant operation, or 2) the valve fails to meet the requirements of a specific test. A review of the repair actions for MOVs indicates that the tendency is to do whatever is necessary to return the operator to service rather than to determine the cause of the failure. Consequently, an adjustment of, for example, a torque switch is made to overcome increased valve operating force, with little or no effort applied

to correcting the source of increased load -- eventually leading to motor failure. Other maintenance actions include motor replacement, lubrication, or repeated cycling of the valve, after which the valve appears to operate satisfactorily.

3.4.1 Aging Issues -- Valves

Five aging issues were addressed for valves. These are discussed in the following subsections.

Vibration

Similar to the discussion for RCPs, the aging mechanisms that may be grouped into this category include high and low cycles, and pressure cycles (where loads are imparted due to periodic pressure transients) even during normal operation of the valves. An external monitoring technique of placing vibration sensors on the affected components may provide "performance signatures" that can detect early changes or damage to the components before any visual evidence is available.

Fatigue

Fatigue is a consideration whenever there is a cyclic variation of service conditions that can put periodic loads on either the system or component of interest. It is to be noted that a variation of temperature without significant change in pressure is also a candidate for fatigue consideration.

As noted for pump fatigue, visual inspections alone have limited value in assessing possible damage from fatigue, even though they are the method called for in the ASME code, in lieu of effective NDE methods. Valve fatigue also progresses in a component with little or no outward indication. It may only become apparent at or near failure when physical evidence, such as small cracks, become obvious. Therefore, fatigue analysis is a required element in the overall evaluation of fatigue-sensitive components.

Corrosion

The aging mechanisms considered in this category are those that cause alterations of the internal

surfaces of flowing systems by the buildup of deposits and scale. Specific mechanisms are:

General Corrosion/Deposition Locations for deposition generally occur in low flow regions. (General corrosion occurs over a wide range of flows.)

Crevice Corrosion Can occur in both high and low temperature conditions. Normally most evident in low velocity regions, weld vicinity, sleeved regions, and welds with backing rings.

Corrosion is similar to erosion with respect to detection, and the problems discussed in the following category are equally applicable. However, there are additional considerations that are unique to corrosion. Monitoring of the key parameter believed to be the principal cause of corrosion may be an important equivalent to an internal inspection. Similarly, performance monitoring may also reduce the frequency of internal inspections.

Two other noteworthy aging effects within this category are IGSCC of SS piping and vessel welds and external SCC, both at high and low temperatures, of components near valves. The early onset of IGSCC can be detected by the detailed volumetric inspection of weld areas for small surface cracks or leaks. The external SCC can be detected through a combination of visual, surface, or volumetric inspections and component functional testing. Periodic water chemistry analysis (for oxygen concentration and water-borne impurities) may also be effective in reducing corrosion, possibly resulting in a reduction in the frequency of internal inspections.

Erosion/Wear

The aging mechanisms associated with erosion and wear are obviously significant, based on serious failures that have occurred (and are anticipated in the future). The mechanisms identified to date are:

Steam Erosion High velocity steam impinging on internal parts causing local wear and thinning of surfaces.

<i>Wet steam</i>	Wet steam impingement causing surface pitting.
<i>Water erosion</i>	Similar to steam erosion, above.
<i>Erosion/Corrosion</i>	A loss of the protective oxides by erosion allowing corrosion of the underlying metal to proceed.

Because erosion and wear both frequently occur on internal surfaces, internal inspections are the best approach to measure their degree of damage. Also, external surface inspections would be considered effective only if the results of erosion (and wear) have become sufficiently severe to develop external leakage or cause substantial degradation of performance. Therefore, an external detection would be too close to failure. Performance testing of the system could be effective in monitoring this aging effect. For example, torque measurements (from surveillance and maintenance records) will provide a relatively accurate measure of the valve wear.

Surveillance

The specific aging mechanisms that are considered to belong to this category include ISI, IST, and other regularly prescribed operation and maintenance procedures. As stated earlier, there is still some controversy regarding interpretation of age-related degradation caused by surveillance; however, there are cases (e.g., EDG fast-start testing and pump testing) that cause identifiable age-related wear of some components (e.g., see Hoopingarner and Zaloudek 1989).

3.4.2 Applicable Aging Management

The following subsections discuss only a few examples of the applicable aging management features, principally implied, as found in the instrument review. The examples shown are typical of the aging features found in the review, which are shown in APPENDIX IV of this report. With the exception of IEEE Standard 382-1980, "Standard Qualification of Safety Related Valve Actuators," the review was

conducted with emphasis on the aging issues related to the coolant loop pressure boundary.

Aging Features Found in the Regulatory Instruments for Valves

The following are typical examples of applications of direct or indirect aging-related directives found in APPENDIX IV, Regulatory Instrument Review for Valves, of this report.

• All Aging Issues

1. As addressed in the RGs

APPENDIX IV, Item 7 - RG 1.8 Rev. 2 provides for NRC guidance on qualification and training of NPP staff that could have an impact on the operation of pumps or valves. The comments of Column 9 indicate that the RG references ANSI/ANS 3.1 1981 "Selection, Qualification and Training of Personnel for NPPs" and provides exceptions or acceptable position to ANS 3.1.

• Vibration

2. As addressed in the SRPs

APPENDIX IV, Item 2 - SRP 3.9.3 and the Appendix A require the design to meet 10 CFR 50 and the GDC in its Appendix A as it relates to components satisfying quality standards, and effects of normal or accident conditions. The Current Initiatives in Column 6 indicate that a draft SRP is being prepared specifically for LR applications, NUREG-1597.

• Fatigue

3. As addressed in the GSIs

APPENDIX IV, Item 27 - GSI C.11 addresses reduced fatigue failures in valves resulting from design error. The comments in Column 9 further indicate that the assessment of failure and reliability of pumps and valves have been resolved.

- **Corrosion**

- 4. As addressed in the RGs

APPENDIX IV, Item 20 - RG 1.37 provides a regulatory position for on-site cleaning of materials and components relating to cleaning of fluid systems. The comments in Column 9 offer examples of nonacceptable chemical compounds, such as those containing chlorides, fluorides, lead, zinc, copper, sulfur, or mercury where such elements are leachable or could be released by breakdown.

- **Wear and Erosion**

- 5. As addressed in the ASME Standards

APPENDIX IV, Item 31 - ASME Section III: Subsections NB and NC provide for design

(nominal) loads affecting the strength and integrity of the pressure boundary. The comments in Column 9 state that the design allowance for normal wear shall be consistent with the specified design life.

- **Surveillance**

- 6. As addressed in the ASME Standards

APPENDIX IV, Item 40 - ASME Section III: Subsections NB-5000 and NC-5000 require that surface examinations of weld joint Categories A, B, C, and D including adjacent base material, be made for at least 1.3 cm (0.5 in.) on each side of the weld. The comments in Column 9 indicate that NDE and nonintrusive examination and testing aids in extending the life of affected components and support structures.

4 Observations, Recommendations, and Conclusions

This section presents PNL's observations, conclusions, and recommendations from the final report for Stage 1 of the regulatory instrument review (Werry 1990) and the work completed in Stage 2. The conclusions and recommendations for the Stage 2 work are essentially the same as those for Stage 1.

4.1 Observations

A number of observations outside the scope of this review were developed during the analysis of the review. The observations are noted below and are presented as general and specific observations. The Stage 2 general observations for the selected components correlate well with those obtained during the Stage 1 review. The Stage 2 specific observations for the containment and basemat, however, are quite specific and primarily address issues associated with areas of the component that are inaccessible to ISI.

4.1.1 General Observations

The general observations that apply to all the components are the following:

- Evaluation of component replacement methods as an aging management procedure should be addressed in the regulatory instruments for augmentation of safety and cost effectiveness. Repair versus replace decisions can and should be made in a life extension context, to the extent that they are consistent with regulations.
- Aging management could be enhanced by improved NDE methods and inspection tools. For example, the ISI requirements for PWR containments and basemats include a tendon surveillance program, integrated leak rate tests, and visual inspection of surfaces. In addition, many of the standard NDE methods employed to carry out ISI were developed for the detection and

qualitative assessment of fabrication-related defects. However, these methods are not entirely adequate for addressing aging issues or for residual life assessment. Effective engineering models are required to implement life assessment strategies and predict the important features of material degradation.

- Studies, such as the ORNL study (Haynes 1989) to identify practical and cost-effective methods of detecting, monitoring, and assessing the severity of time-dependent degradation (aging and service wear) of MOVs in nuclear plants, should be extended to all systems and components identified in these two reports. These methods will then provide capabilities for establishing degradation trends prior to failure and developing guidance for effective inspection, surveillance, and maintenance of NPPs.
- A plant with excellent maintenance methods and procedures is likely to be a plant that is successfully managing the aging of its components and systems.

4.1.2 Specific Observations for Containment and Basemat

Specific observations for managing age-related degradation of the containment and basemat are outlined below:

- Acid attack on below-grade portions of metal containments that may be exposed to sulfate-bearing soils or acidic groundwater is a potentially significant deterioration mechanism that could be more effectively identified by improved inspection and testing procedures, providing the basis to apply mitigating actions when required.
- Interior containment coatings that mitigate corrosion of ferritic steel liners

and free-standing steel containment shells have been subject to plant-specific age-related degradation. Improved bases to inspect and maintain the coatings may be needed.

- The inability to observe the floor liner plate beneath the concrete floor slab can introduce an uncertainty about plausible corrosion effects and their influence on containment leak tightness and structural integrity.
- An enhanced inspection and surveillance program to address those areas of the containment that are inaccessible for normal inspection should be considered on a plant-specific basis. The program should also be influenced by the time remaining under the original operating license, and thus its elements should be prioritized to reflect the amount and significance of the information that can be obtained in that period. This program would supplement the integrated leak rate test conducted to the requirements of Appendix J of 10 CFR Part 50.

4.2 Recommendations and Conclusions

The major recommendation is that the existing body of regulatory instruments should be revised to more directly address aging phenomena and the associated issues, and the management of the aging process. The regulatory instruments should address aging more explicitly and thereby provide guidance to plant staff for improved plant reliability, safety, and, where appropriate, life extension. Sectors of the regulatory instruments that might be considered for more explicit recognition of aging include TS, the SRP, codes and standards that have NRC enforcement, and RGs. The efforts should be scoped to assist the staff to better define the operating and environmental conditions of the plant, providing an improved basis to mitigate effects of age-related degradation. Condition monitoring and strategic maintenance have the potential for reducing forced outages. An important anticipated near-term benefit

of aging management, life extension evaluation, and planning is improved plant reliability.

However, the difficulties with revising regulatory instruments are acknowledged, and it is recommended that a major project plan for revising them be evaluated and defined. The planning should include results and evaluations of ongoing NRC-sponsored aging research as well as industry-initiated age-related research, e.g., by the Electric Power Research Institute or individual utilities, and the development of a realistic time frame for implementation of the revisions.

Conclusions relating to the content of regulatory instruments using cables and containment and basemat, RCPs, and MOVs as reference safety-related components reveal many of the same conclusions as the previous review(s) as reported in Werry (1990). The summary of the report states as its principal conclusion that aging management does exist in the safety-related regulatory instruments; however, the guidance is largely implied. The emphasis of the regulatory instruments appears to be on initial design, construction, qualification, and start-up. Actions to address aging problems frequently develop in response to age-related degradation rather than anticipating it (Werry 1990).

The results of the Stage 2 review do not disclose a major change in the regulatory instrument aging management status. A stronger emphasis on aging management, however, does exist with respect to cables. This is due to the influence of IEEE standards and the influence that environmental equipment qualification of these standards have had on the content of the relevant regulatory instruments. Qualification of equipment for environmental hazards obviously assists in the management of aging; however, qualification before installation does not address the detection of local cable defects during normal operations. Aging management, as noted in Jacobus (1990), is concerned with random failures and how to predict and prevent age-related failures through improved surveillance and maintenance practices.

The operational failure detection mode or an in-situ component aging evaluation is also applicable to containments and basemat. Additionally, lack of age-related data and a comprehensive and standardized ISI program to identify and quantify degradation

in reinforced concrete suggests that increased focus on containment aging is essential. Integration and evaluation of available research results and information from older facilities regarding the degradation of reinforced concrete subjected to long-term exposure to elevated temperatures, radiation, and cyclic loading will support the overall development of an ISI program.

With regard to MOVs in safety-related systems of NPPs, a comprehensive aging assessment has been completed by ORNL (Haynes 1989). Several tests were carried out to investigate the capabilities of monitoring methods (especially motor current signature analysis) for detecting changes in operating conditions and MOV degradation.

5 References

- Copeland, J. F. 1988. *Environmental Effects on Components: Commentary for ASME Section III*. EPRI NP-5775, prepared by Structural Integrity Associates, Inc., San Jose, California, for the Electrical Power Research Institute, Palo Alto, California.
- Gillien, K. T., and R. L. Clough. 1988. *Time-Temperature-Dose Rate Superposition: A Methodology for Predicting Cable Degradation Under Ambient Nuclear Power Plant Aging Conditions*. SAND88-0754-UC-78, prepared by Sandia National Laboratories, Albuquerque, New Mexico.
- Greenstreet, W. L., G. A. Murphy, and D. M. Eissenberg. 1985. *Aging and Service Wear of Electric Motor-Operated Valves Used in Engineered Safety-Feature Systems of Nuclear Power Plants*. NUREG/CR-4234, Volume 1, prepared by Oak Ridge National Laboratory for the U.S. Nuclear Regulatory Commission, Washington, D.C.
- Haynes, H. D. 1989. *Aging and Service Wear of Electric Motor-Operated Valves Used in Engineered Safety-Feature Systems of Nuclear Power Plants*. NUREG/CR-4234, Volume 2, ORNL-6170/V2, prepared by Oak Ridge National Laboratory for the U.S. Nuclear Regulatory Commission, Washington, D.C.
- Hoopingarner, K. R., and F. R. Zaloudek. 1989. *Aging Mitigation and Improved Programs for Nuclear Service Diesel Generators*. NUREG/CR-5057, PNL-6397, prepared by Pacific Northwest Laboratory for the U.S. Nuclear Regulatory Commission, Washington, D.C.
- Hungerford, H. E. 1975. "Concretes, Cements, Mortars and Grouts." In *Engineering Compendium on Radiation Shielding*, New York. Springer Verlag, Inc., New York.
- Jacobus, M. J. 1990. *Aging of Cables, Connections, and Electrical Penetration Assemblies Used in Nuclear Power Plants*. NUREG/CR-5461, prepared by Sandia National Laboratories, Albuquerque, New Mexico. SAND 89-2369, Executive Summary, items a. and b., page 1 and Section 4.2.1 Cables, page 29.
- Jarrell, D. B., A. B. Johnson, Jr., P. W. Zimmerman, and M. L. Gore. 1989. *Nuclear Plant Service Water System Aging Degradation Assessment, Phase I*. NUREG/CR-5379, PNL-6560, Vol. 1, prepared by Pacific Northwest Laboratory for the U.S. Nuclear Regulatory Commission, Washington, D.C.
- Johnson, A. B., D. B. Jarrell, U. P. Sinha, and V. N. Shah. 1991. *Understanding and Managing Corrosion in Nuclear Power Plants*. NACE Paper 289-Corrosion 91, prepared by Pacific Northwest Laboratory, Richland, Washington, and Idaho National Engineering Laboratory, Idaho Falls, Idaho.
- Meyer, L. C., and J. L. Edson. 1989. *Nuclear Plant Aging Research: The IE Power System*. NUREG/CR-5181, EGG-2545, prepared by Idaho National Engineering Laboratory for the U.S. Nuclear Regulatory Commission, Washington, D.C.
- Murphy, G. A., R. B. Gallaher, M. L. Casada, and H. C. Hoy. 1984. *Survey of Operating Experience from LERs to Identify Aging Trends*. NUREG/CR-3543, prepared by Oak Ridge National Laboratory for the U.S. Nuclear Regulatory Commission, Washington, D.C.
- Nuclear Regulatory Commission (NRC). 1987. *Nuclear Plant Aging Research (NPAR) Program Plan*. NUREG-1144, Rev. 1 September 1987, Rev. 2 June 1991. Washington, D.C.

Shah, V. N., R. W. Garner, and D. A. Conley. 1988. *Residual Life Assessment of Major Light Water Reactor Components-Overview Volume 2*. NUREG/CR-4731, EGG-2469. prepared by Idaho National Engineering Laboratory for the U.S. Nuclear Regulatory Commission, Washington, D.C.

Shah, V. N., and P. E. McDonald, eds. 1993. *Aging and Life Extension of Major Light Water Reactor Components*. Elsevier Publishing, New York.

Werry, E. V. 1990. *Regulatory Instrument Review: Management of Aging of LWR Major Safety-Related Components, Volume 1*. NUREG/CR-5490, PNL-7190, prepared by Pacific Northwest Laboratory for the U.S. Nuclear Regulatory Commission, Washington, D.C.

Appendix I

Regulatory Instrument Review for Cables

APP
REGULATORY INSTR

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITI COLUMN 6
CABLES (1)	GENERAL STATEMENTS, APPLIES TO ALL AGING ISSUES	CFRs 10.CFR.50 Appendix B	Provides criteria for quality assur- ance program for design, construc- tion, and operation of NPPs. Provides requirements that applies to CSS.	Provides for docu- mentation of plants historical activi- ties including those needed for consideration for LR, e.g., design control, procure- ment, initial test- ing and inspection.	?
(2)		10-CFR-50 Appendix A Criterion 13	Stipulates that instrumentation and controls shall be provided to monitor "variables and sys- tems" over antici- pated range of all normal and off nor- mal conditions.	These provisions are required for current license and LR applications.	?
(3)		10-CFR-50 Appendix A general design criteria (GDC), § 50.49, § 50.55a.	General design cri- teria covers all principal design features for fab- rication testing construction and performance for safety-related to CSS. This section (50,55a) covers codes and standards required for con- struction permits for structures, systems and compo- nents, e.g., power systems, including cables, and safety- related protection systems.	Yes	See proposed r making, 53 FR for additional ments to GDC.4

General Notes:

1. A "?" indicates further study/investment is needed.
2. For the GSIs, "resolved" means the generic safety issue is resolved, not necessarily the aging issue.
3. For the meaning of abbreviations, acronyms, and initialisms, used throughout, see Acronyms/Abbreviations and xiii of the report.

DIX I ENT REVIEW - CABLES

IVES	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
	None	None	Appendix B defines "quality assurance" as all things planned and actions necessary to provide adequate confidence that the CSS will perform satisfactory service.
	None	None	Cables are a main constituent of instrumentation and controls. For NPP applications refer to to RG 1.97 for instrumentation requirements after an accident regarding 10 CFR 50, GDC, Criterion 13.
e 311, mend-	None	None	10 CFR 50, § 50.49 addresses equipment/cable qualification relating to preconditioning by natural or accelerated aging to its "end-of-installed-life condition. § 50.49 is referenced to the specific aging issues herein, e.g., embrittlement (see item 87).

ns on pages xi, xii.

REGULATORY INSTRUM

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIAL COLUMN 6
CABLES (4)	GENERAL STATEMENTS, Continued	10-CFR-50 § 50.73 (a)(2d(ii) and (v)	Requires a license event report for any event or condition of a NPP, including "its principal safety barriers, being seriously <u>degraded</u> " or any event or condition that alone could have prevented ful- fillment of safety function. Although this section does not specifically identify specific aging issues, an implied aging man- agement exists through the report- ing process, e.g., a reportable failure of cable insulation, may result in a failure of a safety system.	LER provides records of events some of which may be aging related.	A recent report 1990, NUREG/CR- 546/Sand 89-2366 reported, "the r of events (LER failures) as ver small relative (amount of cable typical nuclear plant." See Tabl page 35 of the r report.
(5)		SRP 7.1	Table 7-1 provides a matrix for acceptance criteria and guidelines for instrumentation and controls. References 10-CFR.50 sections and general design criteria, RGs, and Branch Technical Positions (BTPs) applicable to instruments and con- trols.	Yes, the SRP through review of the plant's SAR provides a record of plant status and yearly up-date of status.	?
(6)		8.1	Table 8-1 provides a matrix for acceptance criteria and guidelines for power systems. References 10- CFR.50, GDC Appendix A, RGs and Branch Technical Positions (BTP), with specific applicability to the required SAR sections.	Yes, the SRP through review of the plant's SAR provides a record of plant status and yearly up-date of status.	?

CONTENT REVIEW - CABLES

ITEMS	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
une nber the n a ower 5, ted	None	None	Although implied, the LER Content does not have to specifically address the status or the effect of aging or the contribution of the event may have made to the aging of the components or systems. Current LER cable failure statistics support the conclusion that cables are highly reliable devices under normal operational conditions.
	None	None	Aging management is implied and is not explicitly referenced; the SRP provides a safety review methodology. The SRP applies to all instrumentation and control systems important to safety. References are made to NRC Regulatory Guides (s) and Branch Technical Positions (BTPs) which provide "guidelines", the GDC Criterion and IEEE Standard 279, on the other hand, provide "acceptance criteria" requirements.
	None (See Comments)	See Column 6	Aging management and license renewal features are largely implied; the SRP does not discuss aging of components - the principal purpose of the SRP is to review and insure that safety system power supplies, cables and equipment will perform their safety function. [General Design Criteria (GDC), Criterion 2, 4, 5, 17, 18, and 50 are noted as the applicable 10-CFR.50 requirements.]

REGULATORY INSTRUMENT

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATI COLUMN 6
CABLES (7)	GENERAL STATEMENT FOR THE RG APPLIES TO ALL ISSUES	RG 1.53	Provides reference for criteria and standard for power, instruments and control of NPP safety systems. Principally references are 10- CFR-50 Appendix A criterion and IEEE 603-1980.	Use by NRC provides records of design, reliability, qualification and testability for a plant's con- struction permits - this should aid in the LR process.	?
(8)		RG 1.28	Provides the NRC QA program position for design and construction in addition to the spe- cifics of the ANSI and ASME Standards. (See Comments) Provides specific guidance for records, qualifi- cation of inspection and test staff and audits, e.g., for electrical/cable records lists those needed for lifetime, 3-year and 10-year periods. Provides the NRC QA Program position for NPP Operations. RG stipulates acceptance of ANS 3.2 with additional interpretation and guidance for specific section of the ANS Standard.	Provides records/ documentation for LR application, e.g., preopera- tional and start-up tests records for a lifetime on ISI and maintenance activities.	?

NT REVIEW - CABLES

VES	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
	None	None	
	None	None	References ANSI/ASME N45.2 - 1977, N46.2 - 1978, and ANSI/ASME NQA - 1983.

REGULATORY INSTRUMENT

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATI COLUMN 6
CABLES (9)	GENERAL STATEMENT, Continued	IEEE 308	Standard provides the principle design criteria and design feature for Class IE power systems: (components include cables, raceways, buses, transformers, switchgear, battery chargers and inverters). Design basis shall be provided for off normal operations that would degrade or damage the IE systems. Transient and steady state conductors must address environmental status of the cables installed in NPPs.	Provides design base feature which will support adequacy of the design for license renewal applications.	Past due for action will be reviewed by IEEE in near future (IEEE Standard Office 7/6/90)
(10)		IEEE 323	Provides qualification testing methods for class IE nuclear plant equipment. Qualified life is established for equipment with significant aging mechanisms.	Describes methods of extending the "qualified life" of equipment, e.g., re-evaluation of age conditioning, less severe natural aging conditions, improved surveillance, and maintenance programs that could extend the qualified life. Addresses on a generic basis, "Extension of Qualified Life" in Section 6.9.	Past due for action will be reviewed by IEEE in near future (IEEE Standard Office 7/6/90) IEEE internal Committee is evaluating this standard for applicability to PLEX.

NT REVIEW - CABLES

ES	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
n; y e. ice	None	None	This standard is similar in description and requirements of IEEE 603 which stipulates design criteria for safety systems for NPP, e.g., see comments for 603 found in the issue of Embrittlement (thermal and radiation).

;	See Comments from IEEE Committee.	Generally none, with the exception of initial operation cycles in some BWRs, monitoring of cable environments have shown the actual operating environments are usually much milder than those assumed for qualification. (See Sand-88-1882C report.)	Standard defines:
.	The fundamental EQ concern common cause failure due to environmental condition, the standards should also address random failures and how to predict and prevent them, i.e., increase reliability through maintenance and surveillance purposes.		<u>design life</u> . The time during which satisfactory performance can be expected for a specific set of service conditions. <u>qualified life</u> . The period of time, prior to the start of a design basis event, for which equipment was demonstrated to meet the design requirements for the specified service conditions. <u>service conditions</u> . Environmental, loading, power and signal conditions expected as a result of normal operating requirements, expected extremes (abnormal) in operating requirements, and postulated conditions appropriate for the design basis events of the station.
ce			If significant aging mechanisms are adequately addressed by surveillance/maintenance qualified life determination may not be required.
-			IEEE internal Committee reports that issues relating to LR are not applicable to this standard.

REGULATORY INSTRUMENT

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVE COLUMN 6
CABLES (11)	GENERAL STATEMENT, Continued	IEEE 383	References IEEE Standard 323 which specifies preferred testing methods. This standard provides direction for establishing qualifying tests for class IE cables.	Specifies the cable must, as installed, meet the service conditions for its installed life." Stipulates that modification to the cables must be evaluated for the necessity or non- necessity of addi- tional tests.	Past due for action will be reviewed by IEEE in near future (IEEE Standard Office 7/6/90)
(12)		ANS 3.2	Provides requirements and recommendations for administrative controls and quality assurance program to help ensure activities associated with NPP operations are carried out without undue risk to the health and safety of the public. For example, the standard stipulates that maintenance and modifications shall be performed in a manner consistent with the original design to ensure at least equivalent quality to the original design base.	Activities covered are inspection, testing, mainten- ance and repair that all impact the condition of the NPP. The LR process will be improved with adequate QA documentation for the maintenance/ repair testing etc. of the facility.	(latest standard approved 4/6/89)

REVIEW - CABLES

AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
None	None	<p>Standard defines:</p> <p><u>cable type</u>. A cable type for purposes of qualification testing shall be representative of those cables having the same materials, similar construction, and service rating, as manufactured by a given manufacturer.</p> <p><u>installed life</u>. The interval from installation to removal, during which the equipment or component thereof may be subject to design service conditions and system demands.</p> <p><u>Class IE</u>. The safety classification of the electric equipment and systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling and containment, and reactor heat removal or otherwise are essential in preventing significant release of radioactive material to the environment.</p>
None	None	<p>This standard recognizes that QA encompasses the activities associated with doing a job currently as well as documenting the satisfactory progress of the activities. Work performance is the most fundamental aspect of QA. Standard address principal QA function that impact management of aging, e.g., plant operations, human factors, maintenance and modifications, testing and inspection, procurement and records.</p>

REGULATORY INSTRUMENT

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVE COLUMN 6
CABLES (13)	GENERAL STATEMENT, Continued	ANS 51.1 or 52.1	Establishes the nuclear safety criteria and functional design criteria for structures of system and components for LWRs. Defines existing practices that are consistent with licensing requirements of NRC and the body of regulation instruments so that the plants may be designed, constructed and operational without undue risk to health and safety of the public.	Does not address LR; however, should provide assurance of the adequacy of the original design.	?
(14)	ENVIRON- MENTAL EFFECTS (biological and chemical effects, temperature, humidity, radiation, etc.)	CFRs 10-CFR-50 Appendix A Criterion 4	Requires that CSSs be designed to accommodate the effects of and be compatible with environmental conditions for normal operations, testing, maintenance and off normal events.	These provisions are required for current license and LR applications.	?
(15)		10-CFR-50 Appendix A Criterion 17	On site and off site electrical power systems shall be provided to permit functioning of CSSs important to safety.	These provisions are required for current license and should apply to LR applications.	?
(16)		10-CFR-50 Appendix A Criterion 23	Protection system failure modes shall be designed to fail in a safe state if a loss of energy occurs (e.g., electrical power) or extremes in environmental conditions exist (e.g., heat, cold, fire, steam, radiation, etc.).	These provision are required for current license and should apply to LR applications.	?

REVIEW - CABLES

AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
Contains no explicit language for aging definitions of cables. <u>Cable</u> Aging issues are not addressed. An implied reference exists through testing, inspection, maintenance, and records requirements that stipulate in-service access/monitoring of plant systems and components.	N/A	For cables the standard references the applicable IEEE Standards, e.g., 344, 323 and 308.
The criterion should include deterioration and/or degradation as an operational condition over time.	None	
None	None	The criterion stresses the necessity of redundant systems; it does not specifically address aging over time for the electrical sources.
None (See comments)	None	The criterion discussed factors that can cause aging but does not assess how the design of the protection system would or should compensate for component aging over time.

REGULATORY INSTRUMENT

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIV COLUMN 6
CABLES (17)	ENVIRONMENTAL EFFECTS, Continued	10-CFR-50 Appendix A Criterion 3	CSSs shall be designed and located to minimize the probability and effects of fires and explosions.	These provisions are required for current license and should apply to LR applications.	?
(18)		10-CFR-50 § 50.49	Stipulates what environmental conditions that should be contemplated in the qualification of electrical equipment, includ- ing, temperature and pressure, humidity, chemical effects, radiation and aging. Qualification is required for <u>all</u> electrical equipment important to safety; by definition this includes safety equipment and <u>non</u> safety equipment whose failure may prevent the function of safety-related equipment.	Requires that all replacement equip- ment must be quali- fied in accordance with § 50.49.	?
(19)		SRP 7.2 Sections II and IV	Provides acceptance criteria for reactor trip system (RTS) instrumentation. The RTS system must satisfy the require- ments of Criterion 10-CFR.50, GDC 2 and 4, specifically the system must be designed for "abnormal environ- ments."	Yes, the SRP through review of the plant's SAR provides a record of plant status and yearly up-date of status.	?

REVIEW - CABLES

AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
None	None	
"Significant" degradation (see comments) should be defined.	See Comments on NUMARC report.	Provides the basic requirements for the IEEE equipment qualification standards. Consideration must be given to all significant degradation that could effect the function of the equipment. Equipment subjected to "mild" environmental conditions are not included in the scope of § 50.49.
See Comments	See Comments and Column 6	Aging management implied through GDC Criterion 4 which states design must provide and accommodate normal operations maintenance and testing, i.e., factors that effect aging and the management of aging.

REGULATORY INSTRUMENT

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVE COLUMN 6
CABLES (20)	ENVIRONMENTAL EFFECTS, Continued	8.3.1 Section II AC Power (8.3.2 Applies to DC Power)	Provides acceptance criteria for the SRP review to ascertain compliance with General Design Criterion 4. Criterion 4 requires the NPP power systems shall be capable of withstanding environmental conditions asso- ciated with normal operation and postulated accidents.	Yes, the SRP through review of the plant's SAR provides a record of plant status and yearly up-date of status.	?
(21)		7.3 Sub-Section II	Provides Acceptance Criteria for ESF including instruments and control systems. The ESF protection systems(s) must be compatible with environmental conditions asso- ciated with normal operations, main- tenance, testing, postulated accidents including LOCAs.	Yes, the SRP through review of the plant's SAR provides a record of plant status and yearly up-date of status.	?
(22)		RG 1.131	Describes methods acceptable to NRC staff for qualification tests of electric cables, for example, notes that values of IEEE 323-74 for <u>pressure,</u> <u>temperature,</u> <u>radiation, chemical</u> <u>concentrations,</u> <u>humidity and time</u> do not represent acceptable limit for all NPPs. The users of IEEE 323 are expected to require service limits compatible for the service condition of the specific NPP.	Use of the RG and through the FSAR the qualification tests are part of the license docket and will be useful for documentation in the LR process.	?

T REVIEW - CABLES

S AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
See Comments	See Comments (see Column 6)	SRP reviews are principally concerned with safety and the plant's CSS's ability to perform their safety functions. Should the SRP address aging and life extension, other than the implied aspects now in effect, is a question that should be investigated. As to the question of LR the NRC is moving ahead with a draft LR (only) SRP.
None	None	
None	None	The RG references IEEE Standards and IPCEA Standards and the NRC exceptions to these standards. The RG also lists specific sections of the standard(s) not acceptable and provides specific acceptable methods that "shall" be used for environmental tests.

REGULATORY INSTRUMENT

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVE COLUMN 6
CABLES (23)	ENVIRONMENTAL EFFECTS, Continued	RG 1.89 Rev.1	Describes methods acceptable for the qualification of electric equipment under significant environmental stresses resulting from design base accidents. The regulatory position is given for calculating mass and energy rates for LOCAs and MSLB, effects of spray and chemical in containment, establishment of the radiation environment inside and outside containment, and environmental conditions outside containment.		?
(24)	ENVIRONMENTAL EFFECTS, Continued	RG 1.93	Addresses safest operating mode during limiting conditions for operation (LCO). Continued operation may be a safer course than a shutdown. RG regulatory position is to provide safest operating mode whenever the available power source is less than the LCO. These activities, in general, should not "risk further degradation of electric power systems or in anyway jeopardize plant safety."	None	?

REVIEW - CABLES

AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
None	None	<p>Reference IEEE 323 and 10-CFR-50, § 50.49. The RG recognizes the uncertainty of state-of-the-art preconditioning, e.g., they cannot simulate all significant types of degradation and that neutral pre-aging is difficult and costly. The regulatory positions delineated in the guide reflect the current state-of-the-art and should improve the existing IEEE standard. RG positions may change with additional research in this field.</p> <p>See RG 1.97 for reference to instrumentation for LWR after accident conditions.</p>
None	None	<p>RG is principally a safety guide for limiting conditions when LCO is not met; however, the guide may impact degradation of equipment by restricting operation if the operations or maintenance further degrades the power supplies.</p>

REGULATORY INSTRUMENT

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVE COLUMN 6
CABLES (25)	ENVIRONMENTAL EFFECTS, Continued	RG 1.131	<p>This RG provides a discussion and description of the regulatory position for "ongoing qualification procedure." Some cables may not respond to accelerated aging techniques, thus the qualified life may be <u>less than the required</u> design life. Two methods are suggested to achieve design life:</p> <p>(1) replacement of representative cables, i.e., those having received the most severe conditions of loading, mechanical stress, pressure, temperature and chemical environment, and then testing these which would qualify the remaining cables or</p> <p>(2) additional identical cable installed at conditions that are more harsh than those to be qualified. An identical cable would be removed when it reaches service life of less than the original qualified life and retested using accelerated aging. These results, if successful, could qualify the remaining cable.</p>	Provides qualification records and the replacement option for equipment.	?

REVIEW - CABLES

AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
None	None	The guide states if these methods demonstrate that the qualified life is less than design life a periodic replacement plan should be adopted. Replacement are an effective aging management tool.

REGULATORY INSTRUMENT

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVE COLUMN 6
CABLES (26)	ENVIRONMENTAL EFFECTS, Continued	IEEE 383	Describes significant envi- ronmental conditions that address aging issues, i.e., for corrosion, moisture, and chemicals. Addresses Class IE cables used for power control and instrumentation.	Applies to both normal and design base events. Pro- vides for methods of test documenta- tion for the aging issues.	Past due for action will be reviewed by IEEE in near future (IEEE Standard Of 7/6/90) IEEE internal Com- tee has determined that issues relat- ed to LR are not applicable to this standard.
(27)	NOT USED				
(28)		IEEE 384	Stipulates equipment shall be protected from operational (environmental) hazards, e.g., pipe whip, steam release, radiation, etc., and design base events.	Provides the design base which could qualify life extension for electrical equipment.	Past due for action will be reviewed by IEEE in near future (IEEE Standard Of 7/6/90)
(29)		IEEE 383	Provides Class IE qualifying testing examples specific to the aging issue for normal operation and design base events. Standard provides description of sig- nificant environmental conditions.	Provides for methods of test documentation for the aging issue. Further, the stan- dard suggests radiation exposure and temperature test rates that may provide evidence for extended useful life of the cables.	Past due for action will be reviewed by IEEE in near future (IEEE Standard Of 7/6/90) IEEE internal Com- tee has determined that issues relat- ed to LR are not applicable to this standard.

T REVIEW - CABLES

	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
S ; . ce t-	None	None (See Comments requalification)	<p>Standard defines:</p> <p><u>design basis events</u>. Postulated abnormal events used in the design to establish the performance requirements of the structures, systems, and components.</p> <p>Requalification of equipment is not applicable to this standard.</p> <p>IEEE internal Committee notes that Aging degradation mechanism of "Cable failure due to contact with water; qualification of jacket material on shield cables," are not applicable to this standard.</p>
; . ce	None	None	<p>Use of separation barriers and electrical isolation provides control over design base events that can and will accelerate the aging process. Such as fire, <u>high temperatures</u>, flooding, steam/water spray radiation, etc.</p>
; . ce t-	None	None	<p>Provides that <u>certified</u> evidence from the manufacturer may be used to show the cable is acceptable for normal operation, e.g., the cable was manufactured and tested in accordance with industrial standards of ANSI, NFPA, IPCEA, etc.</p>

REGULATORY INSTRUMENTS

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVE COLUMN 6
CABLES (30)	ENVIRONMENTAL EFFECTS, Continued	IEEE 384	Provides direction that Class IE circuits will be routed to protect them from environmental factors, e.g., radiation, elevated temperatures, flooding, fires, etc., caused by misoperation or failure of mechanical systems.	Provide a design base for license renewal applications.	Past due for action will be reviewed IEEE in near future (IEEE Standard Of 7/6/90)
(31)		IEEE 603	Standard establishes minimum functional and design requirements for safety systems for NPPs. Applicable to power instru- mentation and con- trol portions of safety systems. Systems design base shall establish min- imum criteria to address environmental condition imposed in the operator during normal, abnormal and accident conditions. The design base shall document the range of transient and steady state conditions and the environmental conditions including humidity and ambient temperature effects and the conditions that have the poten- tial for functional degradation of the safety system.		Past due for action will be reviewed IEEE in near future (IEEE Standard Of 7/6/90) IEEE internal Com- tee has determine that issues relat- ed to LR are not applicable to this standard, further Committee does not recommend that the standard be revised to address PLEX at this time, i.e., September 1988.

NT REVIEW - CABLES

VES	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
on; by re. fice	None	None	Standard is principally intended to assure operabil- ity of redundant systems for safe shutdown. Provides implied aging consideration through application of safety concerns.
on; by re. fice	None	None	Standard refers to ANS 4.1 requirements: "Design Basis Criteria for Safety Systems in Nuclear Power Generating Stations." The standard defines the "Operating Elements" of an NPP safety system, including reactor trip and engineered safety features (ESF) and the "General Elements" of an NPP safety system. Also included are the requirements for sense and command and the requirements for execute features. For example, the typical execute features are ESF breakers, motors/starters, pumps and MOVs, and solenoid valves. IEEE internal Committee indicates that "Aging Degradation mechanisms" are not discussed in this standard"; therefore, Aging management as referenced in this review is largely implied through adequate functional design requirements.

REGULATORY INSTRUMENT

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVE COLUMN 6
CABLES (32)	ENVIRONMENTAL EFFECTS, Continued	IEEE 627	Provides basic principles for design qualification of safety equipment used in NPPs over the expected range of normal, abnormal, design base event, post design base event, and in-service test conditions. Service conditions for humidity and ambient temperature effects (local environmental factors) shall be addressed.	Stipulate qualification documentation shall be maintained and recorded. These records will/would be useful in the license renewal process. Notes that periodic maintenance or by periodically changing components (see comments for components) the qualified life of the equipment may be extended.	IEEE Standards Office reports this Standard is now under balloting for reaffirmation (7/6/90).
(33)		GSIs B-9	Thermal effects on electrical cable penetrations.		?
(34)	INSPECTION AND TESTING	CFRs 10-CFR-50 § 50.55a, (h)	Requires construction permits issued after 1/1/91 that the protection systems shall meet the requirements of IEEE Standard 279. "Criteria for Protection Systems for Nuclear Power Generating Stations." (See Comments) This standard, in part, stipulates that the protection system shall be designed to permit checking the operational availability of each protection system during reactor operation.	Must meet these requirements for original license.	?

NT REVIEW - CABLES

ES	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
ice ard t	None	None	Among the equipment "Assembly Components" listed are capacitors, connectors, resistors and wire. The standard stipulates that the qualifying process shall consider significant aging effects as an essential part of the qualification process. To determine its (aging) effect on the operability of the equipment. A criteria is established in the standard that defines a "significant" aging mechanism, see paragraph 4.4.1, "Aging", page 11, STD. 627.
	None	Long-term effects of heat on sealant/insulation used in electrical cable penetrations.	Conductors within penetration module heated by resistance heating; carbon deposits build over time creating short circuit between adjacent conductors.
	None	None	Standard 279 was withdrawn by IEEE on 6-14-84. IEEE Standard 603 has been designated by IEEE as its replacement.

REGULATORY INSTRUMENT

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVE COLUMN 6
CABLES (35)	INSPECTION AND TESTING, Continued	10-CFR-50 Appendix A Criterion 21	Requires, in part, periodic testing of protection systems during reactor operation. Periodic testing is designed to provide status of operability of systems; however, such testing will also provide status and condition, e.g., degradation, of the components.	Provides records for evaluation of protection systems applicable to LR.	?
(36)		10-CFR-50 Appendix A Criterion 18	Electrical power systems, important to safety, shall be designed to permit periodic inspec- tions, including inspections of wiring, insulation and connections.	Requirements are required for cur- rent license and will be required for LR.	?
(37)		10-CFR-50 Appendix J	The appendix provides for reactor containment leak testing. Item II, G, 1. specifically discusses electrical penetrations under "Type B" tests. Except for air locks type B tests are required at refueling outages, but in no case of intervals longer than 2 years.	Tests will provide records of electri- cal penetration leaks. Tests will be required for LR.	See proposed rule: FR 39538 and 53 FR 5985, for proposed amendments to Appendix J.
(38)		10-CFR-50 § 50.49	Requires qualification testing of electric CSS important to safety.	Required for cur- rent licence and will be required for LR.	?

NT REVIEW - CABLES

IES	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
	None	None	Records shall be retained until license is terminated.
	The criteria should state that inspection shall include inspections for aging related deterioration.	None	Stipulate the inspection will verify the function and operability of the power systems with the design.
s 51 r d	None	None	Failure of the leak test requires the licensee to correct the problem which could result in repair/removal of the unacceptable part, e.g., degraded penetration.
	None	None	Provides bases for the IEEE Standards that cover qualification testing, e.g., IEEE 323 and 383.

REGULATORY INSTRUMENT

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATI COLUMN 6
CABLES (39)	INSPECTION AND TESTING, Continued (SURVEILLANCE)	TSs 3/4.3 Instrumentation (3/4.3.1, 3/4.3.2, 3/4.3.3, 3/4.3.4)	Provides specifications for limiting operating conditions and surveillance requirements to demonstrate that the instrumentation is OPERABLE. Includes the Reactor Trip System Instrumentation, Engineered Safety Features Actuation Systems, Monitoring Instruments and Turbine Overspeed Protection. Monitoring instru- mentation covers a variety of safety systems, including radiation, seismic and fire detection systems.	Records for/of Plant performance.	An ongoing NPAR t is underway to ev ate modifications standard TSs to optimize surveill testing to minimi unnecessary deterioration and maximum understan and control of ag (Task 2 FIN No. B28651)
(40)	(SURVEILLANCE)	TSs 3/4.8 Electrical Power Sources (3/4.8.1, 3/4.8.2, 3/4.8.3, 3/4.8.4)	Provides specifications for limiting operating conditions and surveillance required for operat- ing and shutdown conditions for elec- trical power systems to demonstrate the OPERABILITY of systems required for operation, standby or shutdown conditions. Includes AC sources, DC sources, On-site power distribution and electrical equipment protection devices.	Record for/of Plant performance.	An ongoing NPAR t is underway to ev ate modifications standard TSs to optimize surveill testing to minimi unnecessary deterioration and maximum understan and control of ag (Task 2 FIN No. B28651)

ENT REVIEW - CABLES

VES	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
ask alu- of ance ze ding ing.	See Column 6	See Column 6	Although the specifications do not specifically address cables, cables are included in all instrumentation as they provide the carrier for signals and instrument power. The TSs determine the limiting conditions for operations and the surveillance requirements. Both of these requirements, while not mentioning a specific aging issue, support the control of the aging issues by providing the guidelines for operation and if the surveillance/testing does not support continued operation of the instruments remedial action is required to place them in sound and safe working order within the TSs requirements.
ask alu- of ance ze ding ing.	See Column 6	See Column 6	<p>These specifications, with the exception of batteries, (see comments below) do not specifically address the principal aging issues, e.g., corrosion, material embrittlement. The specifications principally describe limiting conditions that dictate intervals allowable and the restoration of power sources. The specification force aging management by providing acceptable limits, in time, for systems to be out of service. If the systems cannot be restored and made operable within the time frame specified, repairs/maintenance, etc., are required to continue operation.</p> <p>TS 3/4.8.2 DC Power Sources provides surveillance requirements that discuss cells, cell plates and racks. The specification calls for a visual inspection at 18 month intervals that shows no indication of physical damage or abnormal deterioration of the battery racks. Further, this TS dictates that the cell-to-cell and terminal connection are clean, tight, and coated with anti-corrosion material.</p>

REGULATORY INSTRUMENTS

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVE COLUMN 6
CABLES (41)	INSPECTION AND TESTING, Continued	SRPs 7.2 Section III and IV	Provides review procedures and evaluation of the RTS for confirmation to 1.22, 1.118 and IEEE Standards 338 and 379 for periodic testing requiremen- ts.	Yes, the SRP through review of the plant's SAR provides a record of plant status and yearly up-date of status.	?
(42)		7.3 Section II	Provides acceptance criteria for Engineered Safety Features (ESF) including instruments and control systems. Acceptance criteria stipulates that the ESF protection systems shall be designed to permit periodic testing while the reactor is in operation.	Yes, the SRP through review of the plant's SAR provides a record of plant status and yearly up-date of status.	?
(43)		8.3.1 Section II AC Power (8.3.2 applies to DC Power)	Provides acceptance criteria for the testability of onsite systems in accordance with GDC Criterion 17, specific guidance of RGs 1.9, 1.32, 175 and 1.108. The SRP also provides accep- tance criteria for periodic testing in accordance with GDC, Criterion 18 and the references RG 1.118 and IEEE 338.	Yes, the SRP through review of the plant's SAR provides a record of plant status and yearly up-date of status.	?
(44)		8.3.1. Section III, Item 7	Stipulates that the SRP review includes review of electrical schematics and functional logic diagrams to assure a "built-in" capability for testing during reactor operations.	Yes, the SRP through review of the plant's SAR provides a record of plant status and yearly up-date of status.	

ENT REVIEW - CABLES

VES	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
	None	See Column 6	
	None	See Column 6	The SRP responses to GDC Criterion 21, "Protection System Reliability and Testability."
	None	See Column 6	Aging management is implied only, e.g., it assumed that if the system or equipment will not satisfactorily test to its design performance specification it will be repaired or replaced; thus the "testability" will provide aging management. The periodic testing of Criterion 18, specifically notes appropriate periodic testing of wiring, insulation, connections and switchboards.
	None	See Column 6	

REGULATORY INSTRUMENT

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATI COLUMN 6
CABLES (44a)	INSPECTION AND TESTING, Continued	RG 1.22	Provides regulatory position or design of system <u>actuation</u> <u>protection functions</u> to allow periodic testing of these devices. Does not directly address aging; however, the guide enhances aging management through its provisions to allow for testing of all protection systems which could in turn determine damage or failure of the equipment.	Provides a design base for LR.	?
(45)		RG 1.33, Rev. 3	Provides NRC QA program position for NPP operations. The RG stipulates acceptance of ANS 3.2 with additional interpretation and guidance for specific section of the ANS Standard.	Provides records and documentation.	?
(46)		RG 1.58	Provides regulatory guidance for qualification of <u>inspection</u> , examination, and testing personnel.	Indoctrination and training programs will provide <u>records/documen-</u> <u>tation</u> of adequacy of staff for plant surveillance activities.	?

NT REVIEW - CABLES

YES	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
	None	None	<p>Defines Protection System, actuation device and actuated equipment all of which may encompass some form of electric signal, power, etc.</p> <p>Testing is specified for either operational or shutdown conditions.</p>
	None	None	
	None	None	<p>Aging of components may best be determined by a well trained and experienced staff. Personnel that has been qualified to an adequate set of guidelines or standards will provide higher level of aging management.</p>

REGULATORY INSTRUMENT

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATION COLUMN 6
CABLES (47)	INSPECTION AND TESTING, Continued	RG 1.32	Describes methods to comply with 10-CFR- Part 50-Appendix A, criterion 17 and 18 with respect to design, operation and testing of safety - related electrical power systems. Standard principally address the IEEE standards 308 and provides additional require- ments and the NRC regulatory position for this standard, e.g., offsite power battery charger supply and battery performance dis- charge tests. Further the RG lists those parts of IEEE 308 that are not acceptable.	The RG has been used for evaluation of construction permits after 4/15/77, i.e., those docketed after this date. Use of this RG provides records/documenta- tion for LR applications.	?
(48)		RG 1.41	Describes regulatory position for verification of the existence of inde- pendence among redundant on site power sources.	Testing records/ documentation.	?
(49)		RG 1.118	Provides regulatory position for interpretation of IEEE Standards for periodic testing of NPP safety systems and those important to safety. Tests should be carried out as close to design as possible, e.g., full operating sequence and operations of protection systems.	Testing records/ documentation.	?

NT REVIEW - CABLES

ES	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
	None	None	RG Does not specifically address aging issues, except through implied additional testing or test intervals that would improve the reliability or availability of equipment. Exclusion of sections of IEEE 308 are referenced to other RGs, e.g., RG 1.81 and 1.03.
	None	None	Could identify aging or degradation problems.
	Guide discusses checking of operational availability, need to determine failures and the demonstration of system performance. The guide could enhance aging management by stating the periodic testing also should gather data on the condition or degradation status of the system and components.	None	The guide's principal discussion pertains to clarification and additional requirements for sections of IEEE 338-1977, e.g., defines electrical systems, limitation on by-pass conditions, independent testing of redundant components and evaluation of failure rate changes.

REGULATORY INSTRUMENT

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATION COLUMN 6
CABLES (50)	INSPECTION AND TESTING, Continued	RG 1.131 Rev.1 (Proposed)	Provides general acceptance of IEEE 383-1974 for standard qualifying tests of cables and field splice and lists exceptions to specific sections. In general the guide stipulates that more extreme testing sequence/conditions should be used, i.e., (a) for steam line breaks, (b) indicates that the environmental conditions given in IEEE 383 are not acceptable in all cases and that limits are not applicable to all NPPs, (c), takes exception to the physical aging properties and stipulates that, "Aging data shall be developed to determine long-term performance of insulation," (e) provides another guide for "radiation exposure," and (d) several exceptions are included for flame testing and/or fire tests.	Testing records/ documentation.	?

NT REVIEW - CABLES

ES	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
	None	None	

REGULATORY INSTRUMENT

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATI COLUMN 6
CABLES (51)	INSPECTION AND TESTING, Continued	IEEE 338	Standard recognizes that test intervals are subject to a number of restraints, among these are, regulatory requirements, operating cycles, plant safety, manpower, impact on radiation exposure, manufacturing specifications, failure data and <u>degradation of equipment caused by testing</u> . The standard, therefore, does recognize that testing can accelerate or cause aging; however, it does not judge the adequacy of current practice of surveillance tests. Changes in the test interval may be made to assure that the performance of the equipment is maintained.	The planned test intervals are intended to maintain the equipment in a satisfactory performance made for their qualified life, i.e., provides documentation of plant equipment status for LR.	None planned by 1 as of July 6, 199 Standard was reaffirmed in 198 An IEEE internal Committee indicates the LR is not applicable to this standard.
(52)		IEEE 308	Provides the principal design criteria and design features for Class IE power systems, and components, including <u>Cables</u> . Design base requirements include consideration for factors that could cause degradation. Periodic testing is required to detect deterioration of system/ components.	Periodic testing will provide records/documentation on status of components proposed for LR.	Past due for action will be reviewed IEEE in near future (IEEE Standards Office 7/6/90)

NT REVIEW - CABLES

ES	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
EE . . s ic-	None	Recent U.S. operating experience indicates a need for cable monitoring technique for local damage in unshielded, low-voltage cables and for techniques to determine satisfactory performance of damaged cables during accident conditions.	<p>Defines: Test interval as "The elapsed time between the initiation (or successful completion) of tests on the same sensor, channel, load group, safety systems, or other specified system or device."</p> <p>A major consideration in determining the test interval is the failure modes and their mechanism of failure and repair distribution. Historical data and engineering judgement is also considered; all of these imply aging management.</p> <p>An IEEE internal Committee reports that "equipment requalification; artificial aging techniques; detection of imminent failure of large power components; high voltage testings; spare parts availability; and technology upgrades" are not applicable to this standard.</p>
n; y e.	None	None	Aging management of system is provided through design and through testing and surveillance.

REGULATORY INSTRUMENT

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATI COLUMN 6
CABLES (53)	INSPECTION AND TESTING, Continued	IEEE 338	General: Provides for periodic testing of NPP safety systems. Scope includes functional tests. Calibration verification and time response to verify the safety systems will meet their defined safety function. One of the program objectives is to identify high failure rates, this strongly implies aging management because of a successful repeated test must be documented by evaluation of maintenance repairs and procedure changes that resulted in the successful repeated test.	The instrument requires written procedures and documentation. The surveillance program should provide records of plant status (current condition) applicable to LR requests.	None planned by I as of July 6, 199
(54)		IEEE 338	Testing is carried out under normal (see comments) operating environmental conditions. Stipulates that safety systems shall be designed to allow testing under operational or shutdown conditions.	Provides current functional testing results of the plants present, as aged, condition, i.e., provides plant equipment status for LR.	None planned by II as of July 6, 1991 Standard reaffirm in 1984.
(55)	(during construction and/or operational phase)	IEEE 383	Describe tests to indicate that Class IE cables may perform under conditions of a design base event including LOCA, fire or other service hazards (see comments).	Provides a design base and qualification for possible life extension of cable systems.	Past due for action will be reviewed I IEEE in near future (IEEE Standard Of 7/6/90)

NT REVIEW - CABLES

ES	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
EE .	None	No improvements are necessary if the surveillance program provides sufficient evidence that the equipment will perform satisfactorily.	Defines: Periodic test "A test performed at <u>Scheduled Intervals</u> to detect failure and verify operability" maintenance is not covered in the standard.
EE 1	None	None	Defines: Functional test, "A test to determine the ability of a component or system to perform an intended purpose." Abnormal testing conditions such as seismic events, high radiation, pressure or temperature and moisture conditions are covered by qualification testing. (See Comments on IEEE for item 51.)
1; ' . ce	None	None	The simulated test conditions of high temperature and radiation for aged and non-aged cables may indicate the ability of the cable types to operate under normal NPP service conditions.

REGULATORY INSTRUMENT

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3.	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATI COLUMN 6
CABLES (56)	INSPECTION AND TESTING, Continued (during construction and/or operational phase)	IEEE 336	Describes pre- installation verification for material and equip- ment. Describes protective measure during storage and physical integrity of material and equipment by visual examination for damage, corrosion, contamination, etc.	Documentation and records for condi- tion of equipment. Requires procedures and documents to be current during installation.	IEEE Standards Of reports, 7/6/90, no action is unde planning for this standard.
(57)	(during construction and/or operational phase)	IEEE 336	Describes installation testing, e.g., electrical tests, physical and chemi- cal tests and mechanical tests.	Documentation and records for condi- tion of material and equipment.	IEEE Standards Of reports, 7/6/90, no action is unde planning for this standard.
(58)	(post- installation)	IEEE 383	Describe post- installation test/inspection, including verification of correct installation requirements and that equipment and materials are not damaged during installation.	Provides documenta- tion and records of installation and condition at time of installation.	Past due for acti will be reviewed IEEE in near futu (IEEE Standard Of 7/6/90)
(59)		ANS 3.1	Provides criteria for selection, training and qualification of personnel for LWR's including those engaged in maintenance, testing and inspection of electrical components, instruments, and equipment related to cables.	Adherence to this standard will pro- vide assurance that competent staff have operated and maintained the facility.	?

NT REVIEW - CABLES

YES	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
ice hat	None	None	Requires qualified staff are to perform inspections. Requires that records of construction become part of plant records such as codes, specifications and project procedures. This standard does not address requalification of equipment, special reference is to IEEE 323 for requalification.
ice hat	None	None	Requires that records of construction become of plant records such as codes, specifications and project procedures.
n; y e. ice	None	None	Require that records of construction become part of plant records such as codes, specifications and project procedures.
	None	None	Standard provides for continuing training programs to improve and maintain acceptable performance.

REGULATORY INSTRUMENT

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATI COLUMN 6
CABLES (60)	OPERATIONAL HAZARDS AND ELECTRICAL AND MECHANICAL LOADS	CFRs 10-CFR-50 Appendix A Criterion 3	CSSs shall be designed and located to minimize the probability and effects of fires and explosions.	These provisions are required for current license and LR applications.	?
(61)		10-CFR-50 Appendix A Criterion 3	Non-combustible and heat resistant materials shall be used insofar as pos- sible throughout the NPP. While this criterion is specific to fire protection and explosives its requirements impact the design with regard to envi- ronmental condition, e.g., high temperatures.	A necessary requirement for existing license and required for LR.	?
(62)		10-CFR-50 Appendix A Criterion 17	On site and offsite electrical power systems shall be provided to permit functioning of CSSs important to safety.	These provisions are required for current license and should apply to LR applications.	?

IT REVIEW - CABLES

ES	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
	None	None	
	None	None	
	None	None	The criterion stresses the necessity of redundant systems; it does not specifically address aging over time of the electrical sources.

REGULATORY INSTRUMENT

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIAT COLUMN 6
CABLES (63)	OPERATIONAL HAZARDS, Continued	TSs 3/4.3	Provides specifications for limiting operating conditions and surveillance requirements to demonstrate that the instrumentation is OPERABLE. Includes the Reactor Trip System Instrumentation, Engineered Safety Features Actuation Systems, Monitoring Instruments and Turbine Overspeed Protection. Monitoring instrumentation covers a variety of safety systems, including radiation, seismic and fire detection systems.	Records for/of Plant performance.	An ongoing NPAR t is underway to ev ate modifications standard TSs to optimize surveill testing to minimi unnecessary deterioration and maximum understar and control of ag (Task 2 FIN No. B28651)
(64)		TSs 3/4.8	Provides specifications for limiting operating conditions and surveillance required for operating and shutdown conditions for electrical power systems to demonstrate the OPERABILITY of systems required for operation, standby or shutdown conditions. Includes AC sources, DC sources, On-site power distribution and electrical equipment protection devices.	Record for/of Plant performance.	An ongoing NPAR t is underway to ev ate modifications standard TSs to optimize surveill testing to minimi unnecessary deterioration and maximum understan and control of ag (Task 2 FIN No. B28651)

NT REVIEW - CABLES

IVES	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
ask alu- of ance ze ding ing.	See Column 6	See Column 6	<p>Although the specifications do not specifically address cables, cables are included in all instrumentation as they provide the carrier for signals and instrument power. The TSs determine the limiting conditions for operations and the surveillance requirements. Both of these requirements, while not mentioning a specific aging issue, support the control of the aging issues by providing the guidelines for operation and if the surveillance/testing does not support continued operation of the instruments remedial action is required to place them in sound and safe working order within the TSs requirements.</p>
ask alu- of ance ze ding ing.	See Column 6	See Column 6	<p>These specifications, with the exception of batteries, (see comments below) do not specifically address the principal aging issues, e.g., corrosion, material embrittlement. The specifications principally describe limiting conditions that dictate intervals allowable and the restoration of power sources. The specification force aging management by providing acceptable limits, in time, for systems to be out of service. If the systems cannot be restored and made operable within the time frame specified, repairs/maintenance, etc., are required to continue operation.</p> <p>TS 3/4.8.2 DC Power Sources provides surveillance requirements that discuss cells, cell plates and racks. The specification calls for a visual inspection at 18 month intervals that shows no indication of physical damage of abnormal deterioration of the battery racks. Further, this TS dictates that the cell-to-cell and terminal connection are clean, tight, and coated with anti-corrosion material.</p>

REGULATORY INSTRUMENT

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIAT COLUMN 6
CABLES (65)	OPERATIONAL HAZARDS, Continued	SRP 8.3.1 Section II AC Power (8.3.2 applies to DC power)	Provides acceptance criteria for electric penetration assemblies associ- ated with the design of reactor contain- ment. These penetra- tion shall be capable of withstanding the maximum fault cur- rent vs. time condition in a given random failure of a circuit O.L. devices - (GD Criterion 50).	Yes, the SRP through review of the plant's SAR provides a record of plant status and yearly up-date of status.	?
(66)		BTP PSB-1	Provides BTP undervoltage protection for Class IE equipment during low grid voltage conditions of off- site power. Path minimum and maximum voltage load conditions that are expected should be analyzed and then verified by testing.	Yes, includes limiting conditions for operation, maximum and minimum trip set joints etc., that should protect and maintain equipment for LR applications.	
(67)		BTP ICSB-3	Provides guidance for control and protection systems. The system should satisfy the require- ments of GDC and Section 50.55a (b) of 10-CFR.50. Section 50.55a (b) references IEEE 279 which stipulates that the protection system design should document normal and abnormal and accident conditions that the system must perform.	Yes, the SRP and the BTP, through review of the plant's SAR pro- vides a record of plant status and yearly up-date of status.	See Comments

NT REVIEW - CABLES

YES	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
	None	See Column 6	Does not address aging, aging management implied by the adequacy of the penetration(s) design.
	None	None	Principally directed toward "electrical equipment", however the BTP does have an impact on the electrical loads of all the system components connected to offsite power.
	None (See Column 6)	None	10-CFR.50 dated November 30, 1988, Section 50.55a (b) refers to IEEE Standard 279, "Criteria for Protection Systems for Nuclear Power Generating Stations." This standard was withdrawn by IEEE on June 14, 1984. IEEE Standard 603, "Criteria for Safety Systems for Nuclear Power Generating Stations" has been identified by IEEE as the replacement standard for Standard IEEE 279.

REGULATORY INSTRUMENT

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIAT COLUMN 6
CABLES (68)	OPERATIONAL HAZARDS, Continued (earthquake)	RG 1.12	Describes seismic instrumentation needed to determine response to an OBE, if or if not the plant can continue to operate. This RG is principally involved with the correct location of instrumentation on piping, foundations, and structural supports to collect seismic data; however, the requirement will provide a measure of control for unsafe continued operation and further degradation to the NPP components, including cables, during a seismic event in excess of OBE.	Provides a design base for LR.	RG prepared Rev. 1 upgraded RG to include requirement of ANS 2.2 dated 9-5-78.
(69)	(earthquake)	RG 1.100	Provides regulatory position for seismic qualification of class IE equipment acceptable to the NRC staff. States that for OBE and SSE conditions qualifying tests should include hydrodynamic loads and other non-seismic vibration loads because they are part of the <u>inplant aging process</u> , and therefore should be included in seismic testing.	Provides a design base for LR. Further, this guide under the value section, page 1.100-3 states that the guide incorporates the latest state-of-the-art technology and this should enhance the <u>licensing process</u> .	This guide was approved in 1988. recognized the revision of IEEE Standard 344-87.

T REVIEW - CABLES

ES	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
	None	None	
s			
nd	None	None	This latest revision to the RG changed the scope to include mechanical equipment (1977 version was for electrical only). The revision is not expected to add new requirements or costs for the licensees.

REGULATORY INSTRUMENT

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATI COLUMN 6
CABLES (70)	OPERATIONAL HAZARDS, Continued	IEEE 383	Stipulates that Class IE cables, as installed, shall be suitable for operation at normal conditions including electrical and physical stresses for their installed life.	Provide a design base for possible life extension of cables.	Past due for acti will be reviewed IEEE in near futu (IEEE Standard Of 7/6/90)
(71)		IEEE 384	Provides guidance for cables and raceways separation in non-hazard areas. Also provides requirements for protection of redundant system for safe shut down.	Provides an ade- quate design base for license renewal applications, e.g., the effects of pipe whip, radiation, high temperatures, humidity and other hazards due to misoperation or failure of mechani- cal equipment shall be considered.	Past due for acti will be reviewed IEEE in near futu (IEEE Standard Of 7/6/90)

NT REVIEW - CABLES

ES	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
n; y e. ice	None	None	Segregation of cable is allowed for ambient service zone, e.g., cables need not meet requirements for all zones, only the ambient service conditions that the cable is intended to be installed.
n; y e. ice	None	None	Discusses area boundaries, routing requirements, hazard areas, equipment (pipe) failure, missile hazard areas, fire hazards and wiring separations at terminals. Principally a standard to assure redundant systems for safe shutdown.

REGULATORY INSTRUMENT

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATI COLUMN 6
CABLES (72)	OPERATIONAL HAZARDS, Continued	IEEE 603	Standard establishes minimum functional and design requirements for safety systems for NPPs. Applicable to power instrumentation and control portions of safety systems. Systems design base shall establish minimum criteria to address environmental condition imposed in the operator during normal, abnormal and accident conditions. The design base shall document the range of transient and steady state conditions and the environmental conditions including inspection and testing and the conditions that have the potential for functional degradation of the safety system.		Past due for action will be reviewed IEEE in near future (IEEE Standard Of 7/6/90)
(73)		IEEE 627	Provides basic principles for design qualification of safety equipment used in NPPs over the expected range of operating conditions including normal, abnormal, design base event, post design base event, and in-service testing. Service conditions for operational hazards, e.g., electrical loads (local environmental factors) shall be addressed.	Stipulate qualification documentation shall be maintained and recorded. These records will/would be useful in the license renewal process. Notes that periodic maintenance or by periodically changing components (see comments for components) the qualified life of the equipment may be extended.	IEEE Standards Office reports this standard is now under ballot for reaffirmation (7/6/90).

IT REVIEW - CABLES

ES	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
<p>ry; / a. ice</p>	<p>None (See Comments)</p>	<p>None</p>	<p>Standard refers to ANS 4.1 requirements: "Design Basis Criteria for Safety Systems in Nuclear Power Generating Stations."</p> <p>The standard defines the "Operating Elements" of an NPP safety system, including reactor trip and engineered safety features (ESF) and the "General Elements" of an NPP safety system. Also included are the requirements for sense and command and the requirements for execute features. For example, the typical execute features are ESF breakers, motors/ starters, pumps and MOVs, and solenoid valves.</p> <p>An IEEE internal Committee has determined that Aging degradation is not discussed and the treatment of Aging is judged to be inadequate for this standard.</p>
<p>ce rd</p>	<p>None</p>	<p>None (See Comments)</p>	<p>Among the equipment "Assembly Components" listed are capacitors, connectors, resistors and wire. The standard stipulates that the qualifying process shall consider significant aging effects as an essential part of the qualification process. To determine its (aging) effect on the operability of the equipment. A criteria is established in the standard that defines a "significant" aging mechanism, see paragraph 4.4.1, "Aging", page 11, STD. 627.</p> <p>PLEX considerations are not directly addressed in this standard.</p>

REGULATORY INSTRUMENT

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATI COLUMN 6
CABLES (74)	WEAR	CFRs 10-CFR-50 Appendix A Criterion 4	Requires that CSSs be designed to accommodate the effects of and be compatible with environmental conditions for normal operations, testing, maintenance and off normal events.	These provisions are required for current license and LR applications.	?
(75)		10-CFR-50 Appendix A Criterion 21	Requires, in part, periodic testing of protection systems during reactor operation. Periodic test is designed to provide status of operability of systems; however, such testing will also provide status and condition, e.g., degradation, of the components.	Provides records for evaluation of protection systems applicable to license renewal.	?
(76)		IEEE 383	Provides qualifying test for significant environmental conditions including mechanical stresses for normal operation and design base events for cables.	Provides a design base and qualification standards for extending the life of cable.	Past due for action will be reviewed IEEE in near future (IEEE Standard OI 7/6/90)
(77)		IEEE 323	Provides limits for in-service age conditioning testing for in-service stresses, e.g., wear.	Documentation that the aging issue has been addressed for IE equipment.	IEEE Standards OI reports this Star now under ballot reaffirmation (7/6/90).

IT REVIEW - CABLES

ES	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
	The criterion should include deterioration and/or degradation as an operational condition over time.	None	
	None	None	Records shall be retained until license is terminated.
n; y e. ice	None	None	Although wear is not specifically mentioned, the factors which enhance wear are noted, e.g., bedding or flexing, vibration and sidewall pressure. This standard recognizes the need for separate requirements for post design base event conditions for momentary or accumulative changes in material properties due to aging, radiation, heat and steam exposure.
ice ard or	None	None	Indicates that natural aging qualification of equipment is the most technically justified method; however, age conditioning testing or analysis is acceptable to account for the difference of natural vs. specified service.

REGULATORY INSTRUMENT

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATION COLUMN 6
CABLES (78)	WEAR, Continued	IEEE 383	Describes significant environmental conditions that address aging issues, i.e., for corrosion, moisture, and chemicals. Addresses Class IE cables used for power control and instrumentation.	Applies to both normal and design base events. Provides for methods of test documentation for the aging issues.	Past due for action will be reviewed by IEEE in near future (IEEE Standard Office 7/6/90)
(79)	FATIGUE AND VIBRATION	CFRs 10-CFR-50 Appendix A Criterion 4	Requires that CSSs be designed to accommodate the effects of and be compatible with environmental conditions for normal operations, testing, maintenance and off normal events.	These provisions are required for current license and LR applications.	?
(80)		IEEE 323	Provides limits for in-service age conditioning testing for in-service stresses, e.g., wear.	Documentation that the aging issue has been addressed for IE equipment.	IEEE Standards Office reports this Standard now under ballot for reaffirmation (7/6/90).
(81)		IEEE 383	Provides qualification tests for significant conditions of operational stressors and design base events including tension and vibration for cables. Service conditions of vibration shall be addressed.	Provides a design base and qualification criteria that will be useful in extending the life of cables.	Past due for action will be reviewed by IEEE in near future (IEEE Standard Office 7/6/90)

T REVIEW - CABLES

S	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
; . ce	None	None	Standard defines: <u>design basis events</u> . Postulated abnormal events used in the design to establish the performance requirements of the structures, systems, and components.
	The criterion should include deterioration and/or degradation as an operational condition over time.	None	
ce rd r	None	None	Indicates that natural aging qualification of equipment is the most technically justified method; however, age conditioning testing or analysis is acceptable to account for the differences of natural vs. specified service.
; . ce	None	None	This standard recognizes the need for separate requirements for post design base event conditions for momentary or accumulative changes in material properties due to aging, radiation, heat and steam exposure.

REGULATORY INSTRUMENT

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVE COLUMN 6
CABLES (82)	FATIGUE AND VIBRATION, Continued	IEEE 383	Describes significant environmental conditions that address aging issues, i.e., for corrosion, moisture, and chemicals. Addresses Class IE cables used for power control and instrumentation.	Applies to both normal and design base events. Pro- vides for methods of test documenta- tion for the aging issues.	Past due for action will be reviewed by IEEE in near future (IEEE Standard Offi 7/6/90)
(83)		IEEE 603	Standard establishes minimum functional and design requirements for safety systems for NPPs. Applicable to power instrumenta- tion and control portions of safety systems. Systems design base shall establish minimum criteria to address environmental condi- tions imposed in the operation during normal, abnormal and accident conditions. The design base shall document the range of transient and steady state conditions and the environment condi- tions including fatigue and vibration and the conditions that have the potential for functional degradation of the safety system.		Past due for action will be reviewed by IEEE in near future (IEEE Standard Offi 7/6/90)

T REVIEW - CABLES

	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
S ; - ce	None	None	Standard defines: <u>design basis events</u> . Postulated abnormal events used in the design to establish the performance requirements of the structure, systems, and components.
; . ce	None	None	Standard refers to ANS 4.1 requirements: "Design Basis Criteria for Safety Systems in Nuclear Power Generating Stations." The standard defines the "Operating Elements" of an NPP safety system, including the reactor trip and engineered safety features (ESF), and the "General Elements" of an NPP safety system. Also included are the requirements for sense and command and the requirements for execute features. For example, the typical execute features are ESF breakers, motors/starters, pumps and MOVs, and solenoids valves.

REGULATORY INSTRUMENT

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATI COLUMN 6
CABLES (84)	FATIGUE AND VIBRATION, Continued	IEEE 323	Provides limits for in-service age conditioning testing for in-service stresses, e.g., wear. Describes significant environmental conditions that address aging issues, i.e., for corrosion, moisture, and chemicals. Addresses Class IE cables used for power control and instrumentation.	Documentation that the aging issue has been addressed for IE equipment.	IEEE Standards Of reports this Stan now under ballot reaffirmation (7/6/90).
(85)		IEEE 627	Provides basic principles for design qualification of safety equipment used in NPPs over the expected range of normal, abnormal, design base event, post design base event, and in-service test conditions. Service conditions for fatigue and vibration (local environment factors) shall be addressed.	Stipulate qualification documentation shall be maintained and recorded. These records will/would be useful in the license renewal process. Notes that periodic maintenance or by periodically changing those components (see comments for components) the qualified life of the equipment may be extended. This standard, however, does <u>not</u> address requalification.	IEEE Standards Of reports this Stan now under ballot reaffirmation (7/6/90).
(86)	EMBRITTLMENT (thermal and radiation)	CFRs 10-CFR-50 Appendix A Criterion 23	Protection system failure modes shall be designed to fail in a safe state if a loss of energy occurs (e.g., electrical power) or extremes in environmental conditions exist (e.g., heat, cold, fire, steam, radiation, etc.).	These provision are required for current license and should apply to LR applications.	?

NT REVIEW - CABLES

YES	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
ice lard or	None	None	Indicates that natural aging qualification of equipment is the most technically justified method; however, age conditioning testing or analysis is acceptable to account for the difference of natural vs. specified service.
ice lard or	None	None	Among the equipment "Assembly Components" listed are capacitors, connectors, resistors and wire. The standard stipulates that the qualifying process shall consider significant aging effects as an essential part of the qualification process to determine its (aging) effect on the operability of the equipment. A criteria is established in the standard that defines a "significant" aging mechanism, see paragraph 4.4.1 Aging, page 11, STD. 627.
	None (See Comments)	None	The criterion discussed factors that can cause aging but does not assess how the design of the protection system would or should compensate for component aging over time.

REGULATORY INSTRUMENT

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVE COLUMN 6
CABLES (87)	EMBRITTLEMENT (thermal and radiation), Continued	10-CFR-50 § 50.49	Stipulates what environmental conditions that should be contemplated in the qualification of electrical equipment, includ- ing, temperature and pressure, humidity, chemical effects, radiation and aging. Qualification is required for <u>all</u> electrical equipment important to safety; by definition this includes safety equipment and <u>non</u> safety equipment whose failure may prevent the function of safety-related equipment.	Requires that all replacement equip- ment must be quali- fied in accordance with § 50.49.	?
(88)		IEEE 323	Provides limits for in-service age conditioning testing for in-service stresses, e.g., wear.	Documentation that the aging issue has been addressed for IE equipment.	IEEE Standards Off reports this Stand now under ballot f reaffirmation (7/6/90).
(89)		IEEE 384	Stipulates equipment shall be protected from operational (environmental) hazards, e.g., pipe whip, steam release, radiation, etc., from design base events.	Provides the design base which could qualify life extension for electrical equipment.	Past due for actio will be reviewed b IEEE in near futur (IEEE Standard Off 7/6/90)

NT REVIEW - CABLES

ES	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
	"Significant" degradation (see comments) should be defined.	See Comments on NUMARC report.	<p>Provides the basic requirements for the IEEE equipment qualification standards. Consideration must be given to all significant degradation that could effect the function of the equipment. Equipment subjected to "mild" environmental conditions are not included in the scope of § 50.49.</p> <p>NUMARC's Report 90-08, July 1990, pages 1-8 states that "The only age-related degradation mechanisms which were identified as being potentially significant in the license renewal term are the embrittlement of insulation for all applications and the embrittlement of jackets for the specific cases where credit is taken for jacket beta radiation shielding, where electrical shield integrity is important to circuit performance, and where the jacket is bonded to the insulation. These potentially significant cable degradation mechanisms are managed by currently practiced, plant-specific cable qualification as described in Section 5.2. Hence, successful extension of cable qualification to encompass the license renewal term will resolve each of these degradation mechanisms, and no further license renewal evaluation is required."</p>
ice ard or	None	None	Indicates that natural aging qualification of equipment is the most technically justified method; however, age conditioning testing or analysis is acceptable to account for the differences of natural vs. specified service.
n; y e. ice	None	None	Use of separation barriers and electrical isolation provides control over design base events that can and will accelerate the aging process. Such as fire, <u>high temperature</u> , flooding, steam/water spray radiation, etc.

REGULATORY INSTRUMENT

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVE COLUMN 6
CABLES (90)	EMBRITTELEMENT, Continued	IEEE 603	Standard establishes minimum functional and design requirements for safety systems for NPPs. Applicable to power instrumentation and control portions of safety systems. Systems design base shall establish minimum criteria to address environmental conditions imposed in the operations during normal, abnormal and accident conditions. The design base shall document the range of transient and steady state conditions and the environmental conditions including radiation and temperatures and the conditions that have the potential for functional degradation of the safety system.		(IEEE 603 has replaced IEEE 279 which was withdrawn 6-14-84 by IEEE.) See Comments. Past due for action will be reviewed by IEEE in near future (IEEE Standard Office 7/6/90)
(91)		IEEE 627	Provides basic principles for design qualification of safety equipment used in NPPs over the expected range of normal, abnormal, design base event, post design base event, and in-service test conditions. Service conditions for radiation and ambient temperatures (local environmental factors) shall be addressed.	Stipulate qualification documentation shall be maintained and recorded. These records will/would be useful in the license renewal process. Notes that periodic maintenance or by periodically changing components (see comments for components) the qualified life of the equipment may be extended.	IEEE Standards Office reports this Standard now under ballot for reaffirmation (7/6/90).

NT REVIEW - CABLES

ES	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
	(See Comments)	None	<p>Standard refers to ANS 4.1 requirements: "Design Basis Criteria for Safety Systems in Nuclear Power Generating Stations."</p> <p>The standard defines the "Operating Elements" of an NPP safety system, including reactor trip and engineered safety features (ESF) and the "General Elements" of an NPP safety system. Also included are the requirements for sense and command and the requirements for execute features. For example, the typical execute features are ESF breakers, motors/starters, pumps and MOVs, and solenoid valves.</p> <p>IEEE 279 is still frequently referenced in current SRP and CFRs instruments, i.e., in 10-CFR-50 § 50.55a, Codes and Standards and in SRP 7.1 and BTP-ICS8-3.</p> <p>An IEEE internal Committee has determined that the treatment of Aging is not adequate for this standard.</p>
	None	None	<p>Among the equipment "Assembly Components" listed are capacitors, connectors, resistors and wire. The standard stipulates that the qualifying process shall consider significant aging effects as an essential part of the qualification process. To determine its (aging) effect on the operability of the equipment. A criteria is established in the standard that defines a "significant" aging mechanism, see paragraph 4.4.1, "Aging", page 11, STD. 627.</p>

REGULATORY INSTRUMENT

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATI COLUMN 6
CABLES (92)	CORROSION	CFRs 10-CFR-50 Appendix A Criterion 23	Protection system failure modes shall be designed to fail in a safe state if a loss of energy occurs (e.g., electrical power) or extremes in environmental conditions exist (e.g., heat, cold, fire, steam, radiation, etc.).	These provision are required for current license and should apply to LR applications.	?
(93)		10-CFR-50 § 50.49	Stipulates what environmental conditions that should be contemplated in the qualification of electrical equipment, including, temperature and pressure, humidity, chemical effects, radiation and aging. Qualification is required for <u>all</u> electrical equipment important to safety; by definition this includes safety equipment and <u>non</u> safety equipment whose failure may prevent the function of safety-related equipment.	Requires that all replacement equipment must be qualified in accordance with § 50.49.	?

NT REVIEW - CABLES

YES	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
	None (See comments)	None	The criterion discussed factors that can cause aging but does not assess how the design of the protection system would or should compensate for component aging over time.
	"Significant" degradation (see comments) should be defined.	None	Provides the basic requirements for the IEEE equipment qualification standards. Consideration must be given to all significant degradation that could effect the function of the equipment. Equipment subjected to "mild" environmental conditions are not included in the scope of § 50.49.

REGULATORY INSTRUMENT

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVE COLUMN 6
CABLES (94)	CORROSION, Continued	IEEE 420	Stipulates that cable or wire insulation that releases toxic or corrosive gases under combustion shall not be used in control board, panel and racks. Refers to IEEE 323 for qualifying requirements stipulated for degradation and hazards associated with the equipment. Describes significant environmental conditions that address aging issues, i.e., for corrosion, moisture, and chemicals. Addresses Class IE cables used for power control and instrumentation.	Provides verification of design requirement for Class IE control board, panels and racks.	Past due for action will be reviewed by IEEE in near future (IEEE Standard Office 7/6/90)
(95)		IEEE 383	Describes significant environmental conditions that address aging issues, i.e., for corrosion, moisture, and chemicals. Addresses Class IE cables used for power control and instrumentation.	Applies to both normal and design base events. Provides for methods of test documentation for the aging issues.	Past due for action will be reviewed by IEEE in near future (IEEE Standard Office 7/6/90)

T REVIEW - CABLES

	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
S ; . ce	None Review of components that need to be renewed is directed to IEEE 323.	None	Promotes management of aging by stipulating the design and installation should provide for ease of repair, maintenance and removal "without degrading the reliability of the control board." Field cable wire-ways "shall have sufficient space to allow maintenance personnel to connect field connections." Qualifications beyond qualified life are directed to IEEE 323.
; . ce	None	None	Standard defines: <u>design basis events</u> . Postulated abnormal events used in the design to establish the performance requirements of the structure, systems, and components.

REGULATORY INSTRUMENT

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVE COLUMN 6
CABLES (96)	CORROSION, Continued	IEEE 627	Provides basic principles for design qualification of safety equipment used in NPPs over the expected range of operating conditions including normal, abnormal, design base event, post design base event, and in-service testing. Service conditions for operational hazards, e.g., electrical loads (local environmental factors) shall be addressed.	Stipulate qualification documentation shall be maintained and recorded. These records will/would be useful in the license renewal process. Notes that periodic maintenance or by periodically changing components (see comments for components) the qualified life of the equipment may be extended.	IEEE Standards Office reports this Standard is now under ballot for reaffirmation (7/6/90).
(97)	CREEP AND SHRINKAGE	IEEE 323	Provides limits for in-service age conditioning testing for in-service stresses. Creep and shrinkage are not specifically mentioned. This aging issue is however implied because the standard addresses the environment of high temperature.	Documentation that the aging issue has been addressed for IE equipment.	Past due for action will be reviewed by IEEE in near future (IEEE Standard Office 7/6/90)

IT REVIEW - CABLES

ES	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
ice ard t	None	None	Among the equipment "Assembly Components" listed are capacitors, connectors, resistors and wire. The standard stipulates that the qualifying process shall consider significant aging effects as an essential part of the qualification process. To determine its (aging) effect on the operability of the equipment. A criteria is established in the standard that defines a "significant" aging mechanism, see paragraph 4.4.1, "Aging", page 11, STD. 627.
1; / 2. ice	None	None	Indicates that natural aging qualification of equipment is the most technically justified method; however, age conditioning testing or analysis is acceptable to account for the difference of natural vs. specified service.

Appendix II

Regulatory Instrument Review for Containment and Basemat

REGULATORY I

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5
CONCRETE CONTAINMENT (BWR)	CORROSION - UNIFORM (OR "NORMAL")	CFRs		
(1)		10 CFR 50 Appendix A Criterion 50, 51, 52, 53 and 54 (Section V Reactor Containment)	Provides general design criteria for reactor con- tainment including provi- sion to allow (Criterion 53) appropriate testing and inspection of all areas of the containment structure/system.	Yes
(2)		Appendix B	Provides Quality Assur- ance requirements for document control, design control, construction control and operations	Yes
(3)		Appendix J	Records document control and defines primary con- tainment leak tests.	Yes
			Leakage Testing (Type A, Type B, and Type C Tests)	(See comments)

General Notes:

1. A "?" indicates further study/investment is needed.
2. For the GSIs, "resolved" means the generic safety issue is resolved, not necessary.
3. For the meaning of abbreviations, acronyms, and initialisms, used throughout, see and xiii of the report.

APPENDIX II

INSTRUMENT REVIEW - CONTAINMENT AND BASEMAT

CURRENT INITIATIVES COLUMN 6	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
See Part 50 proposed Rule Making 45-FR-81602, may impact Appendix A	Corrosion due to differential aeration and microbial influence occurs in the suppression pool steel liner below the water line and at posttensioning tendon wire (Mark I and II) or strand (PCCV).	Improved NDE methods are needed for evaluation of containment facilities.	<p>Applicable components/structures are the reinforced steel drywell liner and the posttensioning system anchors (PCCV).</p> <p>A method to inspect the reinforcing steel for corrosion is required; a ISI method to inspect re-steel is not available. Boroscopic examination of exterior surface near penetrations.</p> <p>Pitting corrosion occurs in posttensioning tendon or wire strand (PCCV).</p> <p>Record and documentation of tests and inspections. QA is to provide action necessary to provide adequate confidence that the facilities' SSC will perform satisfactorily. Aging management implied through documentation of tests and inspections.</p> <p>A general visual inspection of <u>accessible</u> interior and exterior of containment structure and components must be performed prior to a type A to uncover any evidence of structural deterioration. Corrective action is required if deterioration is found. Applicable components/ structures are the reinforced steel drywell liner and the posttensioning system anchors (PCCV containments).</p>
?			
See 51 FR 39528 Proposed Rule Making: to update criteria and clarify questions regarding leakage rate testing of LWRs. See 53 FDR 5985 Alternate Method for Leak Rate Testing. Also see 53 FR 45890 which amended Appendix J and allows statistical data analysis acceptable to the NRC, i.e., the Mass Point method, ANSI/ANS 56.8-1981.			

REGULATORY

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5
CC-BWR Continued (4)	CORROSION Continued	ACI STANDARDS 349-85	This Code provides the minimum requirements for design and construction of nuclear safety related concrete structure and structural elements for NPPs. Covers, flexure and axial loads, shear and torsion and design of slate, walls, footings pre-stressed, precast and shells of concrete.	Yes, refers to 10-CFR-50 Appendix B and 10-CFR-100 Reactor Site Criteria - Appendix Seismic and Geologic Citing Criteria.
(5)		ACI 349-85 Appendix A Thermal Considerations	Contains criteria that address the effects the gradient temperatures, distribution, time dependent variations of temperatures for thermal strains associated with normal operating conditions and accident conditions and thermal stress in considering stiffness and rigidity and restrain of the structure.	Provides limits for operating temperatures of concrete, i.e., 150°F normal areas not to exceed 200°F. Records and calculation of design.
(6)		ACI 349 Appendix B Steel Embedments	Covers general requirements and loading combinations for embedded steel, e.g., shear, compression and bending.	Yes - Records of Design
(7)		ACI 349 Appendix C Special provisions for impulse and impactive effects.	Provides Design Consideration/Criteria for nuclear safety-related concrete structure for impulsive and impactive loads that <u>must be</u> combined with other loads (dual and level loads specified in ACI-349, Section 9.1).	Yes, Records of Design

APPENDIX II STRUMENT REVIEW - CONTAINMENT AND BASEMAT

CURRENT INITIATIVES COLUMN 6	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
Adopted as a standard in 1985 to supersede ACI 349-80.	N/A	N/A	Specially <u>excluded</u> from this Code are those structures covered by ASME BPVC Section III, Division 2. "Code for Concrete Reactor Containments" and the general requirements of ACI Standard 359.
?	N/A	N/A	For accident or other short term conditions concrete surface temperatures may reach 300°F. Local areas may be allowed to reach 650°F in the event of a steam or water jet from pipe failures.
?	N/A	N/A	Notes that fabrication and installation shall be in accordance with good practice to avoid "detrimental cracking or spalling of concrete or excess stress in the embedment anchors."
?	N/A	N/A	<p>Impactive and impulsion loads are treated separately in the appendix/standard because of the effects as well as the response characteristics of the structural elements subjected to the loads.</p> <p><u>impactive loads</u>: time dependent loads due to collision of masses associated with finite amounts of kinetic energy.</p> <p><u>impulsive loads</u>: those <u>not</u> associated with collision of masses including but not limited to, jets impingement, blast pressure, compartment pressurization and pipe whip - restraint reaction.</p>

REGULATORY

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5
CC-BWR Continued (8)	CORROSION Continued	ACI 349.IR-80 ACI 349 Committee Report	This Standard covers re- inforced concrete design for thermal effects on NPP structures intended to conform to the general provisions of Appendix A ACI 349, it covers crack- ing, reinforcement, structural design, temperature gradients, thermal properties and thermal stresses. It is intended to complement Appendix A, ACI-349.	
(9)		ASME BPVC Section III Division 2 Subsection CC	Establishes the rules for material, fabrication, construction, examination and testing for pre- stressed and reinforced concrete containments. Containments that have a design greater than 5 psig and classified as concrete containments (CC) shall be constructed to these rules.	None explicit; yet provides guidance and sets forth rules for repairs and replacements of code components.
(10)		Article CC-3000	Provides general rules of design for CCs and metal- lic liner including load criteria. Load categor- ies include normal service loads, severe and extreme environmental loads, and abnormal loads. Normal service loads include operating loads and thermal effects during operating or shut- down conditions. Table CC 3230-1 provides a list of loads combinations and applicable load factors for the containment design.	Yes, records

APPENDIX II INSTRUMENT REVIEW - CONTAINMENT AND BASEMAT

CURRENT INITIATIVES COLUMN 6	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
Standard reapproved in 1986.			Provides supplemental information for structures, and axisymmetric structures.
?	Implied in the design requirements.	Guidance is required to cover the residual life assessment of components. Wall thickness measure- ments; NDE examinations for extent of corrosion; maintenance records. Water chemistry analysis (oxygen concentration; acidity; presence of chlorides, bacteria, etc.)	Article CC 2140 states that the consideration of <u>deterioration</u> of material caused by the service is outside the scope of this subsection (i.e., concrete containment, CC). It is the responsibility of the <u>owner</u> to select material suitable for the conditions stated in the Design Specifications, with specific attention given to the effects of service conditions upon the properties of the material.
?	Analysis for long term deterioration	Analysis for long term deterioration	Thermal effects and loads shall be considered the most critical transient or steady state condition. Pipe reaction (loads) shall be assessed during normal operation or shutdown conditions based on the most critical transient or steady state condition.

REGULATORY I:

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5
CC-BWR Continued (11)	CORROSION Continued	Article CC-5000 (Construction Testing and Examination)	N/A	Yes, records of successfully design and construction activities
(12)		Article CC- 6000 (Struc- tural Integrity Tests of CC Structures)	N/A	Yes, records of successful initial acceptance of CC structures
(13)		GSIs B.49	Design criteria for corrosion prevention and in-service testing.	
(14)	CREEP/SHRINKAGE (PCC)	CFRs 10 CFR 50 Appendix A	Records GDC 50 and 51. Criterion 51 stipulate the design shall consider seismic conditions of boundary materials during normal operating, test- ing, maintenance and postulated accidents.	Yes
(15)		Appendix B	Provides Quality Assur- ance requirements for document control, design control, construction control and operations	Yes
(16)		Appendix J	Leakage Testing (Type A, Type B, and Type C Tests)	Yes

APPENDIX II STRUMENT REVIEW - CONTAINMENT AND BASEMAT

CURRENT INITIATIVES COLUMN 6	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
---------------------------------	-------------------------	----------------------------------	----------------------

N/A	N/A	N/A	Requires qualified examination and testing personnel as per CC-5122
-----	-----	-----	---

N/A	N/A	N/A	Requires qualified examination and testing personnel as per CC-5122
-----	-----	-----	---

?	Approach to evaluate effects of long-term corrosion on reinforcements, steel liner/concrete contacts, steel surface in contact with water.		Issue also deals with inservice inspection of all types of containments. Designated a "Licensing Issue."
---	--	--	--

Concrete shrinkage due to long-term evaporation of residual water; creep due to increase in concrete strain under sustained stress. Shrinkage increases the magnitude of creep.

Record and documentation of tests and inspections. QA is to provide action necessary to provide adequate confidence that the facilities' SSC will perform satisfactorily. Aging management implied through documentation of tests and inspections.

See 51 FR 39528 Proposed Rule Making: to update criteria and clarify questions regarding leakage rate testing of LWRs. See 53 FDR 5985 Alternate Method for Leak Rate Testing. Also see 53 FR 45890 which amended Appendix J and allows statistical data analysis acceptable to the NRC, i.e., the Mass Point method, ANSI/ANS 56.8-1981.

A general visual inspection of accessible interior and exterior of containment structure and components must be performed prior to a type A to uncover any evidence of structural deterioration. Corrective action required of deterioration is found. Applicable components/structures are the reinforced steel drywell liner and the posttensioning system anchors (PCC containments).

REGULATORY II

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5
CC-BWR Continued (17)	CRACKING/SPALLING (LOSS OF FREE WATER)	10-CFR-50 Appendix A	Records GDC 50 and 51. Criterion 51 stipulate the design shall consider seismic conditions of boundary materials during normal operating, testing, maintenance and postulated accidents.	Yes
(18)		Appendix B	Provides Quality Assur- ance requirements for document control, design control, construction control and operations	Yes
(19)		Appendix J	Leakage Testing (Type A, Type B, and Type C Tests)	Yes
(20)		ACI STANDARDS 349-85 (Appendix A and C)	Leakage Testing	Yes
(21)	FATIGUE	10 CFR 50 Appendix A, B, and J	Recs., Q.A., Leak Test. (Type A, Type B, and Type C Tests)	Yes
(22)		Appendix G	Fracture Toughness (Refer to ASME Codes)	Yes

APPENDIX II STRUMENT REVIEW - CONTAINMENT AND BASEMAT

CURRENT INITIATIVES COLUMN 6	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
See 51 FR 39528 Proposed Rule Making: to update criteria and clarify questions regarding leakage rate testing of LWRs. See 53 FDR 5985 Alternate Method for Leak Rate Testing. Also see 53 FR 45890 which amended Appendix J and allows statistical data analysis acceptable to the NRC, i.e., the Mass Point method, ANSI/ANS 56.8-1981.			Record and documentation of tests and inspections. QA is to provide action necessary to provide adequate confidence that the facilities' SSC will perform satisfactorily. Aging management implied through documentation of tests and inspections.
			A general visual inspection of <u>accessible</u> interior and exterior of containment structure and components must be performed prior to a type A to uncover any evidence of structural deterioration. Corrective action is required of deterioration is found. Applicable components/structures are the reinforced steel drywell liner and the posttensioning system anchors (PCC containments).
?	?	Periodic monitoring of tendon forces required.	
?	?	Check coolant level; flow rate.	Mark I and Mark II suppression pool steel liner, drywell steel liner.
?	?	NDE examination data; design, fabrication and maintenance records.	Mark I and Mark II suppression pool steel liner, drywell steel liner.

REGULATORY II

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5
CC-BWR Continued (23)	FATIGUE Continued	ASME BPVC Section III Division 2 Subsection CC	Establishes the rules for material, fabrication, construction, examination and testing for pre-stressed and reinforced concrete containments. Containments that have a design greater than 5 psig and classified as concrete containments (CC) shall be constructed to these rules.	None explicit; yet provides guidance and sets forth rules for repairs and replacements of code components.
(24)	RELAXATION (PCC)	ACI STANDARDS 349-85 (Appendix B)	Tendon surveillance program, visual inspection.	?
(25)	HYDROGEN EMBRITTLEMENT (PCC)	ACI STANDARDS 349-85 (Appendix B)	Tendon surveillance program, visual inspection.	?
(26)		ASME BPVC Section III Division 2 Subsection CC	Establishes the rules for material, fabrication, construction, examination and testing for pre-stressed and reinforced concrete containments. Containments that have a design greater than 5 psig and classified as concrete containments (CC) shall be constructed to these rules.	None explicit; yet provides guidance and sets forth rules for repairs and replacements of code components.
(27)	GENERAL; ALL POTENTIAL FAILURE MODES	GSIs A.6, A.7, and A.8	Mark I and Mark II programs; ability of Mark I containments to withstand accident loads.	?
(28)	FATIGUE CRACKING	B.5, Pt. I	Ductility of two-way slabs and response to LOCA or high-energy line breaks.	?

APPENDIX II STRUMENT REVIEW - CONTAINMENT AND BASEMAT

CURRENT INITIATIVES COLUMN 6	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
?	Implied in the design requirements.	Guidance is required to cover the residual life, i.e., fatigue life, assessment of components.	Article CC 2140 states that the consideration of <u>deterioration</u> of material caused by the service is outside the scope of this subsection (i.e., concrete containment, CC). It is the responsibility of the <u>owner</u> to select material suitable for the conditions stated in the Design Specifications, with specific attention given to the effects of service conditions upon the properties of the material.
?	?	?	Posttensioning tendon wire or strand requires periodic monitoring of tendon forces.
?	?	?	Posttensioning system anchors require water chemistry analysis (oxygen concentration; acidity; presence of chlorides and sulfates).
?	?	Guidance is required to cover the residual life, of components, vs. material embrittlement properties.	Article CC 2140 states that the consideration of <u>deterioration</u> of material caused by the service is outside the scope of this subsection (i.e., concrete containment, CC). It is the responsibility of the <u>owner</u> to select material suitable for the conditions stated in the Design Specifications, with specific attention given to the effects of service conditions upon the properties of the material.
?	Effects of aging on ability of Mark I containments to withstand accident loads.	?	Aging of Mark I not specifically evaluated.
?	Effects of aging on ability of Mark I containments to withstand accident loads.	?	Aging effects not evaluated; issue deals with behavior of two-way reinforced concrete slabs loaded dynamically in biaxial membrane tension.

REGULATORY I

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5
CC-BWR Continued (29)	INSPECTION & TESTING	ASME BPVC Section XI Subsection IWL	Provides rules for concrete containment. Preservice, inservice and repairs of class cc components.	ISI records. For example, provides ISP schedule 1,3,5 and 10 year intervals. See articles IWL-2410 2420, and 2421, included in 2421-two power units are requirement for 30 & 35 years
(30)		R.G 1.35 Rev. 3	Guide describes a basis for acceptable ISI and surveillance program for ungrouted tendons in prestressed concrete containment structures. Provides regulatory position for the ISI including, <u>frequency</u> , sampling, areas of inspection, criteria, and scope. General regulatory position indicated that the containments shall be designed so that the anchor hardware is <u>accessible for inservice inspection</u> .	Yes. Test records/documentation
(31)		1.35.1	Provides expanded guidance to RG 1.35 for determining prestress forces used for ISI on prestressed concrete containments and discusses the regulatory position on: the measurement of prestressing force, the determination of prestressing losses, the grouping of tendons, and the construction of (tendon) tolerance bands.	Yes. Test records/documentation

APPENDIX II INSTRUMENT REVIEW - CONTAINMENT AND BASEMAT

CURRENT INITIATIVES COLUMN 6	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
Not current approval by NRC. Issued in Dec 31, 1988, addenda of Section XI	?	?	Provides for examination IWL-2000. Provides for acceptance standards IWL-3000. Inservice examination covered in Article IWL 3200.
			RG provides details for inspection interval in years, i.e., 1, 3 and 5 initially and 5 years thereafter and sampling percentage fraction of the posttensioning system. Visual inspection requirements are described including the exterior surfaces which "should be examined to detect areas of large spall, severe scaling, D-cracking in an area of 25 square feet or more, other surface deterioration or disintegration, or grease leakage." ACI 201.1R-60 is identified as the appropriate standard for definitions, e.g., scaling and D- cracking. RG 1.35.1 notes in B. Discussions that "of particular concern in the case of prestressed concrete containment structures is the possible degradation of prestressing tendon system by corrosion." RGs 1.35 and 1.90 are formulated to achieve this basic objective.
	Does not consider time extension beyond 40 years		Discusses the effects of shrinkage of concrete, the effect of concrete creep, the effect of relaxation of prestressing steel, the losses caused by tendon degradation and the effects of variations in the temperature. Understanding of these factors can enhance aging management.

REGULATORY IN

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5
CC-BWR Continued (32)	INSPECTION & TESTING Continued	1.90	Guide provides for acceptable ISI of pre-stressed concrete containments with grouted tendons. Gives the regulatory position for the visual examinations, the interval of inspection in years, sampling routines for both Alternative A and B, sampling areas and reportable conditions.	Yes. Test records/documentation
METAL CON- TAINMENTS (BWR) (33)	CORROSION	CFRs 10 CFR 50 Appendix A Criterion 50, 51, 52, 53 and 54 (Section V Reactor Containment)	Provides general design criteria for reactor containment including provision to allow (Criterion 53) appropriate testing and inspection of all areas of the containment structure/system.	Yes
(34)		Appendix B	Records, document control, QA procedures	Yes
(35)		Appendix J	Reactor Containment leakage testing Type A, Type B, and Type C tests.	Yes

APPENDIX II
TRUMENT REVIEW - CONTAINMENT AND BASEMAT

IRRENT INITIATIVES COLUMN 6	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
--------------------------------	-------------------------	----------------------------------	----------------------

Provides for aging management by requiring that some conditions of the visual inspections are reportable, e.g., crack patterns at structurally critical areas. Also reportable conditions are required for test tendons that indicate a prestress force below the RG acceptable band.

?	ISI method needed for embedded shell.	Wall thickness measurements; NDE examinations for extent of corrosion; maintenance records; e.g., crevice corrosion and pitting of embedded shell.	Exterior surface of Mark I and II drywells and Mark I and II drywells and Mark I drywell base near sand pocket. Embedded shell region, high energy pipe line penetrations hatches, vent lines, stainless steel bellows, submerged portion of suppression pool, different metal welds.
---	---------------------------------------	--	---

?			
---	--	--	--

See 51 FR 39528 Proposed Rule Making: to update criteria and clarify questions regarding leakage rate testing of LWRs. See 53 FDR 5985 Alternate Method for Leak Rate Testing. Also see 53 FR 45890 which amended Appendix J and allows statistical data analysis acceptable to the NRC, i.e., the Mass Point method, ANSI/ANS 56.8-1981.

Aging Management is implied, corrosion not specifically mentioned; yet, a general purpose inspection before a Type A test is required to "uncover any evidence of structural deterioration which may effect either the containment structure integrity or leak tightness" - quoted from 10-CFR-50 Appendix J, Section V, paragraph A, Containment Inspection.

REGULATORY IN

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5
METAL CON- TAINMENT BWR Continued (36)	CORROSION Continued	Article CC-3000	Provides general rules of design for CCs and metallic liner including load criteria. Load categories include normal service loads, severe and extreme environmental loads, and abnormal loads. Normal service loads include operating loads and thermal effects during operating or shut-down conditions. Table CC 3230-1 provides a list of loads combinations and applicable load factors for the containment design.	Yes, records
(37)		Article CC-5000 (Construction Testing and Examination)	N/A	Yes, records of successfully design and construction activities
(38)		Article CC-6000 (Structural Integrity Tests of CC Structures)	N/A	Yes, records of successful initial acceptance of CC structures
(39)	FATIGUE	10 CFR 50 Appendix A Criterion 51	The design shall consider service temperature and other conditions of the containment boundary <u>material</u> including the uncertainties of determining <u>material properties</u> , transient stresses and flaw sizes.	Provides record and QA documentation.

APPENDIX II INSTRUMENT REVIEW - CONTAINMENT AND BASEMAT

CURRENT INITIATIVES COLUMN 6	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
?	Analysis for long term deterioration	Analysis for long term deterioration	Article CC 2140 states that the consideration of <u>deterioration</u> of material caused by the service is outside the scope of this subsection (i.e., concrete containment, CC). It is the responsibility of the <u>owner</u> to select material suitable for the conditions stated in the Design Specifications, with specific attention given to the effects of service conditions upon the properties of the material.
N/A	N/A	N/A	Requires qualified examination and testing personnel as per CC-5122
N/A	N/A	N/A	Requires qualified examination and testing personnel as per CC-5122
?	?	<p>Temperature, pressure, and vibration measurements are needed (from transients, water hammer events, etc.). Coolant level; flow rate required. NDE examination (from surveillance and maintenance records).</p> <p>Fatigue design basis needs to determine the significance of continued cyclic loading beyond established life.</p>	<p>Embedded shell region, high energy pipe line penetrations, hatches, vent lines, submerged portion of suppression pool, transition region from cylindrical to spherical portion of Mark I drywell and different metal welds.</p> <p>Stainless steel bellows (expansion).</p>

REGULATORY IN

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5
METAL CON- TAINMENT BWR Continued (40)	FATIGUE	Article CC-3000	Provides general rules of design for CCs and metallic liner including load criteria. Load categories include normal service loads, severe and extreme environmental loads, and abnormal loads. Normal service loads include operating loads and thermal effects during operating or shutdown conditions. Table CC 3230-1 provides a list of loads combinations and applicable load factors for the containment design.	Yes, records
(41)		Article CC-5000 (Construction Testing and Examination)	N/A	Yes, records of successfully design and construction activities
(42)		Article CC-6000 (Structural Integrity Tests of CC Structures)	N/A	Yes, records of successful initial acceptance of CC structures
(43)	MECHANICAL WEAR AND EROSION	10 CFR 50 Appendix A Criterion 50, 51, 52, 53 and 54 (Section V Reactor Containment)	Provides general design criteria for reactor containment including provision to allow (Criterion 53) appropriate testing and inspection of all areas of the containment structure/system.	Yes
(44)		10 CFR 50 Appendix B	Quality Assurance	Yes

APPENDIX II TRUMENT REVIEW - CONTAINMENT AND BASEMAT

URRENT INITIATIVES COLUMN 6	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
?	Analysis for long term deterioration	Analysis for long term deterioration	Article CC 2140 states that the consideration of <u>deterioration</u> of material caused by the service is outside the scope of this subsection (i.e., concrete containment, CC). It is the responsibility of the <u>owner</u> to select material suitable for the conditions stated in the Design Specifications, with specific attention given to the effects of service conditions upon the properties of the material.
N/A	N/A	N/A	Requires qualified examination and testing personnel as per CC-5122
N/A	N/A	N/A	Requires qualified examination and testing personnel as per CC-5122
?		Visual examination during surveillance and maintenance; transient records.	Contacting surfaces that are in sliding motion - undesired dimensional changes.

REGULATORY II

<u>COMPONENT COLUMN 1</u>	<u>ISSUE COLUMN 2</u>	<u>REG. INSTRUMENT COLUMN 3</u>	<u>AGING FEATURES COLUMN 4</u>	<u>LIFE EXTENSION FEATURES COLUMN 5</u>
METAL CON- TAINMENT BWR Continued (45)	MECHANICAL WEAR AND EROSION Continued	Appendix J	Leakage Testing	Yes
(46)		Article CC-3000	Provides general rules of design for CCs and metallic liner including load criteria. Load categories include normal service loads, severe and extreme environmental loads, and abnormal loads. Normal service loads include operating loads and thermal effects during operating or shut-down conditions. Table CC 3230-1 provides a list of loads combinations and applicable load factors for the containment design.	Yes, records
(47)		Article CC-5000 (Construction Testing and Examination)	N/A	Yes, records of successfully design and construction activities

APPENDIX II INSTRUMENT REVIEW - CONTAINMENT AND BASEMAT

CURRENT INITIATIVES COLUMN 6	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
See 51 FR 39528 Proposed Rule Making: to update criteria and clarify questions regarding leakage rate testing of LWRs. See 53 FR 985 Alternate method for Leak Rate testing. Also see 53 FR 45890 which amended Appendix J and allows statistical data analysis acceptable to the RC, i.e., the Mass point method, NSI/ANS 56.8-1981.			
?	Analysis for long term deterioration	Analysis for long term deterioration	Article CC 2140 states that the consideration of <u>deterioration</u> of material caused by the service is outside the scope of this subsection (i.e., concrete containment, CC). It is the responsibility of the <u>owner</u> to select material suitable for the conditions stated in the Design Specifications, with specific attention given to the effects of service conditions upon the properties of the material.
N/A	N/A	N/A	Requires qualified examination and testing personnel as per CC-5122

5
F
1
4
1
1
C
E
A
7
E
E
C
E
A
E
F
F

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5
METAL CON- TAINMENT BWR Continued (48)	MECHANICAL WEAR AND EROSION Continued	Article CC-6000 (Structural In- tegrity Tests of CC Structures)	N/A	Yes, records of successful initial acceptance of CC structures
(49)	IRRADIATION EMBRITTLEMENT	10 CFR 50 Appendix J	Leakage Testing	Yes
(50)		Article CC-3000	Provides general rules of design for CCs and metal- lic liner including load criteria. Load categor- ies include normal service loads, severe and extreme environmental loads, and abnormal loads. Normal service loads include operating loads and thermal effects during operating or shut- down conditions. Table CC 3230-1 provides a list of loads combinations and applicable load factors for the containment design.	Yes, records

APPENDIX II INSTRUMENT REVIEW - CONTAINMENT AND BASEMAT

CURRENT INITIATIVES COLUMN 6	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
N/A	N/A	N/A	Requires qualified examination and testing personnel as per CC-5122
See 51 FR 39528 Proposed Rule Making: to update criteria and clarify questions regarding leakage rate testing of LWRs. See 53 FDR 5985 Alternate method for Leak Rate testing. Also see 53 FR 45890 which amended Appendix J and allows statistical data analysis acceptable to the NRC, i.e., the Mass Point method, ANSI/ANS 56.8-1981.		Dosimetry readings; design and fabrication data; weld examination and surveillance data; records.	Transition region from cylindrical to spherical portion of Mark I drywell shell at the core horizontal midplane elevation.
?	Analysis for long term deterioration	Analysis for long term deterioration	Article CC 2140 states that the consideration of <u>deterioration</u> of material caused by the service is outside the scope of this subsection (i.e., concrete containment, CC). It is the responsibility of the <u>owner</u> to select material suitable for the conditions stated in the Design Specifications, with specific attention given to the effects of service conditions upon the properties of the material.

REGULATORY IN

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5
METAL CON- TAINMENT BWR Continued (51)	IRRADIATION EMBRITTELEMENT Continued	Article CC-5000 (Construction Testing and Examination)	N/A	Yes, records of successfully design and construction activities
(52)		Article CC- 6000 (Struc- tural Integrity Tests of CC Structures)	N/A	Yes, records of successful initial acceptance of CC structures
(53)	BUCKLING	GSI's B.5, Pt. II	Buckling behavior of steel containments.	
(54)	DYNAMIC LOADS	B.10	Behavior of BWR Mark III containments, particu- larly wet well and sup- pression pool, under LOCA conditions.	
(55)		B.54	Behavior of ice condenser containments under design-basis accident conditions.	
(56)	INSPECTION AND TESTING	ASME BVPV, Section XI, Subsection IWE	Provides for visual inspection of accessible containment shell, vents, flanges, and penetrations.	ISI examination and inspection records.

APPENDIX II STRUMENT REVIEW - CONTAINMENT AND BASEMAT

CURRENT INITIATIVES COLUMN 6	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
N/A	N/A	N/A	Requires qualified examination and testing personnel as per CC-5122
N/A	N/A	N/A	Requires qualified examination and testing personnel as per CC-5122
	Effects of aging on buckling behavior of steel containments.		Aging effects not evaluated, issue deals with design of steel containments subjected to unsymmetrical dynamic loads.
	Effects of aging on the ability of the wet well and suppression pool to withstand dynamic loading conditions.		Aging effects not evaluated.
	Effects of aging on ability of ice condenser containments to withstand DBA loading conditions.		Computer code was developed to model ice condenser containments.
he NRC is currently reviewing the requirements of IWE changes are expected in CFR to mandate ASME requirements.	NRC approved of code inspection methods.	NRC approval of code inspection methods.	

REGULATORY

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5
CONCRETE CONTAIN- MENTS (PWR) (57)	CORROSION	CFRs 10 CFR 50 Appendix A	Records GDC Criterion 50, 51, 52, 53, and 54 address reac- tor containment, e.g., fracture prevention, con- tainment leak testing, provisions for testing and inspection of containment.	Yes
(58)		Appendix B	Provides Quality Assur- ance requirements for document control, design control, construction control and operations	Yes, records.
(59)		Appendix J	Defines primary contain- ment leak test require- ments, i.e., Type A, B, and C tests.	Yes
(60)		ASME BPVC Section III Division 2 Subsection CC	Establishes the rules for material, fabrication, construction, examination and testing for pre- stressed and reinforced concrete containments. Containments that have a design greater than 5 psig and classified as concrete containments (CC) shall be constructed to these rules.	None explicit; yet provides guidance and sets forth rules for repairs and replacements of code components.

APPENDIX II INSTRUMENT REVIEW - CONTAINMENT AND BASEMAT

CURRENT INITIATIVES COLUMN 6	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
See Part 50 proposed Rule Making 45-FR-81602, may impact Appendix A	Additional ISI methods, see Column 8 comments.	Wall thickness measurements; NDE examination for extent of reinforcing steel corrosion; maintenance records.	Dome, wall and base slab reinforcing steel, steel liner over dome and wall, steel liner over base slab are the principal concerns. Current ISI methods are visual inspection and leak tests.
?	?		Record and documentation of tests and inspections. QA is to provide action necessary to provide adequate confidence that the facilities' SSC will perform satisfactorily. Aging management implied through documentation of tests and inspections.
EPRI has supported a study for PWR pre-stressed concrete containments and conventional reinforced containments to evaluate current inspection methods and determine the need for additional inspection to assure contained each tightness and structural integrity.	?		Periodic leak testing is designed to detect leaks when they are still of minor concern. Potentially significant degradation of the intake structures and spray ponds due to corrosion will be kept within acceptable limits by performing the periodic inspections outlined RG 1.127.
?	Implied in the design requirements.	Guidance is required to cover the residual life assessment of components. Wall thickness measurements; NDE examinations for extent of corrosion; maintenance records. Water chemistry analysis (oxygen concentration; acidity; presence of chlorides, bacteria, etc.)	Article CC 2140 states that the consideration of <u>deterioration</u> of material caused by the service is outside the scope of this subsection (i.e., concrete containment, CC). It is the responsibility of the <u>owner</u> to select material suitable for the conditions stated in the Design Specifications, with specific attention given to the effects of service conditions upon the properties of the material.

REGULATORY

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5
CONCRETE CONTAIN- MENTS (PWR) Continued (61)	CRACKING AND SPALLING	CFRs 10 CFR 50 Appendix A	Records GDC Criterion 50, 51, 52, 53, and 54 address reac- tor containment, e.g., fracture prevention, con- tainment leak testing, provisions for testing and inspection and containment system.	Yes
(62)		Appendix B	Provides Quality Assur- ance requirements for document control, design control, construction control and operations	Yes
(63)		Appendix J	Defines primary contain- ment leak test require- ments, i.e., Types A, B, and C.	Yes
(64)		ACI's ACI 349-85 Appendix A Thermal Considerations	Contains criteria that address the effects the gradient temperatures, distribution, time dependent variations of temperatures for thermal strains associated with normal operating condi- tions and accident condi- tions and thermal stress in considering stiffness and rigidity and restrain of the structure.	Provides limits for operating tem- peratures of con- crete, i.e., 150°F normal areas not to exceed 200°F. Records and calcu- lation of design.

APPENDIX II INSTRUMENT REVIEW - CONTAINMENT AND BASEMAT

CURRENT INITIATIVES COLUMN 6	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
?	?	Periodic monitoring of tendon forces. Concrete (dome and wall) and base slab are the principal concerns. The aging issue stresses and aggressive environment and chemical reactions.	
?	?		Record and documentation of tests and inspections. QA is to provide action necessary to provide adequate confidence that the facilities' SSC will perform satisfactorily. Aging management implied through documentation of tests and inspections.
EPRI has supported a study for PWR pre-stressed concrete containments and conventional reinforced containments to evaluate current inspection methods and determine the need for additional inspection to assure contained each tightness and structural integrity.			Periodic leak testing is designed to detect leaks when they are still of minor concern.
?	N/A	N/A	For accident or other short term conditions concrete surface temperatures may reach 300°F. Local areas may be allowed to reach 650°F in the event of a steam or water jet from pipe failures.

REGULATORY 1

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5
CONCRETE CONTAIN- MENTS (PWR) Continued	CRACKING AND SPALLING Continued			
(65)		ACI 349 Appendix C Special provi- sions for impulse and impactive effects.	Provides Design Consider- ation/Criteria for nuclear safety-related concrete structure for impulsive and impactive loads that <u>must be</u> com- bined with other loads (dual and level loads specified in ACI-349, Section 9.1).	Yes, Records of Design
(66)		ACI 349.IR-80 ACI 349 Committee Report	This Standard covers re- inforced concrete design for thermal effects on NPP structures intended to conform to the general provisions of Appendix A ACI 349, it covers crack- ing, reinforcement, structural design, temperature gradients, thermal properties and thermal stresses. It is intended to complement Appendix A, ACI-349.	
(67)		ASME BPVC Section III Division 2 Subsection CC	Establishes the rules for material, fabrication, construction, examination and testing for pre- stressed and reinforced concrete containments. Containments that have a design greater than 5 psig and classified as concrete containments (CC) shall be constructed to these rules.	None explicit; yet provides guidance and sets forth rules for repairs and replacements of code components.

APPENDIX II INSTRUMENT REVIEW - CONTAINMENT AND BASEMAT

CURRENT INITIATIVES COLUMN 6	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
?	N/A	N/A	<p>Impactive and impulsion loads are treated separately in the appendix/standard because of the effects as well as the response characteristics of the structural elements subjected to the loads.</p> <p><u>impactive loads</u>: time dependent loads due to collision of masses associated with finite amounts of kinetic energy.</p> <p><u>impulsive loads</u>: those <u>not</u> associated with collision of masses including but not limited to, jets impingement, blast pressure, compartment pressurization and pipe whip - restraint reaction.</p> <p>Provides supplemental information for structures, and axisymmetric structures.</p>
?	Implied in the design requirements.	<p>Guidance is required to cover the residual life assessment of components.</p> <p>Wall thickness measurements; NDE examinations for extent of corrosion; maintenance records.</p> <p>Water chemistry analysis (oxygen concentration; acidity; presence of</p>	<p>chlorides, bacteria, etc.)Article CC 2140 states that the consideration of <u>deterioration</u> of material caused by the service is outside the scope of this subsection (i.e., concrete containment, CC). It is the responsibility of the <u>owner</u> to select material suitable for the conditions stated in the Design Specifications, with specific attention given to the effects of service conditions upon the properties of the material.</p>

Standard reapproved in 1986.

REGULATORY II

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5
CONCRETE CONTAIN- MENTS (PWR) Continued (68)	CREVICE CORROSION AND PITTING	CFRs 10 CFR 50 Appendix A	Records GDC Criterion 50, 51, 52, 53, and 54 address reac- tor containment, e.g., fracture prevention, con- tainment leak testing, provisions for testing and inspection of containment.	Yes
(69)		Appendix B	Provides Quality Assur- ance requirements for document control, design control, construction control and operations control, design control and quality assurance.	Yes
(70)		Appendix J	Defines primary contain- ment leak test require- ments, i.e., Type A, B, and C tests.	Yes
(71)		ASME BPVC Section III Division 2 Subsection CC	Establishes the rules for material, fabrication, construction, examination and testing for pre- stressed and reinforced concrete containments. Containments that have a design greater than 5 psig and classified as concrete containments (CC) shall be constructed to these rules.	None explicit; yet provides guidance and sets forth rules for repairs and replacements of code components.

APPENDIX II INSTRUMENT REVIEW - CONTAINMENT AND BASEMAT

CURRENT INITIATIVES COLUMN 6	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
See Part 50 proposed Rule Making 45-FR-81602, may impact Appendix A		Improved NDE methods are needed for evaluation of containment facilities.	Principal concern is posttensioning tendon wire or strand.
			Record and documentation of tests and inspections. QA is to provide action necessary to provide adequate confidence that the facilities' SSC will perform satisfactorily. Aging management implied through documentation of tests and inspections.
EPRI has supported a study for PWR prestressed concrete containments and conventional reinforced containments to evaluate current inspection methods and determine the need for additional inspection to assure contained each tightness and structural integrity.			Periodic leak testing is designed to detect leaks when they are still of minor concern.
?	Implied in the design requirements.	Guidance is required to cover the residual life assessment of components. Wall thickness measurements; NDE examinations for extent of corrosion; maintenance records. Water chemistry analysis (oxygen concentration; acidity; presence of chlorides, bacteria, etc.)Article CC 2140	states that the consideration of <u>deterioration</u> of material caused by the service is outside the scope of this subsection (i.e., concrete containment, CC). It is the responsibility of the <u>owner</u> to select material suitable for the conditions stated in the Design Specifications, with specific attention given to the effects of service conditions upon the properties of the material.

REGULATORY I

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5
CONCRETE CONTAIN- MENTS (PWR) Continued (72)	MICROBIOLOGICAL INDUCED CORROSION Continued	ACI STANDARDS 349-85 Appendix B		
(73)	HYDROGEN EMBRITTELEMENT	10 CFR 50 Appendix A	Records	Yes
(74)		Appendix B	Provides Quality Assur- ance requirements for document control, design control, construction control and operations	Yes
(75)	INSPECTION & TESTING	ASME BPVC Section XI Subsection IWL	Provides rules for concrete containment. Preservice, inservice and repairs of class cc components.	ISI records. For example, provides ISP schedule 1,3,5 and 10 year inter- vals. See arti- cles IWL-2410 2420, and 2421, included in 2421- two power units are requirement for 30 & 35 years
(76)		R.G 1.35 Rev. 3	Guide describes a basis for acceptable ISI and surveillance program for ungrouted tendons in pre- stressed concrete containment structures. Provides regulatory position for the ISI including, <u>frequency</u> , sampling, areas of inspection, criteria, and scope. General regula- tory position indicated that the containments shall be designed so that the anchor hardware is <u>accessible for inservice inspection.</u>	Yes. Test records/ documentation

APPENDIX II INSTRUMENT REVIEW - CONTAINMENT AND BASEMAT

CURRENT INITIATIVES COLUMN 6	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
---------------------------------	-------------------------	----------------------------------	----------------------

Principal concern is posttensioning tendon wire or strand.

Water chemistry analysis (oxygen concentration; acidity; presence of chlorides and sulfates) coolant level; flow rate.

Principal concern is posttensioning system anchorage.

Record and documentation of tests and inspections. QA is to provide action necessary to provide adequate confidence that the facilities' SSC will perform satisfactorily. Aging management implied through documentation of tests and inspections.

Not current approval by NRC. Issued in Dec 31, 1988, addenda of Section XI

?

?

Provides for examination IWL-2000. Provides for acceptance standards IWL-3000.

Inservice examination covered in Article IWL 3200.

RG provides details for inspection interval in years, i.e., 1, 3 and 5 initially and 5 years thereafter and sampling percentage fraction of the posttensioning system. Visual inspection requirements are described including the exterior surfaces which "should be examined to detect areas of large spall, severe scaling, D-cracking in an area of 25 square feet or more, other surface deterioration or disintegration, or grease leakage." ACI 201.1R-60 is identified as the appropriate standard for definitions, e.g., scaling and D-cracking. RG 1.35.1 notes in B. Discussions that "of particular concern in the case of prestressed concrete containment structures is the possible degradation of prestressing tendon system by corrosion." RGs 1.35 and 1.90 are formulated to achieve this basic objective.

REGULATORY 1

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5
CONCRETE CONTAIN- MENTS (PWR) Continued (77)	INSPECTION & TESTING Continued	1.35.1	Provides expanded guid- ance to RG 1.35 for determining prestress forces used for ISI on prestressed concrete containments and dis- cusses the regulatory position on: the measurement of prestress- ing force, the determina- tion of prestressing losses, the grouping of tendons, and the construction of (tendon) tolerance bands.	Yes. Test records/ documentation
(78)		1.90	Guide provides for acceptable ISI of pre- stressed concrete containments with grouted tendons. Gives the regulatory position for the visual examinations, the interval of inspec- tion in years, sampling routines for both Alternative A and B, sampling areas and reportable conditions.	Yes. Test records/ documentation
METAL CON- TAINMENTS (PWR) (79)	CORROSION	CFRs 10 CFR 50 Appendix A	Ductility of two-way slabs and response to LOCA or high-energy line breaks.	Yes
(80)		Appendix B	Provides Quality Assur- ance requirements for document control, design control, construction control and operations	Yes

APPENDIX II

INSTRUMENT REVIEW - CONTAINMENT AND BASEMAT

CURRENT INITIATIVES COLUMN 6	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
		Does not consider time extension beyond 40 years	Discusses the effects of shrinkage of concrete, the effect of concrete creep, the effect of relaxation of prestressing steel, the losses caused by tendon degradation and the effects of variations in the temperature. Understanding of these factors can enhance aging management.
	Provides for aging management by requiring that some conditions of the visual inspections are reportable, e.g., crack patterns at structurally critical areas. Also reportable conditions are required for test tendons that indicate a prestress force below the RG acceptable band.		
See Part 50 proposed Rule Making 45-FR-81602, may impact Appendix A		Wall thickness measurements; NDE examinations for extent of corrosion.	Principal areas of concern are shell welds and base metal, discontinuities in the shell, such as hatches and penetrations and the steel bottom of shell embedded in concrete. Record and documentation of tests and inspections. QA is to provide action necessary to provide adequate confidence that the facilities' SSC will perform satisfactorily. Aging management implied through documentation of tests and inspections.

REGULATORY 1

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5
METAL CON- TAINMENTS (PWR) Continued (81)	CORROSION Continued	Appendix J	Defines primary con- tainment leak test requirements.	Yes
(82)		B.49 GSI	Design criteria for corrosion prevention and in-service testing.	
(83)	CRACKS AND SPALLING	10 CFR 50 Appendix A	Records GDC Criterion 50, 51, 52, 53, and 54 address reac- tor containment, e.g., fracture prevention, con- tainment leak testing, provisions for testing and inspection and containment system.	Yes
(84)		Appendix B	Records document control, design control and quality assurance.	Yes

APPENDIX II

INSTRUMENT REVIEW - CONTAINMENT AND BASEMAT

CURRENT INITIATIVES COLUMN 6	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
<p>See 51 FR 39528 Proposed Rule Making: to update criteria and clarify questions regarding leakage rate testing of LWRs. See 53 FDR 5985 Alternate Method for Leak Rate Testing. Also see 53 FR 45890 which amended Appendix J and allows statistical data analysis acceptable to the NRC, i.e., the Mass Point method, ANSI/ANS 56.8-1981.</p>	<p>Approach to evaluate effects of long-term corrosion on reinforcements, steel liner/concrete contacts, steel surface in contact with water.</p>		<p>A general visual inspection is required of exterior and interior surfaces before a Type A leak test. Deterioration must be corrected before testing. Applies to assessable area structures and components.</p> <p>Issue also deals with inservice inspection of all types of containments. Designated a "Licensing Issue."</p> <p>Principal area of concern is the Base Slab Concrete. The use of concretes made with adequate entrained air and aggregates with a competent service history are expected to produce concrete with good-to-excellent resistance to freeze - obtund damage (ACI-306-83, "Recommended Practice for Cold Weather Concreting")</p> <p>Record and documentation of tests and inspections. QA is to provide action necessary to provide adequate confidence that the facilities' SSC will perform satisfactorily. Aging management implied through documentation of tests and inspections.</p>

REGULATORY :

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5
METAL CON- TAINMENTS (PWR) Continued (85)	CRACKS AND SPALLING Continued	Appendix J	Defines primary con- tainment leak test requirements.	Yes
(86)		ACIs ACI 349-85 Appendix A Thermal Considerations	Contains criteria that address the effects the gradient temperatures, distribution, time dependent variations of temperatures for thermal strains associated with normal operating condi- tions and accident con- ditions and thermal stress in considering stiffness and rigidity and restrain of the structure.	Provides limits for operating tem- peratures of con- crete, i.e., 150°F normal areas not to exceed 200°F. Records and calcu- lation of design.
(87)		ACI 349 Appendix B Steel Embedments	Covers general require- ments and loading combi- nations for embedded steel, e.g., shear, com- pression and bending.	Yes - Records of Design
(88)		ACI 349 Appendix C Special provi- sions for impulse and impactive effects.	Provides Design Consider- ation/Criteria for nuclear safety-related concrete structure for impulsive and impactive loads that <u>must be</u> com- bined with other loads (dual and level loads specified in ACI-349, Section 9.1).	Yes, Records of Design

APPENDIX II INSTRUMENT REVIEW - CONTAINMENT AND BASEMAT

CURRENT INITIATIVES COLUMN 6	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
	EPRI has supported a study for PWR pre-stressed concrete containments and conventional reinforced containments to evaluate current inspection methods and determine the need for additional inspection to assure contained each tightness and structural integrity.		Periodic leak tests, designed to detect leaks when they are still of minor concern.
?	N/A	N/A	For accident or other short term conditions concrete surface temperatures may reach 300°F. Local areas may be allowed to reach 650°F in the event of a steam or water jet from pipe failures.
?	N/A	N/A	Notes that fabrication and installation shall be in accordance with good practice to avoid "detrimental cracking or spalling of concrete or excess stress in the embedment anchors."
?	N/A	N/A	<p>Impactive and impulsion loads are treated separately in the appendix/standard because of the effects as well as the response characteristics of the structural elements subjected to the loads.</p> <p><u>impactive loads</u>: time dependent loads due to collision of masses associated with finite amounts of kinetic energy.</p> <p><u>impulsive loads</u>: those <u>not</u> associated with collision of masses including but not limited to, jets impingement, blast pressure, compartment pressurization and pipe whip - restraint reaction.</p>

REGULATORY I

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5
METAL CON- TAINMENTS (PWR) Continued (89)	CRACKS & SPALLING Continued	ACI 349.IR-80 ACI 349 Committee Report	This Standard covers re- inforced concrete design for thermal effects on NPP structures intended to conform to the general provisions of Appendix A ACI 349, it covers crack- ing, reinforcement, structural design, temperature gradients, thermal properties and thermal stresses. It is intended to complement Appendix A, ACI-349.	
CONTAINMENT AND BASEMAT (90)	INSPECTION AND TEST (SURVEILLANCE) GENERAL ISSUE TO ALL DESIGN OF CONTAINMENT	TS 3/4.6	Provides limiting condi- tions and surveillance requirements for various facets of containment and basemat for concrete or metal containment associ- ated with PWRs and BWRs. Typically the technical specification address integrity, leakage, air temperatures, internal pressure, ventilation systems, and structural integrity. For example, the specification requires that containment integrity be maintained; without primary contain- ment the plant must be restored to this condi- tion within the hour, or if not the plant must be at hot standby in 6 hours and it cold shutdown within 30 hours.	Records for/of Plant Performance.

APPENDIX II INSTRUMENT REVIEW - CONTAINMENT AND BASEMAT

CURRENT INITIATIVES COLUMN 6	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
Standard reapproved in 1986.			Provides supplemental information for structures, and axisymmetric structures.
?	N/A	N/A	With minor exceptions the specification do not specifically address NPAR defined aging. The limit- ing conditions and companion surveillance require- ments tend to control aging and force repairs and maintenance of the containment facilities. An example of the minor exceptions, include inspecting tendons for broken or damage wires or strands, and examination of tendons for aging issues of corrosion and cracks.

REGULATORY 1

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5
CONTAINMENT AND BASEMAT Continued (91)	INSPECTION AND TESTING	ASME BVPV, Section XI, Subsection IWE	Provides for visual inspection of accessible containment shell, vents, flanges, and penetrations.	ISI examination and inspection records.
(92)		GSIs II.B.5(3) II.B.7 II.B.8	Effects of hydrogen burn on containment structure.	
(93)		II.E.4 A.23	Containment design to support integrity checks, purging, and isolation capabilities.	
(94)	GENERAL ISSUES THAT APPLY TO ALL TYPES OF CONTAINMENTS	ASME NQA-2-1989 Edition Part 2.5 Section 8	Quality Assurance Requirements for Concrete construction.	None explicit; yet provides require- ments for repairs and replacements of concrete components.
(95)		Part 2.5 Section 9	Quality Assurance Requirements for Steel construction.	None
(96)		SRPs 3.5.1 (Internally Generated Missiles)	General Design Criteria (GDC) 4: Environmental and Missile Design Basis and 2: Design Basis for Protection against Natural Phenomena.	None
(97)		3.5.2 (Externally Generated Missiles)		

APPENDIX II INSTRUMENT REVIEW - CONTAINMENT AND BASEMAT

CURRENT INITIATIVES COLUMN 6	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
The NRC is currently reviewing the requirements of IWE - changes are expected in CFR to mandate ASME requirements.	NRC approved of code inspection methods.	NRC approval of code inspection methods.	
	Determine vulnerability of aged containment structure to hydrogen burn/explosion relative to as-built structure.		Effects of hydrogen burn on aged containment not evaluated.
	Recommend requirements for testing to detect aging of containment structures.		Aging not specifically evaluated; issue directed more towards testing and maintaining containment integrity.
	None	Deterioration limits should be determined.	Explicit guidelines required for replacement of components; long-term deterioration is not covered.
	None	Deterioration limits should be determined.	Explicit guidelines required for finishing and repairs and replacement of components. Long-term deterioration and residual life assessment techniques are necessary.
?	None	?	Evaluate long-term deterioration limits due to environment.

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5
CONTAINMENT AND BASEMAT Continued (98)	GENERAL ISSUES THAT APPLY TO ALL TYPES OF CONTAINMENTS	3.2 (Seismic Classification)	GDC 2	None
(99)		3.3 (Wind and Tornado Loadings)	GDC 2	None
(100)		3.4 (Flood Protec- tion and Analy- sis Procedures)	GDC 2	None
(101)		3.8.1 (Concrete Containment)	GDC 1, 2, 4, 16, and 50. Also refer to ASME Boiler and Pressure Vessel Code Section III Division 2.	
(102)		3.8.2 (Steel Containment)	GDC 1, 2, 4, 16, and 50. Also refer to ASME Boiler and Pressure Vessel Code Section III Division 1.	
(103)		3.8.3 (Concrete and Steel Internal Structure of Steel or Concrete Containments)	GDC 1, 2, 4, 5 and 50. Also refer to ASME Sec- tion III Divisions 1 and 2, ACI 349 AISC and ANSI N45.2.5.	
(104)		3.8.4 (Other Seismic Category I Structures)	GDC 1, 2, 4 and 5. Also refer to ASME Section III Division 1, ANSI N210-76, ACI 349, AISC and 10 CFR 50. § 50.55a	
(105)		3.8.5 (Foundations)	GDC 1, 2, 4, and 5. Also refer to ACI 349, AISC and 10 CFR 50. § 50.55a	

APPENDIX II INSTRUMENT REVIEW - CONTAINMENT AND BASEMAT

CURRENT INITIATIVES COLUMN 6	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
?	None	?	
?	None	?	
?	None	?	

REGULATORY 1

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5
CONTAINMENT AND BASEMAT Continued (106)	GENERAL ISSUES THAT APPLY TO ALL TYPES OF CONTAINMENTS		CFRS 10 CFR 50 Appendix A GDC Criterion 1: Quality Standards and Records	Records - Quality product in keeping with the required safety function.
(107)		Criterion 2: Design Bases for Protection Against Natural Phenomena	Designs to withstand the effects of natural phenomena without loss of capability to perform their safety functions.	
(108)		Criterion 4: Environmental and Missile Design Bases	Designed to accommodate the effects of and to be compatible with the envi- ronmental conditions and appropriately protected against dynamic effects.	
(109)		Criterion 5: Sharing of Structures, Systems, and Components	Structures, systems and components important to safety shall not be shared among nuclear power units unless their ability to perform safety functions is ensured.	
(110)		Criterion 16: Containment Design	Designed to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment.	Limited
(111)		Criterion 50: Containment Design Basis	Containment structure and its internal compartments can accommodate, without exceeding the design leakage rate, and with sufficient margin, the calculated pressure and temperature conditions from any loss-of-coolant accident.	

APPENDIX II
INSTRUMENT REVIEW - CONTAINMENT AND BASEMAT

CURRENT INITIATIVES COLUMN 6	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
---------------------------------	-------------------------	----------------------------------	----------------------

N/A

N/A

REGULATORY 1

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5
CONTAINMENT AND BASEMAT Continued (112)	GENERAL ISSUES THAT APPLY TO ALL TYPES OF CONTAINMENTS		Criterion 57: Closed System Isolation Valves	Containment structure and its internal compartments can accommodate, without exceeding the design leakage rate, and with sufficient margin, the calculated pressure and temperature conditions from any loss-of-coolant accident.
(113)		RGs 1.29: Seismic Design Classification	Defines design methods for Seismic conditions. No explicit reference to aging; provides only construction guidelines.	Limited
(114)		1.60: Design Response Spectra for Seismic Design of NPPs		
(115)		1.61: Damping Values of Seismic Design of NPPs		
(116)		1.59: Design Basis Floods for NPPs	Design methods for other environmental conditions; only construction guidelines.	Limited
(117)		1.102: Flood Protection for NPPs	Design methods for other environmental conditions; only construction guidelines.	Limited

APPENDIX II
INSTRUMENT REVIEW - CONTAINMENT AND BASEMAT

CURRENT INITIATIVES COLUMN 6	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
	N/A	N/A	The RG assists aging and life extension by providing acceptance method for approved materials and construction.
	N/A	N/A	
	N/A	N/A	

REGULATORY 1

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5
CONTAINMENT AND BASEMAT Continued (118)	GENERAL ISSUES THAT APPLY TO ALL TYPES OF CONTAINMENTS	1.115: Protection Against Low- Trajectory Turbine Missiles	N/A	Limited
(119)		1.117: Tornado Design Classification	N/A	Limited
(120)		1.35: Inservice Inspection of Ungouted Tendons in PCC Structures	Effects of fatigue or creep of concrete struc- tures on posttensioning tendon wire or strand.	Limited
(121)		1.70: Standard Format and Content of Safety Analysis Reports for NPPs (1.70 - 1.70.38).	Safety Analysis Reports for various components structures and systems in a NPP.	Limited
(122)		1.90: Inservice Inspection of PCC Structures With Grouted Tendons	ISI requirements to adhere to approved standards and codes.	Limited

APPENDIX II
INSTRUMENT REVIEW - CONTAINMENT AND BASEMAT

CURRENT INITIATIVES COLUMN 6	AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
---------------------------------	-------------------------	----------------------------------	----------------------

Appendix III

Regulatory Instrument Review for Reactor Coolant (or Recirculating) Pumps

APPENDI REGULATORY INSTRUMENT REVIE

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVES COLUMN 6
REACTOR COOLANT PUMPS	VIBRATION, SYSTEM OPERATING TRANSIENTS, AND STRESS	CFRs			
(1)		10 CFR 50, Appendix A, General Design Criteria (GDC)	Provides design cri- teria for the test- ing and performance requirements for various plant components.	Yes, records.	None
(2)		Appendix A, GDC 2	Design of com- ponents should consider natural phenomena such as earthquakes, torna- does and floods.	Existing design is based on histori- cally available data at the time of the original license; the data may have to be modified before license renewal.	None
(3)		SRPs 3.9.3 and 3.9.3 Appendix A	Must meet 10 CFR §50.55a and 10 CFR 50, Appendix A, GDC 1, 2, 4, 14, and 15 as it relates to components satisfy- ing quality stan- dards, effects of normal or accident conditions; meet ASME Section III requirements for pressure boundaries.	Yes. The SRP, through review of the plant's SAR, provides a record of plant status and yearly update of status.	?
(4)	VIBRATION	RGs Design Guide (DG) 1008	Draft guide dis- cussed NRC regula- tory position on the treatment of RCP seal failures.	Yes	See draft guide 1008.

General Notes:

1. A "?" indicates further study/investment is needed.
2. For the GSIs, "resolved" means the generic safety issue is resolved, not necessarily the aging issue.
3. For the meaning of abbreviations, acronyms, and initialisms, used throughout, see Acronyms/Abbreviations and xiii of the report.

K III

W - REACTOR COOLANT PUMPS

AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
Not applicable, since the criteria do not change over time.	The existing design and its criteria (generic or specific) should be made to satisfy the currently acceptable testing and performance standards for extended life of the plant.	Does the existing design meet the currently accept- able aging needs of the plant and the testing requirements for life extension of the plant?
Design require- ments may have to be changed over time, since cer- tain "natural" events may alter the design basis.	Incorporation of the current known conditions as applicable into the design criteria will be required.	Generic or site specific research based on currently available data may be necessary.
None	?	Does not address specific issues; other criteria must be followed. Components which are part of the reactor coolant pressure boundary must meet the requirements for Class 1 components in ASME Section III.
None	None	A seal failure may be classified as LOCA when the seal leakages exceeds the capacity of normal make up systems; RCP LOCA could present serious consequences in plant operations. See generic issues 23 and draft NUREG 1401, "Regulatory Analysis for Generic Issue 23, Reactor Coolant Pump Seal Failures."

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVES COLUMN 6
REACTOR COOLANT PUMPS (contd)	VIBRATION (contd)	RGs (contd)			
(5)		RG 1.14	RG address reactor coolant pump fly-wheel integrity. Provides regulatory position for: <ul style="list-style-type: none"> • material and fabrication • design • testing • inspection. Inspection includes NDE in accordance with ASME III, Par. NB-2540 and NB-2530.	Provides for ISI of high stress concentration on approximately 3 yr. interval, i.e., records for LE and LR.	See comments.
(6)	ALL ISSUES Pumps	RG 1.28 (Rev. 3)	Provides periods for maintaining records of manufacturing that may be used for repair and maintenance of the component, e.g., lifetime retention records are required including: <ul style="list-style-type: none"> a) manufacturing <ul style="list-style-type: none"> • code reports • design reports • materials • repairs • NDE • procurements b) installation <ul style="list-style-type: none"> • welding • heat treatment • cleaning procedures • fabrication procedures c) preoperational tests <ul style="list-style-type: none"> • results • procedures • maintenance activities. 	Yes, records.	?

AGING NEEDS
COLUMN 7

LIFE EXTENSION NEEDS
COLUMN 8

COMMENTS
COLUMN 9

None

None

RG applies to flywheel integrity after January 1, 1976.

The NRC Staff shall be notified for evaluation and examination of the inspection results if an increase in flaw size or flaw growth rate is greater than the predicted service life.

None

None

Also includes time retention intervals of 3 and 10 yrs for selective important activities.

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVES COLUMN 6
REACTOR COOLANT PUMPS (contd)	ALL ISSUES (contd)	RGs (contd)			
(7)		RG 1.29	Specifically mentions that cooling water and seal water systems required for reactor coolant system components, such as the <u>reactor coolant pumps</u> shall be designed to withstand the effects of an SSE.	Yes	?
(8)		RG 1.33	Provides NRC guidance applicable to Appendix B 10 CFR-50 for NPP operating procedures. Specifically Appendix A of RG 1.33 stipulates which <u>safety-related activities</u> should be covered by written procedures including: <ul style="list-style-type: none"> • main steam system • water systems. Also included are <u>maintenance activities</u> : <ul style="list-style-type: none"> • general maintenance • preventative maintenance • repair or replacement. 	Yes	?

AGING NEEDS
COLUMN 7

LIFE EXTENSION NEEDS
COLUMN 8

COMMENTS
COLUMN 9

None

None

Does not specifically mention aging per se; however, the RG requirements to withstand postulate earthquake conditions implies a relationship of long term reliability of the NPP components.

None

None

Does not specifically mention aging or the management of aging; however, written procedures that provide "properly preplanned and performed (maintenance) in accordance with written procedures, documented instructions or drawings appropriate to the circumstance," (quote from paragraph 9, pg. 1.33-7, RG 1.33) will enhance the management of aging and its consequences.

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVES COLUMN 6
REACTOR COOLANT PUMPS (contd)	ALL ISSUES (contd)	RGs (contd)			
(9)		RG 1.58 Rev. 1	Provides the regula- tory position applicable to ANSI N45.2.6 with exceptions for the qualifications of NPP examination, testing and inspec- tion staff, e.g., RT, UT, eddy cur- rent testing and leak testings.	Yes, adequately trained staff contribute to the proper maintenance; thus, contributing to the life exten- sion of components.	?
(10)		RG 1.8 Rev. 2	Provides for NRC guidance on qualifi- cation and training of NPP staff. Included shift supervisors, senior operators, and licensed operators, i.e., staff that could have an impact on the operations of pumps or valves.	Yes, qualified operators manage aging and its consequences more effectively than those poorly trained.	?
(11)		RG 1.68 Rev. 2	Provide guidance for initial preoper- ational tests and start-up tests for SSC, e.g., the reac- tor coolant system, ESF systems, heat removal, etc. The design shall respond to anticipated tran- sients and postu- lated accidents.	Yes, records of initial plant con- ditions. Start of operations and corrective action taken to resolve problems.	?

AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
None	None	<p>The RG was intended for use for those plants (construction permits and operating licenses) after September 30, 1980.</p> <p>Does not specifically mention aging or aging management; however, states that the NRC's regulations require an applicant or licensee to establish a QA program that ensures adequate indoctrination and training of ISI and IST staff for preoperational startup and operational phases of NPPs.</p>
None	None	<p>References ANSI/ANS 3.1 1981 "Selection, Qualification and Training of Personnel for NPP's," and provides exceptions or acceptable position to ANS 3.1.</p>
None	None	<p>RG does not specifically address aging or aging management; yet it does provide implied aging management through verification of the plant status. It is assumed that the tests would discover aging conditions that require correct action before full operation could commence, e.g., excessive vibration of pumps and valves.</p>

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVES COLUMN 6
REACTOR COOLANT PUMPS (contd)	ALL ISSUES (contd)	RGs (contd)			
(12)		RG 1.84	Provides a list of ASME Code cases applicable to design including piping and valves, and fabrication, e.g., welding of valve seats and repair of forgings that are acceptable to NRC staff.	This guide limits and controls the use of analytical procedures in conjunction with ASME Code Case N.411. The ASME Pressure Vessel Research Committee (PVRC) has developed a position that may be acceptable to both NRC and to ASME, providing a more flexible approach to analysis.	RG 1.84 is routinely updated. The current edition is Rev. 27, dated November 1990.
(13)		RG 1.85	Provides a list of ASME Code Cases for materials and testing acceptable to NRC staff.	Yes, see comments on RG 1.84.	RG 1.85 is routinely updated. The current edition is Rev. 27 dated November 1990.
(14)		RG 1.116	Provides methods acceptable to the NRC staff for the QA requirement applicable to installation, testing and inspection of mechanical equipment.	Yes Records	None
(15)	VIBRATION	GSIs 7	Flow-induced vibrations causing valve and pump failure due to high cycle fatigue, impact, and fretting.	None	None

AGING NEEDS
COLUMN 7

LIFE EXTENSION NEEDS
COLUMN 8

COMMENTS
COLUMN 9

None

None

Most Code Cases are eventually superseded by revisions to the Code and then annulled by action of the ASME. Some code cases are annulled because they are no longer acceptable or needed. The Code Case procedure provides aging management by offering alternate methods of design and repair not found in current codes.

None

None

Most Code Cases are eventually superseded by revisions to the Code and then annulled by action of the ASME. Some code cases are annulled because they are no longer acceptable or needed. The Code Case procedure provides aging management by offering alternate methods of design and repair not found in current codes.

None

None

Guide recognizes that not all Quality Assurance requirements are satisfied by the applicable codes, e.g., ASME BPVC Section III. The guide states "Code-Covered activities are primarily intended to assure the integrity of the pressure boundary of an item"; thus, this guide coupled with ANSI 45.2.8 standard and ASME codes provides the NRC staff position on QA required for testing and inspection of mechanical components and systems.

Failures Due to Flow-Induced Vibrations
* Dropped

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVES COLUMN 6
REACTOR COOLANT PUMPS (contd)	VIBRATION (system operaton transients)	T/S			
(16)		3/4.4.1	Provides limiting conditions for the reactor coolant loops and circulator systems for, e.g., operations startups, hot standby, hot shutdown, and cold shutdown.	Yes, provides long term safe operation of the collant loops.	?
(17)	ALL ISSUES	CFRs 10 CFR 50 Appendix A, GDC 2	Design of components should consider natural phenomena such as earthquakes, tornadoes and floods.	Existing design is based on historically available data at the time of the original license; the data may have to be modified before license renewal.	None
(18)	CORROSION	§50.55a Code and Standards and Appendix A, Criterion 1, 2, 4, 14, and 15.	Yes, original design must address corrosion.	Not specific, corrosion allowances may extend beyond the 40-year interval.	?
(19)		Appendix A, GDC 2	Design of components should consider natural phenomena such as earthquakes, tornadoes and floods.	Existing design is based on historically available data at the time of the original license; the data may have to be modified before license renewal.	None
(20)		RGs RG 1.36	Provides an acceptable method for implementing criteria for the selection and use of nonmetallic, thermal insulation that could promote SS cracking in austenitic SS.	Yes	?

AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
None	None	The TS do not specifically mention aging management; however, the limiting condition can provide the incentives to maintain pumps in sound manner. For example, for condition of hot standby the reactor shall have at least two reactor coolant loops <u>operable</u> , similar conditions apply to hot shutdown. Generally the safety implications of the TS limiting conditions promote both safety and aging management.
Design requirements may have to be changed over time, since certain "natural" events may alter the design basis.	Incorporation of the current known conditions as applicable into the design criteria will be required.	Generic or site specific research based on currently available data may be necessary.
Needs to address deterioration over time.	Needs to address deterioration over time.	Components in the pressure boundary must meet the (corrosion) requirements for Class I components in ASME Section III as called for by the regulations noted in 10 CFR 50, §50.55a, Codes and Standards.
Design requirements may have to be changed over time, since certain "natural" events may alter the design basis.	Incorporation of the current known conditions as applicable into the design criteria will be required.	Generic or site specific research based on currently available data may be necessary.
None	None	Fluid systems important to safety, e.g., coolant pumps, systems and valves, manufactured of austenitic SS type 3xx series, should be protected from leachable chlorides and from fluorides that promote SS cracking.

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVES COLUMN 6
REACTOR COOLANT PUMPS (contd)	CORROSION (contd)	RGs (contd)			
(21)		RG 1.37	Provides regulatory position for on-site cleaning of materials and components as related to cleaning of <u>fluid systems</u> , e.g., water systems. Chemical compounds that could contribute to intergranular cracking or SS cracking should not be used with SS or nickel based alloys.	Yes	?
(22)		RG 1.54	Provides for NRC acceptance of ANSI Standards (N101.4-1972 and N45.2 - 1971). These standards set forth the QA requirements for application of protective coatings for NPPs, including coatings for ferritic steels and stainless steels. Coating should perform under the environmental conditions found in an NPP, e.g., radiation.	Yes	?
(23)		Design Guide (DG) 1008	Draft guide discussed NRC regulatory position on the treatment of RCP seal failures.	Yes	See draft guide 1008.

AGING NEEDS
COLUMN 7

LIFE EXTENSION NEEDS
COLUMN 8

COMMENTS
COLUMN 9

None

None

Examples of nonacceptable chemical compounds are those containing chlorides, fluorides, lead, zinc, copper, sulfur or mercury where such elements are leachable or could be released by breakdown. Aging management is not explicitly mentioned; however, these procedures will promote management of the degradation issue of corrosion.

None

None

RG states that "Coating and cleaning materials used with SS should not be compounded from or treated with chemical compounds that could contribute to corrosion intergranular cracking or stress corrosion cracking," i.e., chlorides, fluorides, lead, zinc, copper, sulfur or mercury.

Also, certain known threshold combinations of material stress, material condition and operating (water chemistry) environment must be present. The control of the combinations of these factors as well as the monitoring of the water chemistry and operating temperature are important to minimize RCS corrosion potential.

None

None

A seal failure may be classified as LOCA when the seal leakages exceeds the capacity of normal make up systems; RCP LOCA could present serious consequences in plant operations.

See generic issues 23 and draft NUREG 1401, "Regulatory Analysis for Generic Issue 23, Reactor Coolant Pump Seal Failures."

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVES COLUMN 6
REACTOR COOLANT PUMPS (contd)	CORROSION (contd)				
(24)		ASME Section III Subsection NB, NC	Provides for design loads affecting the strength and integrity of the pressure boundary. See Comments Column 9.	None specific addressed.	ASME has recognized the need to expand treatment of corrosion in Sections III and IX of the BPVC. The PVRC now has assessments underway.
(25)		GSIs B.58	Reduce passive valve failures due to crud and corrosion.	None	None
(26)		ANS Standards 59.1	Provides for sizing and installation criteria affecting the strength and integrity of the pressure boundary.	None	None
(27)	FATIGUE -- LOW CYCLE FATIGUE -- THERMAL AND MECHANICAL	CFRs 10 CFR 50, Appendix A and B. Appendix A Criterion 1, 2, 14, and 15.	Provides design margins.	Not specific, design margin may support life extension beyond 40-year interval.	Visual inspections are required at sufficient intervals to detect leakage so retensioning of bolt can be accomplished prior to the accumulation of significant fatigue damage. (See Comments.)
(28)		SRPs 3.9.3 and 3.9.3 Appendix A	Must meet 10 CFR 50, Appendix B and 10 CFR 50, Appendix A, GDC 1, 2, 14, and 15 (also meet ASME Section III requirements for pressure boundaries).	Design margins provided. The SRP, through review of the plant's SAR, provides a record of plant status and yearly update of status.	?

AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
Implied in the design criteria.	Guidance is needed to cover the limits of component deterioration.	ASME Code for Class 1, 2, or 3 components (safety related pumps) required when corrosion is expected. Rules cover design and construction requirements but do not cover deterioration with time.
Identify aging and service wear effects.		Passive Mechanical Failure <u>* Resolved</u>
Not applicable, since the criteria do not change over time.	The existing design criteria should be made to satisfy the currently acceptable testing and performance standards for the extended life of the plant.	Does the existing design meet the currently acceptable aging needs of the plant and the performance specifications for life extension of the plant?
Needs to address deterioration over time.	Needs to address deterioration over time.	Fatigue analysis must meet ASME Section III requirements. NUMARC Rept. No. 90-09: "BWR Primary Coolant Pressure Boundary License Renewal Industry Report." Stress relaxation is mentioned as a potentially significant age-related degradation mechanism for recirculation pump and valve closure bolting due to potential loss of preload.
None	?	Evaluate transients used in the design and fatigue analysis; ASME Section III must be followed. The role of environmental assisted fatigue (EAF) could be significant and needs to be addressed.

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVES COLUMN 6
REACTOR COOLANT PUMPS (contd)	FATIGUE (contd)	RGs (contd)			
(29)		RG 1.14	RG address reactor coolant pump fly- wheel integrity. Provides regulatory position for: <ul style="list-style-type: none"> • material and fabrication • design • testing • inspection. Inspection includes NDE in accordance with ASME III, Par. NB-2540 and NB-2530.	Provides for ISI of high stress concen- tration on approxi- mately 3 yr. inter- val, i.e., records for LE and LR. Recirculation pump casings could be susceptible both to rapid crack growth from thermal fatigue and to failure from ther- mal embrittlement.	See comments.
(30)	THERMAL DISTORTION	RG 1.100	Provides NRC staff position on their use of IEEE Standard 344. "Rec- ommended Practice for Seismic Qualifi- cation of Class 1E Equipment for Nuclear Power Gen- erating Stations." The RG stipulates use of IEEE-344 to qualify mechanical equipment.	Yes	ASME is currently developing a standard for seismic qual- ification of mechani- cal equipment.
(31)		ASME Section III	Fatigue	Fatigue curve revisions.	Review of fatigue curves in Section III to determine if they can be revised to accommodate plant operation beyond the normal life of 40 years.
(32)	FATIGUE	GSIs C.11	Reduce fatigue failures in valves resulting from design error.	None	None

AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
None	Pump casings should be disassembled and inspected to prevent thermal fatigue cracks from growing to a size that could result in failure of the thermally embrittled component during the license renewal period.	RG applies to flywheel integrity after January 1, 1976. The NRC Staff shall be notified for evaluation and examination of the inspection results if an increase in flaw size or flaw growth rate is greater than the predicted service life.
None	None	Qualification of equipment provides aging management by establishing the suitability of the equipment for the postulated loads in a seismic event. The RG applies to plant granted a construction permit after June 30, 1988 and for plant docketed for operation after December 30, 1988.
Studies to assess the extent of fatigue are required.	Improved end-of-life projections, 40 years and beyond.	Studies to assess the magnitude of the effects of actual environmental conditions. Criteria for thermal embrittlement of PWR reactor coolant system components must include both temperature exposure and shift in the nil-ductility transition (NDT) temperature. The surfaces are more prone to crack initiation. These studies could result in need for Regulatory Guide, Regulatory Instrument, and/or Code revisions.
Identify active pumps and valves that need redesign and replacement.		Assessment of Failure and Reliability of Pumps and Valves. <u>* Resolved</u>

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVE COLUMN 6
REACTOR COOLANT PUMPS (contd)	FATIGUE (contd)	GSIs (contd)			
(33)		78	Monitoring of fatigue transient limits in reactor coolant systems.	None	None
(34)	WEAR AND EROSION	SRPs 3.9.3 and 3.9.3 Appendix A	Reference to ASME Section III and 10 CFR 50, Appendix A, GDC 1, 2, 4, 14 and 15.	Yes. The SRP, through review of the plant's SAR, provides a record of plant status and yearly update of status.	?
(35)		RGs Design Guide (DG) 1008	Draft guide dis- cussed NRC regula- tory position on the treatment of RCP seal failures.	Yes	See draft guide 1008.
(36)		ASME Section III: Subsection NB, NC	Provides for design (nominal) loads affecting the strength and integ- rity of the pressure boundary.	None explicitly; provides rules for testing and repair of code components. Requalification of components.	None
(37)		Section XI	Condition Assessment	Requalification of systems and components.	None
(38)	ALL ISSUES (including wear)	GSIs 4	Determine the end- of-life and main- tenance periodicity for valves and pumps.	None	None

AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
Methods, equipment to monitor fatigue in RCS.		<u>* Scheduled for prioritization</u>
None	?	Loading combinations, System operational transients, and Stress limits.
None	None	A seal failure may be classified as LOCA when the seal leakages exceeds the capacity of normal make up systems; RCP LOCA could present serious consequences in plant operations. See generic issues 23 and draft NUREG 1401, "Regulatory Analysis for Generic Issue 23, Reactor Coolant Pump Seal Failures."
Not applicable, since the criteria do not change over time.	Implied in the design criteria.	The design allowance for normal wear shall be consistent with the specified design life.
Condition Assessment Information.	Requalification of systems and components following expiration of operating license.	End-of-Life and Maintenance Criteria <u>* Resolved</u>

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVES COLUMN 6
REACTOR COOLANT PUMPS (contd)	WEAR	ANS Standards			
(39)		59.1	Provides for sizing requirements to accommodate the anticipated pump wear.	None explicitly; provides guidelines for testing and maintenance of code components.	None
(40)	SURVEILLANCE * TESTING • INSPECTION * MAINTENANCE AND REPAIRS * NDE	CFRs 10 CFR 50, Appendix A, General Design Criteria (GDC)	Provides design criteria for the testing and performance requirements for various plant components.	Yes, records.	None
(41)		SRPs 3.9.6	Must conform to 10 CFR 50, Appendix A, GDC 37, 40, 43, 46, and 54 and 10 CFR 50.55a (i.e., meet ASME Section III requirements for pressure boundaries).	Yes. The SRP, through review of the plant's SAR, provides a record of plant status and yearly update of status.	?
(42)		ASME OM Code-1990	Detection of changes in the hydraulic and mechanical condition of a pump relative to a reference set of parameters established previously.	When a reference or set of values may have been affected by repair, replacement, or routine servicing of a pump, a new reference value or set of values shall be determined or the previous value reconfirmed by an inservice test run before declaring the pump operable.	None. Replaces ASME Section XI. Subsection ISTB (Inservice Testing of Pumps).
			Record-keeping requirements and reports for trend projections.	Records and reports.	?

AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
Condition Assessment Information	Requalification of pumping systems and components following expiration of operating license.	Sizing and other technical specifications should be followed according to stated guidelines for extended life of the plant. Mechanical wear is potentially significant for the RCP seal flange.
Not applicable, since the criteria do not change over time.	The existing design and its criteria (generic or specific) should be made to satisfy the currently acceptable testing and performance standards for extended life of the plant.	Does the existing design meet the currently acceptable aging needs of the plant and the testing requirements for life extension of the plant?
Not applicable since the inspection criteria do not change with time.	The existing design should be made to meet the currently accepted operation and maintenance standards for life extension of the plant.	ASME Section III must be followed.
None	-	<p>The ASME Operational and Maintenance (OM) Code - 1990, Subsection ISTB: Inservice Testing of Pumps in Light-Water Reactor Plants provides the following:</p> <ul style="list-style-type: none"> Inservice Tests for Pumps: nominally every 3 months Reference Values: shall be determined from the results of preservice testing or from the results of the first inservice test; when a reference value or set of values may have been affected by repair, replacement, or routine servicing of a pump, a new reference value or set of values shall be determined or the previous value reconfirmed by an inservice test run before declaring the pump operable. Pumps Lacking Required Fluid Inventory: e.g., pumps in dry sumps, shall be tested at least once every 2 years.
Information and data for trend curves.	Records and reports.	Provide the utility with guidance as to the records and reports needed to support a license renewal request.

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVES COLUMN 6
REACTOR COOLANT PUMPS (contd)	SURVEILLANCE (contd)	ASME (contd)			
(43)		Section III NB-5000, NC-5000	Surface examination of weld joint Cate- gories A, B, C and D including adjacent base material for at least 1/2 in. on each side of the weld (as required by NB-5200 and NB-2500).	Welded joints exam- ined by the radio- graphic and either the liquid pene- trant or magnetic particle method.	None
(44)		IWB-5200 System Test Requirements: * System Leakage Test -- at desig- nated pressures and tempera- tures * System Hydrostatic Test.	Provides test requirements that verify design conditions.	Yes, records.	?
(45)		CFRs 10 CFR 50, Appendix A, General Design Criteria (GDC) 1, 4, 14, 15.	Provides design cri- teria for the qual- ity, testing and performance require- ments for various plant components, including the reac- tor coolant system.	Yes, records.	None

AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
Not applicable, since the criteria do not change over time.	The existing design and its criteria should be made to satisfy the currently acceptable inspection standards for extended life of the plant.	Nondestructive and nonintrusive examination and testing helps in the longer life of the affected components and support structures.
None	Systems tests should be correlated with system deterioration over time.	Repeated harsh testing can contribute to aging.
Not applicable, since the criteria do not change over time.	The existing design and its criteria (generic or specific) should be made to satisfy the currently acceptable testing and performance standards for extended life of the plant.	Does the existing design meet the currently acceptable aging needs of the plant and the testing requirements for life extension of the plant?

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVES COLUMN 6
REACTOR COOLANT PUMPS (contd)	SURVEILLANCE (contd)	CFRs (contd)			
(46)		Appendix A, GDC 37, 40, 43, 46.	The emergency core cooling system, the containment heat removal system, the containment atmos- phere cleanup sys- tem, the cooling water system, and the piping systems penetrating contain- ment shall be designed to permit appropriate periodic inspection of the key components in the systems.	Yes, inspection records from periodic ISI.	None
(47)	TESTING	ANS Standards 59.1	Provides design cri- teria for the <u>test- ing and performance requirements</u> for different pumps and combinations of pumps.	Yes, provides test performance records.	None
(48)	SURVEILLANCE	ANS Standards 59.1	Provisions for pump performance must meet the require- ments of Subsection ISTB in ASME OM Code-1990 (which replaces Subsec- tion IWV in Sec- tion XI of the ASME Boiler and Pressure Vessel Code.	Access to equipment for expected main- tenance shall be adequately provided as required.	None
(49)		TS 4.4.1.1	Provides surveil- lance requirements for coolant loops/systems for <u>hot standby, hot shutdown</u> , and cold shutdown.	Yes, provides long- term surveillance records.	?

AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
Not applicable since the testing criteria do not change over time.	The existing design should be made to meet the currently accepted testing standards for life extension of the plant.	None
Not applicable, since the criteria do not change over time.	The existing design criteria should be made to satisfy the currently acceptable testing and performance standards for the extended life of the plant.	Does not address specific issues; other criteria and design requirements must be followed. Provides provisions for structural integrity of the pumps.
Record-keeping requirements and reports for trend projections.	Records and reports.	Provide guidance as to the design documentation needed to support technical specifications and interface requirements.
None	None	The TS do not specifically mention aging management; however, surveillance requirements do promote aging management by providing minimums allowed for operable units. The pumps must be adequately maintained to provide the minimum operable pumps.

Appendix IV

Regulatory Instrument Review for Valves

APPENDIX
REGULATORY INSTRUMENTS

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVES COLUMN 6
SELECTED VALVES	VIBRATION, SYSTEM OPERATING TRANSIENTS, STRESS LIMITS	CFRs			
(1)		10 CFR 50, Appendix A, General Design Criteria (GDC)	Provides design cri- teria for the test- ing and performance requirements for various plant compo- nents, including all types of valves.	None	None
(2)		SRPs 3.9.3 and 3.9.3 Appendix A	Must meet 10 CFR §50.55a and 10 CFR 50, Appendix A, GDC 1, 2, 4, 14, and 15 as it relates to compo- nents satisfying quality standards, effects of normal or accident conditions; meet ASME Sec- tion III require- ments for pressure boundaries.	The SRP, through review of the plant's SAR, pro- vides a record of plant status and yearly update of status.	?
(3)		IEEE 382	Provides MOV valve actuator qualifica- tion tests for nor- mal expected opera- tional vibrations. "Vibration-Aging Test" for expected actuator qualified life includes repre- sentative system operating transients and other dynamic vibratory environments.	Yes, records.	?

General Notes:

1. A "?" indicates further study/investment is needed.
2. For the GSIs, "resolved" means the generic safety issue is resolved, not necessarily the aging issue.
3. For the meaning of abbreviations, acronyms, and initialisms, used throughout, see Acronyms/Abbreviations and xiii of the report.

IX IV
 NT REVIEW - VALVES

AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
Not applicable, since the criteria do not change over time.	The existing design and its criteria (generic or specific) should be made to satisfy the currently acceptable testing and performance standards for extended life of the plant.	Does the existing design meet the currently accept- able aging needs of the plant and the testing requirements for life extension of the plant?
None	?	Does not address specific issues; other criteria must be followed. Components which are part of the reactor coolant pressure boundary must meet the requirements for Class 1 components in ASME Section III.
None	May need to consider qualification testing beyond 40 years.	See comments for item 44 surveillance issue.

on pages xi, xii,

<u>COMPONENT COLUMN 1</u>	<u>ISSUE COLUMN 2</u>	<u>REG. INSTRUMENT COLUMN 3</u>	<u>AGING FEATURES COLUMN 4</u>	<u>LIFE EXTENSION FEATURES COLUMN 5</u>	<u>CURRENT INITIATIVE COLUMN 6</u>
SELECTED VALVES (contd)	ALL ISSUES Valves	RGs			
(4)		RGs 1.28 (Rev. 3)	Provides periods for maintaining records of manufacturing that may be used for repair and mainte- nance of the compo- nent, e.g., lifetime retention records are required including: a) manufacturing • code reports • design reports • materials • repairs • NDE • procurements b) installation • welding • heat treatment • cleaning procedures • fabrication procedures c) preoperational tests • results • procedures • maintenance activities.	Yes, records.	?

AGING NEEDS
COLUMN 7

LIFE EXTENSION NEEDS
COLUMN 8

COMMENTS
COLUMN 9

None

None

Also includes time retention intervals of 3 and
10 yrs for selective important activities.

Does not explicitly mention aging management;
however, the records noted are clearly required to
manage plant operations, including the aging of
components.

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVES COLUMN 6
SELECTED VALVES (contd)	ALL ISSUES (contd)	RGs (contd)			
(5)		RG 1.33	Provides NRC guidance applicable to Appendix B 10 CFR-50 for NPP operating procedures. Specially Appendix A of RG 1.33 stipulates which <u>safety-related activities</u> should be covered by written procedures including: <ul style="list-style-type: none"> • main steam system • water systems. Also included are <u>maintenance activities</u> : <ul style="list-style-type: none"> • general maintenance • preventative maintenance • repair or replacement. 	Yes	?
(6)		RG 1.58 Rev. 1	Provides the regulatory position applicable to ANSI N45.2.6 with exceptions for the qualifications of NPP examination, testing and inspection staff, e.g., RT, UT, eddy current testing and leak testings.	Yes, adequately trained staff contribute to the proper maintenance; thus, contributing to the life extension of components.	?
(7)		RG 1.8 Rev. 2	Provides for NRC guidance on qualification and training of NPP staff. Included shift supervisors, senior operators, and licensed operators, i.e., staff that could have an impact on the operations of pumps or valves.	Yes	?

AGING NEEDS
COLUMN 7

LIFE EXTENSION NEEDS
COLUMN 8

COMMENTS
COLUMN 9

None

None

Does not specifically mention aging or the management of aging; however, written procedures that provide "properly preplanned and performed (maintenance) in accordance with written procedures, documented instructions or drawings appropriate to the circumstance," (quote from paragraph 9, pg. 1.33-7, RG 1.33) does enhance the management of aging and its consequences.

None

None

The RG was intended for use for those plants (construction permits and operating licenses) after September 30, 1980.

Does not specifically mention aging or aging management; however, states that the NRC's regulations require an applicant or licensee to establish a QA program that ensures adequate indoctrination and training of ISI and IST staff for preoperational startup and operational phases of NPPs.

None

None

References ANSI/ANS 3.1 1981 "Selection, Qualification and Training of Personnel for NPP's," and provides exceptions or acceptable position to ANS 3.1.

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVES COLUMN 6
SELECTED VALVES (contd)	ALL ISSUES (contd)	RGs (contd)			
(8)		RG 1.68 Rev. 2	Provides guidance for initial preoperational tests and start-up tests for SSC, e.g., the reactor coolant system, ESF systems, heat removal, etc. The design shall respond to anticipated transients and postulated accidents.	Yes, records of initial plant conditions. Start of operations and corrective action taken to resolve problems.	?
(9)		RG 1.84	Provides a list of ASME Code cases applicable to design including piping and valves, and fabrication, e.g., welding of valve seats and repair of forgings that are acceptable to NRC staff.	Yes - RG 1.84 is routinely updated. The current edition is Rev. 27, dated November 1990. This guide limits and controls the use of analytical procedures in conjunction with ASME Code Case N.411. The ASME Pressure Vessel Research Committee (PVRC) has developed a position that may be acceptable to both USNRC and to ASME, providing a more flexible approach to analysis.	RG will be revised to reflect current approved NRC code cases.
(10)		RG 1.85	Provides a list of ASME Code Cases for materials and testing acceptable to NRC staff.	Yes, RG 1.85 is routinely updated. The current edition is Rev. 27 dated November 1990.	RG will be revised to reflect current approved NRC code cases.

AGING NEEDS
COLUMN 7

LIFE EXTENSION NEEDS
COLUMN 8

COMMENTS
COLUMN 9

None

None

RG does not specifically address aging or aging management; yet it does provide implied aging management through verification of the plant status. It is assumed that the tests would discover aging conditions that require correct action before full operation could commence, e.g., excessive vibration of pumps and valves.

None

None

Most Code Cases are eventually superseded by revisions to the Code and then annulled by action of the ASME. Some code cases are annulled because they are no longer acceptable or needed. The Code Case procedure provides aging management by provided alternate methods of design and repair not found in current codes.

None

None

Most Code Cases are eventually superseded by revisions to the Code and then annulled by action of the ASME. Some code cases are annulled because they are no longer acceptable or needed. The Code Case procedure provides aging management by provided alternate methods of design and repair not found in current codes.

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVES COLUMN 6
SELECTED VALVES (contd)	ALL ISSUES (contd)	RGs (contd)			
(11)		RG 1.116	Provides methods acceptable to the NRC staff for the QA requirement applicable to installation, testing and inspection of mechanical equipment.	Yes, records.	None
(12)		RG Designed Guide (DG) Task US 901-4	Provides NRC regulatory position that supplements existing ASME Section XI testing, e.g., valves that perform safety function, pressure isolation, containment isolation, should be tested.	Yes, records.	See draft guide issued 11/81.
(13)		IEEE Standard 382-1985	Provides scope for qualification of valve actuators for MOVs. Recognizes aging within contracts - defines: • aging • design life • installed life • qualified life • service conditions.	Yes, qualification of valve actuator can contribute to life extension of components.	?
(14)	VIBRATION	GSIs 7	Flow-induced vibrations causing valve and pump failure due to high cycle fatigue, impact, and fretting.	None	None
(15)	VIBRATION (system operation transients)	TS 3/4.4.1	Provides limiting conditions for the reactor coolant loops and circulator systems for, e.g., operations startups, hot standby, hot shutdown, and cold shutdown.	Yes, provides long term safe operation of the coolant loops.	?

AGING NEEDS
COLUMN 7

LIFE EXTENSION NEEDS
COLUMN 8

COMMENTS
COLUMN 9

None

None

Guide recognizes that not all Quality Assurance requirements are satisfied by the applicable codes, e.g., ASME BPVC Section III. The guide states "Code-Covered activities are primarily intended to assure the integrity of the pressure boundary of an item"; thus, this guide coupled with ANSI 45.2.8 standard and ASME codes provides the NRC staff position on QA required for testing and inspection of mechanical components and systems.

None

None

Guides presents 17 systems for PWR's and BWR Valves (systems) that should be included in a "Comprehensive" IST program."

None

Qualification cases may need to address radiation exposure beyond the current 40-year limit.

References IEEE-323 and 344 which both stipulate requirements for Class 1E equipment for NPP. Also references 10 CFR 100, energy.

The standard is designated for qualification of MOV with safety-related functions.

Failures Due to Flow-Induced Vibrations. (This GSI dropped by the NRC.)

None

None

The TS do not specifically mention aging management; however, the limiting condition can provide the incentives to maintain pumps in sound manner. For example, for condition of hot standby the reactor shall have at least two reactor coolant loops operable, similar conditions apply to hot shutdown. Generally the safety implications of the TS limiting conditions promote both safety and aging management.

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVE COLUMN 6
SELECTED VALVES (contd)	CORROSION	CFRs			
(16)		10 CFR 50, Appendix A, General Design Criteria (GDC)	Provides design criteria for the testing and per- formance require- ments for various plant components, including check valves and other MOVs.	Yes, promotes life extension.	None
(17)		ASME Section III Subsection NB, NC	Provides for design loads affecting the strength and integ- rity of the pressure boundary.	Yes, promotes life extension.	None
(18)		GSIs B.58	Reduce passive valve failures due to crud and corrosion.	Yes, promotes life extension.	None
(19)		RGs RG 1.36	Provides an accept- able method for implementing cri- teria for the selec- tion and use of nonmetallic, thermal insulation that could promote SS cracking in aus- tenitic SS.	Yes	?
(20)		RG 1.37	Provides regulatory position for on-site cleaning of mate- rials and components as related to clean- ing of <u>fluid sys- tems</u> , e.g., water systems. Chemical compounds that could contribute to inter- granular cracking or SS cracking should not be used with SS or nickel based alloys.	Yes	?

AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
Not applicable, since the criteria do not change over time.	The existing design and its criteria (generic or specific) should be made to satisfy the currently acceptable testing and performance standards for extended life of the plant.	Does the existing design meet the currently acceptable aging needs of the plant and the testing requirements for life extension of the plant?
Implied in the design criteria.	Guidance is needed to cover the limits of component deterioration.	ASME Code for Class 1, 2, or 3 components (safety related valves) required when corrosion is expected. Rules cover design and construction requirements but do not cover deterioration with time.
Identify aging and service wear effects.		Passive Mechanical Failure <u>* Resolved</u>
None	None	Fluid systems important to safety, e.g., coolant pumps, systems and valves, manufactured of austenitic SS type 3xx series, should be protected from leachable chlorides and from fluorides that promote SS cracking.
None	None	Examples of nonacceptable chemical compounds are those containing chlorides, fluorides, lead, zinc, copper, sulfur or mercury where such elements are leachable or could be released by breakdown. Aging management is not explicitly mentioned; however, these procedures will promote management of the degradation issue of corrosion.

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVES COLUMN 6
SELECTED VALVES (contd)	CORROSION (contd)	RGs (contd)			
(21)		RG 1.54	Provides for NRC acceptance of ANSI Standards (N101.4-1972 and N45.2 - 1971). These standards set forth the QA requirements for application of protective coatings for NPPs, including coatings for ferritic steels and stainless steels. Coating should perform under the environmental conditions found in an NPP, e.g., radiation.	Yes	?
(22)		ANS Standards 59.1	Provides for maximum design loads affecting the strength and integrity of the pressure boundary under all specified operating conditions.	None	None
(23)	FATIGUE -- LOW CYCLE FATIGUE -- THERMAL AND MECHANICAL	SRPs 3.9.3 and 3.9.3 Appendix A	Must meet 10 CFR 50, Appendix B and 10 CFR 50, Appendix A, GDC 1, 2, 14, and 15 (also meet ASME Section III requirements for pressure boundaries).	Design margins provided. The SRP, through review of the plant's SAR, provides a record of plant status and yearly update of status.	?
(24)	THERMAL DISTORTION	RG 1.100	Provides NRC staff position on their use of IEEE Standard 344. "Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations." The RG stipulates use of IEEE-344 to qualify mechanical equipment.	Yes	ASME is currently developing a standard for seismic qualification of mechanical equipment.

AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
None	None	RG states that " <u>Coating and cleaning</u> materials used with SS should not be compounded from or treated with chemical compounds that could contribute to <u>corrosion intergranular cracking or stress corrosion cracking</u> ," i.e., chlorides, fluorides, lead, zinc, copper, sulfur or mercury.
Implied in the design criteria.	Guidance is needed to cover the limits of physical and functional deterioration of the valves.	Do the existing design criteria meet the currently accepted aging needs and the testing requirements for life extension of the plant?
None	?	Evaluate transients used in the design and fatigue analysis; ASME Section III must be followed.
None	None	Qualification of equipment provides aging management by establishing the suitability of the equipment for the postulated loads in a seismic event. The RG applies to plant granted a construction permit after June 30, 1988 and for plant docketed for operation after December 30, 1988.

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVES COLUMN 6
SELECTED VALVES (contd)	THERMAL DISTORTION (contd)	ASME			
(25)		Section III	Fatigue	Fatigue curve revisions.	Review of fatigue curves in Section II to determine if they can be revised to accommodate plant operation beyond the normal life of 40 years.
(26)	THERMAL CYCLES	GSIs 54	Degradation of motor-driven valve motors from thermal overloads occurring in repeated high- temperature cycles.	None	None
(27)	FATIGUE	C.11	Reduce fatigue failures in valves resulting from design error.	None	None
(28)	THERMAL CYCLES	IEEE 381	Provides MOV valve actuator qualification testing requirements for " <u>thermal aging test</u> " for "desired qualified life" for 10, 20, 40 years at 40°C and accelerated aging at 138°C.	Yes, records.	?
(29)	ALL ISSUES (useful lifetime)	4	Determine the end- of-life and main- tenance periodicity for valves and pumps.	None	None

AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
Studies to assess the extent of fatigue are required.	Improved end-of-life projections, 40 years and beyond.	Studies to assess the magnitude of the effects of actual environmental conditions. The surfaces are more prone to crack initiation. These studies could result in need for Regulatory Guide, Regulatory Instrument, and/or Code revisions.
Development of signature tracing techniques to be incorporated into the in-service testing programs in order to detect valve deterioration and aging effects.		Survey of Valve Operator-Related Events Occurring During 1978, 1979 and 1980 <u>* Met by TMI Action Plan Item II.E.6.1</u>
Identify active pumps and valves that need redesign and replacement.		Assessment of Failure and Reliability of Pumps and Valves. <u>* Resolved</u>
None	May need to consider qualification testing beyond 40 years.	See comments for item _____ surveillance issue.
		End-of-Life and Maintenance Criteria <u>* Resolved</u>

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVES COLUMN 6
SELECTED VALVES (contd)	EROSION/ WEAR	SRPs			
(30)		3.9.3 and 3.9.3 Appendix A	Reference to ASME Section III and 10 CFR 50, Appendix A, GDC 1, 2, 4, 14 and 15.	The SRP, through review of the plant's SAR; pro- vides a record of plant status and yearly update of status.	?
(31)		ASME Section III: Subsection NB, NC	Provides for design (nominal) loads affecting the strength and integ- rity of the pressure boundary.	None explicitly; provides rules for testing and repair of code components. Requalification of components.	None
(32)		ASME Section XI	Condition Assessment	Requalification of systems and components.	None
(33)		ANS Standards 59.1	Provides for maximum design loads affect- ing the strength and integrity of the pressure boundary.	None explicitly; provides sizing and performance cri- teria for design of valves and their operators.	None
(34)	SURVEILLANCE • TESTING • INSPECTION • MAINTENANCE AND REPAIRS • NDE	CFRs 10 CFR 50, Appendix A, General Design Criteria (GDC) 1, 4, 14, 15.	Provides design cri- teria for the qual- ity, testing and performance require- ments for various plant components, including check valves and other MOV's.	None directly, however, the criteria provides for testing/ performance of components that will provide records for LE/LR.	None

AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
None	?	Loading combinations, System operational transients, and Stress limits.
Not applicable, since the criteria do not change over time.	Implied in the design criteria.	The design allowance for normal wear shall be consistent with the specified design life.
Condition Assess- ment Information.	Requalification of sys- tems and components fol- lowing expiration of operating license.	
Condition Assess- ment Information.	Requalification of valve systems and components following expiration of operating license.	The design allowance for normal wear shall be consistent with the specified design life of the component. Also, mechanical wear is potentially significant for the valve closure flange.
Not applicable, since the criteria do not change over time.	The existing design and its criteria (generic or specific) should be made to satisfy the currently acceptable testing and performance standards for extended life of the plant.	Does the existing design meet the currently accept- able aging needs of the plant and the testing requirements for life extension of the plant?

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVES COLUMN 6
SELECTED VALVES (contd)	SURVEILLANCE (contd)	CFRs (contd)			
(35)		Appendix A, GDC 37, 40, 43, 46.	The emergency core cooling system, the containment heat removal system, the containment atmos- phere cleanup sys- tem, the cooling water system, and the piping systems penetrating contain- ment shall be designed to permit appropriate periodic inspection of the key components in the systems.	Yes, convenience of testing is neces- sary for life extension. (See column 8.)	Valve body erosion could occur in BWR systems that operate in a throttled condition for long periods of time such as the RHR system. Guidance for selecting candidate valves, test fre- quency and test mel- tered in addition to the standard ISI and IST procedures are given in a report entitled "BWROG: Report on Caintation and Erosion Assess- ment of Selected Safety Related Valves" (NEDC-31743, Nov. 1990).
(36)		SRPs 3.9.6	Must conform to 10 CFR 50, Appendix A, GDC 37, 40, 43, 46, and 54 and 10 CFR §50.55a (i.e., meet ASME Section III requirements for pressure boundaries).	(See Column 8) Yes. The SRP, through review of the plant's SAR, provides a record of plant status and yearly update of status.	?

AGING NEEDS
COLUMN 7

LIFE EXTENSION NEEDS
COLUMN 8

COMMENTS
COLUMN 9

Not applicable
since the testing
criteria do not
change over time.

The existing design
should be made to meet
the currently accepted
testing standards for
life extension of the
plant.

Not applicable
since the inspec-
tion criteria do
not change with
time.

The existing design
should be made to meet
the currently accepted
operation and mainte-
nance standards for life
extension of the plant.

ASME Section III must be followed.

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVES COLUMN 6
SELECTED VALVES (contd)	SURVEILLANCE (contd)	RGs RG			
(37)		1.28 (Rev. 3)	Provides periods for maintaining records of manufacturing that may be used for repair and maintenance of the component, e.g., lifetime retention records are required including: a) manufacturing <ul style="list-style-type: none"> • code reports • design reports • materials • repairs • NDE • procurements b) installation <ul style="list-style-type: none"> • welding • heat treatment • cleaning procedures • fabrication procedures c) preoperational tests <ul style="list-style-type: none"> • results • procedures • maintenance activities. 	Yes, records.	?
(38)		RG 1.58 Rev. 1	Provides the regulatory position applicable to ANSI N45.2.6 with exceptions for the qualifications of NPP examination, testing and inspection staff, e.g., RT, UT, eddy current testing and leak testings.	Yes, adequately trained staff contribute to the proper maintenance; thus, contributing to the life extension of components.	?

AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
None	None	<p>Also includes time retention intervals of 3 and 10 yrs for selective important activities.</p> <p>Does not explicitly mention aging management; however, the records noted are clearly required to manage plant operations, including the aging of components.</p>
None	None	<p>The RG was intended for use for those plants (construction permits and operating licenses) after September 30, 1980.</p> <p>Does not specifically mention aging or aging management; however, states that the NRC's regulations require an applicant or licensee to establish a QA program that ensures adequate indoctrination and training of ISI and IST staff for preoperational startup and operational phases of NPPs.</p>

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVES COLUMN 6
SELECTED VALVES (contd)	SURVEILLANCE (contd)	ASME			
(39)		Operations and Maintenance (OM) Code-1990	Detection of changes in the hydraulic and mechanical condition of a pump relative to a reference set of parameters estab- lished previously.	When a reference or set of values may have been affected by repair, replace- ment, or routine servicing of a valve, a new refer- ence value or set of values shall be determined or the previous value reconfirmed by an inservice test run before declaring the valve operable.	None. Replaces ASME Section XI. Subsec- tion ISTC (Inservice Testing of Valves).
			Record-keeping requirements and reports for trend projections.	Records and reports.	?
(40)		Section III NB-5000, NC-5000	Surface examination of weld joint Cate- gories A, B, C and D including adjacent base material for at least 1/2 in. on each side of the weld (as required by NB-5200 and NB-2500).	Welded joints exam- ined by the radio- graphic and either the liquid pene- trant or magnetic particle method.	None
(41)	TESTING	ANS Standards 59.1	Provides design criteria for the testing and per- formance require- ments for different types of valves and operators.	None	None
(42)	SURVEILLANCE	59.1	Provisions for valve testing shall meet the requirements of Subsection ISTC of ASME OM Code-1990 (which replaces Subsection IWV in ASME Section XI).	Access to equipment (including valves) for expected main- tenance shall be adequately provided as required.	None

AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
None	?	<p>ASME OM Code - 1990: Code for Operation and Maintenance of Nuclear Power Plants, An American National Standard, (ANSI) includes:</p> <ul style="list-style-type: none"> • Subsection ISTC: Inservice Testing of Valves in Light-Water Reactor Plants. • Mandatory Appendix I: Inservice Testing of Pressure Relief Devices in Light-Water Reactor Power Plants. • The Code requires Valve Position Verification: observed locally at least once every 2 years and Inservice Exercising Tests for Category A and B valves; tested nominally every 3 months.
Information and data for trend curves.	Records and reports.	Provide the utility with guidance as to the records and reports needed to support a license renewal request.
Not applicable, since the criteria do not change over time.	The existing design and its criteria should be made to satisfy the currently acceptable inspection standards for extended life of the plant.	<p>Nondestructive and nonintrusive examination and testing helps in the longer life of the affected components and support structures.</p> <p>NUREG-1344, "Erosion/Corrosion - Induced Pipe Weld Thinning in U.S. Nuclear Power Plants," provides a review of the erosion/corrosion phenomenon and its major occurrences in NPPs and supports the additional regulatory requirement concerning erosion/corrosion monitoring.</p>
Not applicable, since the criteria do not change over time.	The existing design criteria should be made to satisfy the currently acceptable testing and performance standards for the extended life of the plant.	Does not address specific issues; other criteria must be followed. Valves required to perform safety functions should satisfy Subsection ISTC of ASME OM Code-1990. Provides for provisions for structural integrity for valves and valve combinations/types/operators.
Record-keeping requirements and reports for trend projections.	Records and reports.	Provide guidance as to the design documentation needed to support technical specifications and interface requirements.

COMPONENT COLUMN 1	ISSUE COLUMN 2	REG. INSTRUMENT COLUMN 3	AGING FEATURES COLUMN 4	LIFE EXTENSION FEATURES COLUMN 5	CURRENT INITIATIVES COLUMN 6
SELECTED VALVES (contd)	SURVEILLANCE (contd)	T/S 4.4.1.1	Provides surveil- lance requirements for coolant loops/ systems for <u>hot</u> <u>standby</u> , <u>hot shut-</u> <u>down</u> , and <u>cold</u> <u>shutdown</u> .	Yes, provides long- term surveillance records.	?
(43)					
(44)	QUALIFICA- TION TESTING	IEEE 382	Provides baseline operability valve actuator tests that address aging issue equivalent to installed life, including: <ul style="list-style-type: none"> • thermal aging • radiation aging • vibration aging • seismic simulation • pressurization • DBE radiation exposure • DBE environment test. 	Yes, records.	?

AGING NEEDS COLUMN 7	LIFE EXTENSION NEEDS COLUMN 8	COMMENTS COLUMN 9
None	None	The TS do not specifically mention aging management; however, surveillance requirements do promote aging management by providing minimums allowed for operable units. The valves must be adequately maintained to provide the minimum operable valves.
Thermal aging test and limited to 40-year life qualification.	Tests may need to be designed for in-service equipment (used) for qualification beyond planned life - (IEEE 382 - defines normal service conditions as... "expected to operate for a period of approximately 40 yrs and survive a design base event should one occur").	<p>Tests address aging issues for thermal cycles and vibration. These tests are designed to simulate exposure to thermal and vibration environments. In addition, the vibration conditions prior to DBE testing.</p> <p>Management of in-service (operational) aging is implied by qualifying the equipment for a "qualified life" interval; however, the standard does not explicitly address in-service aging of actual plant operations over time.</p> <p>Actuators within generic groups are tested; it is not possible to test all sizes, types and configurations. The standard provides an appendix which describes a suggested procedure for selection of test events.</p>

Appendix V

Glossary of Regulatory Instrument Descriptions

Appendix V

Glossary of Regulatory Instrument Descriptions

The following regulatory instruments were reviewed for this report:

- Standard Review Plan (SRP)
- Code of Federal Regulations (CFR)
- NRC Regulatory Guides (RG)
- American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Sections III and XI
- Generic Safety Issues (GSI)
- Technical Specifications (TS)
- Institute of Electrical and Electronics Engineers, Inc. (IEEE) Standards
- American Nuclear Society (ANS) Standards
- American Concrete Institute (ACI) Standards

Each Regulatory instrument is described below:

- Standard Review Plan. The SRP is published by the NRC and provides guidance for the review of Preliminary Safety Analysis Reports (PSARs) and the Final Safety Analysis Reports (FSARs) for plant design and operations. The SRP contains 18 chapters and covers all facets of NPP operations, including plant descriptions, design and construction, safety features, instrumentation, electrical power, radiation protection, waste management, quality assurance, and human factors engineering. The SRP also presents specific Branch Technical Positions (BTPs) that are developed and published by branches within the NRC, e.g., BTP ISCB 19 by the Information Security Branch found in SRP Chapter 7, Instrumentation and Control.
- Code of Federal Regulations. These regulations codify general and permanent rules published in the U.S. Federal Register by the Executive department and agencies of the U.S. Federal Government. With reference to this report, these codes establish the rules for design, construction licensing, and operating of commercial NPPs. For the NRC, a federal agency, the most significant CFR relating to commercial reactors is Title 10, Chapter 1, Part 50, Domestic Licensing of Production and Utilization Facilities. Part 50 includes important appendixes, such as the General Design Criteria (GDC) requirements, specific material requirements for reactor vessels, emergency planning, and quality assurance criteria for NPPs.
- Regulatory Guides. These guides are published by the NRC in 10 broad divisions, including Division 1, Power Reactors. The guides are available to the public and NPP licensees. They provide general guidance to applicants and describe methods acceptable to the NRC staff, to implement specific parts of the Commission's regulations, and to delineate evaluation techniques used by NRC staff for specific problems and postulated accidents. The RGs often explain and detail acceptable methods for the rules found in the CFRs.
- American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Sections III and XI. The ASME Code establishes the rules of safety governing design, construction, operation, and testing of NPP components and systems. Section III covers a broad range of components and systems including pressure vessels, piping, pumps, valves, supports, and core internals. Section XI provides rules for ISI of components and systems. Section XI is applicable when the requirements of the Construction Code, e.g., Section III, have been satisfied. The ASME code Editions, Addenda

and Code Cases used by the owners of NPPs are subject to acceptance by the NRC. The ASME codes are an integral part of the standards referenced in the CFRs. The ASME Maintenance and Operation committees provide an important function through the testing of pumps, valves, and snubbers.

- Generic Safety Issues. GSIs are published by NRC to identify safety issues generic to NPPs. The GSIs can cover a wide variety of subjects; however, for the purposes of this review, the GSIs reviewed are those that are predominantly concerned with material degradation and operations that cause degradation of plant components. When a solution to the generic issues is found, the usual result is an NRC Generic Letter or an NRC NUREG that provides an industry-wide dissemination of the solution.
- Technical Specifications. TSs are NPP mandatory operational specifications that provide instructions for limiting conditions and surveillance requirements for plant operations. License applicants for authority to operate an NPP are required to supply TSs as enumerated in 10 CFR 50, Section 50.36.
- Institute of Electrical and Electronic Engineers Standards. The IEEE Standards present criteria and requirements for electrical systems that are specifically related to protecting public health and safety. The standards included in this review are principally those prepared by the IEEE Nuclear Power Engineering Committee. The IEEE Standards, through the Equipment Qualification standards, define "qualified life" and provide guidance for requalification or replacement of components when their qualified life is reached. The IEEE standards are not mandatory and use of the standards is wholly voluntary. Use of the standards, as acceptable practice, are subject to the approval of the regulatory agency, i.e., the NRC. Specific IEEE standards are frequently

called out and/or qualified in the Regulatory Position Section of the RGs as acceptable to NRC staff.

- American Nuclear Society Standards. The ANS standards are agreements among designers, engineers, governmental regulatory agencies, manufacturers, and nuclear scientists. The standards are developed to provide current practices on various subjects that affect NPPs. Among the subjects are criteria for earthquake instrumentation, various safety guides, selection and training of personnel, quality assurance, security for NPPs, and auxiliary feed-water systems for PWRs. The ANS standards are frequently published as American National Standards with the approval of the American National Standards Institute (ANSI). ANS standards are guides for prospective use and are not mandatory; the designers are not restricted to the ANS standards and may propose alternative criteria to provide adequate safety.
- American Concrete Institute. ACI standards provide requirements for nuclear safety relating to concrete structures, such as the containment and basemat and external covers. The ACI standards address seismic design, floods, and other natural phenomena.
- Miscellaneous. A number of other regulatory instruments have an impact on the regulatory base. Standards such as the American Society of Testing and Materials (ASTM) are heavily involved with the ASME codes by providing the nonferrous and ferrous material specifications. ANSI provides many standards and is the publishing institution or co-publisher of applicable standards, e.g., ANSI/IEEE Standard 383 is published as a dual ANSI/IEEE Standard. Other codes and standards include the American Institute of Steel Construction, the American Welding Society, and the National Fire Protection Association codes, and the National Fire Codes.

Appendix VI

Glossary of Aging Issues

Appendix VI

Glossary of Aging Issues

corrosion - Corrosion is an aging mechanism that applies to all components in varying degrees and is manifested in different ways. Corrosion in the broad sense is the deterioration of material surfaces, chiefly through electrochemical reactions; it is influenced by the environment of the material or component, e.g., high humidity. A familiar example is the rusting of iron. Corrosion is also linked to other aging mechanisms by causing an acceleration of the degradation, e.g., SCC, irradiation assisted corrosion cracking, corrosion fatigue, and erosion/corrosion. Corrosion is also influenced by microbial effects and differential aeration.

erosion/corrosion - Erosion/corrosion occurs when the eroding fluid (liquid or gas) or particulate matter is in the presence of or contains corrosion-causing products, i.e., the corrosion reaction is accelerated by the relative movement of the corrosive fluid and the metal surface. This mechanism involves the effects of mechanical wear or abrasion. Characteristics of erosion/corrosion are the appearance of grooves, gullies, waves, round holes, and valleys on the surface of the component. An example of erosion/corrosion is tube wall thinning caused by impingement of water droplets containing suspended solids.

crack formation and flaw growth - Material crack formation and fatigue growth of flaws are indicators of material degradation and possible failure by through-wall cracks. A crack is a discontinuity at a particular location in a material as a result of localized excessive elastic/plastic deformation. A crack will propagate as long as the tensile stress acting on the component supplies sufficient energy to sustain a new crack surface (this aging issue is related to fatigue and SCC; i.e., without a driver a crack will not grow). Present methods of flaw and crack detection are not 100% effective and improvements are needed for evaluation of crack initiation and growth. The issue also encompasses human factors; e.g., training of the NDE staff.

cracking and spalling - The breakup of concrete due to cracks and the chipping, flaking, general breakup of the concrete surface, or spalling is associated with concrete containments. It is usually a result of aggressive external environments, internal chemical reactions, and nuclear heat. The consequences of this issue can result in acceleration of other degradation processes; e.g., corrosion of reinforcing steel and loss of stress for BWR prestressed tendons. Degradation of the shielding properties of the concrete may also result from excessive cracking and spalling.

creep/swelling - Creep is defined as the progressive deformation of a material at constant stress. Creep failure (fracture) or stress rupture reflects the influence of relative high temperatures and long-term, load-bearing conditions. The creep temperature is often expressed as a homologous temperature; i.e., the ratio of the operating temperature T to the absolute melting temperature T_m (K). Creep of load-carrying components becomes important at homologous temperatures greater than 0.50. Load-carrying components operating in the temperature range of 538°C to 870°C (1000°F to 1600°F) are susceptible to creep. Generally, creep is not a major problem in LWRs because of their lower operating temperature levels, which are well below the creep range for ASME Code materials specified in ASME BPVC, Section III. This aging mechanism, however, is of interest to the NPAR program because of its impact on electrical components/systems.

Swelling is an increase in the dimensions of components (e.g., fuel elements and fuel cladding) caused by the accumulation of fission product atoms in the structural lattice of components. The fission products occupy a larger volume than the original material resulting in swelling.

element burnout(s) - Element burnouts and the repair and replacement of heaters are identified as an aging issue for the pressurizer vessels.

erosion - Erosion is a broad aging mechanism that is principally associated with fluid flow in components. High velocity water impingement can erode the walls of pipe and fittings. By definition pure erosion is the mechanical or wear action of a fluid and/or particulate matter on the surface of component parts. Erosion is usually accelerated by the presence of solid particles in flowing fluid. The possible consequence of typical erosion is the thinning of a pipe wall to failure.

examination, inspection, maintenance, and repair - These activities may be considered aging issues because the management of aging is affected by these activities and, in some cases, aging is accelerated by poor examination, harsh inspection, and improper maintenance and repair. Also, conditions may exist that do not permit adequate performance of the activities.

excessive testing - Testing programs and procedures that require excessive testing can lead to premature aging; e.g., frequent testing of the EDGs has been identified as a contributor to aging.

fatigue - Fatigue is defined as the phenomenon leading to fracture under repeated or fluctuating stresses having a maximum value less than the tensile strength of the material. For example, stresses induced by stratified flow can cause thermal fatigue. In stratified flow, the cool water flows on the bottom of the pipe while steam flows on the top. These conditions can cause temperature differences that lead to thermal cycling and thermal fatigue of piping systems; e.g., pressurizer spray lines and vessel nozzles.

fracture toughness - Fracture toughness is a material property that reflects the ability to resist fracture. Fracture toughness is related to the unique stress intensity level that causes failure in a component. Its measure is the stress intensity factor, which incorporates both geometrical terms (the crack length appears explicitly, while the crack tip radius is assumed to be very sharp) and the stress level. It is an embrittlement aging issue for NPPs because ferritic materials suffer from a decrease in fracture toughness as a result of irradiation. The CFRs specifically provide limits for the RPV material tests; i.e., initial Charpy tests of material shall be 102J (75 ft-lb) and throughout the life of the vessel shall not be less than 68J (50 ft-lb). Additionally, the

material reference nil-ductility temperature (RT_{NDT}) is limited by the pressurized thermal shock screening rule.

hydrogen embrittlement - Two forms of hydrogen attack that are most relevant to LWR materials and conditions are hydrogen blistering and hydrogen embrittlement. Both are of concern because they produce mechanical damage in the affected component. In each case, atomic hydrogen enters the metal, either as a result of a corrosion reaction at the surface or by cathodic polarization, which results in the evolution of hydrogen gas. In blistering, molecular hydrogen within the metal causes high pressure and local damage in the form of "blistered" regions of the metal surface. Hydrogen embrittlement affects ferritic and martensitic iron-based alloys, and results in low ductility intergranular cracking (similar to SCC). The phenomenon of hydrogen cracking is usually manifested as delayed cracking, at or near room temperature, after stress is applied. Cracking of welds due to hydrogen embrittlement and hydrogen-induced cracking are a common concern. This cracking is more of a problem in higher strength steels, as in the hydrogen cracking of bolts.

intergranular stress corrosion cracking - IGSCC is a preferential corrosion at the grain boundaries of a susceptible metal or alloy in the presence of a chemically aggressive environment; e.g., hot oxygenated water and a tensile stress. IGSCC has occurred in austenitic stainless steel (SS) in oxidizing environments, such as BWRs. It also has occurred in steam generator tubes, safe ends, and type 304 SS piping. The principal concern is that SCC can cause ruptures, leakages, and plant shutdowns.

irradiation embrittlement - Irradiation embrittlement is defined as a decrease in fracture toughness due to long-term exposure to nuclear radiation. High neutron fluence levels can cause embrittlement in the RPV beltline region as well as other reactor internals and core supports. A reduction in tensile ductility is also caused by neutron exposure.

low-flux long-time irradiation of vessels, vessel internals, and supports - This aging issue is principally identified for the RPVs and vessel supports. Long-term flux reduces toughness and initial margins of safety in RPVs.

seismic failure/damage - Aging issues associated with earthquake damage were considered. Although not a classic aging issue, NPPs are required to evaluate seismic conditions and the consequences of natural phenomena and the damage that may result from these events. Aging could weaken a component so that it would not be able to withstand a seismic event.

specific environmental factors, e.g., moisture, oxidation, chemicals, oils, and dust - Environmental factors can induce aging in many NPP components. For this review the factors of moisture, chemicals, oils, dust, etc., are primarily of concern in the operation, testing, and performance of the EDGs. Because these factors can induce aging, physical location, surroundings, housekeeping, and maintenance are important to EDGs. All these factors or elements can accelerate many of the aging mechanisms associated with the operation of diesel engines and their support equipment.

stress corrosion cracking - SCC is cracking accelerated by the combined effect of constant tensile stress, corrosive environments, and susceptible microstructures. The stress may be residual in the material, as from cold working or heat treatment, or it may be externally applied. The observed crack may be transgranular or intergranular, depending on the nature of material and the environment. This term is a broader aging term than IGSCC and has been linked to bolt degradation as a principal aging problem, e.g., leaking steam through gaskets can react with the bolting lubricants and cause SCC.

stud failure - This aging issue is a factor in degradation of closures, flanges, manways, etc., through the cracking of bolts and studs from stress corrosion, fatigue, and corrosion.

thermal cycles - Thermal cycling induces stress through thermal gradients/temperature changes. Thermal cycling induces thermal stress, low-cycle thermal fatigue, and high-cycle fatigue (water temperature fluctuations). Low-cycle fatigue is defined as fatigue caused by high stresses and low numbers of cycles. Significant plastic strains occur during each cycle. Cycle lives are less than 10^4 to 10^6 . High-cycle fatigue is defined as fatigue caused by low stresses and high numbers of cycles. Strain cycles are in the elastic range. Cycle lives are greater than 10^4 to 10^6 .

thermal embrittlement - Thermal embrittlement is defined as a reduction in the ductility of a susceptible material due to a chemical change influenced by high temperature for long periods of time. Thermal embrittlement of cast SS components (pump housing, valve bodies, piping and fittings, etc.) is possible over periods of many years, resulting in increased critical flaw sizes. Thermal aging can significantly reduce fracture toughness and ductility of LWR components. The rate of thermal embrittlement generally increases with increase in temperatures; however, specific material compositions of various steels are factors in thermal embrittlement.

thermally induced bending - This aging issue is principally (for this review) associated with the pressurizer caused by high- and low-cycle thermal loads at the water-steam interface in the vessel wall. (Thermally induced bending stresses can be a factor at other locations).

thermally induced mechanical wear - This aging issue is principally (for this review) associated with the heater failure in pressurizers caused by wearing and thinning of the heater due to rubbing action with supports because of thermal growth.

transient thermal and pressure loads - Transient or short thermal loads have been identified (for this review) as the transients affecting the pressurizer during heatups, cooldowns, testing, and abnormal events. For example, transients can occur in a few seconds and cause temperature changes of 55°C (100°F) in the surge-line nozzle.

tube specific - denting, fretting, and crevices - These aging issues are steam generator tube-specific. Denting results from crevice corrosion of the tube support plate and tubesheet materials; corrosion of carbon steel support plates and tubesheets squeezes the tube's outside diameter and can result in a decrease of thermal efficiency and cracking in the tube. Crevices provide areas for localized corrosion due to stagnant solutions in lap joints, holes, welding surfaces, etc. Fretting in steam generators results principally from wear action between the tube and tube supports and antivibration bars.

vibration - Vibration is a stressor that can cause degradation in many components. Vibration by definition is any physical process that produces cyclic variations or motion. It is an aging issue to

the extent that the vibration may cause failure or damage to a component. Excessive vibration can damage components by wear and promote fatigue and pipe weld fatigue cracking. Vibration is a dynamic action and is associated in this review primarily with piping systems and the EDG. Dynamic loads are caused by response of the component to an oscillating input; e.g., flowing water or rotating equipment.

wear - Wear usually results from the relative motion between two surfaces. Wear is the removal of discrete particles of material from a solid surface by the relative motion between that surface and one or more contacting surfaces. Wear is accelerated by

hard and abrasive particles or action of fluids (erosion), vibration, or corrosive environments. The common result of wear is the reduction of thickness or damage to the mating surfaces. Wear is of concern in rotating machinery; e.g., in pumps and EDGs and the other equipment/parts subject to motion and vibration.

weldments - specific to dissimilar metals of safe ends - Weldments are formed by welding together an assembly of pieces, such as pipe to nozzles. Safe-end weldments have specific problems due to dissimilar metals. Among these are low-cycle thermal and mechanical fatigue, and IGSCC in the heat affected zone in the base metal.

Distribution

**No. of
Copies**

OFFSITE

21 U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
ATTN: S. K. Shaukat (20)
J. P. Vora

2 Idaho National Engineering Laboratory
P.O. Box 1675
Idaho Falls, ID 83415
ATTN: L. Magelby
V. N. Shah

R. L. Lofaro
Brookhaven National Laboratory
Upton, NY 11973

S. P. Nowlen
Sandia National Laboratory
P.O. Box 5800
Albuquerque, NM 87185

**No. of
Copies**

2 Oak Ridge National Laboratory
P.O. Box 2009
Oak Ridge, TN 37831
ATTN: H. D. Haynes
B. Poole

ONSITE

DOE Richland Operations Office

D. C. Langstaff K8-50

24 Pacific Northwest Laboratory

S. H. Bush	K9-04
M. E. Cunningham (10)	K8-43
P. M. Daling	K8-07
A. B. Johnson	K8-34
J. V. Livingston	K8-37
R. D. Orton	K8-33
S. Somasundaram	K5-20
J. C. Spanner	K5-26
Information Release Office (7)	K1-06