Safety Evaluation Report

Related to the License Renewal of Arkansas Nuclear One, Unit 1

Docket No. 50-313

Entergy Operations, Inc.

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

April 2001

ABSTRACT

This document is a safety evaluation report regarding the application to renew the operating license for Arkansas Nuclear One, Unit 1 (ANO-1), which was filed by Entergy Operations, Inc., by letter dated January 31, 2000. The Office of Nuclear Reactor Regulation has reviewed the ANO-1 license renewal application for compliance with the requirements of Title 10 of the *Code of Federal Regulations*, Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," and prepared this report to document its findings.

In its submittal of January 31, 2000, the Entergy Operations, Inc., requested renewal of the ANO-1 operating license (License No. DPR-51), which was issued under Section 104 of the *Atomic Energy Act of 1954*, as amended, for a period of 20 years beyond the current license expiration date of May 20, 2014. ANO-1 is located in Pope County in the central region of Arkansas on the shore of Lake Dardanelle. ANO-1 is a Babcock and Wilcox pressurized-water reactor nuclear steam supply system that is designed to generate 2568 MW thermal, or approximately 836 MW electric.

The NRC ANO-1 license renewal project manager is Robert J. Prato. Mr. Prato may be contacted by calling (301) 415-1147 or by writing to the License Renewal and Standardization Branch, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

CONTENTS

THIS PAGE IS INTENTIONALLY LEFT BLANK

SUMMARY

This report describes the results of a review by the Nuclear Regulatory Commission (NRC) staff of an application to renew the licenses for Arkansas Nuclear One, Unit 1(ANO-1). Under the Atomic Energy Act (AEA), the NRC issues licenses for commercial power reactors to operate for up to 40 years. The AEA also permits the licenses to be renewed. The NRC established license renewal requirements in the regulations. When those requirements are satisfied, a license can be renewed for up to 20 additional years.

Plant owners are interested in license renewal because they need to know what requirements must be satisfied to permit long-term plant operation. This knowledge helps them to predict the cost of plant operation for long-term energy planning.

The requirements for license renewal are presented in Part 54 of Title 10 to the *Code of Federal Regulations* (10 CFR Part 54). When those requirements were developed, the NRC concluded that the existing licensing basis and the regulatory process are adequate to maintain safe plant operation, except for the possible effects of aging on passive systems, structures, and components. Therefore, the requirements in 10 CFR Part 54 focus on managing the effects of aging for such passive structures and components as buildings, tanks, and pipes.

The NRC also established requirements for a license renewal environmental report in 10 CFR Part 51. Those requirements establish the scope of a review of environmental impacts, which is one part of the NRC's responsibilities under the National Environmental Policy Act (NEPA). The results of that review are described in Supplement 3 of NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants Regarding the Arkansas Nuclear One, Unit 1."

In a letter to the NRC dated January 31, 2000, the Entergy Operations, Inc., filed its request for renewal of the ANO-1 operating license (License No. DPR-51), which was issued under Section 104 of the *Atomic Energy Act of 1954*, as amended, for a period of 20 years beyond the current license expiration date of May 20, 2014. If granted the renewed license for ANO-1 would expire May 20, 2034.

ANO-1 is located in Pope County in the central region of Arkansas on the shore of Lake Dardanelle. ANO-1 is a Babcock and Wilcox pressurized-water reactor nuclear steam supply system that is designed to generate 2568 MW thermal, or approximately 836 MW electric.

This document is a safety evaluation report regarding the application to renew the operating license for ANO-1. The Office of Nuclear Reactor Regulation has reviewed the ANO-1 license renewal application for compliance with the requirements of Title 10 of the *Code of Federal Regulations*, Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," and prepared this report to document its findings.

In accordance with Federal regulations under 10 CFR Part 51 and Part 54, and the NRC draft “Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants,” dated September 1997, the staff has completed its review of the Arkansas Nuclear One, Unit 1(ANO-1) license renewal application and supporting documentation, and has documented its finding in this safety evaluation report (SER).

In an SER issued on January 10, 2001, regarding the review of the ANO-1 license renewal application, the staff identified six open items. Those open items have been resolved, as discussed in this SER. On the basis of its evaluation of the ANO-1 license renewal application and the applicant's response to the open items as discussed within this SER, the staff concludes the following:

1. actions have been identified and have been or will be taken with respect to managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require an aging management review under 10 CFR 54.21(a)(1)

2. actions have been identified and have been or will be taken with respect to time-limited aging analyses that have been identified to require review under 10 CFR 54.21(c)

Accordingly, the staff found that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis for ANO-1 for the period of extended operation and, therefore recommends granting the renewed license for an additional 20 years of operation beyond the current licensing term.

ABREVIATIONS

AAC	alternate AC
ABHVS	auxiliary building heating and ventilation systerm
ACI	American Concrete Institute
ACRS	Advisory Committee on Reactor Safeguards
AMP	aging management program
AMR	aging management review
AMSAC	ATWS mitigation system actuation circuit
ANL	Argonne National Laboratory
ANO-1	Arkansas Nuclear One, Unit 1
ANSI	American National Standards Institute
APCSB	auxiliary power conversion system branch
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	anticipated transient without scram
B&W	Babcock and Wilcox
B&WOG	Babcock and Wilcox Owners Group
BACP	boric acid corrosion prevention
BL	bulletin
BTP	branch technical position
BWST	borated water storage tank
CAS	central alarm station
CASS	cast austenitic stainless steel
CBA	core barrel assembly
CEOG	Combustion Engineering Owners Group
CF	core flood
CFR	U. S. Code of Federal Regulations
CLB	current licensing basis
CLQL	component level Q-list
CRA	control rod assembly
CRDM	control rod drive mechanism
CRGA	control rod guide assembly
CRVS	control room ventilation system
CS	condensate system
CSA	core support assembly
CSS	cores support shield assembly
CUF	cumulative usage factor
DB	dry bulb
DBA	design-basis accident
DBE	design-basis events
DG	draft regulatory guide
DH	decay heat
DOE	Department of Energy
DOR	Division of Operating Reactors

ECCS	emergency core cooling system
ECP	emergency cooling pond
ECT	eddy current testing
EDG	emergency diesel generator
EFIC	emergency feedwater initiation and control
EFPY	effective full power years
EFS	emergency feedwater system
EFW	emergency feedwater
EIC	electrical and instrumentation controls
EPRI	Electric Power Research Institute
EQ	environmental qualification
ES	engineered safeguards
ESP	engineering support personnel
FA	fuel assemby
FAC	flow-accelerated corrosion
FERC	Federal Energy Regulatory Commission
FHA	fuel handling accident
FIV	flow-induced vibration
FP	fire protection
FSAR	final safety analysis report
FSER	final safety evaluation report
FTI	Framatome Technologies
GEIS	generic environmental impact statement
GL	generic letter
GSI	generic safety issue
HC	hydrogen control
HELB	high-energy line breaks
HEPA	high-efficiency particulate air (filter)
HPI	high-pressure injection
HUPCAPS	high-fluence supplementary weld metal surveillance capsules
HVAC	heating, ventilation, and air conditioning
IASCC	irradiation-assisted stress-corrosion cracking
IEB	Inspection and Enforcement Bulletin
IEEE	Institue of Electrical and Electronics Engineers
IGA	intergranular attack
IGSCC	intergranular stress corrosion cracking
IMS	in-core monitoring system
IN	information notice
INEEL	Idaho National Engineering and Environmental Laboratory
INPO	Institute of Nuclear Power Operations
IPA	integrated plant assessment
ISI	inservice inspection
IST	inservice testing
ISTS	improved standard technical specification
ITS	improved technical specification

LBB	leak-before-break
LIA	lower internals assembly
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
LPI	low-pressure injection
LRA	license renewal application
MCR	main control room
MCRE	main control room environment
MFS	main feedwater system
MIC	microbiologically influenced corrosion
MRV	minimum required value
MIRVP	master integrated reactor vessel surveillance program
MSS	main steam system
MU	make-up
MUP	make-up and purification
NDE	nondestructive examination
NEI	Nuclear Energy Institute
NEPA	National Environmental Policy Act
NPRDS	Nuclear Plant Reliability Data System (INPO)
NPS	nominal pipe size
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
NSAC	Nuclear Safety Analysis Center
NUMARC	Nuclear Management and Resource Council
NUREG	NRC technical report designation
ODSCC	outside-diameter stress-corrosion cracking
OTSG	once-through steam generator
PA	plenum assembly
PLL	prescribed lower limits
PRVS	penetration room ventilation system
PTS	pressurized thermal shock
PWR	pressurized-water reactors
PWSCC	primary water stress-corrosion cracking
QA	quality assurance
Q-CST	Q-condensate storage tank
RAI	request for additional information
RBCS	reactor building cooling system
RBCP	reactor building cooling and purge
RBI	reactor building isolation
RBIS	reactor building isolation system
RBS	reactor building spray
RCP	reactor coolant pump
RCS	reactor coolant system
RG	regulatory guide

RI-ISI	risk-informed ISI
RPV	reactor pressure vessel
RTD	resistance temperature detector
RTE	resistance temperature element
RVI	reactor vessel internals
RVIAMP	reactor vessel internals AMP
RVLMS	reactor vessel level monitoring program
SAE	Society of Automotive Engineers
SAR	safety analysis report
SC	structures and component
SCBA	self-contained breathing apparatus
SCC	stress corrosion cracking
SCS	structures and component support
SER	safety evaluation report
SFP	spent fuel pool
SFPC	spent fuel pool cooling
SH	sodium hydroxide
SMAW	shielded metal arc welding
SOC	statement of considerations
SPCS	steam and power conversion systems
SRP	standard review plan
SSC	structures, systems, and components
SSHT	surveillance specimen holder tubes
SUPCAPS	supplementary weld metal surveillance capsules
TLAA	time-limited aging analyses
TMI	Three Mile Island
TMI-1	Three Mile Island Unit 1
TMI-2	Three Mile Island Unit 2
TS	technical specification
UFSAR	Updated Final Safety Analysis Report
ULD	upper-level design
USAS	United States of America Standards
USE	upper-shelf energy
UT	ultrasonic testing
VDIL	vents, drains, and instrument lines
VHP	vessel head penetration
WB	wet bulb
WOG	Westinghouse Owners Group

1 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

This document is a safety evaluation report (SER) regarding the application to renew the operating license for Arkansas Nuclear One, Unit 1 (ANO-1), which was filed by Entergy Operations, Inc. (hereafter referred to as Entergy or the applicant), by letter dated January 31, 2000. The applicant submitted its application to the United States Nuclear Regulatory Commission (NRC) for renewal of the ANO-1operating license for an additional 20 years. The Office of Nuclear Reactor Regulation reviewed the ANO-1 license renewal application (LRA) for compliance with the requirements of Title 10 of the *Code of Federal Regulations*, Part 54 (10 CFR Part 54), "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," and prepared this report to document its findings.

In its submittal of January 31, 2000, the applicant requests renewal of the ANO-1 operating license (License No. DPR-51) issued under Section 104 of the *Atomic Energy Act of 1954*, as amended, for a period of 20 years beyond the current license expiration date of May 20, 2014. ANO-1 is located in Pope County in the central region of Arkansas on the shore of Lake Dardanelle. ANO-1 is a Babcock and Wilcox pressurized-water reactor nuclear steam supply system that is designed to generate 2568 MW thermal, or approximately 836 MW electric. Details concerning the plant and the site are found in the updated final safety analysis report (UFSAR) for ANO-1.

The license renewal process requires a technical review of safety issues and an environmental review. The requirements for these reviews are stated in NRC regulations in 10 CFR Parts 54 and Part 51, respectively. The safety review is based on Entergy's LRA, the ANO-1 UFSAR, and the applicant's responses to NRC staff requests for additional information (RAIs). The applicant's answers to the RAIs are documented and docketed in letters to the NRC, and are supplemented by meeting minutes and other docketed correspondence. The public can review the LRA and all pertinent information and material, including the UFSAR, at the NRC Public Document Room, 11555 Rockville Pike, Rockville, MD 20852-2738. In addition, the ANO-1 LRA and other significant information and material relating to the license renewal review are available on the NRC Web page at www.nrc.gov.

This SER summarizes the findings of the staff's safety review of the ANO-1 LRA, and describes the technical details that the staff considered in its safety evaluation of the proposed operation of ANO-1 for an additional 20 years beyond the term of the applicant's current operating license. The staff reviewed the LRA in accordance with NRC regulations and the guidance presented in the NRC draft "Standard Review Plan (SRP) for the Review of License Renewal Applications for Nuclear Power Plants," dated August 2000.

Chapters 2 through 4 of this SER provides the staff's evaluation of the license renewal issues that were considered during the review of the LRA. Chapter 5 contains the report from the Advisory Committee on Reactor Safeguards (ACRS). The conclusions of this report are presented in Chapter 6.

Appendix A is a chronology of NRC's and the applicant's principal correspondence related to the review of the application. Appendix B is a bibliography of the documents used during the review. Appendix C is a list of the NRC staff's principal reviewers and its contractors for this project. Appendix D summarizes the on-site review activities.

In accordance with 10 CFR Part 51, the staff prepared a draft and a final plant-specific supplement to the generic environmental impact statement (GEIS) that discuss the environmental considerations related to renewing the license for ANO-1. The draft and final plant-specific supplement to the GEIS was issued separately from this report. Specifically, a draft and a final Supplement 3 to NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants Regarding Arkansas Nuclear One, Unit 1" were issued in September 2000 and April 2001, respectively.

The NRC ANO-1 license renewal project manager is Robert J. Prato. Mr. Prato may be contacted by calling (301) 415-1147 or by writing to the License Renewal and Standardization Branch, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

1.2 License Renewal Background

Pursuant to the *Atomic Energy Act of 1954*, as amended, and NRC regulations, licenses for the operation of commercial power reactors are issued for 40 years. These licenses can be renewed for up to 20 additional years. The original 40-year license term was selected on the basis of economic and antitrust considerations, rather than technical limitations. However, some plant equipment may have been engineered on the basis of an expected 40-year service life.

In 1982, the NRC anticipated interest in license renewal and held a workshop on aging of nuclear power plants. This workshop led the NRC to establish a comprehensive program for nuclear plant aging research (NPAR). On the basis of the results of that research, a technical review group concluded that many aging phenomena are readily manageable, and do not involve technical issues that would preclude extending the life of nuclear power plants.

In 1986, the NRC published a request for comment regarding a policy statement that would address major policy, technical, and procedural issues related to life extension for nuclear power plants.

In 1991, the NRC published the License Renewal Rule in 10 CFR Part 54. The NRC participated in an industry-sponsored demonstration program to apply the Rule to pilot plants and develop experience to establish implementation guidance. To establish a scope of review for license renewal, the Rule defined age-related degradation unique to license renewal. However, during the demonstration program, the NRC found that many aging mechanisms occur and are managed during the period of the initial license. In addition, the NRC found that the scope of the review did not allow sufficient credit for existing programs, particularly for the implementation of the Maintenance Rule in accordance with 10 CFR 50.65, which also manages plant aging phenomena.

As a result, in 1995 the NRC amended 10 CFR Part 54. The amended License Renewal Rule established a regulatory process that is expected to be simpler, more stable, and more predictable than the previous License Renewal Rule. In particular, 10 CFR Part 54 was clarified to focus on managing the adverse effects of aging rather than on identifying all aging mechanisms. The changes to the Rule were intended to ensure that important systems, structures, and components (SSCs) will continue to perform their intended function during the period of extended operation. In addition, the integrated plant assessment (IPA) process was clarified and simplified to be consistent with the revised focus on passive, long-lived structures and components (SCS).

In parallel with these efforts, the NRC pursued a separate rulemaking effort to amend 10 CFR Part 51 to focus the scope of the review of environmental impacts related to license renewal, and fulfill, in part, the NRC's responsibilities under the *National Environmental Policy Act of 1969* (NEPA).

1.2.1 Safety Reviews

License renewal requirements for power reactors are founded on two key principles:

(1) The regulatory process is adequate to ensure that the licensing bases of all currently operating plants provide and maintain an acceptable level of safety, with the possible exception of the detrimental effects of aging on the functionality of certain SSCs during the period of extended operation, and possibly a few other issues related to safety only during the period of extended operation.

(2) The plant-specific licensing basis must be maintained during the renewal term in the same manner, and to the same extent, as during the original licensing term.

In implementing these two principles, the Rule (in 10 CFR 54.4) defines the scope of license renewal as including those plant SSCs (a) that are safety-related, (b) whose failure could affect safety-related functions, and (c) that are relied on to demonstrate compliance with the Commission's regulations for fire protection, environmental qualification, pressurized thermal shock, anticipated transients without scram, and station blackout.

Pursuant to 10 CFR 54.21(a), the applicant must review all SSCs that are within the scope of the Rule to identify the SCs that are subject to an aging management review (AMR). SCs that are subject to an AMR are those that perform an intended function without moving parts, or without a change in configuration or properties, and that are not subject to replacement based on a qualified life or specified time period. As required by 10 CFR 54.21(a)(3), the applicant must demonstrate that the effects of aging will be managed in such a way that the intended function(s) of the SCs that are within the scope of license renewal will be maintained, consistent with the current licensing basis (CLB), for the period of extended operation.

Active equipment, however, is considered to be adequately monitored and maintained by existing programs. In other words, the detrimental effects of aging that may affect active equipment are more readily detectable and will be identified and corrected through routine surveillance, performance indicators, and maintenance. The surveillance and maintenance programs and activities for active equipment, as well as other aspects of maintaining plant design and licensing bases, are required to continue throughout the period of extended operation.

Pursuant to 10 CFR 54.21(d), each application is also required to include a supplement to the plant’s final safety analysis report (FSAR). This FSAR Supplement must contain a summary description of the programs and activities for managing the effects of aging.

Another requirement for license renewal is the identification and updating of time-limited aging analyses (TLAAs). During the design phase for a plant, certain assumptions are made about the initial operating term of the plant, and these assumptions are incorporated into design calculations for several of the plant's SSCs. In accordance with 10 CFR 54.21(c)(1), these calculations must be shown to be valid for the period of extended operation, or must be projected to the end of the period of extended operation, or the applicant must demonstrate that the effects of aging of these SSCs will be adequately managed for the period of extended operation.

In 1996, the NRC developed and issued a draft regulatory guide, DG-1047, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses." This guide proposes to endorse an implementation guideline prepared by the Nuclear Energy Institute (NEI) as an acceptable method of implementing the License Renewal Rule. The guideline is NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54–The License Renewal Rule," which was issued in March 1996. The NRC prepared a draft SRP for the safety review in September 1997, which was revised and reissued in August 2000. The draft regulatory guide was used, along with the initial draft SRP, to review this application and to assess topical reports involved in license renewal as submitted by industry groups. As experience is gained, the NRC will improve the SRP and clarify the regulatory guidance.

1.2.2 Environmental Reviews

In December 1996, the staff revised the environmental protection regulations in 10 CFR Part 51 to facilitate environmental reviews for license renewal. The staff prepared a "Generic Environmental Impact Statement (GEIS) for License Renewal of Nuclear Plants," NUREG-1437, Revision 1, in which it examined the possible environmental impacts associated with renewing licenses of nuclear power plants. For certain types of environmental impacts, the GEIS establishes generic findings that are applicable to all nuclear power plants. These generic findings are identified as Category 1 issues in 10 CFR Part 51, Subpart A, Appendix B. Pursuant to 10 CFR 51.53(c)(3)(i), an applicant for license renewal may incorporate these generic findings in its environmental report. Environmental impacts for the renewal of a plant's license, which must be analyzed on a plant-specific basis, are identified as Category 2 issues in 10 CFR Part 51, Subpart A, Appendix B. Such analyses must be included in an environmental report in accordance with 10 CFR 51.53(c)(3)(ii).

In accordance with NEPA and the requirements of 10 CFR Part 51, the NRC performs a plant-specific review of the environmental impacts of license renewal, including whether there is new and significant information that was not considered in the GEIS. Two public meetings were held in April 2000, near ANO-1 as part of the NRC's scoping process to identify environmental issues specific to the plant. The results of the environmental review and a preliminary recommendation on the license renewal action were documented in the NRC's draft plant-specific Supplement 3 to the GEIS, which was issued on October 3, 2000. An additional two public meeting were held near the site in November 2000, (during the 75-day comment period for the draft plant-specific Supplement 3 to the GEIS). At these meetings, the staff described the environmental review, and answered questions from members of the public to assist them in formulating any comments that they may have regarding the review. The final Supplement 3 to the GEIS was issued in April 2001.

Supplement 3 presents the NRC's preliminary environmental analysis associated with renewing the ANO-1 operating license for an additional 20 years that considers and weighs the environmental effects, and alternatives that are available to avoid adverse environmental effects. On the basis of (1) the analysis and findings in the "Generic Environmental Impact Statement for License Renewal of Nuclear Power Plants," NUREG-1437; (2) the environmental report submitted by the applicant; (3) consultation with other Federal, State, and local agencies; (4) its own independent review; and (5) its consideration of public comments, the staff recommended, in Supplement 3 to NUREG-1437, that the Commission determine that the adverse environmental impacts of license renewal for ANO-1 are not so great that preserving the option of license renewal for energy planning decision-making would be unreasonable.

1.3 Summary of Principal Review Matters

The requirements for renewing operating licenses for nuclear power plants are described in 10 CFR Part 54. The staff performed its technical review of the ANO-1 LRA in accordance with Commission guidance and the requirements of 10 CFR 54.19, 54.21, 54.22, 54.23, and 54.25. The standards for renewing a license are contained in 10 CFR 54.29.

In 10 CFR 54.19(a), the Commission requires applicants for license renewal to submit general information. Entergy submitted this general information in Enclosure 1 to its letter of January 31, 2000, regarding the application for a renewed operating license for ANO-1. The staff reviewed that enclosure, and found that the applicant submitted the information required by 10 CFR 54.19(a).

In 10 CFR 54.19(b), the Commission requires that LRAs include "conforming changes to the standard indemnity agreement, 10 CFR 140.92, Appendix B, to account for the expiration term of the proposed renewed license." Regarding the standard indemnity agreement, the applicant states the following in its LRA:

> The current Standard Indemnity Agreement for ANO-1 states in Article VII that the agreement shall terminate at the time of expiration of that license specified in Item 3 of the attachment to the Standard Indemnity Agreement. Item 3 of the attachment to the Standard Indemnity Agreement, as revised by Amendment No. 6, lists DPR-51 as an applicable license number. Entergy Operations requested that conforming changes be made to Article VII of the Standard Indemnity Agreement, and/or Item 3 of the attachment to the Standard Indemnity Agreement, specifying the extension of the Standard Indemnity Agreement until the expiration dates of the renewed ANO-1 Operating Licenses. Should the license number be changed upon issuance of the renewed license, Entergy Operations requests that conforming changes be made to Item 3 of the attachment and any other section of Standard Indemnity Agreement, as appropriate.

The staff will use the original license number for the renewed license. Therefore, there is no need to make conforming changes to the indemnity agreement, and the requirements of 10 CFR 54.19(b) have been met.

In 10 CFR 54.21, the Commission requires that each application for a renewed license for a nuclear facility contain the following information: (a) an integrated plant assessment (IPA), (b) current licensing basis changes made during the NRC review of the application, (c) an evaluation of TLAAs, and (d) a final safety analysis report (FSAR) supplement. On January 31, 2000, the applicant submitted the information required by 10 CFR 54.21(a) and (c) in Exhibit A of its LRA, entitled "License Renewal Application, Arkansas Nuclear One, Unit 1." The applicant submitted the information required by 10 CFR 54.21(b) on January 31, 2001. The applicant submitted the information to address the license renewal requirements of 10 CFR 54.21(d) in Appendix A to Exhibit B of its LRA, entitled "Safety Analysis Report Supplement."

In 10 CFR 54.22, the Commission states the requirements regarding technical specifications. The applicant addressed the requirements of 10 CFR 54.22 in Appendix D to Exhibit B of its LRA.

The staff evaluated the technical information required by 10 CFR 54.21 and 54.22 in accordance with the NRC's regulations and the guidance provided in the initial draft SRP. The staff's evaluation of this information is documented in Chapters 2, 3, and 4 of this SER.

The staff's evaluation of the environmental information required by 10 CFR 54.23 is documented in the draft plant-specific supplement to the GEIS (NUREG-1437, Supplement 3), which states the considerations related to renewing the license for ANO-1.

The report by the Advisory Committee on Reactor Safeguards required by 10 CFR 54.25 is contained in Chapter 5 of this SER. The findings required by 10 CFR 54.29 are presented in Chapter 6 of this report.

1.3.1 Babcock and Wilcox Topical Reports

In accordance with 10 CFR 54.17(e), the applicant also references a number Babcock and Wilcox Owners Group topical reports in its LRA. These topical reports were used by the applicant to generically demonstrate that applicable aging effects for reactor coolant system components will be adequately managed for the period of extended operation. Specifically, the applicant incorporated the following topical reports into its application:

- BAW-2241P, "Fluence and Uncertainty Methodologies," May 1997
- BAW-2243A, "Demonstration of the Management of Aging Effects for the Reactor Coolant System Piping," March 1996
- BAW-2244A, "Demonstration of the Management of Aging Effects for the Pressurizer," August 1997
- BAW-2248, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," July 1997
- BAW-2251, "Demonstration of the Management of Aging Effects for the Reactor Vessel," June 1996

The staff issued separate safety evaluations for these topical reports on the following dates: BAW-2243A on March 21, 1996; BAW-2244A on August 18, 1997; BAW-2241P on February 18, 1999; BAW-2251 on April 26, 1999; and BAW-2248 on December 9, 1999. In accordance with procedures established in NUREG-0390, "Topical Report Review Status," the staff requested that the Babcock and Wilcox Owners Group publish the accepted versions of the reports, which incorporates the transmittal letter and the staff's safety evaluation between the title page and the abstract. The accepted versions have an "A" (for "accepted") after the report identification number.

The safety evaluations of the topical reports are intended to be stand-alone documents. An applicant incorporating the topical reports by reference into its LRA must ensure that the conditions of approval stated in the safety evaluations are met. The staff's evaluation of how the topical reports were incorporated into the application is found in Section 3.4 of this SER.

1.4 Summary of Open Items

Upon completing its initial review, the staff identified and documented six open items in an SER dated January 10, 2001. The applicant responded to each of the open items by providing additional information in a letter to the NRC dated March 14, 2001. The following describes each of the six open items, the applicant's response to each item, and the staff's evaluation of the applicant's response.

- Open Item 2.3.2.6.2-1 - The ANO-1 UFSAR, Section 6.2.2.1, identifies an in-line flow orifice as being necessary to ensure the proper sodium hydroxide injection rate for pH control. This flow orifice was not identified as a component of the sodium hydroxide system that was subject to an AMR for its flow control intended function in Table 3.3-6 of the LRA.

  In response to this concern, the applicant added the flow control function for the sodium hydroxide in-line flow orifice to its AMR. This flow orifice is constructed of stainless steel, and is susceptible to cracking and loss of material. The inspection activities used to manage similar applicable aging effects of sodium hydroxide stainless steel components will be used to manage the aging of the in-line flow orifice for the flow control intended function. Aging management activities will be completed as part of the new ASME, Section XI, ISI augmented inspections activities evaluated in this SER, Section 3.3.1.4.9. This information was documented in a letter to the NRC staff dated March 14, 2001. The staff finds this resolution to Open Item 2.3.2.6.2-1 acceptable.

- Open Item 2.3.3.2.2-1 - The applicant does not include the fire protection (FP) jockey pump, carbon dioxide systems, fire hydrants, the water supply to the low level radwaste building FP system, and the piping to the manual hose station (located downstream of FS-43) as being within the scope of license renewal and subject to an AMR. The staff requested additional information for the exclusion of these components; however, at the time the initial SER was issued, the applicant had not provided sufficient justification for the exclusion of these components.

  In a public meeting with the applicant that took place on March 7, 2001, the applicant presented its position as to why the FP jockey pump, carbon dioxide systems, fire hydrants, the water supply to the low level radwaste building FP system, and the piping to the manual hose station (located downstream of FS-43) are not included in the applicant's CLB (as documented in the applicant's F-list) in accordance with the requirements of 10 CFR 50.48. The applicant explained that each of these components is maintained to the National Fire Protection Association standards, and provided a technical justification as to why these components are not required for safe shutdown consistent with General Design Criteria III. The staff presented its position that the requirements of10 CFR 50.48 go beyond safe shutdown, and a number of components beyond those required by GDC III are required by 10 CFR 50.48. As a result of this meeting, the applicant agreed to add the jockey pump and fire hydrants to the scope of SCs subject to an AMR and to its F-list consistent with the requirements of 10 CFR 50.48. At the same time, the applicant provided sufficient justification for not including the carbon dioxide systems, the water supply to the low level radwaste building FP system, and the piping to the manual hose station to the scope of components required by 10 CFR 50.48 for ANO-1. This information was documented in a letter to the NRC dated March 14, 2001. The staff had no additional concerns relating to the scope of FP components subject to an AMR, therefore, this item is considered closed.

- Open Item 3.3-1 - The staff reviewed the applicant's summary descriptions of the aging management programs (AMPs), and the evaluations of the time-limited aging analyses (TLAAs) provided by the applicant in Appendix A, "Safety Analysis Report Supplement," of the LRA, to ensure that they are consistent with the requirements of 10 CFR 54.21(d). The staff identified a number of summary descriptions of AMPs and TLAA evaluations that needed additional information to meet the intent of 10 CFR 54.21(d). The additional information needed include the following:

  - FSAR Item 3.3.1.2.3 - In its revised summary description of Section 16.0 of the FSAR Supplement, the applicant added a summary description of the quality assurance AMP to its FSAR Supplement. This summary description contains an adequate description of the corrective action program that specifically describes corrective actions, the confirmation process, and the administrative controls consistent with 10 CFR Part 50, Appendix B, as it applies to license renewal in accordance with 10 CFR 54.21(d). The staff finds the revised summary description as submitted by the applicant in a letter to the NRC dated March 14, 2001, acceptable.

  - FSAR Item 3.3.1.3.3 - In its revised summary description of Section 16.2.13 of the FSAR Supplement for the Maintenance Rule program, the applicant clarified that this program only applies to external surfaces of the SCs that are managed by this AMP. The staff finds the revised summary description as submitted by the applicant in a letter to the NRC dated March 14, 2001, acceptable.

  - FSAR Item 3.3.1.4.1.3 - A review of the LRA, the applicant's responses to the staff's request for additional information, and the programs credited with managing the aging of fire protection systems buried piping, the staff verified that buried pipe inspection program is not credited, and is not needed to manage the applicable aging effects. The staff finds that no change to Section 16.1.1 of the FSAR Supplemented, as submitted with the LRA, is needed.

  - FSAR Item 3.3.1.4.2.3 - In a letter to the NRC dated September 12, 2000, the applicant states that the Heat Exchanger Monitoring Program does not address fouling. The Heat Exchanger Monitoring Program will inspect heat exchangers to the extent required to ensure seismic qualification is maintained, but it is not intended to monitor for fouling. A staff review of the LRA, the applicant's responses to the staff's request for additional information, and the applicable AMPs, verified that fouling will be adequately managed by other programs such as the Service Water Integrity Program or system surveillance testing. The staff finds that no change to Section 16.1.3 of the FSAR Supplemented, as submitted with the LRA, is needed.

- FSAR Item 3.3.1.4.3.3 - After a review of the LRA, the applicant's responses to the staff's request for additional information, and the applicable AMPs, the staff verified that the wall thinning inspection program was not limited to the chilled water components of penetrations 51 and 59. Other reactor building isolation system carbon steel components credit the Wall Thinning Inspection Program. These other penetrations are correctly listed in the program description in Appendix B of the LRA (Section 3.7) and in the FSAR Supplement as submitted with the LRA. The staff finds that no change to Section 16.1.7 of the FSAR Supplement, as submitted with the LRA, is needed.

- FSAR Item 3.3.2.4.3 - In its revised summary description of Section 16.2.7 of the FSAR Supplement, the applicant states that if an inspection program is determined to be necessary for the CRDM nozzle and other vessel closure penetrations, the applicant will analyze and evaluate axial flaws using NUMARC acceptance criteria, and address circumferential flaws with the NRC on a case-by-case basis. The staff finds the revised summary description as submitted by the applicant in a letter to the NRC dated March 14, 2001, acceptable.

- FSAR Item 3.3.3.3 - In its revised summary description of Section 16.2.3.7 of the FSAR Supplement, the applicant includes a one-time inspection to detect cracking and wall thinning of piping and fittings in the sodium hydroxide system in the summary description of the Augmented Inspection program. The staff finds the revised summary description as submitted by the applicant in a letter to the NRC dated March 14, 2001, acceptable.

- FSAR Item 3.3.7.4 - In its revised summary description of Section 16.1.2 of the FSAR Supplement for inaccessible medium-voltage cables exposed to significant moisture and voltage, the applicant states that it will either test for the presence of aging effects or implement a periodic replacement program for these cables. If periodic replacement of medium-voltage underground cables is determined to be the most effective action for this type of cable, ANO-1 will define the frequency for replacement prior to the expiration of the initial 40-year licensing term. The frequency will be based on site-specific and industry operating experience. The staff finds the revised summary description as submitted by the applicant in a letter to the NRC dated March 14, 2001, acceptable.

- FSAR Item 4.3.4 - In its revised summary description of Section 16.3.2 of the FSAR Supplement, the applicant provides a proposed program to address the environmental effects of fatigue that meet the requirements of 10 CFR 54.21(d). The staff finds the revised summary description as submitted by the applicant in a letter to the NRC dated March 14, 2001, acceptable.

- FSAR Item 4.5.5 - In its revised summary description of Sections 16.2.3.6 and 16.3.4 of the FSAR Supplement, the applicant includes an adequate summary description of the prestress monitoring and trending activities, the acceptance criteria, and corrective actions for managing prestress tendons of the ANO-1 containment in the FSAR Supplement consistent with 10 CFR 54.21(d). The staff finds the revised summary description as submitted by the applicant in a letter to the NRC dated March 14, 2001, acceptable.

- FSAR Item 4.7.3 - In its revised summary description of Section 16.3.6 of the FSAR Supplement, the applicant provides a summary description of the monitoring, evaluation activities, optional corrective actions, and decision criteria for the aging of Boraflex in the spent fuel pool. The staff finds the revised summary description as submitted by the applicant in a letter to the NRC dated March 14, 2001, acceptable.

- Open Item 3.3.7.2-1 - Buried (inaccessible) medium-voltage cables, exposed to ground water typically do not have comparable accessible cables exposed to a similar environment that can serve as a sample for these inaccessible cables. For buried cable exposed to ground water that are within the scope of license renewal and subject to an AMR, visual inspection is not sufficient for managing a reduced insulation resistance to ground, and potential electrical failure due to moisture intrusion, water treeing, and contamination so that the intended function will be maintained consistent with the applicant's CLB for the period of extended operation in accordance with the requirements of 10 CFR 54.21(a)(3).

  In response to this concern, the applicant committed to implement either a test or replacement program for the cables of concern. If a testing program is implemented, inaccessible medium-voltage cables exposed to moisture and voltage will be tested for the presence of aging. The specific type of test that will be performed will be identified and implemented prior to entering the period of extended operation. This test will provide an indication of insulation integrity. Along with this test, the applicant will monitor and manage groundwater in manholes containing in-scope medium-voltage cables to reduce the exposure of these cables to moisture.

  The applicant is also considering a periodic replacement program based on industry and site-specific operational experience, as an alternate approach to testing and monitoring. If the applicant determines periodic replacement to be a more effective means of managing the aging of these cables, the program will be implemented prior to entering the period of extended operation. The staff finds this resolution to Open Item 3.3.7.2-1 acceptable.

- Open Item 4.5.2-1 - In response to an NRC staff RAI, the applicant did not adequately describe the AMP for the prestress forces for the ANO-1 containment. Specifically, the applicant needed to provide additional information regarding the prestress monitoring and trending activities, the acceptance criteria, and corrective actions when acceptance criteria are not met.

In a letter to the NRC dated March 14, 2001, the applicant provided sufficient information regarding the prestress monitoring and trending activities, the acceptance criteria, and corrective actions when acceptance criteria are not met. This information provided by the applicant and the staff's evaluation of this information is discussed in Section 4.5.2 of this SER. The staff finds the additional information regarding prestress tendon forces for the ANO-1 containment acceptable to resolve Open Item 4.5.2-1.

- Open Item 4.7.2-1 - The applicant needed to provide the basis upon which the staff can conclude that there is reasonable assurance that the effects of aging of Boraflex will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1).

  In a letter to the NRC dated March 14, 2001, the applicant acknowledges the analysis of Boraflex in the spent fuel storage racks as a time limited aging analysis. The applicant further states that the existing analysis is not valid through the license renewal period and cannot be acceptably projected to the end of the license renewal period as documented in a letter to the NRC dated September 6, 2000. In accordance with 10CFR54.21(c)(1)(iii), the applicant committed to continue its boraflex monitoring program to provide reasonable assurance that the effects of aging on the intended function will be adequately managed for the period of extended operation.

  In its March 14, 2001 letter, the applicant also provides the additional information regarding the boraflex monitoring program requested by the staff in a letter to the applicant dated May 5, 2000. This information provided by the applicant and the staff's evaluation of this information is discussed in Section 4.7.2 of this SER. The staff finds the additional information regarding the boraflex monitoring program acceptable to resolve Open Item 4.7.2-1.

THIS PAGE IS INTENTIONALLY LEFT BLANK

2 STRUCTURES AND COMPONENTS SUBJECT TO AN AGING MANAGEMENT REVIEW

2.1 Methodology for Identifying Structures and Components Subject to Aging Management Review

2.1.1 Introduction

Title 10 of the *Code of Federal Regulations*, Part 54 (10 CFR Part 54), "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," Section 54.21, "Contents of Application — Technical Information," requires that each application for license renewal contain an integrated plant assessment (IPA). Furthermore, the IPA must list and identify those structures and components (SCS) that are subject to an aging management review (AMR) from the systems, structures, and components (SSCs) that are within the scope of license renewal in accordance with 10 CFR 54.4.

In the LRA, Section 2.1, "Scoping and Screening Methodology," the applicant states that the Arkansas Nuclear One, Unit 1, (ANO-1) IPA was developed along traditional engineering disciplines (that is, mechanical, civil/structural, and electrical). The applicant also states that the scoping and screening methodology used to identify structures and mechanical components subject to an AMR is consistent with the industry guidance in the Nuclear Energy Institute's, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 — The License Renewal Rule" (NEI 95-10), Revision 0; the NRC's "Draft Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants"; and additional correspondence between the NRC, other applicants, and the NEI during the review and development of earlier license renewal applications (LRAs).

2.1.2 Summary of Technical Information in the Application

In the LRA, Sections 2.0 and 3.0, the applicant provides the technical information required by 10 CFR 54.21(a). In Section 2.1, "Scoping and Screening Methodology," of the LRA, the applicant describes the process used to identify the SSCs that meet the license renewal scoping criteria under 10 CFR 54.4(a), as well as the process used to identify the SCs that are subject to an AMR as required by 10 CFR 54.21(a)(1).

Additionally, Section 2.3, "Mechanical Systems Scoping and Screening Results"; Section 2.4, "Structures and Structural Components Scoping and Screening Results"; and Section 2.5, "Electrical and Instrumentation and Control System Scoping and Screening Results"; amplify the process that the applicant uses to identify the SCs that are subject to an AMR. The LRA, Chapter 3, "Aging Management Review Results," contains the following information: Section 3.1, "Common Aging Management Programs"; Section 3.2, "Reactor Coolant System"; Section 3.3, "Engineered Safeguards"; Section 3.4, "Auxiliary Systems"; Section 3.5, "Steam and Power Conversion Systems"; Section 3.6, "Structures and Structural Components"; and Section 3.7, "Electrical and Instrument and Control." Chapter 4, "Time-Limited Aging Analysis," contains the applicant's evaluation of time-limited aging analyses.

2.1.2.1 Technical Information for Identifying Systems, Structures, and Components Within the Scope of License Renewal

In the LRA, Section 2.1.2, “Assessment Using Criteria in 10 CFR 54.4,” the applicant describes the scoping methodology as it relates to the safety-related scoping criteria in accordance with 10 CFR 54.4(a)(1), the non-safety-related criterion in accordance with 10 CFR 54.4(a)(2), and the regulated events scoping criteria in accordance with 10 CFR 54.4(a)(3).

With respect to the safety-related criteria, the applicant states that the scope of license renewal includes those safety-related SSCs that are relied upon to remain functional during and following design-basis events (DBEs) (as defined in 10 CFR 50.49(b)(1)). As such, those safety-related SSCs ensure the following functions: (i) integrity of the reactor coolant pressure boundary; (ii) capability to shut down the reactor and maintain it in a safe shutdown condition; or (iii) capability to prevent or mitigate the consequences of accidents that could result in the potential offsite exposure comparable to the guidelines provided in 10 CFR 50.34(a)(1) or 10 CFR 100.11, as applicable.

The updated final safety analysis report (UFSAR) for ANO-1, Table 1-2, defines “safety-related” or “Q” SSCs on the basis of 10 CFR Part 100, Appendix A, as those SSCs required to ensure the following:

- integrity of the reactor coolant pressure boundary
- capability to shut down the reactor and maintain it in a safe shutdown condition
- capability to prevent or mitigate the consequences of accidents that could result in potential offsite doses comparable to the guideline doses of 10 CFR Part 100

The ANO-1 UFSAR, Table 1-2, also includes a summary-level Q-list for ANO-1 systems and structures. In the mid-1980s, the applicant implemented a Component Level Q-list (CLQL) project, which classified “Q” devices at the component level. The applicant maintains the CLQL in a component database. The ANO-1 summary and CLQL include those SSCs that are relied upon to remain functional during or following DBEs as described in UFSAR Chapter 14, as well as all other design conditions established within the ANO-1 current licensing basis (CLB).

The ANO-1 summary-level Q-list and CLQL were used during the IPA to identify the SSCs that are safety-related and within the scope of license renewal, consistent with 10 CFR 54.4(a)(1).

With respect to the non-safety-related criteria, the applicant states that the majority of SSCs whose failure could prevent satisfactory accomplishment of any of the safety-related functions in 10 CFR 54.4 (a)(1) are classified as safety-related at ANO-1. Therefore, except for a few cases (as described below), the SSCs meeting the criteria in 10 CFR 54.4(a)(1) and (a)(2) are summarized in the ANO-1 Q-list and listed on the CLQL, and are included within the scope of license renewal.

On the basis of a review of the ANO-1 UFSAR and design documents, the applicant identifies a few cases in which passive, long-lived, non-safety-related components could impact safety-related functions. These include spatially related components for which the physical location could result in interaction between components (including seismic or flooding interactions). Additionally, the spent fuel pool liner, although non-safety-related as documented in the ANO-1 UFSAR, has been included in the scope of license renewal, in part, because it protects the concrete walls from borated water and maintains the leak-tight integrity of the pool.

In addition, the following non-safety-related components have been included in the scope of license renewal, although they do not meet the criteria of 10 CFR 54.4(a)(2):

C non-safety-related valves and piping that are part of the pressure boundary for the main steam lines and steam generators inside the reactor building

C non-safety-related valves and piping in the auxiliary building sump system that are credited with preventing offsite releases

The few cases in which ANO-1 non-safety-related components could impact safety-related functions have been identified, and the associated components have been included in the scope of license renewal in accordance with the criterion of 10 CFR 54.4(a)(2).

With respect to other scoping criteria, the applicant reviewed all of the SSCs that are relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commission's regulations for fire protection (10 CFR 50.48), environmental qualification (10 CFR 50.49), pressurized thermal shock (10 CFR 50.61), anticipated transients without scram (10 CFR 50.62), and station blackout (10 CFR 50.63) to ensure that they are adequately accounted for in the scoping methodology. Design documentation to support this review was developed as part of the upper-level design (ULD) process. The ULDs were developed by the applicant during the design configuration documentation project (initiated in April 1988) to support the design basis adequacy verification for the ANO units. The ULDs define the design criteria, requirements, and bases for ANO systems and structures, design-basis accident (DBA) analyses, and topical (generic) areas such as fire protection, environmental qualification, flooding, high energy line break, and other design conditions consistent with the plant's CLB. The internal and external sources of information embodied in the ULDs include regulatory documents, industry codes and standards, design change package information, and general correspondence related to the design of the plant.

In summary, the SSCs relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with NRC regulations for fire protection, environmental qualification, pressurized thermal shock, anticipated transients without scram, and station blackout, have been included in the scope of license renewal in accordance with the criteria of 10 CFR 54.4(a)(3).

2.1.2.2 Technical Information for the Structures and Components Subject to an Aging Management Review

After identifying the SSCs that are within the scope of license renewal, the applicant implemented a process to determine which of the SCs would be subject to an AMR, in accordance with the requirements of 10 CFR 54.21(a)(1). The LRA, Section 2.1.3, "Assessment using criteria in 10 CFR 54.21(a)(1)," describes the screening activities used to determine the SCs that are subject to an AMR. The results of the screening activities are presented in the LRA, Section 2.3 for mechanical components, Section 2.4 for structures, and Section 2.5 for electrical commodities. The staff's evaluation of the screening results is contained in the corresponding sections of this safety evaluation report (SER).

Mechanical Components Review

The list of ANO-1 systems (mechanical and electrical) that are within the scope of license renewal was created from the scoping and screening methodology discussed above, as presented in Section 2.1 of the LRA. The following information contains additional detail relating to the review of mechanical components.

The ANO-1 reactor coolant system (RCS) is a typical Babcock and Wilcox (B&W) pressurized-water nuclear steam supply system with a reactor vessel, two steam generators, four reactor coolant pumps, a pressurizer, and the connecting or interfacing piping as the primary components. The RCS is an American Society of Mechanical Engineers (ASME) Class 1 system that is safety-related and, therefore, within the scope of license renewal. The components that makeup the RCS pressure boundary are within the scope of license renewal.

The non-Class 1 mechanical systems determined to meet the 10 CFR 54.4 criteria were included in the scope of license renewal. Many of these systems (such as high pressure injection, low-pressure injection, core flood, reactor building spray, emergency feedwater, reactor building cooling, emergency diesel generators (EDGs), hydrogen control, penetration room ventilation, control room ventilation, and so forth) have functions important to safety that are required during DBEs, and are clearly within the scope of license renewal. Other systems (such as service water and fuel oil systems) are needed to support the function(s) of safety-related systems, and are also included within the scope of license renewal.

Portions of some non-Class 1 systems required for normal plant operation (such as main feedwater and main steam) can perform one or more safety-related function(s) and, therefore, are included in the scope of license renewal. Portions of the instrument air system that are necessary for the operation of safety-related valves and dampers are included on the Q-list, and are within the scope of license renewal. Portions of the condensate storage and transfer system required to support emergency feedwater system operation are on the Q-list and are within the scope of license renewal. Portions of the chilled water system that support operation of safety-related cooling units are included on the Q-list, and are within the scope of license renewal.

The Halon system, portions of the fire protection system required to support 10 CFR 50.48, and the alternate AC (AAC) diesel generator and supporting equipment are also within the scope of license renewal in accordance with 10 CFR 54.4(a)(3).

Upon identifying the SSCs that are within the scope of license renewal, the applicant performed a screening review to determine which mechanical SCs would be subject to an AMR. The applicant stated that the screening process used in this review is consistent with the guidance in NEI 95-10. The mechanical components subject to an AMR were identified by a review of ANO-1 piping and instrumentation diagrams, the ANO-1 UFSAR, and the ANO-1 ULDs. The applicant determined the applicable intended function(s) for each of these components by reviewing the ANO-1 UFSAR and other design documents. The applicant then identified the mechanical components that perform applicable intended function(s) without moving parts or without a change in configuration or properties, and that are not subject to replacement based on qualified life or specified time period.

Structures and Structural Component Review

In the LRA, Section 2.4, the applicant states that the list of ANO-1 structures that are within the scope of license renewal was identified by reviewing the UFSAR, site plans and general arrangement drawings, and other plant-specific documents. Safety-related and non-safety-related structures whose failure could prevent satisfactory accomplishment of any safety-related function(s) was classified consistent with the CLB as documented in UFSAR, Table 1-2. ANO-1 structures are designated as either seismic Category 1 or seismic Category 2. As defined in the ANO-1 UFSAR, and consistent with the requirements of 10 CFR 54.4(a)(1)(iii), seismic Category 1 structures are those that prevent uncontrolled release of radioactivity, and are designed to withstand design-basis loadings without loss of intended function. Consequently, the applicant determined that ANO-1 seismic Category 1 structures meet the criteria required in 10 CFR 54.4(a)(1) and, therefore, are within the scope of license renewal.

Seismic Category 2 structures are those structures that can withstand limited damage without causing a release of radioactivity, without limiting a controlled plant shutdown, and possibly interrupting power generation. The UFSAR, Chapter 5, states that seismic Category 2 structures do not perform a nuclear safety-related function, but its failure could possibly affect the function(s) of a safety-related system. This is consistent with the scoping requirement of 10 CFR 54.4(a)(2). Consequently, the applicant has determined that some ANO-1 seismic Category 2 structures meet the scoping requirement of 10 CFR 54.4(a)(2) and, therefore, are within the scope of license renewal.

In addition, the applicant reviewed the list of ANO-1 structures that were included within the scope of license renewal in accordance with 10 CFR 54.4(a)(1) and (a)(2), and concluded that this list included the structures that meet the requirements of 10 CFR 54.4(a)(3).

Upon identifying the SSCs that are within the scope of license renewal, the applicant performed a screening review to determine which structures and structural components would be subject to an AMR. In doing so, the applicant divided the structures and structural components into structural component groups for the AMR. Many structural components do not have unique equipment identifiers. Unique identifiers are not needed in this application because the structural components that makeup each grouping are well defined, all structures and structural components within a grouping were reviewed as a group, and the applicable aging management programs (AMPs) were applied to all structural components within each grouping.

The applicant then determined the intended function(s) of the various structures and structural components by reviewing information contained in the ANO-1 UFSAR, engineering documents, and NEI 95-10. The applicant identified the structures and structural components that perform applicable intended function(s) without moving parts or without a change in configuration or properties, and that are not subject to replacement based on qualified life or specified time period.

In summary, the applicant divided the various structural components into three groups on the basis of material of construction and component-level function, with sub-materials indicated, as appropriate. Structural intended functions by structural component groupings were identified and used for the AMR. Bulk commodities were also identified and grouped on the basis of materials of construction, with sub-materials indicated, as appropriate.

Electrical Components Review

The ANO-1 electrical systems include an offsite power supply from the switchyard, two essential trains (red and green) of onsite electrical distribution that supply power to safety-related components, and a non-safety-related power supply for non-safety-related equipment. Upon identifying the SSCs that are within the scope of license renewal, the applicant performed the following screening review to determine which electrical components would be subject to an AMR. As part of this effort, the applicant participated in an industry initiative, coordinated by the NEI, to develop a commodity evaluation approach. The passive, long-lived electrical components were grouped into commodities consistent with NEI 95-10, Appendix B, and the following passive electrical component groups were identified as requiring an AMR: splices, connectors, terminal blocks, and cables. Excluded from these commodities are individual splices, connectors, and terminal blocks that are classified as piece-parts of larger complex assemblies. For example, the wiring, terminal blocks, and connectors located internal to a breaker cubicle were considered piece-parts of the breaker. Because a breaker is an active component not subject to an AMR, the piece-parts that share in the intended function of that component are not subject to an AMR.

2.1.3 Staff Evaluation

In reviewing the ANO-1 LRA, the NRC staff evaluated the scoping and screening activities described in the following sections:

- Section 2.1, “Scoping and Screening Methodology,” and Section 2.2, “Plant-Level Scoping Results,” to ensure that the applicant describes a process for identifying SSCs at ANO-1 that are within the scope of license renewal in accordance with the requirements of 10 CFR 54.4(a)(1), (a)(2), and (a)(3)

- Section 2.3.1.1, “Description of the Process to Identify Reactor Coolant System Components Subject to Aging Management Review“; Section 2.5.1, “Electrical and Instrumentation and Control System Scoping and Screening Results — Purpose and Scope”; Section 3.2.1, “Description of the Process to Identify Aging Effects Requiring Management for Reactor Coolant System Components”; and Section 4.1, “Identification of Time-Limited Aging Analyses,” to ensure that the applicant describes a process for determining structural, mechanical, and electrical components at ANO-1 that are subject to an AMR for renewal in accordance with the requirements of 10 CFR 54.21(a)(1) and 54.21(a)(2)

In addition, the staff performed an onsite audit of the applicant’s processes to ensure that the applicant had developed and implemented adequate guidance to conduct the scoping and screening of ANO-1 SSCs in accordance with the methodologies described in the application.

2.1.3.1 Evaluation of the Methodology for Identifying Systems, Structures, and Components Within the Scope of License Renewal

From May 22 through May 24, 2000, the NRC staff conducted an audit of the ANO-1 license renewal scoping and screening methodology at ANO-1 in Russellville, Arkansas. During this audit, the staff performed a review of the scoping and screening methodology that included detailed discussions with the cognizant engineers on the implementation and control of the program, a review of administrative controls, and a review of design documents used by the applicant during the scoping and screening activities.

As a result of the audit, the staff obtained information regarding the scoping and screening methodology. Specifically, the applicant described in detail the CLQL development process and the ULD document program, which was the basis for verifying safety- and non-safety-related design functions for specific SSCs. The ULD documentation included a series of system specific evaluations, a set of DBA analysis evaluations, and a set of topical (generic) evaluations beyond those in the UFSAR Chapter 14 accident analysis, which provided the basis for initial inclusion of SSCs that are within the scope of license renewal. The combined efforts for developing the ULDs and CLQL were instrumental in identifying the design basis and design conditions considered in implementing the LRA scoping and screening methodology.

The NRC audit team reviewed a sample of the system-level and topical-level ULD reports to better understand the approach that the applicant implemented to determine which SSCs would initially be included within the scope of license renewal. The team found that the ULD documents provided a concise, well-documented discussion of the systems, including safety-related, non-safety-related, and NRC-required functions (i.e., those that were assigned as a result of commitments to the NRC, including those for the Commission regulations identified under 10 CFR 54.4 (a)(3)). Each ULD contained a detailed list of information sources which included both ANO-specific sources (such as the SER, technical specifications, quality assurance manual, and ANO-1 emergency plan), as well as non-ANO sources (such as industry codes and standards, NUREGs, regulatory guides, bulletins, notices, generic letters, and commission orders). The ULD documentation was developed in accordance with site-specific procedure GES-26, “ULD Writers Guide.” The ULD documentation is controlled and maintained in accordance with the applicant’s Nuclear Quality Assurance Program through the implementation of a series of site procedures including NES-16, “Accident Analysis ULD and AIM Basis Document Format and Content”; Procedure 5010.007, “Control of Upper Level Documents”; Procedure 5010.004, “Design Document Changes”; and Procedure 1000.150, “Licensing Document Maintenance.” The NRC audit team reviewed the governing procedures, and determined that they presented adequate guidance for the preparation, control, and maintenance of the ULDs.

With respect to the CLQL process, ULD-0-TOP-22, “ANO Component Classification Topical,” describes the applicant’s CLQL project for the development of the Q-list. The applicant started the CLQL project in 1985 to provide information to support plant operation, and in response to the Salem Anticipated Transient Without Scram event (Generic Letter 83-22).

On the basis of the applicant’s scoping definition, the Q-classification implies that a structure, system, or component is designed to the Class 1 seismic standards, subject to the full scope of the nuclear quality assurance program. In addition to the Q-classification, the applicant’s program defines 16 major system-level intended functions (including reactivity control, reactor core cooling geometry, RCS pressure boundary integrity, RCS inventory, secondary heat removal, containment isolation, containment pressure and temperature control, containment combustible gas control, indirect radioactive release, habitability, spent fuel storage control, display of event information for operator, structural integrity, interaction isolation, essential cooling, environmental support, and essential electrical support) which support the three functional criteria of the Q-scope definition. These system-level intended functions provided further guidance for determining if a component performed a safety-related or non-safety-related function. The CLQL is maintained and controlled in the applicant's component database in accordance with the nuclear quality assurance program through the implementation of a series of site procedures, including Procedure 1409.66, "Component Level Q-List Project Design Review"; Memorandum NEL-057-22, "CLQL Project Implementation - 10 CFR 50.59 Evaluation"; Impell Project Instruction 0260-098-PI-01, "Component-Level Q-list Development"; and procedure 5010.036, "Component Classification Process."

In reviewing the CLQL process, the NRC audit team evaluated a sample of the System Safety Function Review Forms, which were developed by the applicant during the CLQL program to describe each plant system in terms of its safety-related and non-safety-related functions, as defined by the 16 major system-level intended functions. In preparing the review forms, the applicant identified the specific design documentation referenced for each system, including the SER sections and individual design drawings for the system.

During the audit, the applicant further described the process used to incorporate the information from the CLQL and ULD projects into the LRA development process. The applicant referenced ANO-1 Engineering Reports 93-R-1009-01, "ANO-1 License Renewal Project Methodology and Management Plan"; and 93-R-1010-01, "ANO-1 License Renewal Integrated Plant Assessment System and Structures Screening," to describe the detailed process for developing the LRA, and incorporating the ULD and CLQL information into the screening process. These reports outlined the specific use of the ULD and CLQL within the scoping methodology, and presented formal guidance for use during the implementation phase. The applicant's engineering staff were cognizant of the need to use the ULD and CLQL during the scoping development phase of the LRA project.

On the basis of discussions with the applicant's cognizant engineering staff, and a review of selected design documentation in support of the process, the NRC audit team concluded that the applicant's staff understood and adequately implemented the scoping and screening methodology established in the applicant's LRA.

2.1.3.2 Evaluation of Methodology for Identifying Structures and Components Subject to an Aging Management Review

Mechanical Components

During the audit of the ANO-1 license renewal scoping and screening process, the NRC audit team reviewed the methodology used by the applicant to identify and list the mechanical components subject to an AMR, as well as the applicant's technical justification for this methodology. The team also examined the applicant's results from the implementation of this methodology by reviewing an overview of the mechanical systems identified as being within the scope of license renewal, a sample of evaluation boundaries drawn within those systems, the resulting components determined to be within the scope of license renewal, the corresponding component-level intended functions, and the resulting list of mechanical components subject to an AMR.

The methodology for identifying mechanical components that are within the scope of license renewal included the following steps:

- Identify all systems and their intended functions that are relied upon to remain functional during and following a DBE for which the plant must be designed.

- Identify all systems and intended functions whose failure could prevent satisfactory accomplishment of any of the intended functions identified in accordance with the requirements of 10 CFR 54.4(a)(1).

- Identify all systems and intended functions necessary to demonstrate compliance with the regulated events identified in accordance with the requirements of 10 CFR 54.4(a)(3).

Beginning with the results of the CLQL, the applicant identified all of the ANO-1 systems that are within the scope of license renewal. To do so, the applicant included the reactor coolant system Class 1 components without any additional evaluation. For the remaining systems determined to be within the scope of license renewal that contain non-Class 1 components, the applicant used the CLQL to identify the system-level intended functions and evaluation boundaries. The applicant also used system drawings to highlight all of the components for those systems included in the CLQL. In addition, the applicant sampled the components outside of the established evaluation boundary to verify that none of those components contributed to the applicable intended functions. Any such mechanical components were determined to be within the scope of the Rule and subject to an AMR. The applicant also added the fire protection components from the F-list, station blackout components and non-Q-list components whose failure could prevent satisfactory accomplishment of any of the safety-related intended functions from the S-list, equipment qualification components from the EQ-list, and anticipate transient without scram (ATWS) components identified from the review of their 10 CFR 50.62 commitments to the list of components requiring an AMR. The applicant reviewed its 10 CFR 50.61 commitments, and found that no additional components needed to be added to the scope of license renewal for pressurized thermal shock.

The applicant then used the requirements of the Rule and the guidance in NEI 95-10 to identify the components that performed its intended function(s) without moving parts or without a change in configuration or properties, and that are not replaced on the basis of qualified life or specified time period to determine which components are subject to an AMR. The applicant then developed a generic guide using BAW-2270, “Non-Class 1 Implementation Guideline and Mechanical Tools,” to determine the applicable aging effects for each SC subject to an AMR. The mechanical tools include a list of mechanical component types, a description of susceptible materials and environments, related aging effects that need to be managed, and guidance on how to demonstrate that the effects of aging are being managed, as is further evaluated in Sections 2.3, 3.3.2, 3.3.3, 3.3.4, and 3.3.5 of this SER.

Structures

During the audit of the ANO-1 license renewal scoping and screening process, the NRC staff also examined the structures and structural components included within the scope of license renewal, the corresponding structural-level intended functions, and the resulting list of structural components subject to an AMR.

In determining the structures and structural components included within the scope of license renewal, the applicant reviewed the CLQL, F-list, S-list, EQ-list and ATWS to identify any structure that contained any SSC that is within the scope of license renewal and subject to an AMR. Each structure that contains any of these components was included within the scope of license renewal and subject to an AMR. The only identified exception is the turbine building. The shared wall between the auxiliary building and the turbine building is designated as a turbine building wall on the site drawing. As a result of this unique configuration, the shared wall of the turbine building is designated as being within the scope of license renewal and subject to an AMR. In addition, a number of fire doors and walls required by 10 CFR 50.48 are also located in the turbine building and are subject to an AMR. However, due to the fact that fire protection components (pursuant to 10 CFR 50.48) are not required to be seismically qualified, there is no need to include the turbine building itself in the scope of license renewal.

After identifying the structures and structural components subject to an AMR, the applicant reviewed industry operating experience (from the Babcock and Wilcox Owners Group Generic License Renewal Program) to identify the applicable aging effects. This review resulted in report BAW-2279P, “Aging Effects for Structures and Structural Components,” which is referred to as the structural tools. This report is used to evaluate the materials and environments applicable to ANO-1. Accordingly, these structural tools were used in the AMR for ANO-1 structures.

To facilitate the identification of aging effects, the structures and structural component groupings, that were subject to an AMR, were subdivided into the following major groups:

- steel
- threaded fasteners
- concrete
- fire barriers
- elastomers
- earthen structures
- Teflon

The applicant then performed an aging effect evaluation for each material group. The evaluation included the following activities:

- identifying the components and commodities that are within the scope of license renewal on the basis of material type(s).

- determining whether in-scope components and commodities are long-lived and, thus, subject to an AMR.

- identifying plant operating environments
- determining applicable aging effects
- demonstrating the adequacy of AMPs

The AMRs utilize the BAW-2279P methodology, along with existing industry experience, to perform the aging effect evaluation. Only those materials and environments that were determined to result in potential aging effects are evaluated in the AMRs. Potential aging effects identified by this review were determined to be applicable if a plant specific material and environment matched the material and environment of the potential aging effect.

The applicant then prepared site-specific AMRs (engineering reports 93-R-1015 series) for each of the major structures that are within the scope of license renewal (reactor building, reactor building internals, auxiliary building, and the intake structure). The applicant also prepared other reports to document the review of earthen embankments (emergency cooling pond, intake/discharge canals) and yard structures. The applicant prepared a separate report, entitled “Bulk Commodities” to document the review of non-building-specific structural commodities (piping supports, cable trays, electrical cabinets, and so forth). The structural AMR reports are formatted to provide the scope, construction materials, operating environments, applicable aging effects, and a demonstration that the effects of aging are managed as described in Sections 2.4 and 3.3.6 of this SER.

Electrical Components

During the audit of the ANO-1 license renewal scoping and screening process, the NRC staff evaluated the applicant’s implementation of this methodology by reviewing the list of electrical components subject to an AMR.

The audit team reviewed the methodology described in the LRA, Section 2.5.3, entitled “Screening of Electrical SSCs.” The audit team also reviewed ANO-1 engineering report 93-R-1017-1, which described the electrical AMR process. The applicant used the action plan for the generic plant spaces and commodity evaluation methodology developed by the Babcock and Wilcox Owners Group Generic License Renewal Program electrical review group. Passive, long-lived electrical components were categorized and segregated primarily using the NEI 95-10 suggested categorization as a guide.

To review passive electrical components, the applicant used a combination of the “plant spaces” and “commodity” grouping approaches, as listed in the Sandia Report, “Aging Management Guideline for Commercial Nuclear Power Plants – Electrical Cable and Terminations,” as described in Sections 2.5 and 3.3.7 of this SER.

The applicant then prepared site-specific engineering reports to document its review of the passive electrical components that are within the scope of license renewal. The primary engineering report for the electrical components (93-R-1017-01) identifies the component types that the applicant considered to be within the scope of license renewal, as well as the application of the Sandia plant spaces and commodity grouping approaches. The applicant prepared a series of screening reports to identify the passive electrical components that are within the scope of license renewal, and are exposed to the significant stressors identified in the Sandia Report. Plant walkdowns were completed, as required, to identify localized hot spots. Screening of components was performed utilizing the site component (SIMS and WMS) and the cable (PDMS) databases. The applicant then used the intended functions from the scope activities, identified the aging effects, and performed an AMR consistent with the GLRP action plan.

2.1.4 Conclusions

On the basis of the review performed above, the NRC staff finds that there is reasonable assurance that the applicant's methodology for identifying the SSCs that are within the scope of license renewal and SCs that are subject to an AMR is consistent with the requirements of 10 CFR 54.4 and 10 CFR 54.21(a)(1), respectively, and is, therefore, acceptable.

2.1.5 References for Section 2.1

1. ULD-0-TOP-22, "ANO, Unit 1 and 2 Component Classification Topical," Revision 0.
2. 93-R-1009-01, ANO-1 License Renewal Project Methodology and Management Plan, Revision 0
3. ANO-1 Updated Final Safety Analysis Report
4. 93-R-1010-01, "ANO-1 License Renewal Integrated Plant Assessment System and Structures Screening," Revision 0
5. Letter from C. Randy Hutchinson, Entergy Operations, Inc., to the U.S. Nuclear Regulatory Commission, "Response to NRC Request Under 10 CFR 50.54(f) Regarding Adequacy and Availability of Design Bases Information." February 7, 1997
6. NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 — The License Renewal Rule," Revision 0, March 1996
7. Working Draft, "NRC Generic Aging Lessons Learned Report (GALL)," August 2000.
8. Procedure GES-26, "ULD Writers Guide," Revision 1
9. Procedure NES-16, "Accident Analysis ULD and AIM Basis Document Format and Content," Revision 1
10. Procedure 1000.150, "Licensing Document Maintenance," Revision 2
11. Procedure 1409.66, "Component Level Q-List Project Design Review," Revision 0
12. Procedure 5010.004, "Design Document Changes," Revision 3
13. Procedure 5010.007, "Control of Upper Level Documents," Revision 3
14. DG-1047, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," Working Draft, April 21, 2000

2.2 Plant Level Scoping Results

2.2.1 Introduction

The statement of considerations (SOC) for the License Renewal Rule (60 FR 22478) indicate that an applicant has the flexibility to determine the set of SCs for which an AMR is performed, provided that this set encompasses the SCs for which the Commission has determined an AMR is required. Accordingly, the staff focused its review on verifying that the implementation of the applicant's methodology discussed in Section 2.1 of this SER did not result in the omission of SCs that are subject to an AMR in accordance with 10 CFR 54.21(a)(1). The staff performed the following two-step evaluation:

- The first step was to determine whether the applicant has properly identified the SSCs that are within the scope of license renewal, in accordance with 10 CFR 54.4. As described in more detail below, the staff reviewed selected SSCs that the applicant did not identify as being within the scope of license renewal to verify that they do not meet any of the scoping criteria in 10 CFR 54.4(a).

- The second step was to determine whether the applicant has properly identified the SCs that are subject to an AMR from among the SSCs identified in the first step in accordance with 10 CFR 54.21(a)(1). As described in more detail below, the staff evaluated the evaluation boundaries for the systems and structures included within the scope of license renewal to verify that all the SCS, that contributed to the intended function(s) within the scope of license renewal, were considered during the AMR. The staff also evaluated the SCs within the evaluation boundaries to verify that all passive/long-lived SCs were subject to an AMR. The staff did not review SCs that the applicant had identified as subject to an AMR because it is an applicant's option to include more SCs than those required by 10 CFR 54.21(a)(1).

The staff performed the following scoping and screening review to determine if there is reasonable assurance that the applicant has identified and listed those SCs that are subject to an AMR to meet the requirements stated in 10 CFR 54.21(a)(1).

2.2.2 Summary of Technical Information in the Application

In Sections 2.3 through 2.5 of the LRA, the applicant describes the SCs that are within the scope of license renewal and subject to an AMR in accordance with 10 CFR 54.4, and 54.21(a)(1), respectively.

Component supports for equipment and piping, which are common to two or more components and within the scope of license renewal, are presented as bulk commodities in Section 2.4.6.2 of the LRA. Electrical components that support the operation of the systems presented in Sections 2.3 are presented in Section 2.5 of the LRA. The staff evaluated component supports that are identified as "bulk commodities" and electrical components for all systems and structures in Section 2.4.6.2 and 2.5 of this SER, respectively.

The staff used the ANO-1 UFSAR in performing its review. Pursuant to 10 CFR 50.34(b)(2), the UFSAR contains "[a] description and analysis of the SSCs of the facility, with emphasis upon performance requirements, the bases, with technical justification therefor, upon which such requirements have been established, and the evaluations required to show that safety functions will be accomplished." The UFSAR is required to be updated periodically pursuant to 10 CFR 50.71(e). Thus, the UFSAR contains updated plant-specific licensing-basis information regarding the systems, SSCs and their functions.

The staff reviewed Sections 2.3 through 2.5 of the LRA to determine if there is reasonable assurance that the applicant has appropriately identified and listed those SCs that are subject to an AMR to meet the requirements stated in 10 CFR 54.21(a)(1).

2.2.3 Staff Evaluation

In the LRA, Section 2.1, the applicant describes its methodology for identifying the SCs that are within the scope of license renewal and subject to an AMR. This IPA methodology typically consists of a review of all plant SSCs to determine those that are within the scope of license renewal in accordance with the requirements of 10 CFR 54.4. From those SSCs that are within the scope of license renewal, an applicant will identify and list those SCs that perform intended function(s) without moving parts, or without a change in configuration or properties, and that are not replaced based on qualified life or specified time period. The staff reviewed the scoping and screening methodology, and provided its evaluation in Section 2.1 of this SER. The applicant documents the implementation of that methodology in Sections 2.3 through 2.5 of the LRA.

To ensure that the scoping and screening methodology described in Section 2.1 of the LRA was implemented properly and identified the SCs that are subject to an AMR, the staff performed the following additional review. The staff sampled the contents of the UFSAR based on the listing of systems and structures on Tables 2.2-1 and 2.2-2 of the LRA to identify systems or structures that may have intended functions that meet the scoping requirements of 10 CFR 54.4 that the applicant does not include within the scope of license renewal. The staff selected several systems and structures, such as structures that support the ultimate heat sink, and the atmospheric vent system, and in a letter to the applicant dated May 5, 2000, the staff requested additional information about these systems and structures. In a letter to the NRC dated August 30, 2000, the applicant provides its response to the staff's requests for additional information (RAIs).

Specifically, the staff requested that the applicant provide justification for omitting the Dardanelle Dam and certain components in the atmospheric vent and main chiller cooling water systems. In a letter to the NRC dated August 30, 2000, the applicant states that although the Dardanelle Dam is part of the ultimate heat sink complex, the dam was in existence before the construction of ANO-1, and is under the jurisdiction of the U.S. Army Corps of Engineers. They are responsible for the inspection and maintenance programs that are expected to adequately manage the aging effects on the dam for the period of extended operations. The staff stated in a letter to NEI dated May 5, 1999, "License Renewal Issue 98-0100, Crediting Federal Energy Regulatory Commissions (FERC) – Required Inspection and Maintenance Programs for Dam Aging," that dam inspections and maintenance performed under the jurisdiction of FERC or the U.S. Army Corps of Engineers, continued through the period of license renewal, will be adequate for the purpose of aging management. Other structures that comprise the ultimate heat sink complex that are under the control of the applicant (e.g., earthen embankments) are reviewed in Section 2.4 of this document.

The applicant also states that the components of concern in the atmospheric vent system and the main chiller cooling water system do not meet the criteria for being within the scope of license renewal. The staff reviewed the applicant's responses, the LRA, and ANO-1 UFSAR, and agreed with the applicant that these systems do not have components that are within the scope of license renewal.

2.2.4 Conclusions

The NRC staff reviewed the information submitted by the applicant in the LRA, information in the ANO Unit 1 FSAR, and additional information in the applicant's August 30, 2000, response to the NRC, and did not identify any SSCs with intended functions that were not already evaluated in the LRA. Therefore, the staff finds that there is reasonable assurance that the applicant has appropriately identified the SSCs that are within the scope of license renewal in accordance with 10 CFR 54.4. The NRC staff's review of the SCs that are subject to an AMR is provided in Section 2.3 through 2.5 of this SER.

2.2.5 References for Section 2.2

1. 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."
2. DG-1047, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," Working Draft, April 21, 2000.
3. "Arkansas Nuclear One - Unit 1, License Renewal Application," dated January 31, 2000.
4. "ANO-1 Updated Final Safety Analysis Report."

THIS PAGE IS INTENTIONALLY LEFT BLANK

2.3 Mechanical Systems Scoping and Screening Results

2.3.1 Reactor Coolant System

In the ANO-1 LRA, Section 2.3.1, “Reactor Coolant System Mechanical Components,” the applicant describes the mechanical components of the reactor coolant system (RCS) that are within the scope of license renewal and subject to an AMR. The NRC staff reviewed this section of the LRA to determine whether the applicant has adequately demonstrated that requirements of 10 CFR 54.4, 10 CFR 54.21(a)(1), and 10 CFR 54.21(a)(2) have been met for the mechanical components of the RCS.

2.3.1.1 Technical Information in the Application

As described in the LRA, the following components are within the reactor coolant pressure boundary: reactor vessel, once-through steam generators (primary side), pressurizer, reactor coolant pump, main coolant piping, and portions of other systems attached to these components. The attached systems that contain Class 1 components include the core flood system, makeup/high-pressure injection system, and decay heat/low-pressure injection system. In addition, RCS vents, drains, and instrumentation lines also contain Class 1 components. RCS piping includes piping, fittings, branch connections, safe ends, thermal sleeves, pressure-retaining parts of RCS valves, and bolted closures and connections.

Non-Class 1 portions of the systems attached to the RCS are discussed in the following sections of the LRA:

- 2.3.2.1 - Core Flood
- 2.3.2.2 - Low-Pressure Injection/Decay Heat
- 2.3.2.3 - High-Pressure Injection/Makeup and Purification
- 2.3.2.7 - Reactor Building Isolation

Reactor Coolant System Piping

The NRC staff has reviewed the Babcock & Wilcox Owner Group (B&WOG) topical report BAW-2243A, “Reactor Coolant System Piping,” and has approved its use by participating applicants for license renewal. The applicant participated in the development of BAW-2243A by providing ANO-1-specific design and operational information. The applicant has subsequently reviewed the current design and operation of the ANO-1 RCS piping, and confirms that the ANO-1 Class 1 piping is bounded by the description of Class 1 piping contained in BAW-2243A, with regard to materials and operating environment. ANO-1 specific evaluations of RCS piping components not addressed in BAW-2243A include the fast response resistance temperature element (RTE) thermowell, the letdown coolers, and the reactor vessel leakage monitoring piping connected to the reactor vessel. The staff’s review of these evaluations is discussed below.

The fast-response RTE connections include a thermowell mounted within the mounting boss. The thermowell, which is constructed from Type 304 austenitic stainless steel, was not evaluated in BAW-2243A. In addition, the evaluation boundary in BAW-2243A did not include the non-Class 1 instrumentation tubing that connects the second isolation valve to the instrumentation. These items are part of the RCS pressure boundary at ANO-1, are constructed from austenitic stainless steel, and are evaluated in Section 3.3.2 of this SER. Thus, the thermowell is included within the scope of license renewal and subject to an AMR.

The letdown coolers are heliflow shell and tube heat exchangers with spiral Type 304 stainless steel tubes and manifolds, carbon steel casing shells, and carbon steel casing end plates. The tube side was designed in accordance with ASME Section III, Class C, and the shell side was designed in accordance with ASME Section VIII. The primary water enters the tubes at approximately 555EF during normal plant operation, and is cooled to approximately 120EF by intermediate cooling water (treated water) flowing through the shell. The intermediate cooling water enters at approximately 95EF, and exits at less than 175EF. Both coolers are in service during normal plant operation, with a relatively constant intermediate cooling water flow rate. The total letdown flow rate that is split between the coolers is manually varied anywhere between 45 and 140 gpm, as required for RCS inventory control. The letdown flow through the coolers may be manually or automatically terminated. The letdown coolers are within the scope of license renewal, and are subject to an AMR.

The reactor vessel leakage monitoring system piping at ANO-1 is 1-inch, Schedule 160, Class 3 piping. The lines do not support the RCS pressure boundary, and were not addressed in either BAW-2243A (RCS Piping Report) or BAW-2251A (Reactor Vessel Report). If the reactor vessel closure flange O-rings fail and RCS fluid is introduced into the monitoring piping, leak flow would be limited since the ½-inch diameter hole in the vessel flange, which connects the region between the O-rings to the monitoring pipe, is less than the inside diameter of the monitoring pipe. Therefore, the reactor vessel leakage monitoring piping is not within the scope of license renewal because it does not directly support the RCS pressure boundary, nor does it meet any of the other scoping criteria under 10 CFR 54.4(a).

Pressurizer

The pressurizer is a vertical cylindrical vessel with a penetration connecting the surge line to the hot-leg piping. The pressurizer contains electric heaters in its lower section, and a water spray nozzle in its upper section. Since sources of heat in the RCS are interconnected by piping with no intervening isolation valves, relief protection is provided on the pressurizer. Over-pressure protection consists of two code safety valves and one power-operated relief valve.

Piping attached to the pressurizer is Class 1 up to and including the second isolation valve (with the exception of the pressurizer code safety valve), and is discussed in Section 2.3.1.3 of the LRA. Additional descriptions of the pressurizer and related components are contained in BAW-2244. The applicant has reviewed the current design and operation of the ANO-1 pressurizer, and has confirmed that the pressurizer is bounded by the description contained in BAW-2244A. The ANO-1 pressurizer and related components are within the scope of license renewal and subject to an AMR.

Reactor Vessel

The reactor vessel consists of the cylindrical vessel shell, lower vessel head, closure head, nozzles, interior attachments, and associated pressure-retaining bolting. Coolant enters the reactor vessel through the inlet nozzles, passes down through the annulus between the thermal shield and vessel inside wall, reverses at the lower head, passes up through the core, turns around through the plenum assembly, and leaves the reactor vessel through the outlet nozzles.

The reactor vessel has two outlet nozzles that allow reactor coolant to enter the steam generators, and four inlet nozzles that allow reactor coolant to enter the reactor vessel from the discharge of the reactor coolant pumps. Two smaller nozzles located between the inlet nozzles serve as inlets for decay heat removal and emergency core cooling water injection lines, and instrumentation nozzles penetrate the lower vessel head. Piping attached to the reactor vessel is discussed in BAW-2251A and in Section 2.3.1.3 of the LRA. The reactor vessel support skirt and control rod drive service structure are discussed in Section 2.4.2.1 of the LRA, and are evaluated in Section 2.4 of this SER.

Control rod drive mechanisms (CRDMs) are attached to flanged nozzles that penetrate the closure head. The active portions of the CRDMs are not within the scope of license renewal, however, the control rod drive motor tube assemblies, closure insert, and vent assemblies are within the scope of license renewal and subject to an AMR. These components are discussed in Section 2.3.1.9 of the LRA. In addition, one of the ANO-1 CRDMs was removed to install a reactor vessel level monitoring probe. The reactor vessel level monitoring probe is discussed in Section 2.3.1.6 of the LRA. Additional reactor vessel components are discussed in BAW-2251A.

The NRC staff has approved the use of BAW-2251A by participating applicants for license renewal. The applicant has reviewed the current design and operation of the reactor vessel, and has confirmed that the ANO-1 reactor vessel is bounded by the description contained in BAW-2251A.

Reactor Vessel Internals

The reactor vessel internals consist of two structural subassemblies, the plenum assembly and the core support assembly. These subassemblies can be removed during refueling outages, when necessary. A description of the reactor vessel internals are provided in BAW-2248A.

The applicant reviewed the current design and operation of the ANO-1 reactor vessel internals, and determined that they have the following additional intended functions that were not addressed in BAW-2248A:

- provide support for the reactor vessel level monitoring probe
- provide gamma and neutron shielding
- provide support for the surveillance specimen assemblies in the annulus between the thermal shield and the reactor vessel wall

One of the two ANO-1 CRDMs was removed, and a control rod guide assembly in the reactor vessel plenum was modified to accept a level monitoring probe. Support for this probe is an additional intended function of the reactor vessel internals. The items that support the reactor vessel level monitoring probe are fabricated from Type 304L austenitic stainless steel, and are evaluated in Section 3.3.2.5 of this SER.

The thermal shield, thermal shield upper restraint, and associated bolting are all fabricated from austenitic stainless steel, and support the intended function to “provide gamma and neutron attenuation.” These items are within the scope of license renewal, are subject to an AMR, and are further evaluated in Section 3.3.2.5 of this SER.

In addition, portions of the ANO-1 surveillance specimen holder tubes are attached to the reactor vessel internals. Although all of the specimens have been removed, portions of the shroud tube and its supports are bolted to the core support shield. These items only have the function of remaining secured to prevent loose parts in the RCS. The applicant states that this function is applicable to the remaining portions of the surveillance specimen holder tubes.

Once-Through Steam Generators

ANO-1 has two once-through steam generators (OTSGs). Each is a vertical, straight-tube, once-through, counter-flow, shell-and-tube heat exchanger with shell-side boiling. The steam generator consists of upper and lower hemispherical heads welded to tubesheets that are separated by a shell assembly. Over 15,000 straight Alloy 600 tubes are held in alignment by 15 tube support plates.

Primary coolant from the reactor enters the steam generator through a single inlet nozzle in the top of the upper head. Coolant flows downward through the straight parallel tubes, is cooled by the secondary coolant on the shell side, and then exits through two outlet nozzles in the lower head. The cooling medium enters through a ring of ports that penetrate the shell approximately midway up the shell assembly. The feedwater travels downward through an annulus between the lower baffle and the shell. Near the lower tubesheet, the feedwater turns inward, and then flows upward around the tubes and through the tube support plates. As the feedwater absorbs heat from the primary coolant, it boils and then becomes superheated. The dry steam exits the steam generator through two steam outlet nozzles just above the feedwater inlet ports.

The intended functions of the OTSGs include maintaining the primary and secondary pressure boundaries, transferring heat from the primary fluid to the secondary fluid, and providing for reactor building isolation. The OTSG items that are within the scope of license renewal and subject to an AMR include the hemispherical heads, secondary shell, tubes, plugs, mechanical sleeves, tubesheets, primary nozzles, primary manway and inspection port assemblies, main and auxiliary feedwater nozzles, main and auxiliary feedwater header and riser piping, steam outlet nozzles, instrumentation nozzles, temperature sensing connections, drain nozzles, secondary manway and inspection port covers, associated pressure retaining bolting, and integral attachments inspected in accordance with ASME Section XI, Subsections IWB and IWC. Class 1 RCS piping attached to the primary once-through steam generator nozzles, including the welded joints, is addressed in Section 2.3.1.3 of the LRA. Secondary piping attached to the OTSG nozzles, including the main and auxiliary feedwater headers and riser piping, is addressed in Section 2.3.4.2 of the LRA. The steam generator supports are addressed in Section 2.4.2 of the LRA.

The OTSG items that do not support an intended function and that are not subject to an AMR include weld deposit pads on the external shell of the generator that are used for insulation supports, shell thermocouples, and grounding lugs; an internal support ring that is attached to the inside shell of the secondary side, secondary internal baffles, support plates, variable orifice plate, and tube stabilizers; and gaskets used in bolted connections at manways inspection ports, and main and auxiliary feedwater inlet piping.

The OTSG items that are fabricated from low-alloy steel include the hemispherical heads, transition ring, tubesheets, and pressure-retaining bolting. Items that are fabricated from carbon steel include primary inlet and exit nozzles, secondary shell, secondary outlet nozzles, main and auxiliary feedwater header and riser piping, primary and secondary manway covers and inspection port covers, secondary vent nozzles, drain nozzles, level-sensing nozzles, and main and auxiliary feedwater nozzles. Items that are fabricated from Alloy 600 include the primary drain nozzle, nozzle dam support rings, tubes, plugs, sleeves, and secondary temperature sensing connections. The OTSGs were designed as Class A vessels in accordance with ASME Section III, 1965 Edition, with Addenda through Summer of 1967.

Reactor Coolant Pumps

Reactor coolant pumps propel reactor coolant through the reactor core, piping, and steam generators. The four reactor coolant pumps at ANO-1 are required during normal full-power operation. These pumps were manufactured by Byron-Jackson, and were designed, fabricated, tested, and inspected as Class A vessels, in accordance with ASME Section III, 1968 Edition.

The intended function of the reactor coolant pumps is to maintain the RCS pressure boundary. The reactor coolant pump components that perform or support this function are within the scope of license renewal, and are subject to an AMR. These components include the casing, cover, integral seal injection heat exchangers, and pressure-retaining bolting. Non-Class 1 piping, instrumentation, and other components attached to the reactor coolant pump are addressed in Section 2.3.2 of the LRA. Class 1 piping connected to the pump, including the welded joints, is discussed in Section 2.3.1.3 of the LRA. The portions of the reactor coolant pump rotating elements above the pump coupling, including the electric motor and the flywheel, are not subject to an AMR in accordance with 10 CFR 54.21(a)(1).

The reactor coolant pump casing also includes the bolted closures and connections. These are constructed of stainless steel, except for the pressure-retaining bolting, which is fabricated from low-alloy steel. The upper and lower halves of the Byron-Jackson pump casings are cast austenitic stainless steel.

The pump cover is a generic term used to describe the pressure-retaining closure of a pump casing. The reactor coolant pumps cast austenitic stainless steel covers serve as housing for the mechanical seals, radial bearings, thermal barriers, and recirculating impellers. They are clamped between the carbon steel driver mounts, and the stainless steel pump casings.

Bolts that are used to secure the covers to the casings include cover-to-case studs and nuts, which are fabricated from low-alloy steel. Bolting used to secure the seal housings and/or seal glands to the pump covers include studs and nuts. These bolting materials are less than two inches in diameter and are fabricated from low-alloy steel.

Each reactor coolant pump is supported by the cold leg piping during all modes of operation. The weight of each reactor coolant pump motor is supported by two vertical constant load supports, which are addressed in Section 2.4.2.1 of the LRA.

Control Rod Drive Mechanisms Pressure Boundary

The ANO-1 CRDM motor tube assemblies, closure insert assemblies, and vent assemblies provide the reactor coolant pressure boundary around the CRDMs. During normal operation, the CRDM motor tube assemblies are filled with borated reactor coolant at the system operating pressure. Thermal barriers in the lower motor tube mechanism and the CRDM cooling system maintain the temperatures in the housings below system temperature.

The CRDM motor tube assemblies are designed, fabricated, tested, and inspected in accordance with ASME Section III, 1965 Edition and the Summer 1967 Addendum. The material of construction is stainless steel or Alloy 82/182 clad low-alloy steel.

Two different designs of CRDMs are currently in use at ANO-1, Type B and Type C. The CRDMs themselves are active and not subject to an AMR. The CRDM items subject to an AMR include the motor tube assemblies, closure insert and vent assemblies, associated bolting, and the reactor vessel level monitoring system adapter flange assembly.

2.3.1.2 Staff Evaluation

The NRC staff reviewed the information in Section 2.3.1 of the LRA to determine whether there is reasonable assurance that the RCS components and supporting structures that are within the scope of license renewal and subject to an AMR have been identified in accordance with the requirements of 10 CFR 54.4 and 54.21(a)(1).

As part of the evaluation, the staff reviewed portions of the ANO-1 UFSAR for the RCS, and the associated pressure boundary components, and compared the information in the UFSAR with the information in the LRA to identify any instances where the applicant failed to identify SSCs that are required to be included within the scope of license renewal. The staff then evaluated the evaluation boundaries for the systems and structures included within the scope of license renewal to verify that all the SCS, that contributed to the intended function(s) within the scope of license renewal, were considered during the AMR. The staff also evaluated the SCs within the evaluation boundaries to verify that all passive/long-lived SCs were subject to an AMR.

In a letter date June 1, 2000, the staff requested that the applicant provide additional information and/or clarifications for a selected group of RCS SCs excluded from the scope of license renewal, or determined not to be subject to an AMR to verify the following:

- For those SSCs excluded from the scope of license renewal, verify that they do not have any of the intended functions delineated under 10 CFR 54.4(a).

- For those SCs that have an applicable intended function(s), but determined not to be subject to an AMR, verify that they either perform this function(s) with moving parts or a change in configuration or properties, or that they are subject to replacement based on a qualified life or specified time period, as described in 10 CFR 54.21(a)(1).

The staff also reviewed the UFSAR to identify any function(s) delineated under 10 CFR 54.4 (a) that is not identified as an intended function(s) in the LRA, to verify that the effects of aging of

SCs with such function(s) will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the extended period of operation.

The applicant responded to the staff's RAIs in letters to the NRC dated August 24, 2000, and September 6, 2000. On September 13, 2000, the staff had a follow-up telephone conference with the applicant to discuss some additional concerns and to obtain additional clarification. This telephone conference, and the applicant's response are documented in a letter to the applicant dated October 11, 2000, and a letter from the applicant dated October 3, 2000, respectively.

The staff had a concern regarding the exclusion of the reactor vessel head leakage monitoring piping from the scope of license renewal, and requested verification as to whether the ½-inch diameter hole in the vessel flange, as mentioned in Section 2.3.1.3 of the LRA, would limit leakage to less than normal RCS makeup capacity, and thereby minimize the consequences of failure of the reactor vessel monitoring pipes. In response, the applicant verifies that in the event of leakage past the inner O-ring, the leak flow rate during normal operation (i.e., heatup, cooldown, and power operation) through the ½-inch-diameter penetration downstream in the reactor vessel flange has been estimated to be within the makeup system capacity at ANO-1. The applicant, therefore, concluded that the leakage monitoring piping need not be included within the scope of license renewal and, therefore, need not be subject to an AMR. The staff found the applicant's response acceptable.

The staff also requested that the applicant provide a technical justification for the exclusion of the pressurizer and OTSG manhole gaskets from the scope of license renewal, consistent with the Rule and staff guidance. In response, the applicant states that the pressurizer and OTSG manway gaskets are not within the scope of license renewal, in accordance with the NRC's SER for BAW-2244A. In Section 3.1.1 of that SER, the staff concluded that the gasket was part of the bolted connection, exists to minimize leakage, and is not responsible for providing the pressure boundary or supporting a structural load. Furthermore, the applicant indicates that the Boric Acid Corrosion Prevention Program includes components that are exposed to boric acid leakage. The applicant states that if a gasket is the source of leakage, it would be addressed in the program regardless of its exclusion from the scope of license renewal. The staff found the applicant's response acceptable.

The staff also requested that the applicant clarify whether the reactor vessel level monitoring system probe itself is subject to an AMR and, if not, provide a justification for excluding the level probe from an AMR. In its response, the applicant states that the reactor vessel level monitoring system probe was installed to monitor the fluid level in the upper plenum and head of the reactor vessel as part of the post-Three Mile Island modifications. The applicant further states that this component was excluded from the scope of license renewal because it does not support an intended function required to satisfy the criteria in 10 CFR 54.4(a)(1), (2), or(3). The staff questioned the fact that, if the component was added as a Post-Three Mile Island requirement, it should have been predicated on an intended safety function. The staff requested additional discussion as to why the intended function of the reactor vessel level monitoring system probe does not meet the criteria defined in 10 CFR 54.4(a). In response, the applicant states that these monitors are used as an alternative/backup means of determining if a bubble has formed in the reactor vessel. However, the applicant also states that these instruments are not credited for making this determination in any design basis event (DBE) analyses. Rather, accident mitigation for the formation of a bubble in the reactor vessel is determined by the subcooling margin in the case of a DBE. The applicant, therefore, reconfirmed its conclusion that the subject SSCs need not be included within the scope of license renewal. Upon reviewing the above information, the staff did not find any omissions in the RCS SSCs included within the scope of license renewal and subject to an AMR for ANO-1.

2.3.1.3 Conclusions

On the basis of the staff's review of the information presented in Section 2.3.1 of the LRA, the supporting information in the ANO-1 UFSAR, and the applicant's response to the staff's RAIs and clarifications, the staff finds that there is reasonable assurance that the applicant has adequately identified those portions of the RCS that are within the scope of license renewal, and the associated SCs that are subject to an AMR, in accordance with the requirements of 10 CFR 54.4(a) and 54.21(a)(1), respectively.

2.3.2 Engineered Safeguards Scoping and Screening

In the LRA, Section 2.3.2, "Engineered Safeguards," the applicant describes the SSCs of the engineered safeguards (ES) system that are within the scope of license renewal and subject to an AMR. The NRC staff reviewed this section to determine whether the applicant has adequately demonstrated that the requirements of 10 CFR 54.4, 54.21(a)(1), and 54.21(a)(2) have been met for ES SSCs.

The following systems makeup the ES systems that are within the scope of license renewal:

- core flood
- low pressure injection/decay heat (LPI/DH)
- high pressure injection/makeup and purification (HPI/MUP)
- reactor building spray
- reactor building cooling and purge (including reactor building heating and ventilation and portions of the reactor building purge)
- sodium hydroxide (including chemical addition)
- reactor building isolation
- hydrogen control (including hydrogen purge and hydrogen recombiners)

2.3.2.1 Core Flood

In the LRA, Section 2.3.2.1, "Core Flood," the applicant describes the core flood system and the components therein that are within the scope of license renewal. The applicant also identifies the SCs that are subject to an AMR in Table 3.3-1 of the LRA. The design of the core flood system is described in Section 6.1of the ANO-1 UFSAR.

2.3.2.1.1 Technical Information in the Application

ES systems consist of SSCs designed to function under accident conditions to minimize the severity of an accident, or to mitigate the consequences of an accident. In the event of a loss-of-coolant accident (LOCA), the core flood system provides emergency coolant to ensure the structural integrity of the core, to maintain the integrity of the reactor building, and to reduce the concentration of fission products expelled to the reactor building atmosphere.

Specifically, the safety function of the core flood system is to provide core cooling after intermediate and large-break LOCAs. The core flood system is within the scope of license renewal, and its SCs that are subject to an AMR include two core flood tanks, piping, and other components up to the reactor coolant system boundary. The intended function of these SCs is to maintain the integrity of the system pressure boundary.

2.3.2.1.2 Staff Evaluation

The NRC staff reviewed Section 2.3.2 of the LRA to determine whether there is reasonable assurance that the applicant has identified the core flood system SCs that are within the scope of license renewal and subject to an AMR in accordance with the requirements of 10 CFR 54.4 and 54.21(a)(1).

The staff reviewed portions of the ANO-1 UFSAR for the core flood system and associated pressure boundary components, and compared the information in the UFSAR with the information in the LRA to identify any instances where the applicant failed to identify SSCs that are required to be included within the scope of license renewal. The staff then evaluated the evaluation boundaries for the systems and structures included within the scope of license renewal to verify that all the SCS, that contributed to the intended function(s) within the scope of license renewal, were considered during the AMR. The staff also evaluated the SCs within the evaluation boundaries to verify that all passive/long-lived SCs were subject to an AMR. .

Upon completing its initial review, the staff requested that the applicant provide additional information and/or clarifications for a selected number of these SCs in a letter dated June 1, 2000, to verify the following information:

- the selected SCs do not have any of the intended functions identified in 10 CFR 54.4(a)

- the SCs that have an applicable intended function(s), perform this function(s) with moving parts or with a change in configuration or properties, or are subject to replacement based on a qualified life or specified time period, in accordance with 10 CFR 54.21(a)(1)

The staff also reviewed the UFSAR to identify any function(s) delineated under 10 CFR 54.4(a) that was not identified as an applicable intended function(s) in the LRA. The purpose of this part of the evaluation was to verify that the SSCs with such a function(s) will be included within the scope of license renewal.

The staff also requested that the applicant provide a justification for excluding from an AMR the thermal insulation of the tanks and pipes which carry borated water for ECCS injection. The concern was boron precipitation from borated water, resulting in the reduction of required boron concentration. In a letter to the NRC dated August 30, 2000, the applicant states that the borated water storage tank (BWST) is located outdoors and exposed to ambient weather conditions. The piping of concern runs through the tank bottom, the tank foundation oiled sand, concrete, and portions of the auxiliary building. The applicant further states that the ANO-1 technical specification (TS) 3.3.1(G) requires that the boron concentration in the BWST be maintained at 2,470 +/- 200 ppm boron at a temperature not less than 40EF. A TS limiting condition of operation is entered if this requirement is not met. The applicant also states that for a concentration of 3,000 ppm, boron will not precipitate from solution until water temperature falls below 22EF. As a result of this TS requirement, the applicant will have to take corrective actions if the water temperature falls below 40EF (which is well above the boron precipitation temperature) for any reason, including from the degradation of insulation, age-related or otherwise. The applicant, therefore, concludes that the insulation material of the tank and piping is not required to support any system function that is required to satisfy the criteria of 10 CFR 54.4(a) during or following any DBE. The staff found the applicant’s assessment acceptable.

The NRC staff also requested that the applicant clarify whether the foundations or pads of the ECCS tanks are included within the scope of license renewal and subject to an AMR, or to provide a justification for the exclusion of these structural components from an AMR. The applicant verifies that foundations of tanks, including the ECCS tanks, are included within the scope of license renewal and are addressed in Section 2.4.6.1 of the LRA. The AMR of tank foundations is presented in Table 3.6-7 of the LRA.

The NRC staff also requested a technical justification as to why the limiting mass flow rate during postulated breaks is not an applicable intended function of the orifices identified in LRA, Table 3.3-1, in accordance with 10 CFR 54.4(a)(1)(iii). In response to the staff’s request, the applicant states that the orifices listed in Table 3.3-1 do not have a safety-related function in accordance with its CLB to limit mass flow rate during postulated breaks, and that maintaining pressure boundary integrity is the only intended function these components are required to maintain during the period of extended operation. The staff found the applicant’s assessment acceptable.

Upon reviewing the above information, the staff did not identify any omissions in the core flood SSCs included within the scope of license renewal, and the SCs that are subject to an AMR.

2.3.2.1.3 Conclusions

On the basis of the review described above, the staff finds that there is reasonable assurance that the applicant adequately identified those portions of the core flood system that are within the scope of license renewal, and the associated SCs that are subject to an AMR, in accordance with the requirements of 10 CFR 54.4(a) and 54.21(a)(1), respectively.

2.3.2.2 Low Pressure Injection/Decay Heat

In the LRA, Section 2.3.2.2, “Low Pressure Injection/Decay Heat,” the applicant describes the low pressure injection/decay heat (LPI/DH) system, and the component therein that are within the scope of license renewal. The applicant also identifies the SCs that are subject to an AMR in Table 3.3-2 of the LRA. The design of the LPI/DH system is described in Section 6.1of the ANO-1 UFSAR.

2.3.2.2.1 Technical Information in the Application

The LPI/DH system is a dual-purpose system. This system operates as the DH system whose intended function is to remove decay heat from the core and sensible heat from the RCS during the latter stages of cooldown. The LPI System injects borated water into the reactor vessel to cool the core in the event of a LOCA.

The LPI system has the following safety functions:

- Inject borated water from the borated water storage tank (BWST) during postulated large-break LOCA.

- Provide long-term cooling following a LOCA by recirculating injection water from the reactor building sump.

- Supply recirculated water from the reactor building sump to the suction of the high-pressure injection pumps if RCS pressure is too high to allow the LPI pumps to function

- Supply injection water from the BWST to the DH/LPI pumps as well as the high pressure injection and the reactor building spray pumps. The BWST floods the reactor building basement to a level that will allow for recirculation from the reactor building sump under accident conditions.

- Provide water that is free of entrained air from the screened reactor building sump, when the BWST is depleted.

The DH system is credited in the fire protection analysis (10 CFR 50.48) with the capability of attaining cold shutdown. Therefore, the DH system has a function to remove decay heat from the reactor core and sensible heat from the RCS during the latter stages of cooldown such that fuel design limits and design conditions of the RCS pressure boundary are not exceeded. The DH system also supports the following functions:

- Circulate reactor coolant to prevent boron stratification and to minimize the effects of a boron dilution event.

- Provide an alternate supply of borated water from the BWST for volume contraction during cooldown to cold shutdown.

- Provide cooling, inventory addition, and instrumentation for loss of decay heat removal events.

The following LPI/DH components are within the scope of license renewal and are subject to an AMR:

- the DH system piping that passes through the reactor building penetrations, including the injection lines, drop line, pressurizer auxiliary spray line, and emergency sump lines (These portions of the DH system perform a reactor building isolation function and are within the scope of license renewal.)

- the DH drop line valves, coolers, cooler isolation valves, and pumps

- the BWST, BWST supply header, and injection lines up to the outboard RCS pressure boundary valve of the low-pressure injection lines, and the suction supply piping to the high pressure injection system

- piping and components from the reactor building sump, including some piping and components from the Post-Accident Sampling System (PASS) (used for post LOCA sump sampling), and sump screens and the vortex breakers (Sump screens and vortex breakers are reviewed in Section 2.3.3.4 of the LRA.)

- the oil side of the LPI pump lube oil coolers (The service water side of the coolers is evaluated in Section 2.3.3.10 of the LRA.)

The intended function that is within the scope of license renewal is to maintain pressure boundary integrity. For the LPI/DH heat exchangers that are within the scope of license renewal, the heat transfer intended function is performed without moving parts, or without a change in configuration or properties, and is subject to an AMR.

2.3.2.2.2 Staff Evaluation

The NRC staff reviewed Section 2.3.2 of the LRA to determine whether there is reasonable assurance that the applicant has identified the LPI/DH SCs that are within the scope of license renewal and subject to an AMR in accordance with the requirements of 10 CFR 54.4 and 54.21(a)(1).

The staff reviewed portions of the ANO-1 UFSAR for the LPI/DH system and associated pressure boundary components, and compared the information in the UFSAR with the information in the LRA to identify any instances where the applicant failed to identify SSCs that are required to be included within the scope of license renewal. The staff then evaluated the evaluation boundaries for the systems and structures included within the scope of license renewal to verify that all the SCS, that contributed to the intended function(s) within the scope of license renewal, were considered during the AMR. The staff also evaluated the SCs within the evaluation boundaries to verify that all passive/long-lived SCs were subject to an AMR.

Upon completing its initial review, the staff requested that the applicant provide additional information and/or clarifications for a selected number of these SCs in a letter dated June 1, 2000, to verify the following information:

- selected SCs do not have any of the intended functions identified in 10 CFR 54.4(a)

- SCs that have an applicable intended function(s), perform this function(s) with moving parts or with a change in configuration or properties, or are subject to replacement based on a qualified life or specified time period, as described in 10 CFR 54.21(a)(1)

The staff also reviewed the UFSAR to identify any function(s) delineated under 10 CFR 54.4(a) that was not identified as an applicable intended function(s) in the LRA. The purpose of this part of the evaluation was to verify that the SSCs with such a function(s) will be included within the scope of license renewal.

The staff also requested that the applicant provide a justification for excluding from an AMR the thermal insulation of the tanks and pipes which carry borated water for ECCS injection. Because the SCs in question are common to the ECCS systems, the staff's evaluation of these components is discussed in the core flood system evaluation, above.

Upon reviewing the above information, the staff did not identify any omissions in the LPI/DH SSCs included within the scope of license renewal, and the SCs that are subject to an AMR.

2.3.2.2.3 Conclusions

On the basis of the review described above, the staff finds that there is reasonable assurance that the applicant adequately identified those portions of the LPI/DH S systems that are within the scope of license renewal, and the associated SCs that are subject to an AMR, in accordance with the requirements of 10 CFR 54.4(a) and 54.21(a)(1), respectively.

2.3.2.3 High-Pressure Injection/Makeup And Purification

In the LRA, Section 2.3.2.3 "High-Pressure Injection/Makeup and Purification," the applicant describes the high-pressure injection/makeup and purification (HPI/MUP) system, and the components therein that are within the scope of license renewal. The applicant also identifies the SCs that are subject to an AMR in Table 3.3-3 of the LRA. The design of the HPI/MUP system is described in Section 6.1of the ANO-1 UFSAR.

2.3.2.3.1 Technical Information in the Application

The safety function of the HPI system is to provide high-pressure injection into the RCS during emergency conditions. This system is normally operated as part of the MUP system. During normal operations, the MUP system performs various functions in support of the RCS. The HPI/MUP systems have the following safety functions:

- Inject borated water from the BWST during postulated accidents, such as the small-break LOCA.

- Provide long-term cooling following small-break LOCAs by recirculating injection water from the reactor building sump.

The HPI/MUP systems are credited in the fire protection analysis (10 CFR 50.48) with the capability to provide RCS makeup and pressure control. Some of the system valves must remain closed to prevent a direct RCS leak path in the event of a fire. The HPI/MUP systems also support the following functions:

- Provide inventory to the RCS during operational transients, such as reactor trips and overcooling events.

- Provide a backup inventory supply to the RCS during a loss of decay heat removal event.

- Provide core cooling following a total loss of feedwater event via feed-and-bleed cooling of the RCS.

- Provide an auxiliary means to spray the pressurizer steam space when normal spray is not available.

The following HPI/MUP components are within the scope of license renewal, and are subject to an AMR:

- the seven mechanical reactor building penetrations necessary for meeting reactor building isolation requirements

- the Class 1 RCS pressure boundary that extends to the second isolation valve off of the RCS (For the letdown line, this is downstream of the letdown coolers. The letdown coolers and the Class 1 valves are reviewed in Section 2.3.1.3 of the LRA.)

- the HPI piping from the BWST supply header to the outboard RCS pressure boundary valve of the injection lines, and all portions of the system needed to support high-pressure injection, including the suction supply from the low pressure injection system

- the oil-side of the HPI pump oil coolers (The service water side of the coolers is evaluated in Section 2.3.3.10 of the LRA.)

The intended function of the HPI/MUP systems that are within the scope of license renewal is to maintain the integrity of the system pressure boundary. For the HPI/MUP heat exchangers that are within the scope of license renewal, the heat transfer intended function is performed without moving parts, or without a change in configuration or properties, and is subject to an AMR.

2.3.2.3.2 Staff Evaluation

The NRC staff reviewed Section 2.3.2 of the LRA to determine whether there is reasonable assurance that the applicant has identified the HPI/MUP SCs that are within the scope of license renewal and subject to an AMR in accordance with the requirements of 10 CFR 54.4 and 54.21(a)(1).

The staff reviewed portions of the ANO-1 UFSAR for the HPI/MUP system and associated pressure boundary components, and compared the information in the UFSAR with the information in the LRA to identify any instances where the applicant failed to identify SSCs that are required to be included within the scope of license renewal. The staff then evaluated the evaluation boundaries for the systems and structures included within the scope of license renewal to verify that all the SCS, that contributed to the intended function(s) within the scope of license renewal, were considered during the AMR. The staff also evaluated the SCs within the evaluation boundaries to verify that all passive/long-lived SCs were subject to an AMR.

Upon completing its initial review, the staff requested that the applicant provide additional information and/or clarifications for a selected number of these SCs in a letter dated June 1, 2000, to verify the following information:

- selected SCs do not have any of the intended functions identified in 10 CFR 54.4(a)

- SCs that have an applicable intended function(s), perform this function(s) with moving parts or with a change in configuration or properties, or are subject to replacement based on a qualified life or specified time period, as described in 10 CFR 54.21(a)(1)

The staff also reviewed the UFSAR to identify any function(s) delineated under 10 CFR 54.4(a) that was not identified as an applicable intended function(s) in the LRA. The purpose of this part of the evaluation was to verify that the SSCs with such a function(s) will be included within the scope of license renewal.

The staff also requested that the applicant provide a justification for excluding from an AMR the thermal insulation of the tanks and pipes which carry borated water for ECCS injection. Because the SCs in question are common to the ECCS systems, the staff evaluation of these components is discussed in the core flood system evaluation, above.

In response to another staff RAI, the applicant verifies that Drawing LRA-M-231, Sheet 3, is incorrect. Specifically, the drawing should indicate valves MU-1210E, MU-1210F, MU-1210G, and MU-1210H, as well as the associated tubing are within the scope of license renewal, and that screens or vortex breakers are not installed in the tanks from which ECCS water is drawn. The AMR for the makeup and purification system stainless steel valves and piping components is provided in Table 3.3-3 of the LRA.

Upon reviewing the above information, the staff did not identify any omissions in the HPI/MUP SSCs included within the scope of license renewal, and the SCs that are subject to an AMR for ANO-1.

2.3.2.3.3 Conclusions

On the basis of the review described above, the staff finds that there is reasonable assurance that the applicant has adequately identified those portions of the HPI/MUP system that are within the scope of license renewal, and the associated SCs that are subject to an AMR, in accordance with 10 CFR 54.4(a) and 54.21(a)(1), respectively.

2.3.2.4 Reactor Building Spray System

In the LRA, Section 2.3.2.4, “Reactor Building Spray,” of the LRA, the applicant describes the reactor building spray system and the components therein that are within the scope of license renewal. The applicant also identifies the SCs that are subject to an AMR in Table 3.3-4 of the

LRA. The design of the reactor building spray system is described in Section 6.2 of the ANO-1 UFSAR.

2.3.2.4.1 Technical Information in the Application

The system safety function of the reactor building spray system is to remove heat from the reactor building atmosphere following a DBA. The system also removes the fission product iodine and reduces pressure from the post-accident reactor building atmosphere. The components of the reactor building spray system that are within the scope of license renewal and subject to an AMR consist of two redundant trains that include two pumps, two reactor building spray headers, and the supporting equipment (lube oil coolers and seal water cyclone separators), piping, and valves. In addition, a tank (T10) containing sodium hydroxide is supplied for iodine removal, and for pH adjustment of the borated water. The tank is evaluated with the sodium hydroxide system in Section 2.3.2.6 of this SER. The reactor building spray system also includes the interfacing systems, which form part of the reactor building spray system pressure boundary. The interfacing system components include the valves from the sodium thiosulphate tank, the interfaces with the service air system, and the vents and drains off the spray system pump casings.

The intended function of the components that are within the scope of license renewal is to maintain the reactor building spray system pressure boundary integrity. Heat transfer is also an intended function of the heat exchangers that are within the scope of license renewal.

2.3.2.4.2 Staff Evaluation

The staff reviewed Section 2.3.2.4 of the LRA, Section 6.2 of the USAR, and the associated flow diagrams (P&ID drawings) to determine whether there is reasonable assurance that the applicant has identified the reactor building spray SSCs that are within the scope of license renewal in accordance with10 CFR 54.4(a) and SCs that are subject to an AMR in accordance with the requirements of 10 CFR 54.21(a)(1). After completing its initial review, the staff requested additional information regarding the information provided by the applicant for the reactor building spray system in a letter dated April 18, 2000.

In a letter to the NRC dated August 30, 2000, the applicant highlights the portions of the reactor building spray system that are within the scope of license renewal on the system drawings listed in Table 2.3-6 of the LRA and identifies the components that are subject to an AMR and its intended functions in Table 3.3-4 of the LRA. The staff reviewed the components in the table and verified them with the highlighted portions of the drawings. The following component commodity groups were identified in the table as being subject to an AMR:

- bolting
- external valve parts
- piping, tubing, and valves
- separators
- pump casings
- heat exchanger (for the lube oil coolers)

However, the spray nozzles and orifices in the reactor building spray system were not listed in the table. The nozzle and orifice perform system safety functions of spraying and throttling, respectively. The staff requested that the applicant justify excluding these nozzles and orifices from an AMR. In addition, the staff also found that the sodium thiosulfate storage tank and its piping connected to the spray system, as seen in LRA-M-236, Sheet 1, are not highlighted as being within the scope of license renewal and are not included in Table 3.3-4 of the LRA. The staff requested that the applicant justify excluding the sodium thiosulfate storage tank and its piping from an AMR.

In its response, the applicant states that orifices and nozzles were added to in the component commodity group listed as piping in Table 3.3-4 of the LRA and were subject to an AMR. The sodium thiosulfate storage tank and its piping that are connected to the spray system, are not within the scope of license renewal because they are isolated and no longer in service. Therefore, they are not required to meet any of the scoping criteria of 10 CFR 54.4(a), and are not subject to an AMR. The staff's review found that the sodium thiosulfate storage tank and its piping to the spray system do not perform the intended function of the spray system and do not require an AMR. The staff found the applicant's response acceptable.

In its submittal, the applicant also identifies a number of license renewal interface boundaries within the reactor building spray system. The interface systems include the interfacing valves of the sodium thiosulfate tank, the interfaces with the service air system and the vent and drains of the spray system pump casings. On one side of the interface boundary, the SCs are within the scope of license renewal; on the other side of the interface boundary, the SCs are not within the scope of license renewal. Appropriate isolation, which is part of the existing licensing basis for the system, is provided at each of the license renewal interfaces. Isolation capability is not evaluated for license renewal because, other than the valve body, valves are excluded from an AMR in accordance with 10 CFR 54.21(a)(1)(i). The staff reviewed the ANO-1 UFSAR to determine if any of the interface systems had a system functions that met this scoping requirement in 10 CFR 54.4 or if there were any SCs that might have been omitted from consideration as being within the scope of license renewal. The staff did not identify any omissions as a result of this review.

2.3.2.4.3 Conclusions

On the basis of the review described above, the staff did not identify any omissions by the applicant. Therefore, the staff finds that there is reasonable assurance that the applicant has adequately identified those portions of the reactor building spray system that are within the scope of license renewal, and the associated SCs that are subject to an AMR, in accordance with the requirements of 10 CFR 54.4(a) and 54.21(a)(1), respectively.

2.3.2.5 Reactor Building Cooling and Purge Systems

In the LRA, Section 2.3.2.5, "Reactor Building Cooling and Purge," the applicant describes the reactor building cooling and purge systems and the components therein that are within the scope of license renewal. The applicant also identifies the components that are subject to an AMR. The reactor building cooling and purge systems are also described in Sections 5.2.6 and 6.3 of the ANO-1 UFSAR. A flow diagram of the reactor building cooling and purge systems is shown in Figure 5-7 of the ANO-1 UFSAR.

2.3.2.5.1 Technical Information in the Application

The system function of the reactor building cooling system is to provide cooling to the reactor building that limits the reactor building pressure and temperature to the design value following a LOCA. The system accomplishes this by continuously recirculating the air-steam mixture through cooling coils that transfers heat from the reactor building to the service water system. During normal plant operation, the system is required to maintain the reactor building temperature below the maximum allowed for equipment qualification, and below accident analyses initial temperature assumptions. The reactor building purge system has no defined system function that meets any of the scoping criteria in 10 CFR 54.4, but its penetrations are required to maintain the reactor building integrity under accident conditions. All the components of the reactor building purge system are located outside of the reactor building except interior ducts and two reactor building isolation valves. The applicant identifies the following components as being within the scope of license renewal and subject to an AMR:

- four safety-related reactor building coolers

- service water cooling coils, the fan/cooler housings, and the discharge duct work, including the duct relief valves that prevent damage to the duck work during a rapid building pressurization

- reactor building isolation valves and piping at the two penetrations in the reactor building purge system

The intended function of these SCs that needs to be considered during the AMR is to maintain the pressure boundary integrity. For the heat exchangers, heat transfer is an intended function that needs to be considered during the AMR, as well.

2.3.2.5.2 Staff Evaluation

The staff reviewed Section 2.3.2.5 of the LRA, Sections 5.2.6 and 6.3 of ANO-1 USAR, and associated drawings to determine whether there is reasonable assurance that the applicant has identified the reactor building cooling and purge system and its SCs that are subject to an AMR in accordance with the requirements of 10 CFR 54.21(a)(1). After completing its initial review, the staff requested additional information in a letter to the applicant dated April 18, 2000.

In a letter to the NRC dated August 30, 2000, the applicant identifies the portions of the reactor building cooling and purge systems that are within the scope of license renewal on the flow diagrams (highlighted on some of the drawings listed in Table 2.3-6 of the LRA). The applicant listed the component commodity groups subject to an AMR in Table 3.3-5 of the LRA. In this table, the applicant identifies the duct, dampers, pipe, valves, fan and cooler housings, and heat exchangers as the component commodity groups that require an AMR. For these component commodities subject to an AMR, maintaining the pressure boundary integrity is identified as the intended function. Heat transfer was also identified as an additional intended function for the system heat exchangers.

In Section 6.3.2 of the ANO-1 UFSAR, the applicant states that the normal cooling system consists of five reactor building cooling fans and their associated chilled water cooling coils. The post-accident cooling system uses four of the five reactor building cooling fans and four associated service water cooling coils. However, in the LRA, Section 2.3.2.4, only the four safety-related reactor building cooling units for post accident cooling were included in the scope of license renewal. The five non-safety-related reactor building cooling units, which are used for normal plant operation, are not included in the scope of license renewal. The five normal cooling units are used to remove heat from equipment, piping, and reactor cavity during normal operation. However, the normal duty cooling units are not required to meet any of the scoping requirements in 10 CFR 54.4 and, therefore, are not in the scope of license renewal.

In P&ID LRA-M-261, Sheet 1, the staff found that the 2-inch temporary duct of the reactor building cooling system from the supply air plenum to the box of temperature detectors was not identified as being subject to an AMR. The staff requested the applicant to justify excluding this component from an AMR, as well.

In its response to the NRC, the applicant states that the 2-inch temporary duct of the reactor building cooling system from the supply air plenum to the box of temperature detectors is not within the scope of license renewal. According to its CLB, this duct is not required for the system to perform the function of reducing post-accident temperature and pressure in the reactor building or providing mixing of the reactor building atmosphere following a LOCA. Its failure would not prevent the remainder of the system from performing its intended function. The staff found the applicant’s response acceptable.

In Tables 3.3-5 and 3.3-8 of the LRA, tubing is not listed as a component group that is subject to an AMR even though tubing is used in the system, and is addressed in the notes of the tables in the LRA. The staff asked the applicant to justify excluding tubing from an AMR. In its response, the applicant states that there is no tubing in the reactor building cooling and purge systems that are within the scope of license renewal. The tubes referenced in the notes of Table 3.3-5 are referring to heat exchanger tubes. Heat exchanger tubes are evaluated in the heat exchangers AMR. The staff found the applicant’s response acceptable.

The staff also reviewed Section 5.2.6 of the ANO-1 UFSAR to verify that the applicant identified all the system functions that meet the scoping criteria in 10 CFR 54.4(a). Except for the intended function of the reactor building normal cooling units, which was determined not to be in the scope of license renewal, the staff did not identify any omissions.

2.3.2.5.3 Conclusions

On the basis of the review described above, the staff finds that there is reasonable assurance that the applicant has adequately identified those portions of the reactor building cooling and purge systems that are within the scope of license renewal, and the associated SCs that are subject to an AMR in accordance with the requirements of 10 CFR 54.4(a) and 54.21(a)(1), respectively.

2.3.2.6 Sodium Hydroxide

In the LRA, Section 2.3.2.6, “Sodium Hydroxide,” the applicant describes the sodium hydroxide system, and identifies the SCs that are within the scope of license renewal and subject to an AMR in Table 3.3-6 of the LRA. The sodium hydroxide system is described in Section 6.2 of the ANO-1 UFSAR.

2.3.2.6.1 Technical Information in the Application

The system function of the sodium hydroxide system is to provide a solution of sodium hydroxide to the ECCS suction headers to improve iodine absorption and retention in the water as a result of increased pH, thereby minimizing the gaseous iodine, and the offsite dose following a LOCA. The applicant determines that the sodium hydroxide tank (T10) and its associated piping, and the components from the tank to the ECCS suction headers are within the scope of license renewal. The applicant identifies the following SCs as being subject to an AMR: pipe and valves, bolting, and tank. The intended function of these SCs that are within the scope of license renewal is to maintain the system pressure boundary integrity.

2.3.2.6.2 Staff Evaluation

The staff reviewed Section 2.3.2.6 of the LRA to determine whether there is reasonable assurance that the applicant has identified the sodium hydroxide system and its SCs that are subject to an AMR in accordance with the requirements of 10 CFR 54.21(a)(1). After completing its initial review, the staff requested additional information in a letter to the applicant dated April 18, 2000. The applicant responded to the staff's RAIs in a letter to the NRC dated August 30, 2000.

In the LRA, Table 3.3-6, the applicant identifies piping, valves, bolting, external valve parts, and tanks as component commodities of the sodium hydroxide system that are subject to an AMR. Section 6.2.2.1 of the ANO-1 UFSAR states that the content of the tank (T10) is proportioned so that the proper quantity of sodium hydroxide is injected for pH control. Flow orifices in the discharge lines from the sodium hydroxide tank assist in assuring the proper injection rate. However, the flow orifice was not listed in Table 3.3-6 as the component requiring AMR. The flow orifice has the intended function to throttle the flow and should have been included in the table for the AMR. This was Open Item 2.3.2.6.2-1.

In a letter to the NRC dated March 14, 2001, the applicant states that the flow control function for the sodium hydroxide in-line flow orifices has been added to the scope of license renewal and subject to an AMR. The AMR activities as a result of adding the flow control function of this in-line flow orifice is evaluated in this SER, Section 3.3.1.4.9. The staff found this resolution to Open Item 2.3.2.6.2-1 acceptable.

The staff reviewed the system drawings listed in Table 2.3-6 of the LRA that contain the sodium hydroxide system. In Drawing LRA-M-233, Sheet 1, the sodium hydroxide recirculating pump line from the chemical addition system to the sodium hydroxide storage tank was not highlighted as the component being in-scope. The staff asked the applicant to justify excluding the pump line from the components that are subject to an AMR. In its response to the NRC, the applicant states that the sodium hydroxide recirculating pump line from the chemical addition system to the sodium hydroxide storage tank is not required for the sodium hydroxide system to perform its function of providing sodium hydroxide to the ECCS suction headers. Therefore, this line is not within the scope of license renewal. As stated in Note 2 on Drawing M-233, Sheet 1, this pump line allows recirculation prior to sampling. The sodium hydroxide recirculating line enters the tank above the normal level, and is isolated during normal plant operation. The pump line to the sodium hydroxide storage tank, therefore, does not meet any of the scoping criteria of 10 CFR 54.4, and is not in the scope of license renewal, or subject to an AMR. The staff found the applicant's response acceptable.

2.3.2.6.3 Conclusions

On the basis of the review described above, the staff finds that there is reasonable assurance that the applicant has adequately identified those portions of the sodium hydroxide system that are within the scope of license renewal, and the associated SCs that are subject to an AMR, in accordance with the requirements of 10 CFR 54.4(a) and 54.21(a)(1), respectively.

2.3.2.7 Reactor Building Isolation System

In the LRA, Section 2.3.2.7, "Reactor Building Isolation," the applicant describes the reactor building isolation system and the components therein that are within the scope of license renewal. The applicant also identifies those SCs that are subject to an AMR in Table 3.3-7 of the LRA. The design of the reactor building isolation system is described in Section 5.2.5 of the ANO-1 UFSAR.

2.3.2.7.1 Technical Information in the Application

As listed in Table 3.3-7 of the LRA, the reactor building isolation system includes the isolation valves and associated piping, bolting and penetrations necessary to isolate the reactor building in the event of a LOCA. The system function of the reactor building isolation system is to provide closure of all fluid penetrations not required for operation to prevent the leakage of uncontrolled or unmonitored radioactive materials to the environment.

In the LRA, Section 2.3.27, the applicant states that the portions of the reactor building isolation system that are within the scope of license renewal are the 20 penetration mechanical components and piping that are not covered by other sections of the LRA. These penetrations include the following:

- intermediate cooling water, nitrogen, breathing air, plant heating, and gaseous radwaste
- core flood system - tank sampling and makeup and nitrogen pressurization
- sampling system - steam generator secondary sampling and quench tank sampling
- condensate storage and transfer - condensate transfer supply to quench tank
- liquid radwaste - quench tank drain
- heater vents system - steam generator secondary drains
- integrated leak rate test connection

Other system penetrations that provide the reactor building isolation function not included in the above list are discussed separately in the applicable system description. The intended function of the reactor building isolation system SCs is to maintain system pressure boundary integrity.

2.3.2.7.2 Staff Evaluation

The staff reviewed Section 2.3.2.7 of the LRA, and the ANO-1 UFSAR to determine whether there is reasonable assurance that the applicant has identified the reactor building isolation system and its SCs that are subject to an AMR in accordance with the requirements of 10 CFR 54.21. After completing the initial review, the staff requested additional information in a letter to the applicant dated April 18, 2000. The applicant responded to the staff's RAIs by a letter to the NRC dated August 30, 2000.

The staff reviewed the system diagrams listed in Table 2.3-6 of the LRA, which highlight the portions of the reactor building isolation system that are within the scope of license renewal. In Drawing LRA-M-230, Sheet 1, the reactor building penetration boundary and the isolation valves of the high pressure nitrogen line are not clearly defined in the flow diagram. The staff requested the applicant to provide additional information on this portion of the reactor building isolation. In its response to the NRC, the applicant states that, due to an administrative error, a license renewal boundary flag was omitted from Drawing LRA-M-230, Sheet 1. There should be a license renewal boundary flag at valve N2-61. The line continues, as shown, to Drawing LRA-M-236, Sheet 1, in Zone 6, where it ties into Pipe FCB-1-1, just inside the reactor building penetration. In Table 5-1 of the ANO-1 UFSAR, the isolation valves for penetration 31 are identified as MU-35A, N2-3, N2-61, AND MU-36A. The applicant corrected Drawing LRA-M-230, Sheet 1. The staff found the applicant response acceptable.

In Drawing LRA-M-237, Sheet 1, the redundant isolation valves (SS-1017B, SS-1018B) for the test connections of the sampling system are not highlighted as being within the scope of license renewal. However, containment isolation provisions require double isolation at the test connections for greater assurance of containment integrity. The staff asked why the second isolation valve on each test connection were not identified as being subject to an AMR. In its response to the NRC, the applicant states that this penetration is associated with the secondary side of the steam generator, and is not required to meet General Design Criteria (GDC) 57 of Appendix A to 10 CFR Part 50. The reactor building boundary or barrier against fission product leakage to the environment is the inside surface of the steam generator tubes, the outer surface of the line emanating from the steam generator, and the outer surface of the steam generator below the lower and above the upper tube Sheet. Valves SS-1017B and SS-1018B are not within the scoping of license renewal because they do not meet any of the scoping criteria in 10 CFR 54.4(a). The staff found the applicant's response acceptable.

In the LRA, Section 2.3.2.7, the applicant states that the reactor building isolation system also seals the penetrations that are not required for operation to provide a fission product barrier between the inside of the reactor building and outside environment. However, Table 3.3-7 of the LRA only lists the piping, bolting, and valves as the components of the reactor building isolation system as being within the scope of license renewal. There should be other types of components used for containment isolation, such as leak-testable blank flanges, weld end caps, orifices, and flow monitors. Also, valve types, such as check, motor-operated, remote, manual, or hand valves, used for the reactor building isolation purposes, should be identified in the table. The staff requested the applicant to list all the isolation barriers and valve types that are subject to an AMR for the license renewal. In its response to the NRC, the applicant states that the component commodity grouping in Table 3.3-7 designated as "piping" includes pipe, fittings and flanges. The leak testable blank flanges, weld end-caps, and orifices are considered to be "fittings and flanges" that are included in the piping component commodity group. There are no flow monitors in the reactor building isolation system. The valve types are identified in the system drawings associated with the reactor building isolation system. The legends for the drawings are provided in Drawing LRA-M-200, Sheets 1, 2, and 3. Additionally, Table 5-1 of ANO-1 UFSAR identifies the valve types for all the reactor building isolation valves. The staff reviewed these drawings and Table 5-1 of the ANO-1 UFSAR, and found the applicant's response acceptable.

The staff also reviewed Section 5.2.5 of the UFSAR to determine if the applicant should have identified any additional portions of the reactor building isolation system as being within the scope of license renewal. However, Section 2.3.2.7 of the LRA did not include all the reactor building isolation penetrations in scope. Only the 20 penetration mechanical components and piping are addressed in the section. Other components that perform the reactor building isolation function in systems not included in this section are included in other sections of the LRA. The staff compared the descriptions of the 20 penetrations in Section 2.3.2.7 to Section 5.2.5 of the ANO-1 UFSAR to verified the SCs with the drawings, and found that the SCs that are subject to an AMR are properly selected.

2.3.2.7.3 Conclusions

The staff has reviewed the information presented in Section 2.3.2.7 of the LRA, the information in the UFSAR, and the additional information provided by the applicant in response to the staff's RAI. On the basis of this review, the staff finds that there is reasonable assurance that the applicant has adequately identified those portions of the reactor building isolation system that are within the scope of license renewal, and the associated SCs that are subject to an AMR, in accordance with the requirements of 10 CFR 54.4(a) and 54,21(a)(1), respectively.

2.3.2.8 Hydrogen Control

In the LRA, Section 2.3.2.8, "Hydrogen Control," the applicant describes the hydrogen control system and the components therein that are within the scope of license renewal. The applicant also identifies which of those SCs are subject to an AMR in Table 3.3-8 of the LRA. The design of the hydrogen control system is described in Section 6.6 of the ANO-1 UFSAR.

2.3.2.8.1 Technical Information in the Application

The system safety function of the hydrogen control system is to provide a direct measure of the hydrogen concentration in the reactor building using the hydrogen analyzer, and to reduce the hydrogen concentration following a LOCA using the hydrogen recombiner. The SSCs of the hydrogen control system that are within the scope of license renewal are the reactor building penetrations, the mechanical components of the hydrogen samplers, and the piping to and from the hydrogen samplers. The piping to the hydrogen analyzers uses a portion of the hydrogen purge system and one of the boundary valves in the gas collection header system. These mechanical components associated with the hydrogen recombiner are also within the scope of license renewal. The control power cabinets in the penetration room and the electrical components of the hydrogen recombiners are also within the scope of license renewal, and are reviewed in Section 2.5 of this report.

2.3.2.8.2 Staff Evaluation

The staff reviewed Section 2.3.2.8 of the LRA to determine whether there is reasonable assurance that the applicant has identified the hydrogen control system, and its SCs that are subject to an AMR in accordance with the requirements of 10 CFR 54.21(a)(1). After completing its initial review, the staff requested additional information in a letter to the applicant dated April 18, 2000. In a letter to the NRC dated August 30, 2000, the applicant provides its response to the staff's RAIs.

In the LRA, Table 3.3-8, the applicant identifies the component commodity grouping for the hydrogen control system and lists piping, valves, recombiners, heat exchangers, and sample stations as the components subject to an AMR. The intended function of these SCs is to maintain the system pressure boundary integrity. Heat transfer is also an intended function of hydrogen control system heat exchangers.

As discussed in Section 2.3.2.5 of this report, the staff also states that tubing is not listed in Table 3.3-8 of the LRA as the component subject to an AMR, even though tubing is used in the system and is discussed in the notes of the table. The staff requested the applicant to justify excluding tubing from an AMR. In its response, the applicant states that tubing is included within the "sample stations" component commodity grouping that is subject to an AMR. The staff found the applicant's response acceptable.

The staff also reviewed the drawings listed in Table 2.3-6 of the LRA that contain the components of the hydrogen control system. In Drawing LRA-M-261, Sheet 3, the staff found that some of the system lines attached to the hydrogen control system outside containment and the gas sampling system are not highlighted as being within the scope of license renewal. The staff requested the applicant to justify excluding these system lines from the scope of license renewal. In its response to the NRC the applicant states that the system lines attached to the hydrogen control system outside containment, such as the hydrogen purge air system and post-accident gas sampling system, as seen in Drawing LRA-M-261, Sheet 3, are not within the scope of license renewal because these lines are not part of the pressure boundary of the hydrogen control system and are not required to meet the scoping criteria in 10 CFR 54.4. The hydrogen purge air system is abandoned and isolated in place. The staff found the applicant's response acceptable.

2.3.2.8.3 Conclusions

On the basis of the review described above, the staff finds that the applicant has adequately identified those portions of the hydrogen control system that are within the scope of license renewal, and the associated SCs that are subject to an AMR, in accordance with the requirements of 10 CFR 54.4(a) and 10 CFR 54.21(a)(1), respectively.

2.3.3 Auxiliary Systems

2.3.3.1 Spent Fuel

In the LRA, Section 2.3.3.1, "Spent Fuel," the applicant describes the components of the spent fuel system that are within the scope of license renewal and subject to an AMR. This system is further described in Section 9.4.1 of the ANO-1 UFSAR.

2.3.3.1.1 Technical Information in the Application

The system functions of the spent fuel cooling system are to remove decay heat from the spent fuel stored in the spent fuel pool (SFP), to maintain clarity and chemistry at acceptable levels, and to transfer water within the systems. The spent fuel pool cooling system consists of two circulating water pumps, two spent fuel coolers (heat exchangers), a demineralizer, filters, and a borated water recirculation pump. The borated water recirculation pump assists operators in performing various demineralizing and filtering functions for the spent fuel pool, the transfer canal, and the borated water storage tank. The spent fuel coolers reject decay heat to the nuclear intermediate cooling water system.

The applicant describes its process for identifying the mechanical components that are within the scope of license renewal in Section 2.1.2, “Assessment Using Criteria in 10 CFR 54.4,” of the LRA. The applicant determines that the cooling and purification functions of the spent fuel cooling systems do not provide any DBE mitigation functions that warrant inclusion of the system within the scope of license renewal. However, the safety functions of the spent fuel system are to maintain adequate water level in the spent fuel pool for cooling and shielding and to maintain subcritical margin. Therefore portions of the spent fuel pool cooling system piping, the stainless steel pool liner, the spent fuel storage racks, the spent fuel pool gates, the transfer tube, and other components that meet the scoping criteria of 10 CFR 54.4 are identified as being within the scope of license renewal and subject to an AMR.

Some components normally associated with the spent fuel system were identified by the applicant as being evaluated in other sections of the LRA. These components are Boraflex neutron absorbing material (Section 4.7), mechanical reactor building penetration (Section 2.4.1.1), and the spent fuel pool structure (Section 2.4.3).

On the basis of its methodology described above, the applicant identifies portions of the spent fuel system that are within the scope of license renewal on the flow diagrams listed in Table 2.3-7 of the LRA. Using the methodology described in Section 2.1.3, “Assessment Using Criteria in 10 CFR 54.21(a)(1),” of the LRA, the applicant compiled a list of mechanical component commodity groupings within the license renewal boundaries that are subject to an AMR and identified their intended functions. In the LRA, Table 3.4-1, the applicant lists the following nine component commodity groups as being subject to an AMR: liner plate, gates, racks, piping, valves, fuel transfer tube, blind flanges, bolting, and external valve parts. The applicant states that maintaining the pressure boundary integrity is the only intended function of the SCs that are subject to an AMR, with the exception of the racks which provide structural support for the stored fuel.

2.3.3.1.2 Staff Evaluation

The staff reviewed Section 2.3.3.1 of the LRA to determine whether there is reasonable assurance that the applicant appropriately identified the spent fuel system SCs that are within the scope of license renewal in accordance with 10 CFR 54.4, and subject to an AMR in accordance with the requirements of 10 CFR 54.21(a)(1).

The staff reviewed the text and diagrams submitted by the licensee in Section 2.3.3.1 of the LRA and the ANO-1 UFSAR to determine if the applicant adequately identified the SSCs of the spent fuel system that are within the scope of license renewal. The staff verified that those portions of the spent fuel system that meet the scoping requirements of 10 CFR 54.4 are included within the scope of license renewal, and are identified as such by the licensee in Section 2.3.3.1 of the LRA. The staff then focused its review on those portions of the spent fuel system that were not identified as being within the scope of license renewal to verify that they do not meet the scoping requirements of 10 CFR 54.4. The staff also reviewed the FSAR to determine if there were any additional system functions that were not identified in the LRA, and verified that those additional functions did not meet the scoping requirements of 10 CFR 54.4. The staff did not identify any omissions by the applicant are, therefore, there is reasonable assurance that the applicant adequately identified all portions of the spent fuel system that should be included within the scope of license renewal in accordance with 10 CFR 54.4.

The staff then determined whether the applicant had properly identified the SCs that are subject to an AMR from among those portions of the spent fuel system that are identified as being within the scope of license renewal. The applicant identifies and lists the SCs that are subject to an AMR for the spent fuel systems in Table 3.4-1 of the LRA using the screening methodology described in Section 2.1 of the LRA. The staff evaluated the scoping and screening methodology, and documented its findings in Section 2.1 of this SER. The staff performed its review by sampling the SCs that the applicant determined to be within the scope of license renewal, but not subject to an AMR, to verify that these SCs perform its intended function(s) with moving parts or with a change in configuration or properties or were subject to replacement based on qualified life or specified time period.

In the LRA, Table 2.3-7, the applicant lists two detailed flow diagrams, LRA-M-232 and 235, of the spent fuel system, and identifies the mechanical components subject to an AMR and its intended functions in Table 3.4-1 of the LRA. The detailed flow diagrams were highlighted to identify those portions of the system that are within the scope of license renewal. The applicant highlighted those components that perform at least one of the intended functions associated with the scoping criteria of 10 CFR 54.4(a). The staff compared the LRA flow diagrams to the system drawings and the descriptions in the UFSAR to ensure they were representative of the spent fuel system. The staff sampled portions of the flow diagrams that were not highlighted to verify that these components did not meet any the scoping criteria in 10 CFR 54.4.

On the basis of this review, in a letter to the applicant dated May 5, 2000, the staff requested additional information regarding several components in the spent fuel system. In its response to the NRC dated August 30, 2000, the applicant provides its response to the staff's RAI regarding certain piping segments, strainers, and flanges that may have met the scoping requirements but were not identified as being within the scope of license renewal by the applicant. In each case, the applicant justified the exclusion of the component, or identified where in the application the component was included within the scope of license renewal.

The staff reviewed the applicant's responses, and the information contained in the LRA and the UFSAR, and found the applicant's responses acceptable for the components of concern.

2.3.3.1.3 Conclusions

On the basis of the staff's review of the information contained in Section 2.3.3.1 of the LRA, the August 30, 2000, response to the staff's RAIs and the supporting information in the ANO-1

UFSAR, as described above, the staff did not identify any omissions by the applicant. Therefore, the staff find that there is reasonable assurance that the applicant has adequately identifies those portions of the spent fuel system that are within the scope of license renewal, and the associated SCs that are subject to an AMR, in accordance with the requirements of 10 CFR 54.4(a) and 54.21(a)(1), respectively.

2.3.3.2 Fire Protection System

In the LRA, Section 2.3.3.2, "Fire Protection," the applicant identifies the fire protection (FP) SSCs that are required for compliance with 10 CFR 50.48, and that are within the scope of license renewal in accordance with 10 CFR 54.4(a)(3) and subject to an AMR. The applicant also identifies the SCs for the FP system that are subject to an AMR in Tables 3.4-2, and 3.4-6 of the LRA. In letters to the applicant dated May 2, 2000, May 5, 2000, and June 1, 2000, the NRC requested additional information regarding the FP system. In letters to the NRC dated August 30, 2000 and November 2, 2000, the applicant provides additional information in response to the staff's RAIs.

2.3.3.2.1 Technical Information in the Application

In accordance with 10 CFR 54.4(a)(3), the SSCs that are relied on in safety analyses or plant evaluation to demonstrate compliance with 10 CFR 50.48, the FP Rule, are within the scope of license renewal. The FP system is relied upon to meet the requirements of 10 CFR 50.48.

In accordance with 10 CFR 50.48, the applicant is required to implement and maintain an FP program. As stated in the ANO-1 UFSAR, Section 9.8.1, "Design Basis," the applicant's FP program is needed to satisfy Appendix A of Branch Technical Position (BTP) APCSB 9.5-1, "FP for Nuclear Power Plants," and Appendix R to 10 CFR Part 50, and other staff positions. In response to an RAI, the applicant states that a fire area analysis was performed at ANO-1 to evaluate the plant equipment required to place the plant in a safe shutdown condition for any single fire scenario. The fire analysis contains a listing of the ANO-1 components that can be used to place the plant in a safe shutdown condition following a fire. The applicant identifies these SCs as being within the scope of license renewal and subject to an AMR. In the LRA, Section 2.1.2, "Assessment Using Criteria in 10 CFR 54.4," the applicant identifies the ANO-1 component database as another means of identifying the SSCs used to fulfill the requirements of 10 CFR 50.48.

The purpose of the FP system is to minimize the effects of fires on SSCs important to safety as required by Appendix A to 10 CFR Part 50. On the basis of the methodology described above, the applicant identifies the highlighted portions of the flow diagrams LRA-M-2219, Sheet 5, and LRA-M-219, Sheet 1, as the evaluation boundaries for the portions of the FP system that are included within the scope of license renewal.

In the LRA, Section 2.3.3.2, the applicant identifies the following FP system components that are within the scope of license renewal and subject to an AMR:

- electric motor-driven fire pump

- diesel-driven fire pump, including the engine gearbox oil cooler, the jacket water heat exchanger and the lube oil cooler (The fuel oil portions of the system are discussed in Section 2.3.3.7, “Fuel Oil,” of the LRA.)

- fire water distribution system, including the portion of the outside loop, hose stations, standpipes, sectional control valves, and isolation valves that are required for protection of safety-related areas sprinkler systems required to meet 10 CFR 50.48 requirements, including piping, control valves, and sprinkler heads.

- sprinkler system required to meet 10 CFR 50.48, including piping, control valves, and sprinkler heads

The intended function of the FP mechanical components, identified by the applicant, is to maintain the system pressure boundary integrity. In the LRA, Table 3.4-2, the applicant shows that the following FP mechanical component groups have pressure boundary intended functions, and are subject to an AMR: pumps, piping, valves, intake air, exhaust air, lube oil, cooling water, and heat exchangers.

2.3.3.2.2 Staff Evaluation

The Commission’s regulations in 10 CFR 54.21(a)(1), states that for those SSCs that are within the scope of this part, as delineated in 10 CFR 54.4, the applicant must identify and list those SCs that are subject to an AMR. The staff reviewed Section 2.3.3.2 of the LRA, as supplemented by a letter to the NRC dated August 30, 2000, to determine whether there was reasonable assurance that the applicant has appropriately identified the SCs that serve FP intended functions that are within the scope of license renewal in accordance with 10 CFR 54.4, and are subject to an AMR in accordance with the requirements of 10 CFR 54.21(a)(1).

The applicant searched its component database for SSCs required to meet 10 CFR 50.48. In a letter to the NRC dated November 2, 2000, the applicant states that the component database uses an F-List designation to identify FP equipment that is part of the ANO-1 CLB for compliance with 10 CFR 50.48. The applicant also states that the F-List was created in the mid- to late-1980's as part of the development of the overall component database. Prior to that time, no comprehensive list existed at ANO-1 to identify components relied upon for compliance with 10 CFR 50.48. The F-List was the source used by the applicant to identify components that are within the scope of license renewal.

The staff sampled portions of Section 9.8.1, "Design Basis,” and Section 9.8.2, “System Description and Evaluation,” of the ANO-1 UFSAR to identify any additional FP system function that met the scoping requirements of 10 CFR 54.4, but that was not identified as an intended function in the LRA. The UFSAR, Section 9.8.1, states that the ANO-1 FP program satisfies the NRC’s criteria documented in Appendix A to BTP APCSB 9.5-1. The NRC staff also reviewed the August 22, 1978, “FP Safety Evaluation Report,” which summarizes the FP program at ANO-1 using the guidelines of Appendix A to BTP 9.5-1. In addition, the staff reviewed the letter from the applicant dated September 17, 1976, which describes Appendix A to BTP 9.5-1, to verify that the function(s) of the FP components relied upon to satisfy the provisions of Appendix A to BTP 9.5-1 were identified as intended functions in the LRA.

The staff then compared the FP SSCs identified in the system flow diagrams LRA-M-2219, Sheet 5, and LRA-M-219, Sheet 1, to verify that the required components were highlighted as being within the evaluation boundaries on the flow diagram, and were not excluded from the scope of license renewal. As part of the evaluation, the staff also sampled portions of the same flow diagrams for the FP system to determine if there were any additional portions of the system piping or components located outside of the evaluation boundary that should have been identified as being within the scope of license renewal.

In a letter to the applicant dated May 5, 2000, the staff requested additional information regarding the exclusion of some FP components required for compliance with 10 CFR 50.48. The applicant's F-List, which designates both safety-related and non-safety-related SSCs required for compliance with 10 CFR 50.48, did not appear to include some non-safety-related SSCs, which the staff views as being required for compliance with Appendix A to BTP 9.5-1. The scope of 10 CFR 50.48 includes those FP components required to meet the provisions of Appendix A to BTP 9.5-1. Since the F-List was created in the mid-1980s, and has never been reviewed or evaluated by the NRC staff, the staff has concerns that the F-List may not adequately capture the FP SSCs required for compliance with Appendix A to BTP 9.5-1. A more detailed discussion is provided below for the components identified by the NRC as not being, but needing to be, included in the F-List.

The staff also asked the applicant to provide the technical basis for the exclusion of selected SCS, including the jockey pump casing, the carbon dioxide system, and the fire hydrants from being within the scope of license renewal. In the letter to the NRC dated November 2, 2000, the applicant states that these systems are not required for compliance with 10 CFR 50.48 for the following reasons:

- The FP jockey pump (casing) is not required on the basis that the only function of the jockey pump is to minimize cycling of the main fire pumps, and is not required to protect safe-shutdown equipment.

- The carbon dioxide system is not required on the basis that it is not needed to protect safe-shutdown equipment.

- The fire hydrants are not required on the basis that they are not a primary source of FP needed to support safe-shutdown in the event of a fire.

- The water supply to the low level radwaste building FP system is not required to protect safe-shutdown equipment.

- The piping to the manual hose station (located downstream of FS-43) is not required because it is not used to protect safe-shutdown equipment.

The staff disagreed with the basis for the exclusion of these SSCs from being within the scope of license renewal. These components do not perform its intended function(s) with moving parts or with changes in configuration or properties, and are not replaced based on qualified life or specified time period, and should be subject to an AMR in accordance with 10 CFR 54.21. The August 22, 1978, NRC-approved SER for FP states that these components satisfied the provisions of Appendix A to BTP 9.5-1. Furthermore, in a letter to the NRC dated

September 17, 1976, which describes its FP program as meeting the guidelines of Appendix A to BTP 9.5-1, the applicant identified these components as part of its FP program.

The exclusion of any FP SSC on the basis that its intended function is not required for the protection of safe-shutdown equipment is not acceptable to the staff, in itself. Compliance with 10 CFR 50.48 requires a FP program that goes beyond safe shutdown, and includes such requirements as a means to limit fire damage to SSCs that are important to safety so that the capability to safely shutdown the plan is ensured as is described in BTP APSCB 9.5-1. In the event that these components are determined to be required for compliance with 10 CFR 50.48, they will need to be subject to an AMR in accordance with 10 CFR 54.21(a).

In a letter to the applicant dated May 5, 2000, the staff requested additional information regarding the exclusion of the pipes and valves connected to the outside FP loop, shown in flow diagram LRA-M-2219, Sheet 5. In a letter to the NRC dated August 30, 2000, the applicant states that the piping and valves that are not highlighted, are not required for the protection of safety-related areas, and that their failure would not affect the capability of the portion of the outer fire water loop, that is required for compliance with 10 CFR 50.48, to perform its intended function.

The staff disagreed with the applicant's response because the piping, which is not included within the scope of license renewal, supplies water to the FP system in the low-level radwaste building. This piping is required to meet the requirements of 10 CFR 50.48 as described in BTP APSCB 9.5-1, and should be subject to an AMR in accordance with 10 CFR 54.21. Flow diagram LRA-M-2219 shows that the piping leading to the radwaste building supplies a wet and dry pipe suppression system within the radwaste building, and is required for compliance with the provisions of Appendix A to BTP 9.5-1 for the protection of areas where a fire could result in the release of radioactive materials to the environment. Furthermore, in a letter dated September 17, 1976, the applicant states in Section 14, "Radwaste Building (Auxiliary Building)," that automatic sprinklers were provided for protection of areas in the radwaste building where combustible materials are located. Therefore, in the event that this suppression system is determined to be required for compliance with 10 CFR 50.48, it will be included within the scope of license renewal and subject to an AMR in accordance with 10 CFR Part 54.

In addition, the staff requested additional information regarding the exclusion of the following FP suppression SSCs, as shown in flow diagram LRA-M-219, Sheet 1:

- lube oil tank deluge system
- lube oil storage tank T-26
- fuel oil tank sprinkler system
- MFW pump deluge system
- basement sprinkler system
- piping located off of FS-43 and FS-90
- hydrogen seal oil unit deluge system
- outside firewater loop to wall sprinkler system

In a letter to the NRC dated August 30, 2000, the applicant states that in accordance with ANO-1 CLB, the FP suppression systems listed above are not required for compliance with

10 CFR 50.48. In a letter dated November 2, 2000, the applicant provided the technical basis for the exclusion of these systems from within the scope of license renewal. On the basis of the staff's review of the letters dated September 17, 1976, and November 2, 2000, the staff agrees that the following suppression systems are not required for compliance with 10 CFR 50.48. The September 17, 1976, letter shows that the applicant never committed to providing suppression systems for the following systems to satisfy Appendix A to BTP 9.5-1:

- lube oil tank deluge system (D-3)
- lube oil storage tank T-26 (D-1)
- fuel oil tank sprinkler system (D-7)
- MFW pump deluge system (E-3)
- basement sprinkler system (E-3)
- hydrogen seal oil unit deluge system (F-3)
- outside firewater loop to wall sprinkler system (Column 1)

For the piping located off of FS-43 and FS-90, the applicant states (in its November 2, 2000 response) that the piping downstream of FS-43 supplies water to turbine building hose stations located on the east side of the structure. The applicant excludes this piping from being within the scope of license renewal on the basis that the types of fires that these hose stations would be utilized to combat would not prevent a safe shutdown of the plant.

The staff disagrees with this response. Failure of the FP piping leading to this portion of the fire suppression system would prevent the hose stations from functioning as designed. Also, hose stations are subject to an AMR in accordance with 10 CFR 54.21. In the August 22, 1978, NRC approved SER, Section 5.17.5, the applicant stated that manual hose station are provided throughout the turbine building. In addition, in Section 3(d) of their September 17, 1976, submittal, the applicant stated that hose stations are provided in the turbine building at 100 foot intervals. Furthermore, exclusion of FP SSCs on the basis that it's intended function are not required for the protection of safe shutdown equipment is not acceptable if that SSC is required for compliance with 10 CFR 50.48. This piping, which supplies the hose stations in the turbine building, is required to fulfill the manual fire suppression requirement of 10 CFR 50.48(a). Therefore, these hose stations should be included within the scope of license renewal and subject to an AMR.

With respect to the piping downstream of FS-90 that provides water to the laundry area of the auxiliary building, this piping is not required for compliance with 10 CFR 50.48 and, therefore, is not within the scope of license renewal.

At the time the initial SER was issued, the applicant did not provide sufficient justification for the exclusion of the FP jockey pump, carbon dioxide systems, fire hydrants, the water supply to the low level radwaste building FP system, and the piping to the manual hose station (located downstream of FS-43). This was Open Item 2.3.3.2.2-1.

In a public meeting with the applicant that took place on March 8, 2001, the NRC staff heard the applicant's position as to why the FP jockey pump, carbon dioxide systems, fire hydrants, the water supply to the low level radwaste building FP system, and the piping to the manual hose station are not included in the applicant's CLB (as documented in the applicant's F-list) in accordance with the requirements of 10 CFR 50.48. The applicant explained that each of these components are maintained to the National Fire Protection Association standards, and provided a technical justification as to why these components are not required for safe shutdown consistent with General Design Criteria III. The staff presented its view that 10 CFR 50.48 goes beyond safe shutdown, and that a number of select components beyond those required by General Design Criteria III are required by 10 CFR 50.48. As a result of this meeting, the applicant agreed to add the jockey pump and fire hydrants to the scope of SCs subject to an AMR and to its F-list consistent with the requirements of 10 CFR 50.48. At the same time, the applicant provided sufficient justification for excluding the carbon dioxide systems, the water supply to the low level radwaste building FP system, and the piping to the manual hose station from the scope of components required to fulfil the requirements of 10 CFR 50.48 (as documented in the applicant's F-list) based on the following additional information.

- The carbon dioxide system is not required on the basis that it was not a requirement under BTP 9.5-1 and was never considered part of the applicant's CLB.

- The water supply to the low level radwaste building FP system is not required on the basis that a fire in the low level radwaste building will not result in the release of radioactive material that would exceed 10 CFR Part 100 limits.

- The piping to the manual hose station (located downstream of FS-43) is not required on the basis that the single manual hose station in question is located on top of the turbine building and is not used to protect equipment important to safety.

This information was documented in a letter to the NRC dated March 14, 2001. The staff had no additional concerns relating to the scope of FP components subject to an AMR, therefore, this item is considered closed.

After determining which components were within the scope of license renewal, the staff reviewed the components the applicant identified as being subject to an AMR. The staff reviewed select components that the applicant identified as being within the scope of license renewal to verify that the applicant determined those SCs that performed its intended functions without moving parts or without a change in configuration or properties, and that are not subject to replacement based on qualified life or specified time period were subject to an AMR.

In a letter to the applicant dated May 5, 2000, the staff requested additional information regarding the exclusion of system filters, fire extinguishers, fire hoses, and air packs from being subject to an AMR. In a letter to the NRC dated August 30, 2000, the applicant states that the system filters, fire extinguishers, fire hoses, and air packs (i.e., self-contained breathing apparatus) are within the scope of license renewal. However, based on the NRC letter from C.I. Grimes to D.J. Walters, NEI, "Consumables," dated March 10, 2000, filters, fire extinguishers, fire hoses, and air packs were excluded from an AMR because the applicant replaces them based on a qualified life. In its RAI, the staff noted that the exclusion of a structure or component from an AMR based on a qualified life determined by performance or condition monitoring required that each SC be identified and listed, and a site-specific evaluation for each of these SCs be included in the LRA.

In its response to the NRC dated August 30, 2000, the applicant states that filters are within the scope of license renewal at ANO-1. Furthermore, they are tested or inspected periodically and replaced as part of ANO-1 TS or preventive maintenance activities; therefore, these filters are replaced based on a qualified life determined by performance monitoring and are not subject to an AMR. Fire extinguishers and fire hoses are routinely monitored and replaced in accordance with National Fire Protection Association (NFPA)-10 and NFPA-1962, respectively, and are also within the scope of license renewal, but not subject to an AMR. In addition, air packs are maintained and replaced in accordance with the self-contained breathing apparatus program contained in 42 CFR Part 84, 29 CFR 19.10, and 19.26, NUREG-41, and ANSI-Z88.2 and, therefore are not subject to an AMR. The staff found the applicant's response consistent with the staff's letter on consumables and, therefore, acceptable.

The staff also reviewed mechanical components, from flow diagrams LRA-M-2219, Sheet 5 and LRA-M-219, Sheet 1, and compared them to the list of components and corresponding intended function(s) presented in Table 3.4-2 of the LRA. On the basis of this review, the staff did not identify any omissions in the SCs identified by the applicant as being subject to an AMR.

2.3.3.2.3 Conclusions

On the basis of the review described above, the staff finds that there is reasonable assurance that the applicant has adequately identified those portions of the FP system that are within the scope of license renewal, and the associated SCs that are subject to an AMR, in accordance with the requirements of 10 CFR 54.4(a) and 54.21(a)(1), respectively.

2.3.3.3 Emergency Diesel Generator

In the LRA, Section 2.3.3.3, "Emergency Diesel Generator," the applicant describes the components of the EDG system that are within the scope of license renewal and subject to an AMR. This system is further described in Section 8.3.1.1.7, "Emergency Power Supply System," of the ANO-1 UFSAR.

2.3.3.3.1 Technical Information in the Application

The system function of the emergency power supply system is to supply emergency power to the engineered safeguards bus loads following a DBA. The emergency power supply system at ANO-1 consists of two diesel generators, each connected to one of the 4160-volt engineered safeguards buses, and their associated support systems. The EDGs are required for event mitigation, and to be available following a fire and are considered included within the scope of license renewal in accordance with 10 CFR 54.4(a)(1) and (a)(3). The applicant identifies the following support systems of the EDG system as being within the scope of license renewal and subject to an AMR:

- safety-related portions of the EDG starting air subsystem
- EDG lubrication subsystem components
- EDG combustion air intake and exhaust subsystem components
- EDG cooling water subsystem components

Some components normally associated with the EDG system were identified by the applicant as being evaluated in other sections of the LRA. These components are the fuel oil system, including the EDG fuel oil components (Section 2.3.3.7), and the service water side of the EDG heat exchangers (Section 2.3.3.10).

On the basis of its methodology described above, the applicant identified portions of the EDG system that are within the scope of license renewal on flow diagrams listed on Table 2.3-7 of the LRA. Using the methodology described in Section 2.1.3, "Assessment Using Criteria in 10 CFR 54.21(a)(1)," of the LRA, the applicant lists the mechanical component commodity groupings that are subject to an AMR and identified its intended functions in Table 3.4-3 of the LRA.

The applicant identifies the following component commodity groups for the four support systems as being subject to an AMR:

- starting air - valves (two types), bolting, external valve parts, piping, tanks, strainers, and tubing

- air intake and exhaust - piping, filters, expansion joints, turbo chargers, valves, and heat exchangers

- lube oil - piping, valves (three types), filters (two types), pumps, strainer, heat exchanger, and sight glass

- cooling water - piping, valves (two types), pumps, tanks, thermowells, and level glass

The applicant states that maintaining the pressure boundary integrity is the only intended function for the listed components, with the exception of the heat exchangers, which also perform a heat transfer intended function.

2.3.3.3.2 Staff Evaluation

The staff reviewed Section 2.3.3.3 of the LRA to determine whether there is reasonable assurance that the applicant appropriately identified the EDG system SCs that are within the scope of license renewal in accordance with 10 CFR 54.4 and subject to an AMR in accordance with the requirements of 10 CFR 54.21(a)(1).

The staff reviewed the text and diagrams submitted by the licensee in Section 2.3.3.3 of the LRA, and the ANO-1 UFSAR to determine if the applicant adequately identified the SSCs of the EDG system that are within the scope of license renewal. The staff verified that those portions of the EDG system, and its support systems that meet the requirements of 10 CFR 54.4 are included within the scope of license renewal, and are identified as such by the applicant in Section 2.3.3.3 of the LRA. The staff then focused its review on those portions of the EDG systems that were not identified as being within the scope of license renewal to verify that they do not meet the scoping requirements of 10 CFR 54.4. The staff also reviewed the UFSAR to determine if there were any additional system functions that were not identified as intended functions in the LRA and verified that those additional functions did not meet the scoping requirements of 10 CFR 54.4. The staff did not identify any omissions by the applicant, therefore, there is reasonable assurance that the applicant has adequately identified all portions of the EDG systems that should be included within the scope of license renewal in accordance with 10 CFR 54.4.

The staff then determined whether the applicant had properly identified the SCs that are subject to an AMR from among those portions of the EDG systems identified as being within the scope of license renewal. The applicant identifies and lists the SCs that are subject to an AMR for the EDG systems in Table 3.4-3 of the LRA using the screening methodology described in Section 2.1 of the LRA. The staff evaluated the scoping and screening methodology and documented their findings in Section 2.1 of this SER. The staff performed their review by sampling the SCs that the applicant identifies as being within the scope of license renewal, but not subject to an AMR to verify that these SCs perform its intended function(s) with moving parts or with a change in configuration or properties or were subject to replacement based on qualified life or specified time period.

In the LRA, Table 2.3-7, the applicant lists three detailed flow diagrams, LRA-M-217, Sheets 2, 3, and 4, of the EDG system, and identifies the mechanical components subject to an AMR and their intended functions in Table 3.4-3 of the LRA. The detailed flow diagrams were highlighted to identify those portions of the system that are within the scope of license renewal. The applicant highlighted those components that performs at least one of the intended functions associated with the scoping criteria of 10 CFR 54.4(a). The staff compared the LRA flow diagrams to the system drawings and the descriptions in the UFSAR to ensure they were representative of the EDG system. The staff sampled portions of the flow diagrams that were not highlighted to verify that those components did not perform any of the intended functions associated with the scoping criteria of 10 CFR 54.4(a).

On the basis of this review, in a letter to the applicant dated May 5, 2000, the staff requested additional information regarding several components in the EDG systems. In its response to the NRC dated August 30, 2000, the applicant provides its response to the staff's RAIs regarding room drains in the EDG building design to protect the diesel generator from flooding that were not included within the scope of license renewal. In addition, the staff identifies several components such as the turbo charger, crankcase ejector, expansion joints, and exhaust silencer that the applicant identified as being within the scope of license renewal, but not subject to an AMR. The applicant clarifies that sufficient drainage of the EDG rooms is provided by a 10-inch, through wall opening located behind a curb and, therefore the room drains are not needed for event mitigation. The structure was included within the scope of license renewal and evaluated in Section 2.4.3 of the LRA. The staff reviewed the applicant's evaluation of this auxiliary building structure in Section 2.4.3 of this SER. The applicant also clarifies where in the LRA the turbo charger, crankcase ejector, expansion joints, and exhaust silencer were evaluated in an AMR. The staff reviewed the applicant's responses and the information contained in the LRA and the UFSAR, and found the applicant's responses acceptable for the components of concern.

2.3.3.3.3 Conclusions

On the basis of the staff's review of the information contained in Section 2.3.3.3 of the LRA, the August 30,2000, response to the staff's RAIs, and the supporting information in the ANO-1 UFSAR, as described above, the staff did not identify any omissions by the applicant. Therefore, the staff finds that there is reasonable assurance that the applicant has adequately identified those portions of the EDG system and associated subsystems that are within the scope of license renewal, and the associated SCs that are subject to an AMR, in accordance with the requirements of 10 CFR 54.4(a) and 54.21(a)(1), respectively.

2.3.3.4 Auxiliary Building Sump and Reactor Building Drains

In the LRA, Section 2.3.3.4, "Auxiliary Building Sump and Reactor Building Drains," the applicant describes the components of the auxiliary building and reactor building sump and drain system that are within the scope of license renewal and subject to an AMR.

2.3.3.4.1 Technical Information in the Application

The overall function of the auxiliary building and reactor building sump and drain system is to collect liquids from the reactor building and auxiliary building for processing and disposal. The following specific system functions are safety-related and consistent with the scoping criteria in 10 CFR 54.4(a)(1):

- prevents flow of radioactive material from reactor building following a LOCA (reactor building penetrations)
- prevents debris from interfering with post-LOCA recirculation (system screens)
- prevents reactor building sump vortexing that could occur under accident conditions (anti-vortex device)
- prevents radioactive liquids that may be present in the decay heat pump room post-LOCA from spreading throughout the auxiliary building drains and isolation valves
- collect reactor coolant pump (RCP) motor oil leakage to reduce the chance of a fire (collection tanks, piping and valves)

The applicant describes its process for identifying the mechanical components that are within the scope of license renewal in Section 2.1.2, "Assessment Using Criteria in 10 CFR 54.4," of the LRA. On the basis of this methodology, the applicant identifies the portions of the auxiliary building and reactor building sump and drain system that are within the scope of license renewal on the flow diagrams that are listed in Table 2.3-7 of the LRA. Using the methodology described in Section 2.1.3, "Assessment Using Criteria in 10 CFR 54.21(a)(1)," of the LRA, the applicant compiles a list of mechanical component commodity groupings that are subject to an AMR and identified their intended functions, in Table 3.4-4 of the LRA.

2.3.3.4.2 Staff Evaluation

The staff reviewed Section 2.3.3.4 of the LRA to determine whether there is reasonable assurance that the applicant has appropriately identified the auxiliary building and reactor building sump and drain system SCs that are within the scope of license renewal in accordance with the requirements of 10 CFR 54.4, and subject to an AMR in accordance with the requirements of 10 CFR 54.21(a)(1).

The staff reviewed the text and diagrams submitted by the licensee in Section 2.3.3.4 of the LRA and the ANO-1 UFSAR to determine if the applicant adequately identified the SSCs that are within the scope of license renewal. The staff verified that those portions of the auxiliary building and reactor building sump and drain system that meet the scoping requirements of 10 CFR 54.4 are included within the scope of license renewal and are identified as such by the licensee in Section 2.3.3.4 of the LRA. The staff then focused its review on those SCs of the auxiliary building and reactor building sump and drain systems that were not identified as being within the scope of license renewal to verify that they do not meet the scoping requirements of 10 CFR 54.4. The staff also reviewed the UFSAR to determine if there were any additional system functions that were not identified in the LRA and verified that those additional functions did not meet the scoping requirements of 10 CFR 54.4. The staff found no omissions by the applicant, therefore, there is reasonable assurance that the applicant adequately identified all portions of the auxiliary building and reactor building sump and drain systems that should be included within the scope of license renewal in accordance with 10 CFR 54.4.

The staff then determined whether the applicant had properly identified the SCs that are subject to an AMR from among those portions of the system that are identified as within the scope of license renewal. The applicant identifies and lists the SCs that are subject to an AMR for the auxiliary building and reactor building sump and drain system in Table 3.4-4 of the LRA using the screening methodology described in Section 2.1.3 of the LRA. The staff evaluated the scoping and screening methodology, and documented its findings in Section 2.1 of this SER. The staff performed its review by sampling the SCs that the applicant determines as being within the scope of license renewal but not subject to an AMR to verify that these SCs perform its intended function(s) with moving parts or with a change or configuration or properties or were subject to replacement based on qualified life or specified time period.

In the LRA, Table 2.3-7, the applicant lists six detailed flow diagrams, LRA-M-213, Sheets 1 and 2, LRA-M-214, Sheet 3, LRA-M-232, Sheet 1, and LRA-M-238 Sheets 1 and 2, of the auxiliary building and reactor building sump and drain systems and identifies the mechanical components subject to an AMR and its intended functions in Table 3.4-4 of the LRA. The detailed flow diagrams were highlighted to identify those portions of the system that are within the scope of license renewal. The applicant highlights those components, which perform at least one of the scoping requirements of 10 CFR 54.4. The staff compared the LRA flow diagrams to the system drawings and the descriptions in the UFSAR to ensure they were representative of the auxiliary building and reactor building sump and drain systems. The staff sampled portions of the flow diagrams that were not highlighted to verify that these components did not meet any of the scoping criteria in 10 CFR 54.4.

On the basis of this review, in a letter to the applicant dated May 5, 2000, the staff requested additional information regarding several components in the auxiliary building and reactor building sump and drain system. In its response to the NRC dated August 30, 2000, the applicant responded to the staff's RAIs, regarding the inclusion of the drain lines located in the decay heat removal pump rooms within the scope of license renewal. The decay heat removal pump rooms are credited as pressure boundaries for offsite dose calculations. The applicant states that the components in question had been included within the scope of license renewal and subject to an AMR, and should have been highlighted on the drawing. The staff find the applicant's response acceptable.

2.3.3.4.3 Conclusions

On the basis of the staff's review of the information contained in Section 2.3.3.4 of the LRA, and the supporting information in the ANO-1 UFSAR, as described above, the staff did not identify any omissions by the applicant. Therefore, the staff finds that there is reasonable assurance that the applicant has adequately identified those portions of the auxiliary building and reactor building sump and drain system that are within the scope of license renewal, and the associated SCs that are subject to an AMR, in accordance with the requirements of 10 CFR 54.4(a) and 54.21(a)(1), respectively.

2.3.3.5 Alternate AC Diesel Generator

In the LRA, Section 2.3.3.5, "Alternate AC Diesel Generator," the applicant describes the components of the AAC diesel generator and its support systems that are within the scope of license renewal and subject to an AMR.

2.3.3.5.1 Technical Information in the Application

The system function of the AAC generator is to provide backup power in the event of a station blackout at ANO-1 or ANO-2. The AAC generator is a 4400 kW diesel generator and is credited with supplying power during a loss off site power concurrent with the loss of the EDGs. The applicant identifies the following support systems of the AAC generator as being within the scope of license renewal and subject to an AMR:

- portions of the AAC generator starting air subsystem
- AAC generator lubrication subsystem components
- AAC generator combustion air intake and exhaust subsystem components
- AAC generator cooling water subsystem components
- engine room exhaust fans and the corresponding inlet air dampers
- switchgear room exhaust fan and its associated inlet air damper

Some components normally associated with the AAC generator were identified by the applicant as being evaluated in other sections of the LRA. These components are the AAC generator building (Section 2.4.6.1), and fuel oil system including the AAC generator (Section 2.3.3.7).

On the basis of its methodology described above, the applicant identifies the portions of the AAC generator that are within the scope of license renewal on flow diagrams listed on Table 2.3-7 of the LRA. Using the methodology described in Section 2.1.3, "Assessment Using Criteria in 10 CFR 54.21(a)(1)," of the LRA, the applicant lists the mechanical component commodity groupings that are subject to an AMR and identified their intended functions, in Table 3.4-5 of the LRA.

The applicant identifies the following component commodity groups for the four support systems as subject to an AMR:

- starting air - valves (four types), piping, tanks, filters (two types), and motor casing
- air intake and exhaust - piping, filters, expansion joints, turbo chargers, valves (two types), muffler, and heat exchanger
- lube oil - piping, valves (three types), pumps, and heat exchanger
- cooling water - piping, valves (three types), pumps, tanks, thermowells, heaters, orifices, filters, heat exchanger, and level glass
- AAC building ventilation - fans and dampers/louvers

The applicant states that maintaining the pressure boundary integrity is the only intended function for the listed components, with the exception of the heat exchangers, which also perform a heat transfer intended function.

2.3.3.5.2 Staff Evaluation

The staff reviewed Section 2.3.3.5 of the LRA to determine whether there is reasonable assurance that the applicant appropriately identified the AAC generator SCs that are within the scope of license renewal in accordance with 10 CFR 54.4 and subject to an AMR in accordance with the requirements of 10 CFR 54.21(a)(1).

The staff reviewed the text and diagrams submitted by the licensee in Section 2.3.3.5 of the LRA, and the ANO-1 UFSAR to determine if the applicant adequately identified the SSCs of the system that are within the scope of license renewal. The staff verified that those portions of the AAC generator system that meet the scoping requirements of 10 CFR 54.4 are included within the scope of license renewal, and are identified as such in Section 2.3.3.5 of the LRA. The staff then focused its review on those portions of the AAC generator systems that were not identified as being within the scope of license renewal to verify that they do not meet the scoping criteria of 10 CFR 54.4. The staff also reviewed the UFSAR to determine if there were any additional system functions that were not identified as intended functions in the LRA and verified that those additional functions did not meet the scoping requirements of 10 CFR 54.4. The staff did not identify any omissions by the applicant, therefore, there is reasonable assurance that the applicant has adequately identified all portions of the AAC generator systems that are within the scope of license renewal in accordance with 10 CFR 54.4.

The staff then determined whether the applicant had properly identified the SCs that are subject to an AMR from among those parts of the systems identified as being within the scope of license renewal. The applicant identifies and lists the SCs that are subject to an AMR for the AAC diesel generator systems in Table 3.4-5 of the LRA using the screening methodology described in Section 2.1 of the LRA. The staff performed its review by sampling the SCs that the applicant determined as being within the scope of license renewal, but not subject to an AMR to verify that these SCs perform its intended functions with moving parts or with a change in configuration or properties or were subject to replacement based on qualified life or specified time period.

In the LRA, Table 2.3-7, the applicant lists five detailed flow diagrams of the AAC generator on LRA-M-2241, Sheets 1, 2, 4 and 5, and LRA-M-2260, Sheet 4, and identified the mechanical components subject to an AMR and their intended functions in Table 3.4-5 of the LRA. The detailed flow diagrams were highlighted to identify those portions of the system that are within the scope of license renewal. The applicant highlights those components which meet at least one of the scoping criteria of 10 CFR 54.4. The staff compared the LRA flow diagrams to the system drawings and the descriptions in the UFSAR to ensure they were representative of the AAC generator systems. The staff sampled portions of the flow diagrams that were not highlighted to verify that these components did not meet any of the intended functions associated with the scoping criteria of 10 CFR 54.4.

On the basis of this review, in a letter to the applicant dated May 5, 2000, the staff requested additional information regarding several components in the AAC generator systems. In a letter to the NRC dated August 30, 2000, the applicant provides its response to the staff's RAI regarding pipe segments that were not identified as being within the scope of license renewal. In addition, several other components were identified by the applicant as being within the scope of license renewal, but the staff could not determine whether these components were identified by the applicant as being subject to an AMR. In its response the applicant clarified that some of the pipe segments in question were incorrectly identified on Drawing LRA-M-2241 and should have indicated that those pipe segments were included within the scope of license renewal and subject to an AMR. The applicant states that the crankcase vent lines and pressure sensing lines did not perform an intended function and were not within the scope of license renewal. The applicant also identifies nine additional components that were evaluated in its AMR of the AAC generator systems. The staff reviewed the applicant's responses to the RAIs, and the information in the LRA and the UFSAR, and found the applicants responses acceptable in addressing these concerns.

2.3.3.5.3 Conclusions

On the basis of the staff's review of the information contained in Section 2.3.3.5 of the LRA, the August 30, 2000, response to the staff's RAIs, and the supporting information in the ANO-1 UFSAR, as described above, the staff did not identify any omissions by the applicant. Therefore, the staff finds that there is reasonable assurance that the applicant has adequately identified those portions of the AAC diesel generator that are within the scope of license renewal, and associated SCs that are subject to an AMR, in accordance with 10 CFR 54.4(a) and 54.21(a)(1), respectively.

2.3.3.6 Halon System

In the LRA, Section 2.3.3.6, "Halon," the applicant describes the Halon fire suppression system equipment, which protects the areas above the ceiling tiles, and below the false floor of the ANO-1 control room as required by 10 CFR 50.48. The applicant identifies the Halon system as being within the scope of license renewal, and identifies the SCs that are subject to an AMR. In letters to the applicant dated May 2, May 5, and June 1, 2000, the NRC requested additional information concerning the ANO-1 Halon system. The applicant responded to the staff's RAIs in letters to the NRC dated August 30, 2000 and November 2, 2000.

2.3.3.6.1 Technical Information in the Application

SSCs that are relied on in a safety analysis or plant evaluation to perform a function that demonstrates compliance with the Commission's regulations described in 10 CFR 54.4(a)(3) are within the scope of license renewal. The Halon system is relied upon to meet the Commission regulation for fire protection, 10 CFR 50.48, in accordance with 10 CFR 54.4(a)(3).

This regulation, 10 CFR 50.48, requires that an applicant implement and maintain an FP program. The ANO-1 FP program is required to satisfy the NRC criteria in Appendix A to Branch Technical Position (BTP) APCSB 9.5-1, "FP for Nuclear Power Plants," and Appendix R to 10 CFR 50, and other staff positions. In a letter to the NRC dated August 30, 2000, the applicant states that a fire area analysis was performed at ANO-1 to evaluate the plant equipment required to place the plant in a safe shutdown condition for any single fire scenario. The fire analysis contains a listing of the ANO-1 components that can be used to place the plant in a safe shutdown condition following a fire. These components are within the scope of license renewal. Additional sources used to identify 10 CFR 50.48 SSCs were the ANO-1 component database as discussed in the LRA, Section 2.1.2, "Assessment Using Criteria in 10 CFR 54.4."

In the LRA, Section 2.3.3.6, the applicant identifies the following portions of the Halon system as being within the scope of license renewal:

- Halon cylinders
- actuation valves
- pilot piping
- manual actuator cylinders and valves
- discharge piping
- outlet nozzles

The intended function identified by the applicant that was considered during the AMR of these SCs was to maintain the system pressure boundary integrity. The electrical portions of the Halon system were evaluated in LRA Section 2.5, "Electrical and Instrumentation and Controls System Scoping and Screening Results." The bottle racks, and structural and component supports, as well as ceiling tiles, marinite boards, concrete walls, concrete and false floor components that are required to enclose selected areas to allow effective use of the Halon system were addressed in Section 2.4.3 of the LRA, "Auxiliary Building."

License renewal flow diagram LRA-M-219, Sheet 2, shows the evaluation boundaries for the portions of the Halon system that are within the scope of license renewal. In the LRA, Table 3.6-4, the applicant identifies the following Halon system mechanical components that are needed to maintain system pressure boundary integrity, and that are subject to an AMR:

- valves
- pipe
- tanks

- discharge nozzles
- discharge tube
- pilot header discharge tube flexible connectors

2.3.3.6.2 Staff Evaluation

In accordance with 10 CFR 54.21(a)(1), the NRC staff reviewed Section 2.5.11 of the LRA, as supplemented by letter dated February 8, 1999, and the other documentation discussed below, to determine whether there is reasonable assurance that the applicant has appropriately identified the SSCs that serve FP-intended functions as being within the scope of license renewal in accordance with 10 CFR 54.4, and the corresponding SCs that are subject to an AMR in accordance with the requirements of 10 CFR 54.21(a)(1).

The applicant searched its licensing documents for commitments to 10 CFR 50.48 and to evaluate plant equipment required to place the plant in a safe shutdown condition for any single fire scenario. In the response to NRC dated August 30, 2000, the applicant states that the scope of SSCs required by 10 CFR 50.48 is consistent with the ANO-1 CLB, and that the component database includes SSCs required to meet 10 CFR 50.48 and Appendix R, Section III.G, III.J., and III.O.

The staff sampled portions of Section 9.8.1, “Design Basis,” and Section 9.8.2, “System Description and Evaluation” of the ANO-1 UFSAR to determine if there were any Halon system functions that were not identified in the LRA during the scoping of SSCs for the Halon system. The staff then compared the Halon SCs identified within the UFSAR to the Halon flow diagram, LRA-M-219, Sheet 2, to verify that the required SCs were subject to an AMR. As part of the evaluation, the staff also reviewed the same flow diagram for the Halon system to determine if there were any additional portions of the system piping or components located outside of the evaluation boundary, with intended functions that should have been identified as being within the scope of license renewal.

For the Halon system, the staff determined that the applicant identified all the SSCs that are within the scope of license renewal. In addition, the applicant identified the SCs that perform a pressure boundary intended function and are, therefore, subject to an AMR. These components include enclosures, flex hoses, pipes, tubing, and valve bodies.

The staff did identify a concern that smoke detectors located on flow diagram LRA-M-219, Sheet 2, for the Halon system, were not included within the highlighted evaluation boundaries. In a letter to the NRC dated August 30, 2000, the applicant states that smoke detectors are included within the scope of license renewal at ANO-1, but were not highlighted on the applicable drawing because the drawings were primarily intended to show the pressure boundary portions of systems that are within the scope of license renewal, and not the electrical components that are within the scope of license renewal. The staff found the applicant’s response acceptable.

On the basis of the review described above, the staff determined that there is reasonable assurance that the applicant adequately identified the portions of the Halon system that are within the scope of license renewal in accordance with 10 CFR 54.4.

After determining which SSCs were within the scope of license renewal, the staff sampled the SCs that the applicant identified as being subject to an AMR. The staff sampled portions of mechanical components, from flow diagram LRA-M-219, Sheet 2, and compared them to the list of SCs and the intended functions identified by the applicant in Table 3.6-4, of the LRA to verify that there were no omissions in the SCs identified by the applicant as being subject to an AMR.

The staff was concerned that certain components, which provide an enclosure for the effective use of the Halon system, were excluded from an AMR. In a letter to the applicant dated June 1, 2000, the staff requested additional information to verify that the control room Halon system supports listed in the LRA, Table 3.6-4, included the ceiling tiles, marinite boards, concrete walls, and concrete and false floor components referred to in Section 2.3.3.6 of the LRA. The applicant's response and staff's overall evaluation of these SCs are provided in Section 3.3.6 of this SER. On the basis of the review described above, the staff did not find any omissions in the FP SCs identified by the applicant as being subject to an AMR.

2.3.3.6.3 Conclusions

On the basis of its review, the staff finds that there is reasonable assurance that the applicant has adequately identified the portions of the Halon system that are within the scope of license renewal, and the associated SCs that are subject to an AMR, in accordance with the requirements of 10 CFR 54.4(a) and 54.21(a)(1), respectively.

2.3.3.7 Fuel Oil

In the LRA Section 2.3.3.7, "Fuel Oil," the applicant describes the components of the fuel oil system that are within the scope of license renewal and subject to an AMR. This system is further described in Section 8.3.1.1.7.2 of the ANO-1 UFSAR.

2.3.3.7.1 Technical Information in the Application

The system function of the fuel oil system is to store and supply fuel oil to diesel-driven safety related and non-safety-related components. The system includes the emergency diesel fuel tanks and the EDG day tank, which have the safety-related function of storing and supplying the EDGs with fuel oil. Also included in this system is the bulk fuel oil storage tank which supplies fuel oil to non-safety-related equipment including the AAC generator and the diesel fire pump day tanks. In addition to the tanks, the equipment and piping required to transfer the fuel oil to these various components are also within scope for license renewal. The applicant identified these components as being within the scope of license renewal because they meet the requirements of 10 CFR 54.4(a).

The applicant describes its process for identifying the mechanical components that are within the scope of license renewal in Section 2.1.2, "Assessment Using Criteria in 10 CFR 54.4," of the LRA. On the basis of this methodology, the applicant identifies portions of the fuel oil system that are within the scope of license renewal on flow diagrams that are listed in Table 2.3-7 of the LRA. Using the methodology described in Section 2.1.3, "Assessment Using Criteria in 10 CFR 54.21(a)(1)," of the LRA, the applicant compiles a list of mechanical component commodity groupings that are subject to an AMR and identified their intended functions in Table 3.4-7 of the LRA. Specifically, the applicant identifies the following fourteen component commodity groups as subject to an AMR: piping, valves (three types), filters (two types), pumps (two types), tubing (two types), thermowells, strainers, tanks, and heat exchangers. The applicant states that maintaining pressure boundary integrity is the only intended function, with the exception of the heat exchangers, which provide heat transfer function for various components.

2.3.3.7.2 Staff Evaluation

The staff reviewed Section 2.3.3.7 of the LRA to determine whether there is reasonable assurance that the applicant has appropriately identified the fuel oil system SCs that are within the scope of license renewal in accordance with 10 CFR 54.4 and subject to an AMR in accordance with the requirements of 10 CFR 54.21(a)(1).

The staff reviewed the text and diagrams submitted by the applicant in Section 2.3.3.7 of the LRA and the ANO-1 UFSAR to determine if the applicant has appropriately identified the SSCs that are within the scope of license renewal. The staff verified that those portions of the fuel oil system that meet the scoping requirements of 10 CFR 54.4 are included within the scope of license renewal, and these SCs are identified as such in Section 2.3.3.7 of the LRA. The staff then focused its review on the SCs of the fuel oil system that were not identified as being within the scope of license renewal to verify that these SCs do not meet the scoping requirements of 10 CFR 54.4. The staff also reviewed the UFSAR to determine if there were any additional system functions that were not identified in the LRA and verified that those additional functions did not meet the scoping requirements of 10 CFR 54.4. The staff did not identify any omissions by the applicant, therefore, there is reasonable assurance that the applicant adequately identified all portions of the fuel oil system that should be within the scope of license renewal in accordance with 10 CFR 54.4.

The staff then determined whether the applicant had properly identified the SCs that are subject to an AMR from among those portions of the systems that are identified as being within the scope of license renewal. The applicant identifies and lists the SCs that are subject to an AMR for the fuel oil system in Table 3.4-7 of the LRA using the screening methodology described in Section 2.1 of the LRA. The staff evaluated the scoping and screening methodology and documented its findings in Section 2.1 of this SER. As described in more detail below, the staff performed its review by sampling SCs that were within the scope of license renewal but not subject to an AMR to verify that these SCs performed their intended functions with moving parts or without a change in configuration or properties, or were subject to replacement based on qualified life or specified time period.

In the LRA, Table 2.3-7, the applicant lists five detailed flow diagrams, LRA-M-217, Sheets 1, 2, and 3, LRA-M-219, Sheet 1, and LRA-M-2241, Sheet 3, of the fuel oil system, and identifies the mechanical components subject to an AMR and their intended functions in Table 3.4-7 of the LRA. The detailed flow diagrams were highlighted to identify those portions of the system that are within the scope of license renewal. The applicant highlighted those components which meet at least one of the scoping criteria of 10 CFR 54.4(a). The staff compared the LRA flow diagrams to the system drawings and descriptions in the UFSAR to ensure they were representative of the fuel oil system. The staff sampled portions of the flow diagrams that were not highlighted to verify that these components did not meet any of the intended functions associated with the scoping criteria of 10 CFR 54.4(a).

In a letter to the applicant dated May 5, 2000, the staff requested additional information regarding several components in the fuel oil system. In a letter to the NRC dated August 30, 2000, the applicant responded to those RAIs. Specifically, the staff questioned whether the vent lines on the tanks in the fuel oil system should be included within the scope of license renewal. The applicant responded that no credible aging effect including the complete loss of the vent line could prevent a tank from being vented. The staff concluded that the vent lines do not perform any of the scoping criteria in 10 CFR 54.4(a). The staff also questioned whether pipe segments (tubing) for the fuel oil day tank and the governors' instrumentation should be in scope for license renewal. The applicant states that the tubing to the tank level switch is within the scope of license renewal; however, the tubing used to vent fuel oil system components are not within scope. The components identified in the governors' instrumentation were mechanical linkage and not tubing. The mechanical linkage requires a change in configuration to perform its intended function and, therefore, is not subject to an AMR.

The staff reviewed the applicant's responses, the information in the LRA, and the UFSAR, and found the applicant's responses acceptable in addressing these concerns.

2.3.3.7.3 Conclusions

On the basis of the staff's review of the information contained in Section 2.3.3.7 of the application, the August 30, 2000, response to the staff's information request, and the supporting information in the ANO-1 UFSAR, as described above, the staff did not identify any omissions by the applicant. Therefore, the staff finds that there is reasonable assurance that the applicant has adequately identified those portions of the fuel oil system that are within the scope of license renewal, and the associated SCs that are subject to an AMR, in accordance with the requirements of 10 CFR 54.4(a) and 54.21(a)(1), respectively.

2.3.3.8 Instrument Air

In the LRA, Section 2.3.3.8, "Instrument Air," the applicant describes the components of the instrument air system that are within the scope of license renewal and subject to an AMR. This system is further described in Section 9.9 of the ANO-1 UFSAR.

2.3.3.8.1 Technical Information in the Application

The system function of the instrument air system is to provide a reliable supply of dry, oil-free, compressed air for pneumatic equipment operation. Most of the system is not safety-related and does not meet the scoping criteria for license renewal. However, some safety-related components utilize instrument air for operation of their pneumatic components. While many of the pneumatic components fail in the desired post-accident position upon loss of air supply the following components require that pressure boundary integrity be maintained following an accident and, therefore, are within the scope of license renewal:

- reactor building penetrations for the instrument air system

- reactor coolant pump (RCP) motor and lube oil cooling water supply valves' air supply
- letdown coolers and RCP seal coolers cooling water supply and return valves' air supply
- control room ventilation emergency fan filter unit air damper control air supply

The applicant describes it process for identifying the mechanical components that are within the scope of license renewal in Section 2.1.2, "Assessment Using Criteria in 10 CFR 54.4," of the LRA.

On the basis of its methodology described above, the applicant identifies portions of the instrument air system that are within the scope of license renewal on flow diagrams listed on Table 2.3-7 of the LRA. Using the methodology described in Section 2.1.3, "Assessment Using Criteria in 10 CFR 54.21(a)(1)," of the LRA, the applicant lists the mechanical component commodity groupings that are subject to an AMR and identified their intended functions, in Table 3.4-8 of the LRA. The applicant identifies the following nine component commodity groups as subject to an AMR: piping, valves (three types), tubing (two types), tanks, flanges, and regulators. The applicant also identifies maintaining pressure boundary integrity as the only intended function.

2.3.3.8.2 Staff Evaluation

The staff reviewed Section 2.3.3.8 of the LRA to determine whether there is reasonable assurance that the applicant appropriately identified the instrument air system SCs that are within the scope of license renewal in accordance with 10 CFR 54.4 and subject to an AMR in accordance with the requirements of 10 CFR 54.21(a)(1).

The staff reviewed the text and diagrams submitted by the licensee in Section 2.3.3.8 of the LRA, and the ANO-1 UFSAR to determine if the applicant adequately identified the SSCs of the system that are within the scope of license renewal. The staff verified that those portions of the instrument air system that meet the scoping requirements of 10 CFR 54.4 are included within the scope of license renewal and are identified as such in Section 2.3.3.8 of the LRA. The staff then focused its review on those portions of the instrument air system that were not identified as being within the scope of license renewal to verify that they do not meet the scoping requirements of 10 CFR 54.4. The staff also reviewed the UFSAR to determine if there were any additional system functions that were not identified as intended functions in the LRA and verified that those additional functions did not meet the scoping requirements of 10 CFR 54.4. The staff did not find any omissions by the applicant, therefore, there is reasonable assurance that the applicant has adequately identified all portions of the instrument air system that are within the scope of license renewal in accordance with 10 CFR 54.4.

The staff then determined whether the applicant had properly identified the SCs that are subject to an AMR from among those portions of the system identified as being within the scope of license renewal. The applicant identifies and lists the SCs that are subject to an AMR for the instrument air system in Table 3.4-8 of the LRA using the screening methodology described in Section 2.1 of the LRA. The staff performed its review by sampling the SCs that the applicant identifies as being within the scope of license renewal, but not subject to an AMR to verify that these SCs perform its intended functions with moving parts or with a change in configuration or properties, or were subject to replacement based on qualified life or specified time period.

In the LRA, Table 2.3-7, the applicant lists eighteen detailed flow diagrams for the instrument air system, and identified the mechanical components subject to an AMR. The applicant also identifies the intended functions in Table 3.4-8 of the LRA. The detailed flow diagrams were highlighted to identify those portions of the system that are within the scope of license renewal. The applicant highlights those components, which meet at least one of the scoping criteria of 10 CFR 54.4(a). The staff compared the LRA flow diagrams to the system drawings and the descriptions in the UFSAR to ensure they were representative of the instrument air system. The staff sampled portions of the flow diagrams that were not highlighted to verify that these components did not perform any of the intended functions associated with the scoping criteria of 10 CFR 54.4(a).

In a letter to the applicant dated May 5, 2000, the staff requested additional information regarding the instrument air system. In a letter to the NRC dated August 30, 2000, the applicant provides its response to the staff's RAI regarding the tubing that provides control air to containment isolation valves, and whether they should be included within the scope of license renewal. The applicant states that failure of the tubing would place the valve in the required position in the event of an accident. Therefore the tubing does not meet the scoping criteria in 10 CFR 54.4(a)(1). The staff reviewed the applicant's response and found it acceptable.

2.3.3.8.3 Conclusions

On the basis of the staff's review of the information contained in Section 2.3.3.8 of the LRA , the August 30, 2000, response to the staff's RAIs, and the supporting information in the ANO-1 UFSAR, as described above, the staff did not identify any omissions by the applicant. Therefore, the staff finds that there is reasonable assurance that the applicant has adequately identified those portions of the instrument air system that are within the scope of license renewal, and the associated SCs that are subject to an AMR, in accordance with the requirements of 10 CFR 54.4(a) and 54.21(a)(1), respectively.

2.3.3.9 Chilled Water

In the LRA, Section 2.3.3.9, "Chilled Water," the applicant describes the components of the chilled water system that are within the scope of license renewal and subject to an AMR. This system is also described in Sections 6.3, 8.3.2.1.7, 9.7.2.1 of the ANO-1 UFSAR.

2.3.3.9.1 Technical Information in the Application

The primary function of the chilled water system is to provide chilled water to the cooling coils of a variety of room and area ventilation units.

The applicant describes its process for identifying the mechanical components that are within the scope of license renewal in Section 2.1.1, "Assessment Using Criteria in 10 CFR 54.4," of the LRA. The applicant identifies the portion of the chilled water system that supplies the electrical, switchgear, and battery room emergency coolers, as being within the scope of license renewal. The safety-related function of these SSCs is to supply chilled water for emergency cooling to coolers that service the safety-related electrical equipment located in the above-mentioned rooms. Two emergency chillers, the internal surfaces of the six cooling coils supplied by the chillers, the associated valves, and piping are within the scope of license renewal and subject to an AMR. Also, the main chilled water system reactor building penetrations piping and valves are included in the scope of license renewal and subject to an AMR because they perform a safety function for reactor building isolation.

The applicant identifies some components associated with the chilled water system as being evaluated in other section of the LRA, including the fan/coil housing assemblies, the external surfaces of the cooling coils, the ductwork, and fire dampers in the ductwork.

Consistent with its methodology, the applicant identifies the portions of the chilled water system that are within the scope of license renewal on the flow diagrams listed in Table 2.3.7 of the LRA. Using the methodology described in the LRA, Section 2.1.3, "Assessment Using Criteria in 10 CFR 54.21(a)(1)," the applicant compiles a list of mechanical component commodity groupings that are subject to an AMR and identifies their intended functions in Table 3.4.9 of the LRA. The following twelve component groups are subject to an AMR: piping, valves, thermowells, tanks, pumps, tubing, coils, sight glasses, filters, compressors, mufflers, and heat exchangers (condensers and evaporators). The applicant states that pressure boundary and heat transfer are the intended functions of the chilled water system mechanical components.

2.3.3.9.2 Staff Evaluation

The staff reviewed Section 2.3.3.9 of the LRA to determine whether there is reasonable assurance that the applicant appropriately identifies the chilled water system components and supporting structures as being subject to an AMR in accordance with the requirement of 10 CFR 54.21(a)(1). The staff reviewed the text and diagrams submitted by the applicant in Section 2.3.3.9 of the LRA and the ANO-1 UFSAR to identify any SCs that may have been omitted from an AMR.

The applicant identifies and lists the SCs that are subject to an AMR for the chilled water system in Table 3.4.9 of the LRA, using the screening methodology described in Section 2.1 of the LRA. The staff evaluated the scoping and screening methodology and documented their findings in Section 2.1 of this SER. The staff then performed a review of the implementation of that methodology for the chilled water system by sampling the SCs that were within the scope of license renewal, but not subject to an AMR to verify that these SCs perform its intended functions with moving parts or with a change in configuration or properties, or were subject to replacement based on a qualified or specified time period.

In the LRA, Table 2.3.7, the applicant lists two detailed flow diagrams, LRA-M-221, Sheet 2, and LRA-M-222, Sheet 1, of the chilled water system. The applicant also identifies the mechanical components subject to an AMR and its intended functions in Table 3.4.9 of the LRA. The detailed flow diagrams were highlighted to identify those portions of the system that were included within the scope of license renewal. The applicant highlighted those components, which meet at least one of the scoping criteria of 10 CFR 54.4(a). The staff compared the LRA flow diagrams to the system drawings and descriptions in the UFSAR to ensure that they were representative of the chilled water system. The staff also sampled portions of the flow diagrams that were not highlighted to ensure these components did not perform any of the functions associated with the scoping criteria of 10 CFR 54.4(a).

In a letter to the applicant dated May 5, 2000, the staff requested additional information regarding the emergency feedwater (EFW) pump room unit coolers that were not considered in the scope of license renewal. In its response to the NRC dated August 30, 2000, the applicant confirms that the EFW pump room unit coolers do not meet any of the scoping requirements in 10 CFR 54.4(a) and, therefore, were not included in the scope of license renewal. In a telephone conference with the applicant on October 31, 2000, the staff raised a question regarding the appropriate cross reference to drawing LRA-M-221 of the UFSAR. The applicant indicates that the current UFSAR does not include a drawing of the chilled water system. The staff, therefore, was unable to verify the content of LRA-M-221. However, during the license renewal scoping inspection as documented in inspection report IR 00-17, the staff performed a review of site controlled piping and instrument drawing of the chilled water system and walked-down portions of that system to verify the accuracy of LRA-M-221.

2.3.3.9.3 Conclusions

On the basis of the review described above, the staff finds that there is reasonable assurance that the applicant has adequately identified those portions of the chilled water system that are within the scope of license renewal, and the associated SCs that are subject to an AMR, in accordance with the requirements of 10 CFR 54.4(a) and 54.21(a)(1), respectively.

2.3.3.10 Service Water

In Section 2.3.3.10, “Service Water,” of the LRA, the applicant described the components of the service water system that are within the scope of license renewal and subject to an AMR. This system is further described in Section 9.3 of the ANO-1 UFSAR.

2.3.3.10.1 Technical Information in the Application

The primary function of the service water system is to transfer heat from safety-related components to an ultimate heat sink. Lake Dardanelle and the emergency cooling pond (ECP) serve as the plant’s ultimate heat sink. If the water supply from Lake Dardanelle is lost, water from the ECP can be fed by gravity through the supply line to the service water compartment in the intake structure. The service water system consists of two independent but interconnected, 100 percent redundant trains to insure continuous heat removal. In the event of a loss offsite power supply, the service water pumps are powered from the diesel generators.

The applicant describes its methodology for identifying the mechanical components that are within the scope of license renewal in Section 2.1.1, “Assessment Using Criteria in 10 CFR 54.4,” of the LRA. The applicant states that the safety-related service water system provides the emergency supply of water to the emergency feedwater pumps and the spent fuel pool. The service water system is also credited in the fire analysis, and is required to meet the requirements of 10 CFR 50.48. All passive, long-lived safety-related components and piping in the service water system, in addition to the piping to and from the ECP and the sluice gates, are within the scope of license renewal and subject to an AMR.

The applicant identifies some of the components associated with the service water system that are evaluated in other sections of the LRA. These components are the intake structure and the ECP (Sections 2.4.4 and 2.4.5, respectively), and the penetration assembly (section 2.4.3).

On the basis of its methodology described above, the applicant identified portions of the service water system that are within the scope of license renewal, and are shown on the flow diagrams listed on Table 2.3.7 of the LRA. Using the methodology described in Section 2.1.3, "Assessment Using Criteria in 10 CFR 54.21(a)(1)," of the LRA, the applicant lists the mechanical component commodity groupings that are subject to an AMR, and identifies its intended function(s) in Table 3.4.10 of the LRA. The applicant identifies the following eight component commodity groups that are subject to an AMR: piping, pumps, strainers, valves, flow elements, thermowells, sluice gates, and heat exchangers. The applicant also identifies maintaining the pressure boundary and heat transfers as the intended functions.

2.3.3.10.2 Staff Evaluation

The staff reviewed Section 2.3.3.10 of the LRA to determine if the applicant has adequately identified the SSCs of the service water system that are within the scope of license renewal, and the SCs that are subject to an AMR in accordance with 10 CFR 54.4(a), and 54.21(a)(1), respectively. The staff reviewed the text and diagrams submitted by the applicant in Section 2.3.3.10 of the LRA and the ANO-1 UFSAR to identify any SSCs of the service water system that may have been omitted from the scope of license renewal that meet the scoping criteria in 10 CFR 54.4. The SSCs of the service water system that meet the license renewal scoping criteria are included within the scope of license renewal, and are identified as such by the applicant in Section 2.3.3.10 of the LRA.

The applicant identifies and lists the SCs that are subject to an AMR for the service water system in Table 3.4.10 of the LRA using the screening methodology described in Section 2.1 of the LRA. The staff evaluated the scoping and screening methodology and documented their findings in Section 2.1 of this SER. The staff then performed a review of the implementation of the methodology for the service water system by sampling the SCs that were identified as being within the scope of license renewal but not subject to an AMR to verify that these SCs perform its intended functions with moving parts or with a change in configuration or properties, or are subject to replacement based on qualified life or specified time period.

In the LRA, Table 2.3-7, the applicant lists four detailed flow diagrams, LRA-M-204, Sheet 3; LRA-M-209, Sheet 1; LRA-M-210, Sheet 1; and LRA-M-221, Sheet 2, of the service water system. The applicant also identifies the mechanical components subject to an AMR and its intended functions in Table 3.4.10 of the LRA. The detailed flow diagrams were highlighted to identify those portions of the system that were included within the scope of license renewal. The applicant highlighted those components, which meet at least one of the scoping criteria of 10 CFR 54.4(a). The staff compared the LRA flow diagrams to the system drawings and descriptions in the UFSAR to ensure that they were representative of the service water system. The staff also sampled portions of the flow diagrams that were not highlighted to ensure these components did not perform any of the intended functions associated with the scoping criteria of 10 CFR 54.4(a).

In a letter to the applicant dated May 5, 2000, the staff requested additional information regarding certain valves, piping, and strainers that were not highlighted as being within the scope of license renewal, and certain orifices that were identified as being within the scope of license renewal, but not subject to an AMR. In its response to the NRC dated August 30, 2000, the applicant confirms that valves, piping, and orifices identified by the staff were subject to an AMR. The strainers, more commonly known as traveling water screens, are within the scope of license renewal but are not subject to an AMR because they perform the intended function within the scope of license renewal using moving parts and change in configuration. During the scoping inspection, the inspection team evaluated the potential of the trash racks being within the scope of license renewal but determined that these components do not meet any of the scoping criteria in 10 CFR 54(a) and, therefore, are not in the scope of license renewal and not subject to an AMR.

2.3.3.10.3 Conclusions

The staff reviewed the information submitted by the applicant in the LRA, information in the ANO-1 FSAR, and the additional information provided by the applicant in the letter dated August 30, 2000. On the basis of the review described above, the staff finds that there is reasonable assurance that the applicant identified those portions of the service water system that are within the scope of license renewal, and the associated SCs that are subject to an AMR, in accordance with the requirements of 10 CFR 54.4(a) and 54.21(a)(1), respectively.

2.3.3.11 Penetration Room Ventilation

In the LRA, Section 2.3.3.11, “Penetration Room Ventilation,” the applicant identifies portions of the penetration room ventilation system (PRVS), and the components that are within the scope of license renewal and subject to an AMR. In this section of the LRA, the applicant also states that the PRVS is further described in Section 6.5 of the UFSAR.

The applicant evaluates component supports for equipment, piping, fire damper, and motor operated valves that are associated with the PRVS in Section 2.4.6.2 and Table 3.6-8 of the LRA. The applicant also evaluates electrical components that support the operation of the PRVS in Section 2.5 of the LRA. The staff evaluated component supports and electrical components in Sections 2.4.6.2 and 2.5 of this SER. The PRVS instrument lines are individually highlighted as being within the scope of license renewal on flow diagram, LRA-M-264 Rev. 0, Sheet 1. The applicant evaluates these instrument line components with the PRVS system in Section 2.3.3.11 of the LRA.

2.3.3.11.1 Technical Information in the Application

The PRVS schematic and characteristics are shown in the UFSAR, Figure 6-10. Penetration rooms are formed adjacent to the outside surface of the reactor building by enclosing the area around the majority of the penetrations. The only penetrations that do not pass through one of the penetration rooms are the two main steam lines, the permanent equipment hatch, the emergency personnel access lock, the refueling tube and purge lines.

The PRVS is made up of two trains, each train contains a filter assembly, fan, and duct work. Each filter assembly and each fan is designed to handle 2,000 scfm, the second filter assembly and second fan being a full-size standby (redundant) unit. Normal system flow is approximately 1800 cfm. Particulate filtration is achieved by a medium-efficiency prefilter and a high-efficiency particulate air (HEPA) filter. Adsorption filtration is accomplished by an activated charcoal filter. The design-basis requirement for each charcoal filter is to remove 25 percent of the core iodine inventory. The 25 percent core iodine inventory release was derived using the standard assumption that during a DBA, 50 percent of the halogens are released from the core and that 50 percent of the iodine released plates out within the reactor building.

Following a loss-of-coolant accident (LOCA), a reactor building isolation signal places the lead system in operation by starting the fan and opening the power-operated butterfly damper on the outlet of the filter assembly. If the lead system does not achieve proper flow within 20 seconds, the lead system is automatically stopped, and the standby system automatically starts 5 seconds later. In the event of an excessive pressure drop across any filter, or a high radiation reading on the filter assembly discharge, the standby system can be started remotely.

Penetration room vacuum is displayed in the control room, and low vacuum is annunciated. Fan operating status and the radiation level of filter effluent are displayed in the control room, and high radiation is annunciated. Filter differential pressure (?P) is displayed locally and annunciated in the control room. Filter high temperature is annunciated in the control room. The system flow rate is displayed adjacent to the remote control valve stations. The system may be manually actuated from the control room.

In Section 2.3.3.11 of the LRA and Section 6.5 of the UFSAR, the applicant identifies the following intended function and system functions, respectively, for the PRVS that relate to 10 CFR 54.4(a)(1) and 54.4(a)(2):

Section 2.3.3.11 of the LRA -

- The safety function of the PRVS is to collect and process the radioactivity released to the penetration areas due to post LOCA reactor building leakage to ensure that the 10 CFR Part 100 dose values are not exceeded. The intended function of the PRVS, which needs to be considered during the AMR, is to maintain the system pressure boundary integrity.

Section 6.5 of the UFSAR -

- Control and minimize the release of radioactive materials from the reactor building to the environment in post-accident conditions .

- Maintain a negative pressure in the penetration room (with respect to outside atmosphere and auxiliary building) to ensure that any leakage goes into the penetration room when the system is in (normal) operation (to prevent uncontrolled releases).

- Collect and process potential reactor building penetration leakage to minimize environmental activity levels resulting from post-accident reactor building leaks.

- Withstand a single failure without loss of function.

On the basis of the functions identified above, the applicant determined that all PRVS safety-related components (electrical, mechanical, and instrument) are within the scope of license renewal. The applicant described its process for identifying the mechanical components that are subject to an AMR in Section 2.5.2 of the LRA. The applicant uses this methodology, to identify the portions of the PRVS that are within the scope of license renewal, and that are highlighted on flow diagrams listed in Table 2.3-7 of the LRA. Using the methodology described in Section 2.2.1 of the LRA, the applicant lists the mechanical components and component types and the intended function(s) that are within the scope of license renewal and subject to an AMR. The applicant provides this list in Table 3.4-11 of the LRA.

Specifically, the applicant identifies the following 10 device types as being within the scope of license renewal and subject to an AMR: duct (carbon steel), dampers (carbon steel), valves (carbon steel), expansion joints (carbon steel), exhaust stack (carbon steel), exhaust stack screen (stainless steel), blowers (carbon steel), filters (carbon steel), flow element (stainless steel), and tubing (copper and brass).

In LRA Table 3.4-11, the applicant further notes that the PRVS pressure boundary is the only applicable intended function associated with components of the PRVS that are subject to an AMR.

2.3.3.11.2 Staff Evaluation

The NRC staff reviewed the above information to verify that the applicant identified the components of the PRVS that are within the scope of license renewal and subject to an AMR, in accordance with 10 CFR 54.4 and 54.21(a)(1). The staff also reviewed the information in the UFSAR, Section 6.5. After completing the initial review, the staff issued a RAI, by letter dated June 1, 2000, regarding the PRVS. The applicant responded to that RAI in a letter dated August 30, 2000.

In the LRA, Section 2.1, "Scoping and Screening Methodology," the applicant discusses the process for identifying mechanical components that are subject to an AMR. The NRC staff evaluated the applicant's scoping methodology in Section 2.1 of this SER, "Scoping and Screening Methodology."

In its review of the PRVS, the NRC staff reviewed the flow diagrams listed in LRA Table 2.3-7 (which show the evaluation boundaries for the highlighted portions of the PRVS that are within the scope of license renewal), and Table 3.4-11 (which lists those mechanical components and their intended functions that are subject to an AMR).

The NRC staff also reviewed the UFSAR, Section 6.5, to determine if there were any portions of the PRVS that met the scoping criteria in 10 CFR 54.4(a), but were not identified as being within the scope of license renewal. The staff also reviewed the UFSAR to determine if any system function was not identified as an intended function(s) in the LRA, and to determine if any SCs that have an intended function were omitted from the scope of SCs that are subject to an AMR. The staff also reviewed the PRVS flow diagrams identified in Table 2.3-7 of LRA to determine if any SCs that are within the evaluation boundaries were omitted from the scope of components that are subject to an AMR, in accordance with 10 CFR 54.21(a)(1). The NRC staff also compared the functions described in the UFSAR to those identified in the LRA, and then considered whether the applicant had properly identified the SCs that are subject to an AMR from among those identified as being within the scope of license renewal.

The applicant identifies the SCs that are subject to an AMR for the PRVS using the screening methodology described in Section 2.1 of the LRA, and lists them in Table 3.4-11 of the LRA. The NRC staff evaluated the scoping and screening methodology, and documented its findings in Section 2.1 of this SER. The NRC staff sampled the SCs from Table 3.4-11 to verify that the applicant accurately identified the SCs that are subject to an AMR. The staff also sampled the SCs that were within the scope of license renewal, but not subject to an AMR, to verify that these SCs perform their intended functions with moving parts or with a change in configuration or properties, and are subject to replacement based on qualified life or specified time period.

To help ensure that those portions of the PRVS that the applicant identifies as not being within the scope of license renewal do not perform any of the scoping functions in 10 CFR 54.4, the NRC staff requested additional information on the basis of the information in the UFSAR and the LRA. The NRC staff noted that the LRA, Section 2.3.3.11 presents a summary description of the system functions, Table 3.4-7 flow diagrams highlight the evaluation boundaries of the PRVS, and Table 3.4-11 tabulates the PRVS components that are within the scope of license renewal and subject to an AMR. The corresponding drawings for these systems in the UFSAR, however, show additional components that were not listed in Table 3.4-11 of the LRA.

The NRC staff requested specific information concerning the exclusion of the following components from the scope of license renewal and/or an AMR:

- the piping, piping reducers, piping rectangular to round transitions, plugged pitot tube connections

- bird screen or wire mesh for an exhaust stack

In a letter dated August 30, 2000, the applicant provides the following responses: the piping, piping reducers, piping rectangular to round transitions, and plugged pitot tube connections are included in the AMR in the component commodity grouping “duct” in Table 3.4-11 of the LRA; and new component commodity grouping category, “exhaust stack screen,” is added in Table 3.4-11 of the LRA for the “bird screen or wire mesh” which was excluded due to an administrative error.

On the basis of the additional information provided by the applicant, the staff did not identify any omission in the component commodity groupings that were included within the scope of PRVS components requiring an AMR.

The NRC staff also requested the following specific information on the following: does the "filter" commodity group include the filter housings, prefilters, absolute HEPA filters and charcoal absorbers (as shown in P&ID M-264); and does the "blower" commodity group include the exhaust fans (VEF-38A/B) and fan housings, as stated in the text of LRA, Section 2.3.3.11, and as shown in P&ID M-264, Sheet 1.

In a letter dated August 30, 2000, the applicant provides the following responses: the "filter" component commodity grouping listed in Table 3.4-11 of the LRA includes the housings for the prefilters, absolute (HEPA) filters; and the charcoal absorbers; prefilters, absolute (HEPA) filters, and charcoal absorbers are considered short-lived; in accordance with the NRC letter from C.I. Grimes to D.J. Walters, NEI, dated March 10, 2000, regarding License Renewal Issue No. 98-12, "Consumables." This NRC staff guidance states that system filters may be excluded, on a plant-specific basis, from an AMR under 10 CFR 54.21(a)(1)(ii); and because the performance and condition of these filters are periodically tested and the filters are replaced in accordance with ANO-1 TS 4.11, “Penetration Room Ventilation System Surveillance” they are not subject to an AMR.

On the basis of the additional information provided by the applicant, the NRC staff determined that the exclusion of the prefilters, absolute (HEPA) filters, and charcoal absorbers (except their filter housings) from the list of SCs subject to an AMR is consistent with the requirements of 10 CFR54.21(a)(1)(ii). The applicant also states that the "blower" component commodity grouping listed in Table 3.4-11 of the LRA includes the housings for the exhaust fans (VEF-38A/B), and these exhaust fans are active components and thus are not subject to an AMR. The NRC staff found the exclusion of the exhaust fans from the scope of license renewal to be acceptable because they do not meet the scoping criteria in 10 CFR 54.4(a).

Some components that are common to many systems, including the PRVS, have been separately evaluated in the LRA as commodity groups with similar components from other systems, and are evaluated by the NRC staff in other sections throughout this SER.

In Section 2.4 of the SER, the staff evaluated component supports for piping, cables, and equipment that are discussed in LRA Section 2.4 "Structures and Structural Component Scoping and Screening Results." In Section 2.5 of the SER, the staff evaluated electrical components that support the operation of the PRVS which are discussed in the LRA. Section 2.5 "Electrical and Instrumentation and Controls System Scoping and Screening Results." The PRVS instrumentation lines are evaluated with the PRVS, and are listed as “tubing” in LRA Table 3.4-11, of the LRA.

The NRC staff reviewed Exhibit A of the LRA, supporting information in the UFSAR, and the applicant's responses to the staff's RAIs. In addition, the NRC staff sampled several components in the PRVS flow diagrams (Table 2.3-7 of LRA) to determine whether the applicant properly identifies the components that are within the scope of license renewal and subject to an AMR. No omissions were identified.

2.3.3.11.3 Conclusions

On the basis of this review, the staff finds that there is reasonable assurance that the applicant has adequately identified those portions of the PRVS components that are within the scope of license renewal, and the associated SCs that are subject to an AMR, in accordance with the requirements of 10 CFR 54.4(a) and 54.21(a)(1), respectively.

2.3.3.12 Auxiliary Building Heating and Ventilation

In the LRA, Section 2.3.3.12, “Auxiliary Building Heating and Ventilation,” the applicant identifies the portions of the auxiliary building heating and ventilation system (ABHVS) and its components that are within the scope of license renewal and subject to an AMR. In this section of the LRA, the applicant states that the ABHVS is further described in Section 9.7.2.1 of the UFSAR.

The applicant evaluates component supports for equipment, piping, fire damper, and motor-operated valves within Section 2.4.6.2 and Table 3.6-8 of the LRA. The applicant evaluates electrical components that support the operation of the ABHVS in Section 2.5 of the LRA. The NRC staff evaluated component supports and electrical components in Sections 2.4.6.2 and 2.5 of this SER, respectively. The ABHVS instrument lines are individually highlighted as being within the scope of license renewal on flow diagrams LRA-M-262, Revision 0, Sheets 3 and 4, LRA-M-263, Revision 0, Sheets 2 and 3. The applicant evaluated these instrument line components with the ABHVS in Section 2.3.3.12 of the LRA.

2.3.3.12.1 Technical Information in the Application

The portions of the ABHVS that are seismic Category 1, include the decay heat removal rooms unit coolers, makeup pump rooms unit coolers, switchgear rooms unit coolers, EDG rooms ventilation system, and auxiliary building electrical rooms unit coolers. Seismic Category 1 components are designed to ensure that this equipment is capable of performing its safety-related function(s) during and following a design-basis earthquake.

Each decay heat removal engineered safeguard room is cooled by two 100-percent capacity fan coil unit coolers (one redundant) installed in each of the three interconnected makeup pump rooms. The makeup pumps remain operable without any of the unit coolers in operation. The unit coolers use service water and, therefore, the loss of the cooling system is not anticipated. The coolers are designed to maintain room temperature below 110 °F dry bulb (DB) under normal conditions. The unit coolers are automatically energized when decay heat removal equipment is in operation. Service water is supplied to these unit coolers on a continuous basis. Purge systems are available for each room, and are used when access to any room is needed. The air is discharged through the radwaste exhaust system.

Each engineered safeguard switchgear room is normally cooled by air circulated through fan coil unit coolers supplied with chilled water from the control room chillers. Emergency cooling is provided by separate fan coil unit coolers supplied by emergency chillers VCH 4A and 4B. The north vital electrical equipment room is normally cooled by the radwaste auxiliary building ventilation system and the south vital electrical equipment room is normally cooled by an air-cooled refrigeration unit. Emergency cooling for both vital electrical equipment rooms is provided by fan coil unit coolers supplied by emergency chilled water.

Each EDG room is ventilated by two exhaust fan units. Makeup air to these rooms is 100 percent outside air. The capacity of each fan is predicated on the equipment heat gains within the respective room, and the capability to maintain ambient room temperature at or below 110°F DB, with two fans running. In the event that one fan fails, the room temperature is maintained by the other fan at a temperature not to exceed 120°F DB. The emergency diesel electrical equipment and controls are derated to operate at 100 percent of the diesel load, without being affected by an ambient design temperature of 120°F.

The auxiliary building ventilation systems that are seismic Category 2, include the rad waste area ventilation system, reactor building penetration rooms normal ventilation system (reactor building ventilation system is described in UFSAR Chapter 5 and shown on UFSAR Figure 5-7), fuel handling floor area ventilation system, battery room ventilation system, cable spreading room cooling system, relay room cooling system, heating and ventilating equipment rooms ventilation system, and boiler room exhaust fans.

The auxiliary building ventilation systems primarily uses outside air. The auxiliary building is served by separate ventilation systems for the fuel handling area, the radwaste area, and the non-radioactive area. These systems are shown in the UFSAR, Figure 9-13. The radwaste area ventilation system and fuel handling area ventilation system are not required to meet the single-failure criterion. One supply unit serves the fuel handling area, and the second supply unit serves the radwaste area. The ventilation air from these areas is discharged to the reactor building flutes (plant vents) through multiple filter units. The ventilation air from the fuel handling and radwaste areas are continuously discharged through the exhaust filters. The reactor building penetration rooms emergency exhaust air discharges into the atmosphere through an exhaust stack. The penetration rooms, fuel handling, rad waste, and reactor building purge exhaust lines are monitored separately for radiation by an isokinetic sample taken downstream of each filtering unit. These samples feed a common station vent radiation monitoring system.

The redundant battery rooms have independent exhaust fans (VEF 33, VEF 34). Both rooms are normally cooled by air-cooled refrigeration units. Emergency cooling is provided by emergency cooling units (VUC 14A & C) that are cooled by VCH-4A or 4B.

During operation, the cable spreading room is cooled by a recirculation unit located outside the room. The relay room has two recirculation type cooling units, one of which is a standby unit. The rooms are designed for an ambient temperature of 85 °F during normal operation. Also, a small portion of air is supplied to the relay room from Unit-2 supply fan 2VSF-6 for pressurization of the relay and cable spreading room to prevent any in-leakage from the turbine building. The air leaks to the turbine building area and is not recirculated through the system.

In the LRA, Section 2.3.3.12, and Section 9.7.1 of the UFSAR, the applicant identifies the following intended functions for the ABHVS, consistent with 10 CFR 54.4 and 54.21(a)(1):

Section 2.3.3.12 of the LRA -

- provide a suitable environment for those areas of the auxiliary building that contain equipment requiring post accident cooling

- close some of the fire dampers in the ABHVS during the unlikely event of a fire to meet the requirements of 10 CFR 50.48

- maintain the system pressure boundary integrity for the decay heat removal rooms unit coolers, makeup pump rooms unit coolers, and switchgear rooms unit coolers is the intended function that needs to be considered during the AMR

- maintain the system heat transfer integrity for the decay heat removal rooms unit coolers and switchgear rooms unit coolers is the intended function that needs to be considered during the AMR

Section 9.7.1 of the UFSAR -

- provide a suitable environment for equipment and personnel

- provide maximum safety and convenience for operating personnel, with equipment arranged in zones so that potentially contaminated areas are separated from clean areas to inhibit the spread of any radioactive contamination

- direct the flow path of the ventilation air in the auxiliary building from clean or low-activity areas toward areas of higher activity

- direct all exhaust air from the auxiliary building to the reactor building flutes (plant vents), and to monitor each exhaust separately for radiation using an isokinetic sampling which in turn feeds a common station vent radiation monitoring system.

- Maintain temperature limits in the rad waste and fuel handling areas to 105°F and 109°F, respectively (during summer), and 60 °F during the winter

On the basis of the intended functions identified above, the portions of the ABHVS that are identified by the applicant as being within the scope of license renewal include all ABHVS safety-related components (electrical, mechanical, and instrument). The applicant describes its methodology for identifying the mechanical components that are subject to an AMR in Section 2.2.1 of the LRA. On the basis of this methodology, the applicant identifies the portions of the ABHVS that are within the scope of license renewal on the flow diagrams listed in Table 2.3-7 of the LRA. Using the methodology described in Section 2.2.1 of the LRA, the applicant lists the mechanical components and component types subject to an AMR that are within the evaluation boundaries highlighted on the flow diagrams, and identified their intended functions. The applicant provides this list in Table 3.4-12 of the LRA.

The following nine device types are identified as being within the scope of license renewal and subject to an AMR: exterior duct (carbon steel), louvers (carbon steel), fans (carbon steel), ductwork (carbon steel), dampers (carbon steel), heat exchangers (carbon steel), tubing for heat exchangers (copper), tubing for heat exchangers (90/10 CuNi), and sealants.

The applicant further noted in Table 3.4-12 that the ABHVS pressure boundary and heat transfer functions are the only applicable intended functions of ABHVS components that are subject to an AMR.

2.3.3.12.2 Staff Evaluation

The NRC staff reviewed the above information to verify that the applicant identified the components of the ABHVS that are within the scope of license renewal and subject to an AMR, in accordance with 10 CFR 54.4 and 54.21(a)(1). The staff also reviewed the information in the UFSAR, Section 9.7.1. After completing the initial review, the NRC staff issued RAIs by letter dated June 1, 2000 regarding the ABHVS. The applicant provides its response to staff the RAIs in letter dated August 30, 2000.

In the LRA, Section 2.1, "Scoping and Screening Methodology," the applicant discusses the process for identifying mechanical components that are subject to an AMR. The NRC staff evaluated the applicant's methodology in Section 2.1 of this SER, "Scoping and Screening Methodology."

In its review of the ABHVS, the NRC staff reviewed the flow diagrams listed in LRA Table 2.3-7 (which show the evaluation boundaries for the highlighted portions of the ABHVS that are within the scope of license renewal), and Table 3.4-12 (which lists the mechanical components and the applicable intended functions that are subject to an AMR).

The NRC staff also reviewed the UFSAR, Section 9.7, to determine if there were any portions of the ABHVS that met the scoping criteria in 10 CFR 54.4(a), but were not identified as being within the scope of license renewal. The staff also reviewed the UFSAR to determine if there were any safety-related system function(s) that were not identified as intended function(s) in the LRA, and to determine if there were any SCs that have intended function(s) that might have been omitted from the scope of SCs that are subject to an AMR. The staff also reviewed the system flow diagrams identified in Table 2.3-7 of the LRA to determine if any SCs that are within the evaluation boundaries were omitted from the scope of components that are subject to an AMR in accordance with 10 CFR 54.21(a)(1). The NRC staff compared the functions described in the UFSAR to those identified in the LRA. The NRC staff then determined whether the applicant had properly identified the SCs that are subject to an AMR from among those identified as being within the scope of license renewal.

The applicant identifies the SCs that are subject to an AMR for the ABHVS using the screening methodology described in Section 2.1 of the LRA, and lists them in Table 3.4-12 of the LRA. The NRC staff evaluated the scoping and screening methodology, and documented its findings in Section 2.1 of this SER. The NRC staff sampled the SCs listed in Table 3.4-12 of the LRA to verify that the applicant accurately identified the SCs that are subject to an AMR. The staff also sampled the SCs that the applicant identified as being within the scope of license renewal, but not subject to an AMR, to verify that these SCs perform their intended functions with moving parts or with a change in configuration or properties, and are subject to replacement on the basis of a qualified life or specified time period.

To help ensure that those portions of the ABHVS that the applicant identifies as not being within the scope of license renewal do not perform any intended functions, the NRC staff requested additional information on the basis of the information in the UFSAR and the LRA. The staff noted that in the LRA, Section 2.3.3.12, the applicant presents a summary description of the system functions, Table 3.4-7 flow diagrams highlight the evaluation boundaries of the ABHVS, and Table 3.4-12 tabulates the components that are within the scope of license renewal and subject to an AMR for the ABHVS. However, the corresponding drawings for these systems in the UFSAR, show additional components that were not listed in Table 3.4-12 of the LRA.

The NRC staff requested specific information concerning the exclusion of the following components from the scope of license renewal and/or an AMR:

- damper bodies, blower housings, and cooler housings
- auxiliary building electrical room unit coolers

- fan coil unit housings for fan coil units VUC-1A/B/C/D
- fan coil unit housings for fan coil units VUC-14A/C/D and VUC-2D/B
- diesel generator exhaust fan housings for exhaust fans VEF-24A/B/C/D
- fan coil housing for fan coil unit VUC-14B
- valve bodies for solenoid valves 2100-2108 and 2111-2116
- control valves 7621, 7622, 7635, 7636, and 7638
- fire dampers
- sealant materials

In a letter dated August 30, 2000, the applicant provided the following responses:

- The damper bodies are included in the component commodity grouping “dampers”; cooler housings are included in the component commodity grouping category “heat exchangers.”
- The auxiliary building electrical room unit coolers are included in the component commodity grouping “heat exchangers” (switch gear room coolers and auxiliary building electrical room coolers).
- The fan coil housings for fan coil units VUC-1A/B/C/D are included in the component commodity grouping “heat exchangers” (decay heat room coolers).
- The fan coil housings for fan coil units VUC-14A/C/D and VUC-2D/B are included in the component commodity grouping “heat exchangers” (switch gear room coolers).
- The diesel generator exhaust fan housings for exhaust fans VEF-24A/B/C/D were included in an AMR and are included in the component commodity grouping “fans.
- The fan coil housings for fan coil units VUC-14A/B/C/D are included in the component commodity grouping “heat exchangers” (auxiliary building electrical room coolers).
- The valve bodies for solenoid valves 2100-2108 and 2111-2116 are part of the instrument air system and included in the component commodity grouping “valves” in Table 3.4-8 of the LRA and are subject to an AMR.
- The valve bodies for control valves 7635 and 7636 are part of the instrument air system, and are included in the component commodity grouping “valves” in Table 3.4-8 of the LRA and are subject to an AMR. Control valves 7621, 7622, and 7638 are dampers in the ABHVS, and the bodies of these dampers are included in the component commodity grouping “dampers” in Table 3.4-8 of the LRA and subject to an AMR.

- 10 CFR 50.48 fire dampers are within the scope of license renewal and the passive portions of the fire dampers (i.e., the mountings) were included in the AMR in the structural portions of the LRA, Section 3.6.1 and Table 3.6-8 except in the cases where fire dampers formed part of a pressure boundary.

- Sealant materials are within the scope of license renewal when they are part of components or commodities that are within the scope of license renewal and when they are important in maintaining the integrity of the component or commodity. Preventive maintenance activities are credited for managing sealant aging effects, as identified in Table 3.4-12 of the LRA. The component commodity groupings for the ABHVS are included in Table 3.4-12 of the LRA except those groupings identified in Table 3.4-8 of the LRA.

The applicant further clarifies that the fan coil units VUC-1A/B/C/D, VUC-14A/B/C/D, and VUC-2D/B, and the filters associated with those units are considered short-lived, as discussed in an NRC letter from C.I. Grimes to D.J. Walters, NEI, dated March 10, 2000, regarding License Renewal Issue No. 98-12, "Consumables." This NRC staff guidance states that the screening process allows the exclusion of component filters because they are inspected and replaced during preventive maintenance activities, and therefore, these filters are not subject to an AMR.

The staff disagreed with this statement. The guidance in the March 10, 2000, letter on consumables required the applicant to identify any SC that is excluded under 10 CFR 54.21 (a)(1)(ii) based on performance or condition monitoring, and that an applicant must provide a site-specific evaluation to justify the exclusion of any structure or component based on performance or condition monitoring. However, the applicant has provided sufficient additional information such that the NRC staff found the exclusion of the filters that are categorized as "Consumables" from the list of SCs that are subject to an AMR consistent with the requirements of 10 CFR 54.21(a) (1)(ii).

In addition, the NRC staff requested specific information regarding the exclusion of the fuel handling floor exhaust filtration system components (exhaust fans, exhaust filters, flow element, control valves, associated ductwork, and flue) from the scope of license renewal and/or an AMR. In a letter dated August 30, 2000, the applicant clarified that the fuel handling ventilation system is not within the scope of license renewal, since it is not safety-related; its failure would not prevent the satisfactory accomplishment of a safety-related function; and it is not relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the NRC regulations for fire protection, environmental qualification, pressurized thermal shock, anticipated transient without scram, or station blackout. The applicant also stated that this system is not needed to mitigate the consequences of the fuel handling accident (FHA) on the basis of the following discussion. As stated in ANO-1 SAR Section 14.2.2.3.2, the criterion for reactor protection for an FHA is that the resultant doses from such an incident shall not exceed 25 percent of the 10 CFR 100 limits. The 10 CFR 100 limits at the exclusion boundary are 25 rem to the whole body and 300 rem to the thyroid, and the numerical criteria for an FHA are 25 percent of these values (based on the ANO-1 SAR criterion above), or 6.25 rem to the whole body and 75 rem to the thyroid. The UFSAR, Table 14-25, shows that an FHA with an unfiltered release would only result in a dose of 0.27 rem to the whole body and 63.599 rem to the thyroid at the exclusion boundary. Since these doses are below the criteria, having a filtered ventilation path for an FHA is not necessary. Thus, operation of the fuel handling ventilation system is not required to meet the ANO-1 UFSAR criterion for an FHA, and the system does not meet the 10 CFR 54.4(a) criteria for inclusion within the scope of license renewal. On this basis, the NRC staff has no objections to the exclusion of the fuel handling floor exhaust filtration system components from the scope of license renewal.

The NRC staff also requested more specific information on the exclusion of the following components from the scope of license renewal and/or an AMR:

- air bottles (VRA 2 through VRA 8)
- exhaust ductwork
- fan coil units (VUC-2A/C, VUC-3, VUC 4 A/B, VUC-11, VUC-13A/B, VE-1A/B), and VUE-32 through VUE-35
- exhaust filtration units (VEF-8A/B, VEF-33, and VEF-34)
- vent (VPH-6) with associated ductwork
- fire dampers
- flow element (FE8001)
- valve bodies for control valves (CV 7603 and CV 7604)

In a letter dated August 30, 2000, the applicant states that the air bottles, fan coil units, exhaust filtration units, vent and associated ductwork, flow element, and valve bodies for control valves are not within the scope of license renewal because they do not meet the scoping criteria in 10 CFR 54.4(a). Specifically, these units are not safety-related; the failure of any of these components would not prevent the satisfactory accomplishment of a safety-related function; and they are not relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the NRC regulations for fire protection, environmental qualification, pressurized thermal shock, anticipated transients without scram, and station blackout. On that basis, the NRC staff found the exclusion of the above referenced air bottles, fan coil units, exhaust filtration units, vent with associated ductwork, flow element, and valve bodies from the scope of license renewal to be acceptable.

The applicant also stated that the exhaust ductworks for the north and south electrical penetration rooms (two zones each), the north and south piping penetration rooms, and the electrical equipment room, are subject to an AMR, and are included under the “duct” component commodity grouping “duct” in Table 3.4-11 of the ANO-1 LRA. The absence of highlighting on Drawings LRA-M-262, Sheets 1 and 2, to indicate that these exhaust ductworks (up to the associated isolation dampers) are within the scope of the license renewal was an administrative error.

The applicant also informed the NRC staff that the “fire dampers” in question do not form part of a pressure boundary of a system that is within the scope of license renewal, and are treated generically in the structural portions of the LRA, Section 3.6.1 and Table 3.6-8. On the basis of the above information, the NRC staff found the exclusion of the fire dampers in question from the scope of license renewal to be acceptable because they do not meet the scoping criteria in 10 CFR 54.4(a).

Some components that are common to many systems, including the ABHVS, have been separately evaluated in the LRA as commodity groups with similar components from other systems, and are evaluated by the NRC staff in other sections throughout this SER. In Section 2.4 of the SER the staff evaluated component supports for piping, cables, and equipment, that are discussed in LRA Section 2.4 "Structures and Structural Component Scoping and Screening Results." In Section 2.5 of the SER, the staff evaluated electrical components that support the operation of the ABHVS; these components are discussed in the LRA Section 2.5, "Electrical and Instrumentation and Controls System Scoping and Screening Results." The ABHVS instrumentation lines are evaluated with the ABHVS, and are listed as “tubing” in Table 3.4-12 of the LRA.

The NRC staff reviewed the LRA, supporting information in the UFSAR, and the applicant's responses to the staff's RAI. In addition, the NRC staff sampled several components from the ABHVS flow diagrams (Table 2.3-7 of LRA) to determine whether the applicant properly identified the components that are within the scope of license renewal and subject to an AMR. No omissions were identified.

2.3.3.12.3 Conclusions

On the basis of this review, the staff finds that there is reasonable assurance that the applicant has adequately identified those portions of the ABHVS that are within the scope of license renewal, and the associated SCs that are subject to an AMR, in accordance with the requirements of 10 CFR 54.4(a) and 54.21(a)(1), respectively.

2.3.3.13 Control Room Ventilation

In the LRA, Section 2.3.3.13, “Control Room Ventilation,” the applicant identifies portions of the control room ventilation system (CRVS), and its components that are within the scope of license renewal and subject to an AMR. The applicant states in Section 2.3.3.13 of the LRA that the CRVS is further described in Section 9.7 of the UFSAR.

The applicant also evaluates component supports for equipment, piping, fire damper, and motor operated valves within this system in Section 2.4.6.2 and Table 3.6-8 of the LRA. The applicant evaluated electrical components that support the operation of the CRVS in Section 2.5 of the LRA. The NRC staff evaluated component supports and electrical components in Sections 2.4.6.2 and 2.5 of this SER. The CRVS instrument lines are individually highlighted as being within the scope of license renewal on flow diagrams LRA-M-2221, Rev. 0, Sheet 2, and LRA-M-263, Rev. 0, Sheet 1. The applicant evaluated these instrument line components with the CRVS system in Section 2.3.3.13 of the LRA.

2.3.3.13.1 Technical Information in the Application

Normal Ventilation System

The normal control room ventilation system serves the control room and computer room only. The control room and computer room are normally air conditioned by two 100-percent capacity air conditioning units that receive chilled water from two 100-percent capacity chillers. One air conditioning unit is normally running, with the other in standby status, and isolated from the system by shutoff dampers. The standby unit is available for manual actuation in the event of failure of the operating unit. Fan failure is monitored at the unit by a flow switch with an indicating light in the control room. The CRVS is designed to maintain ambient room temperature at 75°F DB and 43 percent relative humidity given the space load, lighting load, and equipment load. Computer equipment cooling is achieved by two-out-of-three packaged air conditioning units (one standby), which are located in the computer room and circulate air through a false floor.

Control Room Isolation System

The control room air is continuously monitored, and alarmed for high radiation. The control room inlet air radiation monitor consists of an auto ranging digital rate-meter, pre-amplifier, and Beta-Gamma sensitive scintillation detector. Redundant quick-acting chlorine detectors are presently in place in the control room fresh air inlets, which initiates closure of the isolation dampers if chlorine (Cl_2) levels exceed 5 ppm in the incoming air.

Cl_2 detection system design features are consistent with the recommendations of Regulatory Guide 1.95, “Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release,” February 1975. However, since elemental Cl_2 is no longer stored or used on site, or within a 5 mile radius of the plant site, the regulatory guide recommendation for seismic Category 1 designation is not necessary at ANO-1. A postulated seismic event concurrent with transport failure and offsite release of Cl_2 or other toxic gas is not considered a credible event.

In the event of high radiation or Cl_2 levels, the normal air conditioning system is automatically de-energized, and the normal control room ventilation system is completely isolated from both the outside air and the rest of the building within 5 seconds after detection. The actuation level for high radiation is sufficiently below hazardous radiation levels to minimize operator doses during an accident, and is sufficiently above normally experienced background levels to minimize spurious actuations. The control room isolation dampers in the supply and return ductwork are spring loaded, such that they fail closed upon loss of air or power. The single supply and single return isolation dampers are both actuated by either one of two solenoid valves.

To ensure initiation of control room emergency air filtration within 10 seconds after a step increase in Cl_2 concentration that results in a control room isolation, the HVAC isolation damper is interlocked with the control room emergency air filtration system. Under these conditions, control room air is recirculated by the emergency air filtering system. The emergency air filtering system consists of two redundant filter trains. One filter train consists of a fan, roughing filters, HEPA filters, and a 4-inch deep-bed charcoal absorber rated for 2,000 cfm. The other train consists of a fan, one filter unit assembly rated for 2,000 cfm with an outside air filter unit rated for 333 cfm, each with the necessary roughing filters, HEPA filter, and 2-inch charcoal tray absorber. For either train, 333 cfm of outside air, that is used for pressurization, will be filtered through 4 inches of charcoal absorber, and the recirculating air will go through at least 2 inches of charcoal bed. Fan flow is monitored by a flow switch with an indicating light in the control room. On an indication of low flow, the standby unit will be manually started.

The safety-related high-efficiency particulate filters and charcoal filters are tested in accordance with the guidance in Regulatory Guide 1.52, as specified in ANO-1 TS 4.10.

Emergency Ventilation System

The original DBA maximum emergency cooling pond (ECP) temperature of 129.5 °F discussed in the UFSAR, Section 9.3.2.4, exceeded earlier estimates of past ECP temperature. This change required the replacement of the Unit 2 emergency air conditioning units with larger units capable of operating with 129.5 °F cooling water. The new control room emergency air conditioning units are located in the Unit 2 control room, where they provide emergency air conditioning to both Unit 1 and Unit 2 control rooms and provide for air mixing during a control room isolation condition. Seismically supported ductwork has been added for air distribution to the Unit 1 control room.

In conjunction with installation of new control room emergency air conditioning units, a cross-connect between the Unit 1 service water system (Loop 2), and the air conditioning condensing units, was added to provide an alternate source of service water to the emergency air conditioning system.

The worst-case outside environment assumed for this analysis is a maximum of 103 °F DB, 83 °F wet bulb (WB), and 43 percent relative humidity. These environmental conditions are predicated on records of ambient conditions at the site. The capacity of the control room emergency recirculation system is based on a minimum of three room air changes per hour for the combined control room volume. The filter banks are sized in accordance with the manufacturer's recommendations for maximum efficiency. The control room operator has manual control for selecting fan, filter, and air conditioning unit operations, in order to ensure satisfactory control room conditions following an accident. Self-contained breathing apparatus (SCBA) are available for use following a toxic gas release.

All portions of the reactor protection and engineered safeguards actuation systems located in the control room are designed to operate at the ambient conditions of 110 °F and 80-percent-relative humidity.

Fire or smoke in the control room could be visually detected by the operator. The valves that isolate the control room from the other areas close in 5 seconds. This prevents significant quantities of smoke from entering the control room from the outside. In the unlikely event of a fire in the control room, smoke is exhausted outside of the building, and makeup air is supplied to the room by the normal ventilation system. The system is sized such that it provides 15 air-changes per hour. This ventilation rate rapidly dissipates any smoke generated or admitted to the control room. A failure analysis was performed to demonstrate the ability of the control room emergency air conditioning system to meet single-failure criterion. The analysis is documented in the UFSAR Table 9-20.

In Section 2.3.3.13 of the LRA and Section 9.7 of the UFSAR, the applicant identifies the following intended functions for the CRVS, consistent with 10 CFR 54.4(a)(1) and 54.4(a):

Section 2.3.3.13 of the LRA -

C Isolate the control room under accident conditions.

C Provide a suitable environment for the control room operators and for equipment that requires post-accident cooling.

C Credit the fire dampers and temperature elements on the charcoal filters in the CRVS to meet the requirements of 10 CFR 50.48.

C Remove smoke from the control room during and after a fire.

Section 9.7 of the UFSAR -

C Provide a suitable environment for equipment and personnel.

C Provide maximum safety and convenience for operating personnel with equipment arranged in zones, so that potentially contaminated areas are separated from clean areas to inhibit the spread of any radioactive contamination.

C Isolate the control room on detection of high radiation or high Cl_2 (5 ppm level) in fresh air supply inlets.

C Withstand a single failure without loss of function for the control room emergency air conditioning system (failure analysis is shown in Table 9-20 of the UFSAR).

C Maintain ambient conditions of 110 °F and 80 percent relative-humidity inside the control room to protect the portions of the reactor protection and engineered safeguards actuation systems that are located in the control room.

On the basis of the above intended functions, the portions of the CRVS system that were identified by the applicant as being within the scope of license renewal include all CRVS safety-related electrical, mechanical, and instrument components. The applicant describes its methodology for identifying the mechanical components that are subject to an AMR in Section 2.5.2 of the LRA. On the basis of that methodology, the applicant identified the portions of the CRVS that are within the scope of license renewal on the highlighted flow diagrams listed in Table 2.3-7 of the LRA. Using the methodology described in the LRA, Section 2.2.1, the applicant compiles a list of the mechanical components and component types that are within the license renewal evaluation boundaries, and that are subject to an AMR in Table 3.4-13 of the LRA. This table also contains the intended functions identified by the applicant.

Specifically, the applicant identifies 12 device types as being within the scope of license renewal and subject to an AMR: duct work (carbon steel), dampers (carbon steel), heat exchangers (carbon steel), fans (carbon steel), filters (carbon steel), tubing (copper, brass and admiralty), valves (carbon steel), evaporators (carbon steel), evaporator tubing (copper), condenser (carbon steel), condenser tubing (90/10 CuNi), and compressor (carbon steel).

In the LRA, Table 3.4-13, the applicant also notes that maintaining pressure boundary and heat transfer are the only applicable intended functions associated with the components of the CRVS that are subject to an AMR.

2.3.3.13.2 Staff Evaluation

The NRC staff reviewed the above information to verify that the applicant identified the components of the CRVS that are within the scope of license renewal and subject to an AMR, in accordance with 10 CFR 54.4 and 54.21(a)(1). The staff also reviewed the information in the UFSAR, Section 9.7.1. After completing the initial review, the NRC staff issued RAIs by letter dated June 1, 2000, regarding the CRVS. The applicant responded to that RAI in a letter dated August 30, 2000.

In the LRA, Section 2.1, "Scoping and Screening Methodology," the applicant discusses the process for identifying mechanical components subject to an AMR. The applicant's scoping methodology is evaluated in Section 2.1 of this SER, "Scoping and Screening Methodology."

In its review of the CRVS, the NRC staff reviewed the flow diagrams listed in Table 2.3-7 (which show the evaluation boundaries for the highlighted portions of the CRVS that are within the scope of license renewal) and Table 3.4-13 (which lists the mechanical components and their intended functions that are subject to an AMR).

The NRC staff also reviewed the UFSAR, Section 9.7 to determine if there were any portions of the CRVS that met the scoping criteria in 10 CFR 54.4(a) that the applicant did not identify as being within the scope of license renewal. The staff also reviewed the UFSAR to determine if there were any system function(s) that were not identified as intended function(s) in the LRA, and to determine if there were any SCs that have intended function(s) that might have been omitted from the scope of SCs that are subject to an AMR. The staff also reviewed the CRVS flow diagrams identified in Table 2.3-7 of the LRA to determine if any SCs that are within the evaluation boundaries were omitted from the scope of components that are subject to an AMR, in accordance with 10 CFR 54.21(a)(1). The NRC staff compared the functions described in the UFSAR to those identified in the LRA. The staff then determined whether the applicant had properly identified the SCs that are subject to an AMR from among those identified as being within the scope of license renewal.

The applicant identifies and lists the SCs that are subject to an AMR for the CRVS in Table 3.4-13 of the LRA using the screening methodology described in Section 2.1 of Exhibit A of the LRA. The NRC staff evaluated the scoping and screening methodology, and documented its findings in Section 2.1 of this SER. The NRC staff sampled the SCs listed in Table 3.4-13 to verify that the applicant accurately identified the SCs that are subject to an AMR. The staff also sampled the SCs that were within the scope of license renewal, but not subject to an AMR to verify that these SCs perform their intended functions with moving parts or with a change in configuration or properties, or are subject to replacement based on qualified life or specified time period.

To help ensure that those portions of the CRVS identified as not being within the scope of license renewal do not perform any applicable intended function, the NRC staff requested additional information. The NRC staff noted that the LRA, Section 2.3.3.13, presents a summary description of the system functions, Table 2.3-7 flow diagrams highlight the evaluation boundaries of the CRVS, and Table 3.4-13 tabulates the CRVS components that are within the scope of license renewal and subject to an AMR. However, the corresponding drawings for these systems in the UFSAR, show additional components that were not listed in Table 3.4-13 of the LRA.

The NRC staff requested more specific information concerning the exclusion of the following components and component groupings from the scope of license renewal:

- damper bodies, blower housings, and cooler housings
- control room emergency unit coolers
- electrical equipment room 2150 emergency cooling units
- filtration unit housings for emergency filter and fan units (VSF-9 and 2VSF-9)
- valve bodies for control valves (CV-7905, CV-7907, and CV-7910)
- air-operated dampers and operators (2PCD-8605, 2PCD-8607, 2UCD-8609, and 2UCD-8683)
- air handling unit housings, and heating and cooling coils for 2VUC-27A/B
- sealants
- radiation monitors and Cl_2 detectors

In a letter dated August 30, 2000, the applicant provided the following responses:

- The carbon steel damper bodies are included in the “dampers” component commodity group, the blower housings are included in the “fans” component commodity group, and the cooler housings are included in the “heat exchangers (evaporators), component commodity group with a "pressure boundary intended function,” and are subject to an AMR. These components are listed in Table 3.4-13 of the LRA. Although aluminum damper bodies are not listed in Table 3.4-13 due to an administrative error, they are also subject to an AMR.
- The control room emergency unit coolers are included in the "heat exchangers (evaporators)" component commodity grouping in Table 3.4-13 of the LRA, and are subject to an AMR.
- Copper tubes in the electrical equipment room 2150 emergency cooling units that maintain the Freon pressure boundary are included in the "heat exchangers (evaporators)" component commodity grouping in Table 3.4-13 of the LRA, and are subject to an AMR.

- Housings for the emergency filter and fan units VSF-9 and 2VSF-9 and the outside air emergency filter unit are shown in the fans and "filters" component commodity group in Table 3.4-13 of the LRA, and are subject to an AMR.

- Valve bodies for control valves (dampers) CV-7905, CV-7907, and CV-7910 are included in Table 3.4-13 of the LRA under the "dampers" component commodity group and are subject to an AMR.

- Bodies for air-operated dampers 2PCD-8607A/B, 2UCD-8609, 2PCD-8685, and 2UCD-8683 are included in the component "dampers" commodity grouping for damper bodies made of aluminum and carbon steel (similar to item 1 above) in Table 3.4-13 of the LRA, and are subject to an AMR. Exceptions to this grouping include the dampers and their operators, which are considered to be active components, and are not subject to an AMR.

- Housings and cooling coils for the fan and cooling units 2 VUC-27A/B (which each contains a filter, cooler, and fan) are included in Table 3.4-13 of the LRA in the "heat exchangers (evaporators)" component commodity grouping with a pressure boundary intended function. (The coolers contained in these units are also evaluated in the table with regard to their heat transfer function), and are subject to an AMR.

- Sealant materials used to maintain positive pressure in the control room are within the scope of license renewal and are subject to an AMR.

- Supply vent radiation detectors 2RE8001A/B and 2RE-8750-1A/1B for the Unit 1 and 2 main control room (MCR), respectively, isolate the control rooms on high inlet air radiation, are safety-related, and are within the scope of license renewal. However, these detectors are active components, and thus are not subject to an AMR except for the passive, long-lived electrical, and instrumentation and controls components associated with these detectors which are evaluated in Sections 2.5 and 3.7 of this report. The area radiation monitor (RE-8001) inside the Unit 1 MCR and Cl_2 detectors (2CLS-8760-2, 2CLS-8761-1, 2CLS-8762-2, and 2CLS-8763-1) do not meet the scoping criteria of 10 CFR 54.4(a), and are not within the scope of license renewal. The applicant explained that these monitors are not safety-related; do not prevent the satisfactory accomplishment of a safety-related function if they were to fail; and are not relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the NRC regulations for fire protection, environmental qualification, pressurized thermal shock, anticipated transients without scram, and station blackout.

- As stated in UFSAR Section 9.7.2.1 5, the Cl_2 detection system design features are consistent with the recommendations of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Cl_2 Release," February 1975. However, elemental Cl_2 is no longer stored or used on site, or within a 5-mile radius of the plant site; therefore, the regulatory guide recommendation for Seismic Category I designation is not necessary for ANO-1. In addition, a postulated seismic event concurrent with transport failure and offsite release of Cl_2 or other toxic gas is not considered a credible event.

The staff also requested a verification that the CRVS components (including air handling units and fan coil units with their associated ductwork, fire damper and control valves, air intake, and exhaust fan with purge ductwork) inside the main control room environment (MCRE) are within the scope of license renewal and subject to an AMR, or for the applicant to provide a justification for excluding these components from the scope of license renewal and an AMR. In a letter dated August 30, 2000, the applicant responded that the ANO-1 MCRE includes the auxiliary building walls, floor, ceiling, and doors that encompass the control room; piping penetrations; fire dampers; and the safety-related components in the control room ventilation system. The walls, floor, ceiling, and doors of the control room and the piping penetrations were included in the structural review (see Section 2.4.3 and 3.6 of the LRA); fire dampers also were included in the structural review, except for those that form part of the pressure boundary for the safety-related portions of the control room ventilation system; these fire dampers were included in an AMR for the CRVS. The above CRVS components, which are relied on to perform the safety-related cooling and filtration functions for the MCRE, are included in the LRA, Table 3.4-13, as being within the scope of license renewal and subject to an AMR. The staff reviewed the applicant's response, and did not identify any omissions from the CRVS components that are relied on to perform the safety-related cooling and filtration functions for the MCRE.

Some components that are common to many systems, including the CRVS, have been separately evaluated in the LRA as commodity groups with similar components from other systems, and are evaluated by the NRC staff in other sections throughout this SER.

In Section 2.4 of the SER the staff evaluated component supports for piping, cables, and equipment, which are discussed in LRA Section 2.4, "Structures and Structural Component Scoping and Screening Results." In Section 2.5 of the SER, the staff evaluated the electrical components that support the operation of the CRVS; these components are discussed in the LRA Section 2.5, "Electrical and Instrumentation and Controls System Scoping and Screening Results." The CRVS instrumentation lines are listed as “tubing” in Table 3.4-13 of the LRA.

The NRC staff reviewed the LRA, supporting information in the UFSAR, and the applicant's responses to the staff's RAI. In addition, the NRC staff sampled several components from the CRVS flow diagrams (Table 2.3-7 of the LRA) to determine whether the applicant properly identified the components that are within the scope of license renewal and subject to an AMR. No omissions were identified.

2.3.3.13.3 Conclusions

On the basis of this review, the staff finds that there is reasonable assurance that the applicant has adequately identified the portions of the CRVS that are within the scope of license renewal, and the associated SCs that are subject to an AMR, in accordance with the requirements of 10 CFR 54.4(a) and 54.21(a)(1), respectively.

2.3.4 Steam and Power Conversion Systems

The ANO-1 steam and power conversion systems are designed to remove heat from the reactor coolant system, and include the following systems: main steam system, main feedwater system, emergency feedwater system, and condensate storage and transfer system. In the LRA, Section 2.3.4, “Steam and Power Conversion System,” the applicant describes these systems, and identifies and lists the components from these systems that are within the scope of license renewal and subject to an AMR. The applicant describes its process for identifying the mechanical components that are within the scope of license renewal and subject to an AMR in the LRA, Section 2.1 “Scoping and Screening Methodology,” and Section 2.2.1, “Mechanical and Electrical Systems.”

2.3.4.1 Summary of Technical Information in the Application

In the LRA, Section 2.3.4, the applicant describes the steam and power conversion systems, and identifies the following four portions of the steam and power conversion systems that are within the scope of license renewal:

- main steam
- main feedwater
- emergency feedwater
- condensate storage and transfer

In the LRA, Table 2.3-8, the applicant provides a list of scoping drawings, consisting of the P&IDs, for the four steam and power conversion systems that are within the scope of license renewal. The applicant provided a highlighted set of these drawing with the LRA, to show the portions of these systems that are within the scope of license renewal. From the components highlighted in these drawing, the applicant provided lists of the mechanical component groups that are subject to an AMR in the LRA, Table 3.5-1 through Table 3.5-4 for the main steam system, main feedwater system, emergency feedwater system, and condensate storage and transfer system, respectively.

The ANO-1 main steam system is primarily a non-safety-related system, with the majority of the system components outside the scope of license renewal. However, the non-safety-related small-bore piping and components that are attached to the steam generator shell, which perform a system pressure boundary function, are in scope. This includes valves that are part of the heater vent system. In addition, the safety-related portion of the main steam system piping between the steam generators and the main steam isolation valves, including the steam supply to the emergency feedwater (EFW) turbine, as well as the nitrogen supply to the steam generators is in the scope of license renewal.

The safety-related functions of the main steam system is to remove heat from the RCS to protect the RCS and the steam generators from over pressurization, to provide isolation for the steam generators during a postulated steam line break, and to supply steam to the emergency feedwater turbine. The components in the main steam system that are subject to an AMR include the piping between the steam generators and the main steam isolation valves, the piping from the nitrogen supply to the steam generators, the vent and drain valves from the steam generators to the main steam isolation valves, and EFW turbine steam supply piping. This includes the main steam safety valves, the atmospheric dump and block valves, and the main steam isolation valves. The primary intended function for the components that are within the scope of license renewal is pressure boundary integrity. The following drawings are identified for the main steam system: LRA-M-204, Sheet 6, LRA-M-206, Sheets 1 and 2. The portions of the system that are within the scope of license renewal are highlighted on the main steam system drawings.

The main feedwater system is a two train system consisting of pumps, feedwater heaters, associated piping and valves. This system supplies feedwater to the steam generators to support normal plant operation. The ANO-1 main feedwater system is largely a non-safety-related system, and therefore, the majority of the system components are outside of the scope of license renewal. The piping between the main feedwater isolation valves and the steam generators is the portion of the main feedwater system that is safety-related. Other portions of the main feedwater system are non-safety-related and are outside of the scope of license renewal. The main feedwater isolation valves isolate the feedwater line during a main steam or a main feedwater line break. The components that move to provide the necessary isolation are active. However, the valve bodies and piping in the safety-related portions of the system are within the scope of license renewal and subject to an AMR because they are required to maintain the main feedwater system pressure boundary integrity.

The components in the main feedwater system that are subject to an AMR include the main feedwater isolation valves and the piping, vent, and drain valves in the piping from the isolation valves to the steam generator ring headers. The intended function of the main feedwater system components that are subject to an AMR is to maintain pressure boundary integrity. The main feedwater system drawing is LRA-M-206, Sheet 1, which is highlighted to show the portions of the system that are within the scope of license renewal. The mechanical component groups for the main feedwater system that are subject to an AMR are identified in Table 3.5-2 of the LRA, and includes piping, tubing and valves.

The EFW system is a two train system that contains pumps, associated piping, and valves used to supply emergency feedwater to the steam generators upon failure of the main feedwater system. One EFW pump is motor-driven and the other is turbine-driven. Both pumps are capable of taking suction from the safety-related condensate storage tank, the non-safety-related condensate storage tank, the service water system, or the ANO-2 condensate storage tanks. Both EFW pumps supply feedwater to both steam generators.

The EFW system provides a backup source of feedwater to the steam generators to ensure the removal of decay heat from the reactor core, and the removal of residual heat from the primary system. The EFW system removes decay heat until the plant has been cooled and depressurized sufficiently to permit use of the decay heat system.

The EFW system components that are subject to an AMR include the EFW discharge piping and valves, the EFW pumps, the safety-related portion of the minimum recirculation lines, and the pump discharge piping and valves to the steam generators EFW headers. The main steam supply valves to the EFW turbine and the steam supply piping downstream of the valves are also within the scope of license renewal and are included within the scope of this review. The steam generators EFW headers and the associated nozzles are included within the scope of the steam generator AMR.

The EFW system's primary intended function is pressure boundary integrity. For the heat exchangers included within the scope of this review, heat transfer is also an intended function that needs to be considered during the AMR.

The EFW system drawings are LRA-M-204, Sheets 3 and 6, and LRA-M-206, Sheet 1. These drawings are highlighted to show the portions of the EFW system that are within the scope of license renewal. The EFW system mechanical components are divided into commodity groups that are subject to an AMR. These mechanical components commodity groups are identified in Table 3.5-3 of the LRA, and include piping, valves, tubing, pump casings, orifice plates, steam traps, turbine casing, expansion joints, filters, and heat exchangers.

The ANO-1 condensate storage and transfer system consists of the condensate storage tank, the safety-related condensate storage tank, and the system piping and valves that are needed to supply water from the condensate storage tanks to the secondary plant systems. This system is the primary source of demineralized water to the secondary plant. The safety-related condensate storage tank and the associated piping serves as the safety-related initial (preferred) source of water to the emergency feedwater pumps. The condensate storage and transfer system mechanical components that are within the scope of license renewal and subject to an AMR include the safety-related condensate storage tank, and the piping that maintains the pressure boundary of the system from the condensate storage tanks to the emergency feedwater pumps. The primary intended function of the system is pressure boundary integrity. The condensate storage and transfer system drawings are LRA-M-204 Sheets 3 and 5. These drawings are highlighted to show the portions of the system that are within the scope of license renewal. The applicant identifies the mechanical component commodity groups that are subject to an AMR in Table 3.5-4 of the LRA, and includes piping, tubing, valves, appurtenances, tanks, and heaters.

2.3.4.2 Staff Evaluation

The staff reviewed the above information to verify that the applicant identified the components of the steam and power conversion systems that are within the scope of license renewal and subject to an AMR, in accordance with 10 CFR 54.4 and 54.21(a)(1).

The applicant identified and listed the components subject to an AMR for the steam and power conversion systems in Table 3.5-1 through Table 3.5-4 of the LRA using the screening methodology described in Sections 2.1 and 2.2.1 of the LRA. The screening methodology is evaluated by the NRC staff in Section 2.1 of this SER.

The NRC staff reviewed the ANO-1 UFSAR, Chapter 10, “Steam and Power Conversion System,” to determine if there were any system functions, not identified as intended function in accordance with 10 CFR 54.4. The NRC staff then reviewed the system drawings (LRA-M-204 Sheets 3, 5, and 6, and LRA-M-206, Sheets 1 and 2) to verify that the applicant identified all the components that are within the scope of license renewal in accordance with 10 CFR 54.4(a). Further, the NRC staff verified the accuracy of the system drawings, and completeness of Table 3.5-1 through Table 3.5-4 by sampling the components adjacent to, but outside the highlighted portion of the system to verify that all the components that are within the scope of license renewal were included in the applicant’s evaluation. In addition, the NRC staff sampled the components that are within the scope of license renewal, but not subject to an AMR to verify that all of the components that meet the requirements of 10 CFR 54.21(a)(1) were subject to an AMR.

As a result of this review, the NRC staff requested additional information in a letter to the applicant dated May 5, 2000. The applicant responded to the NRC staff’s RAI in a letter to the NRC dated, August 20, 2000.

In its responses, the applicant verified that the EFW valve body for valve MS-2652 and the associated instrument tubing are within the scope of license renewal and were included in the AMR. These components, which are made of carbon steel, are included in the component commodity groupings “valves” and “tubing” in Table 3.5-1 of the ANO-1 LRA. The lack of highlighting on Drawing LRA-M-206, Sheet 1, to indicate that these components are within the scope of license renewal was an administrative error. In addition, the applicant also verified that the EFW turbine casing, which is part of the EFW system, is within the scope of license renewal and was included in an AMR. The results of the AMR for this casing are provided in Table 3.5-3 under the steam supply and exhaust subsystem.

The NRC staff had also asked the applicant to explain the exclusion of the emergency feedwater initiation and control (EFIC) system in Drawing No. LRA-M-206, Sheet 2, from the scope of license renewal. The applicant responded that the emergency feedwater initiation and control (EFIC) system is not a mechanical system, and thus it is not color coded on the drawings as being within the scope of license renewal. Drawing LRA-M-206, Sheet 2, shows three solenoid-operated valves located in the instrument air system that are within the scope of license renewal. Operation of these valves is controlled by the EFIC system. As indicated in Table 2.2-1 of the ANO-1 LRA, the EFIC system is within the scope of license renewal and was included in the review of ANO-1 electrical and instrumentation and controls systems as described in Sections 2.5 and 3.7 of the LRA.

For the main steam system, the applicant verified that no filters or expansion joints are within the scope of license renewal. Two orifices in the main steam system are within the scope of license renewal and were included in the AMR. These orifices, which are made of stainless steel, serve a pressure boundary intended function, and are included in the “piping” component commodity grouping in Table 3.5-1 of the ANO-1 LRA.

For the condensate storage and transfer system, the NRC staff asked the applicant to justify the exclusion of the demineralizer from the scope of license renewal. The applicant responded that although the condensate storage and transfer system is filled with demineralized water, it does not contain a demineralizer. The source of make-up water to the condensate storage and transfer system is from the mobile water treatment facility via the makeup water degasification system. The makeup line to the safety-related condensate storage tank is not safety-related, and its failure would not prevent the tank from performing its intended function. Thus, the makeup line is not within the scope of license renewal. Condensate demineralizers are part of another ANO-1 system (i.e., the condensate demineralizer system), and do not interface with the safety-related portions of the condensate storage and transfer system. The applicant also verifies that the condensate storage and transfer system has no filters, expansion joints, orifices, or strainers that are within the scope of license renewal.

On the basis of the NRC staff’s review of the LRA and associated drawings, the ANO-1 UFSAR, and the applicant’s responses to RAIs, the staff did not identify any omissions from the components highlighted in the diagrams that identify the system level scoping boundaries. The NRC staff also compared the components listed in Tables 3.5-1 through 3.5-4 of the LRA and the components highlighted in the drawings, and found them consistent.

2.3.4.3 Conclusions

On the basis of the review described above, the NRC staff finds that there is reasonable assurance that the applicant has adequately identified those portions of the steam and power conversion systems that are within the scope of license renewal, and the associated SCs that are subject to an AMR, in accordance with the requirements of 10 CFR 54.4(a) and 54.21(a)(1), respectively.

2.3.5 References for Section 2.3

1. 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."
2. DG-1047, "Standard Review Plan for the Review of License Renewal Application for Nuclear Power Plants," Working Draft, April 21, 2000.
3. Arkansas Nuclear One - Unit 1, License Renewal Application dated January 31, 2000.
4. ANO-1 Updated Final Safety Analysis Report.

THIS PAGE IS INTENTIONALLY LEFT BLANK

2.4 Structures and Structural Components Scoping and Screening Results

2.4.1 Reactor Building

In the LRA, Section 2.4.1, "Reactor Building," the applicant described the reactor building structure, and identified its structural components that are within the scope of license renewal and subject to an AMR. The design of the reactor building structure is described in Sections 5.1 and 5.2 of the ANO-1 UFSAR. The NRC staff reviewed this information to determine whether the applicant has adequately demonstrated that the requirements of 10 CFR 54.4, 54.21(a)(1), and 54.21(a)(2) have been met for the reactor building structure and structural components.

2.4.1.1 Technical Information in the Application

In the LRA, Section 2.4.1, the applicant states that the reactor building is a seismic Category 1 structure that completely encloses the reactor and RCS, as well as other electrical, mechanical, and structural SSCs. Seismic Category 1 structures are designed to prevent the uncontrolled release of radioactive material as a result of a specified seismic event, and to withstand all applicable loads without loss of function.

The reactor building structure consists of a post-tensioned concrete cylindrical shaped wall, a shallow domed roof, and a flat reinforced concrete foundation. The internal surfaces of the wall, roof, and foundation are lined with a carbon steel liner to maintain a high degree of leak tightness. Various penetrations through the cylindrical wall are provided for the passage of piping, ducts, and electrical conduits that are sealed at the penetration to ensure the reactor building integrity. The reactor building and its structural components meet the intent of 10 CFR 54.4(a) for license renewal because they perform one or more of the following functions:

- provide a leak tight barrier to prevent uncontrolled release of radioactivity
- provide structural support or functional support to safety-related SSCs
- provide shelter or protection to safety-related equipment (including radiation shielding)
- provide rated fire barriers to confine or retard a fire from spreading to or from adjacent areas
- serve as external missile barriers
- provide structural or functional support to non-safety-related equipment, failure of which could directly prevent satisfactory accomplishment of required safety-related functions
- provide a heat sink during a DBA or station blackout

In the LRA, Table 3.6-2, the applicant lists the SCs of the reactor building in the following material groupings: steel (including welds), concrete (including non-shrink grout, epoxy grout, embedment, and reinforcement), and post-tensioning system. In this table, the applicant further divides these material groupings into a total of 15 structural components or unique commodities that are subject to an AMR. Some of the components in the reactor building are common to other buildings, and these components are listed as bulk commodities in Table 3.6-8 of the LRA. The bulk commodities are discussed in Section 2.4.6.2 of the LRA, and are evaluated in Section 2.4.7 of this SER.

The structural components listed in Table 3.6-2 of the LRA are subject to an AMR in accordance with 10 CFR 54.21(a)(1) because applicable intended functions are performed without moving parts or without a change in configuration or properties, and they are not replaced based on a qualified life or specified time period.

2.4.1.2 Staff Evaluation

The NRC staff reviewed Section 2.4.1 of the LRA, and the ANO-1 UFSAR to determine if the applicant has adequately identified the SSCs of the reactor building that are within the scope of license renewal in accordance with 10 CFR 54.4(a), and the SCs that require an AMR in accordance with 10 CFR 54.21(a)(1). After completing its initial review, the staff requested additional information in a letter to the applicant dated April 18, 2000. The applicant responded to the staff's RAIs in a letter to the NRC dated August 30, 2000.

In the LRA, Section 2.4.1, the applicant states that many components are not typically associated with a unique equipment identifier and, therefore, were not individually identified as being subject to an AMR. The staff requested that the applicant provide examples of reactor building components without unique identifier, and to explain how these components were included within the scope of license renewal, were included in an AMR, and will be maintained during the period of extended operation.

In its response, the applicant states that an AMR was typically performed for component groupings, rather than for individual components. For example, "structural shapes" are made of steel, and "columns" are made of concrete. Structural shapes and columns associated with the reactor building have been included in Table 3.6-3 of the LRA, and are considered to be within the scope of license renewal and subject to an AMR. During the AMR, the aging effects were determined by the materials of construction and the environment to which, the SCs of each component grouping are exposed. As summarized in the LRA, Table 3.6-3, the aging effects for components requiring an AMR will be managed during the period of extended operation by a proposed AMP. The staff reviewed this information and did not identify any concerns relating to the scoping and screening of reactor building SCs without unique identifiers.

In the LRA, Table 3.6-2, the applicant identifies the following reactor building SCs and groupings:

- The liner plate, threaded fasteners, personnel air-lock, emergency personnel hatch, equipment hatch, mechanical penetrations, electrical penetrations, fuel transfer tube, and anchorage/embedment/attachment are identified as the steel components or unique commodities.

- The reactor building dome, cylinder wall, floor, and foundation are identified as the concrete components.

- The tendon wires and tendon anchorage are identified as the post-tensioning system.

The applicant combined many of these structures and structural components as component or commodity groupings for the AMR. For example, embedment includes plates and bolts below the concrete, reinforcement includes embedded bars, wires, and strands. In addition, the anchor, embedment, and attachments (such as angles, anchor studs) welded to the liner of the concrete cylinder wall are included within the steel component grouping. Certain anchors and attachments directly welded to the liner to support various SCs (i.e., the polar crane bracket) are also included in the steel component grouping. These attachments to the liner are integral with the liner and concrete structure at the inside surface of the reactor building. In addition, the liner plate is thickened at these attachments by a welded plate assembly embedded in the concrete.

There are two personnel air-locks in the reactor building including a personnel access lock and a personnel escape lock. The personnel air-locks are double-door, welded steel assemblies that are designed to withstand reactor building design conditions with either or both doors closed and locked. Quick-acting, equalizing valves are provided for each personnel air-lock to equalize the pressures on either side of the air-lock door to allow operation of the door. The equalizing valves are active components and, therefore, are not subject to an AMR. The applicant also considers other operating mechanisms (such as gears, latches, hinges, linkages, etc.) that are used to open and close the air-lock doors to be active components that are not subject to an AMR. The applicant states that the operating mechanism components perform its function(s) with moving parts and, therefore, are not subject to an AMR in accordance with 10 CFR 54.21. However, the staff also recognizes that the gears, latches, and hinges are needed for proper alignment of the hatches, which is an intended function that may be performed without moving parts, or without a change of configuration or properties and, therefore, may require that these components be subject to an AMR. After a number of discussions with the applicant relating to the gears, latches, and hinges, the applicant agreed to perform an AMR on these components. In a letter to the NRC dated December 20, 2000, the applicant identifies the ASME Section XI, Subsection IWE inspection activities as the program that will be used to manage the aging of these components. The staff found this acceptable because it is consistent with the current requirements for these components, and with the recommendations from the generic aging lessons learned issued by the NRC staff in August 2000.

The inner and outer doors on each of the personnel air-locks are interlocked to maintain the reactor building integrity during normal plant operation. The interlock system is an active component that is not subject to an AMR. Serviceability of the interlock system is verified during periodic inspection and maintenance. Each personnel air-lock door has flexible seals, which are short-lived components and are subject to periodic replacement. Therefore, the flexible seals do not require an AMR.

In the LRA, Section 2.4.1, the applicant identifies a single equipment hatch in the reactor building that allows passage for large items or equipment. The applicant states that the hatch is furnished with a double-sealed flange and a bolted, dished head. The space between the double seals of the hatch flange can be pressurized for local leakage testing. The seals are not subject to an AMR because they are replaced based on a qualified life or specified time period.

In the LRA, Section 2.4.1, the applicant also identifies various penetrations in the reactor building pressure boundary that are designed with leak-tight barriers to prevent uncontrolled release of radioactive material. These mechanical penetrations allow for the movement of liquids or gases across the reactor building boundary through piping. The portion of the mechanical penetrations that is within the scope of license renewal includes the entire penetration assembly and typically the process piping in the penetration. The applicant also identifies spare penetrations with welded-end caps or bolted blind flanges as also being within the scope of license renewal and subject to an AMR because they are part of the reactor building pressure boundary. The electrical penetrations provide the means for electrical and instrumentation conductors to cross the reactor building pressure boundary. The metallic components of the electrical penetration are identified as being within the scope of license renewal because they are part of the reactor building pressure boundary. The fuel transfer tube is the underwater pathway for moving fuel assemblies into and out of the reactor building during refueling operations. The closure between the transfer tube and the sleeve, that is welded to the reactor building liner, is part of the reactor building pressure boundary. The applicant identifies this portion of the transfer tube as being within the scope of license renewal and subject to an AMR. The transfer tube, blind flange, and gate valve are part of the spent fuel pool system, and are reviewed in Section 2.3.3.1 of this report. The staff reviewed the above information and did not identify any omissions relating to the scoping and screening of the personnel air-locks, the equipment hatch, and other penetrations in the reactor building.

The reinforced concrete dome and cylindrical walls are prestressed by the post-tensioning system. The cylindrical portion is prestressed by a post-tensioning system with horizontal and vertical tendons. The dome has a three-way post-tensioning system. Reinforcing steel is provided near the surface of the cylinder wall and dome. Additional reinforcing is provided at structural discontinuities to resist local loads and thermal stresses. A reinforced concrete foundation slab is designed to support the reactor building. The reactor building has a reinforced concrete floor above the embedded portion of the liner plate. The reinforced concrete structures and the post-tensioning system of the reactor building are within the scope of license renewal and subject to an AMR.

A reinforced concrete enclosure under the foundation slab (known as the lower tendon access gallery) is provided for tendon installation and surveillance activities. The applicant has determined that the lower tendon access gallery does not perform a reactor building pressure boundary function, or any other function under 10 CFR 54.4 and, therefore, is not included within the scope of license renewal. The staff reviewed this information, and determined that the post-tensioning system is the primary means of containing the internal pressure of the reactor building during DBEs. The tendon gallery protects the bottom anchorages of the tendons and provides access for tendon anchorage inspections. The staff agrees that the tendon gallery does not perform the intended function required by 10 CFR 54.4(a).

The staff reviewed the above information and did not identify any omission relating to the scoping and screening of the reactor building reinforced concrete dome, cylindrical walls, foundation and floor, and the reactor building post-tensioning system.

2.4.1.3 Conclusions

On the basis of the staff's review of the information presented in Section 2.4.1 of the LRA, Sections 5.1 and 5.2 of the ANO-1 UFSAR, the additional information submitted by the applicant in response to the staff's RAI, and the design drawings submitted by the applicant for this review, the staff finds that there is reasonable assurance that the applicant has adequately identified those portions of the reactor building that are within the scope of license renewal, and the associated SCs that are subject to an AMR, in accordance with the requirements of 10 CFR 54.4(a) and 54.21(a)(1), respectively.

2.4.2 Reactor Building Internals

In the LRA, Section 2.4.2, "Reactor Building Internals," the applicant describes the reactor building internal structures and identifies the structural components that are within the scope of license renewal and subject to an AMR. In addition, the design of the reactor building internals is described in Sections 5.1 and 5.3.1 of the ANO-1 UFSAR. The NRC staff reviewed this information to determine whether the applicant has adequately demonstrated that the requirements of 10 CFR 54.4, 54.21(a)(1), and 54.21(a)(2) have been met for the reactor building internals.

2.4.2.1 Technical Information in the Application

In the LRA, Section 2.4.2, the applicant states that the reactor building internal structures consist of the reactor cavity, two steam generator compartments, and a fuel transfer canal (located between the two steam generator compartments and above the reactor cavity). The reactor cavity, which serves as the primary shield wall, houses the reactor pressure vessel. Structural steel components, such as platforms, ladders, and grating are also in the reactor cavity that provide access for inspection and maintenance activities. Each of the two steam generator compartments houses a steam generator, reactor coolant pump and the associated reactor coolant system piping. The pressurizer is located in a compartment adjacent to and integral to one of the steam generator compartments. The primary function of the steam generator compartment, known as the D-ring walls, is to serve as the secondary shield walls that resist pressure and jet impingement as a result of a high energy line rupture. The reactor building internals are seismic Category 1 structures, and are included within the scope of license renewal in accordance with 10 CFR 54.4(a)(1). Seismic Category 1 structures are designed to prevent the uncontrolled release of radioactive material as a result of a specified seismic event, and to withstand all applicable loads without loss of function.

In the LRA, Section 2.4.2, the applicant also states that the reactor building internals include various structural components, which support or protect SSCs that are within the scope of license renewal. In the LRA, Table 3.6-3, the applicant identifies anchorages, embedment attachments, threaded fasteners, structural shapes, steam generator support steel, pressurizer support steel, main fuel handling bridge, jib cranes, polar crane, control rod drive service structure, and reactor vessel support skirt as the steel components or unique commodities. The applicant also identifies the primary and secondary shield walls, reinforced concrete columns, walls, hatches, reactor missile shield, and fuel transfer canal as concrete components or unique commodities. Other components that are common to other buildings are listed as bulk commodities in Table 3.6-8 of the LRA, and are evaluated in Section 2.4.7 of this SER.

The applicant states that the components listed in Table 3.6-3 are within the scope of license renewal because they perform one or more of the following intended functions:

- provide structural support or functional support to safety-related equipment
- provide shelter or protection to safety-related equipment (including radiation shielding)
- provide rated fire barriers to confine or retard a fire from spreading to or from adjacent areas
- serve as internal missile barriers
- provide structural or functional support to non-safety-related equipment, failure of which could directly prevent satisfactory accomplishment of required safety-related functions
- provide a heat sink during DBE or station blackout

The applicant also states that these components perform the intended functions listed above without moving parts or without a change in configuration or properties, and are not subject to replacement based on qualified life or specified time period and, therefore, are subject to an AMR in accordance with 10 CFR 54.21(a)(1).

2.4.2.2 Staff Evaluation

The NRC staff reviewed Section 2.4.2 of the LRA, and the ANO-1 UFSAR to determine if there is reasonable assurance that the applicant has identified the structures and structural components that comprise the reactor building internals, and that are within the scope of license renewal and subject to an AMR in accordance with 10 CFR 54.4(a), and 54.21(a), respectively. After completing its initial review, the staff requested additional information relating to reactor building internals in a letter to the applicant dated April 18, 2000. The applicant responded to the staff's questions in a letter to the NRC dated August 30, 2000.

As part of the evaluation, the staff reviewed the portions of the ANO-1 UFSAR and the applicable drawings for the reactor building internals, and compared this information with the information in the LRA to identify any instances where the LRA did not identify SCs as being within the scope of license renewal and subject to an AMR.

As a result of this review, the staff identified the following concerns:

- The incore instrumentation tunnel and reactor building sump are not discussed in Section 2.4.2, and are not listed in Table 3.6-3. The staff requested that the applicant identify where in the LRA are these components are addressed, or to provide a technical justification as to why these components are not within the scope of license renewal.
- In the LRA, Section 2.4.2.1, the applicant states that structural steel is provided for supporting several nuclear components (i.e., the core flood tanks, reactor building cooling units, emergency core cooling system piping). The staff asked the applicant if the lateral support steel that holds the snubbers and turnbuckles for the steam

generators and reactor coolant pumps are included within the scope of license renewal, or to provide a technical justification for the exclusion of these components from an AMR.

In its response, the applicant states that the incore instrument tunnel and the reactor building sump are part of the basement floor slab that is within the scope of license renewal. The component grouping for the instrumentation tunnels and the reactor building sump was deleted from the last row of Table 3.6-3 of the LRA as a result of an administrative error. In the table, under the column heading "Component/Commodity Grouping," it should read "a basement floor slab" rather than "reinforced concrete." Reinforced concrete should be under the column heading "material."

In addition, the applicant states that the support steel for snubbers associated with the steam generators and reactor coolant pumps is included in the bulk commodity grouping "piping and tubing supports," in Table 3.6-1 of the LRA. Table 3.6-1, general note "G," states that this grouping includes mounting brackets for snubbers. Steel supporting equipment, such as the lateral support steel for turnbuckles for the steam generators and reactor coolant pumps are considered bulk commodities, and are included under the commodity grouping "equipment supports" in Table 3.6-8 of the LRA.

Major portions of the reactor building internals are reinforced concrete structures, which include basement floor slab (cover over the liner plate), columns, the walls surrounding the steam generators, reactor, and the pressurizer, the valve pits and pipe chases and the slabs on top of them, missile shields, fuel transfer canal, and removable concrete hatches and covers. The reactor building internals also contain reinforced concrete floors and galvanized steel gratings at various elevations that are supported by columns, or attached to the exterior surface of the secondary shield wall. Structural steel that is welded to the liner plate also provides grating support. The applicant identified a total of 17 commodity groupings in Table 3.6-3 of the LRA. These commodity groupings are further combined into two material groups (i.e., steel and concrete). All these steel and concrete components are in scope and subject to an AMR for license renewal because they are passive and long-lived and provide structural support or functional support to safety-related components and equipment.

In addition, the reactor building has a number of cranes that are used for different maintenance activities. The applicant determined that the following reactor building crane components are within the scope of license renewal and subject to an AMR, because of the potential for failure when lifting or carrying heavy loads, or the potential impact on safety-related SSCs:

- main fuel handling bridge
- auxiliary fuel handling bridge
- jib cranes
- polar crane

The applicant identifies other crane components, such as the fuel tilt machine and control rod drive crane, that are seismic Category 2 structures. The applicant states that failure of these components is not expected to impact safety-related SSCs and, therefore, are not within the scope of license renewal.

The control rod drive service structure, which supports the control rod drive mechanism, is within the scope of license renewal. This structure is located above the reactor vessel and consists of the following five major assemblies:

- lower control rod drive service structure skirt, which provides a seating surface to support the upper control rod drive service structure

- upper control rod drive service structure skirt, which is a carbon steel cylindrical shell that connects to the lower control rod drive service structure skirt

- closure head service structure shell, which is a carbon steel cylinder attached to the upper control rod drive service structure skirt to support the control rod drive service structure platform assembly

- control rod drive service structure strut support assembly, which is the horizontal steel beams oriented in a radial direction that are welded to the closure head service structure shell on one end and supported by angled beams on the other

- control rod drive service structure platform assembly, which is a horizontal platform that is made of steel beams, and used to restrain the lateral movement of the top ends of the control rod drive mechanisms during design basis loading

The applicant states that these assemblies are within the scope of license renewal and subject to an AMR because they provide structural support to safety-related components and equipment without any moving parts, or without a change in configuration or properties, and are not replaced based on qualified life or specified time period.

The reactor vessel supports include a support skirt and a support flange. The reactor vessel support skirt, which supports the reactor vessel, is a steel cylindrical structure. The support skirt sits on a sole plate, which is fixed to a reinforced-concrete pedestal by a steel flange that is bolted to the pedestal. The steel cylindrical structure is welded to the bottom of the reactor vessel transition forging. The cylinder has holes for ventilation of the reactor cavity. The applicant identifies the evaluation boundary for the reactor vessel support skirt to include the structural components between the weld of the skirt at the reactor vessel transition forging to the bottom of the skirt flange. The anchor bolts and shear pins are also within the scope of license renewal and subject to an AMR.

The staff reviewed the above information and did not identify any omissions by the applicant relating to the scoping and screening of reactor building internals.

2.4.2.3 Conclusions

On the basis of the staff's review of the information presented in Section 2.4.2 of the LRA, the ANO-1 UFSAR, the additional information submitted by the applicant in response to the staff's RAIs, and the design drawings submitted by the applicant for this review, the staff finds that there is reasonable assurance that the applicant has adequately identified those portions of the reactor building internals that are within the scope of license renewal, and the associated SCs that are subject to an AMR, in accordance with the requirements of 10 CFR 54.4(a) and 54.21(a)(1), respectively.

2.4.3 Auxiliary Building

In the LRA, Section 2.4.3, “Auxiliary Building,” the applicant describes the auxiliary building and identifies the SCs in the auxiliary building that are within the scope of license renewal and subject to an AMR. The design of the auxiliary building is described in Sections 5.1 and 5.3.2 of the ANO-1 UFSAR.

2.4.3.1 Technical Information in the Application

The auxiliary building is located adjacent to the reactor building and turbine building, and houses the safety-related SSCs that support normal operation, shutdown, and accident conditions. It is a free-standing reinforced concrete structure founded on bedrock. The structure and structural components of the auxiliary building are designed as seismic Category 1. Seismic Category 1 structures are designed to prevent uncontrolled release of radioactivity, and to withstand system and seismic loading without loss of function. The applicant has determined that seismic Category 1 structures meet the intent of 10 CFR 54.4(a)(1).

Several structural components within the auxiliary building (i.e., the liner plate within the spent fuel pool and the small pipe chase at elevation 341') are classified as seismic Category 2 structures. The seismic Category 2 structures are those structures whose failure would not result in a release of radioactivity and would permit a controlled plant shutdown, but could interrupt power generation. The applicant has determined that seismic Category 2 structures meet the intent of 10 CFR 54.4(a)(2).

The applicant also determined that some areas in the auxiliary building (i.e., areas with 10 CFR 50.48-required fire barriers) meet the scoping requirements of 10 CFR 54.4(a)(3). The fire barriers and fire doors are grouped as steel components, while fire walls and slabs are grouped as the concrete components. In the LRA, Section 2.4.3, the applicant states that the turbine building itself is not within the scope of license renewal, but the fire doors and fire walls and slabs of the turbine building are within the scope of license renewal and subject to an AMR. These are addressed along with those for the auxiliary building.

The auxiliary building was built partially below grade. The construction joints of the exterior concrete wall contain water-stops at the joints below the plant’s design flood level that are subject to an AMR. The boron holdup tank vault is located below grade and is structurally connected to the auxiliary building. The borated water storage tank sits on top of the vault. The post-accident sampling system building is anchored to the top of the ANO-1 and ANO-2 tank vaults. The building and vaults are designed to seismic Category 1 criteria. These SCs are within the scope of license renewal in accordance with 10 CFR 54.4(a)(1).

In the LRA, Table 3.6-4, the applicant lists the SCs and unique commodities of the auxiliary building that are subject to an AMR. The SCs in the auxiliary building that meet one of the scoping criteria in 10 CFR 54.4(a) are within the scope of license renewal because they perform at least one of the following intended functions, as noted in the table:

- provide essentially leak tight barriers to prevent uncontrolled release of radioactivity

- provide structural or functional support to safety-related equipment

- provide rated fire barriers to confine or retard a fire from spreading to or from adjacent areas

- serve as missile (internal or external) barriers

- provide structural or functional support to non-safety-related equipment, failure of which could directly prevent satisfactory accomplishment of required safety-related functions

- provide protective barriers for internal and external flood events

- provide for storage of spent fuel assemblies

Some of the components in the auxiliary building are common to many other buildings that are listed as the bulk commodities in Table 3.6-8 of the LRA. The bulk commodities have been reviewed by the applicant in Section 2.4.6.2 of the LRA. The SCs and commodities in the auxiliary building are subject to an AMR because they perform its intended function(s) without moving parts or without change in configuration or properties, and are not subject to periodic replacement based on qualified life or specified time limit.

2.4.3.2 Staff Evaluation

The staff reviewed Section 2.4.3 of the LRA and the supporting information in ANO-1 UFSAR to determine whether there is reasonable assurance that the SCs and commodities comprising the auxiliary building have been properly identified as being within the scope of license renewal and subject to an AMR. After completing its initial review, the staff requested additional information in a letter to the applicant dated April 18, 2000, regarding the information provided in the LRA. The applicant responded to the staff's RAIs by a letter to the NRC dated August 30, 2000.

The applicant lists the passive components and unique commodities of the auxiliary building in Table 3.6-4 of the LRA and the bulk commodities in Table 3.6-8 of the LRA. The applicant further combined these components and commodities into three groups based on their construction materials, i.e., (1) steel (including welds), (2) threaded fasteners (including structural bolts, expansion anchors and undercut anchors), and (3) concrete (including non-shrink grout, epoxy grout, embedments, and reinforcement, but not including prestressed concrete). The staff reviewed the component groupings in Table 3.6-4 to determine if there were any other components in the auxiliary building that meet the scoping criteria of 10 CFR 54.4(a), and were not included within the scope of license renewal. As a result of this review, the staff requested additional information regarding the auxiliary building and its structural components that serve as missile barriers. In the LRA, Table 3.6-4, only the missile shield doors and walls are listed. The staff asked whether any missile protective devices for resisting internal missiles are installed in the auxiliary building, such as missile barriers to protect safety-related SSCs from pipe whipping or jet forces due to main steam line ruptures or pressure relief valve failures.

In its response to this RAI, the applicant states that there are other missile protective devices in the auxiliary building in addition to missile-shield doors and walls. In LRA, Table 3.6-4, the control room extension substructure is a missile barrier. As stated in Section 2.4.3 of the LRA, the commodities considered common to the auxiliary building and other in-scope structures are listed as bulk commodities in Table 3.6-8 of the LRA. These include missile-protected hatches that are under the commodity grouping “hatch frames/covers” for steel, or under the commodity grouping “hatch covers/plugs” for concrete. Piping whip restraints and impingement barriers are also addressed in Table 3.6-8 of the LRA. The staff’s review found that the applicant did include the missile barriers in the scope of components that are subject to an AMR.

In the LRA, Section 2.4.3 the applicant states that the turbine building itself is not within the scope of license renewal, some fire doors, fire walls, and slabs within the turbine building are in scope and subject to an AMR. These components are addressed along with those for the auxiliary building. The staff considers that these in-scope components of the turbine building provide a rated fire barrier to confine a fire from spreading to adjacent areas of the plant. The staff felt that turbine building should be added to the scope of license renewal because it contains components that were subject to an AMR and, therefore, asked the applicant to justify excluding the turbine building from the scope of license renewal. The staff also asked the applicant to identify any safety-related piping or cable routed through the basement of the turbine building that needs to be sheltered or protected.

In its response to this RAI, the applicant states that there are no safety-related pipes or cables in the turbine building. The turbine building has been included in the scope of license renewal as identified in Sections 2.4.3 and 2.4.6.2 of the LRA. In a letter to the NRC dated August 30, 2000, the applicant further clarified that Section 2.4 of the LRA should have included the turbine building as being within the scope of license renewal because it contains 10 CFR 50.48 SCs and commodities that are subject to an AMR. The staff reviewed the applicant’s response and found that the applicant satisfied the initial questions. However, there is no place in Section 2.4 of the LRA that describes the turbine building, and therefore, the staff does not have the needed information to verify with reasonable assurance that the applicant has identified all the components in the turbine building that require an AMR. However, during the license renewal scoping inspection, the NRC evaluated the potential for the SCs that should have been included within the scope of license renewal. As documented in NRC Inspection Report, IR0017, the inspection team identified additional cables that are required to support station blackout, and the reactor protection systems (reactor-turbine trip function) that are within the scope of license renewal and subject to an AMR. These cable are not seismically qualified and were included in the applicant’s AMR, therefore, no change is needed to the applicant’s program as a result of the inspection team’s finding.

In the LRA, Section 2.4.3, the applicant states that for the material group elastomers, none of the components or unique commodities are subject to an AMR and there are no components or unique commodities associated with the material groups earthen structures or Teflon. However, some of the components or commodities associated with the elastomers or Teflon group in the auxiliary building are listed in Table 3.6-8 of the LRA as bulk commodities that are subject to an AMR. The staff asked that the applicant explain this inconsistency.

In its response to this RAI, the applicant states that the commodities considered common to the auxiliary building are the bulk commodities discussed in Section 2.4.6.2 of the LRA. For the material group elastomers, none of the auxiliary building's elastomer components or "unique" commodities were subject to an AMR. In contrast, the water-stops as indicated in Table 3.6-8 of the LRA are subject to an AMR because they are common to other structures and are considered in the AMR of the bulk commodity. For the material group Teflon, there are no components or unique commodities associated with this material group in the auxiliary building. However, there are several bulk commodities in the auxiliary building, as well as in other structures, constructed with polytetrafluoroethylene materials (Teflon) that are subject to an AMR. In the auxiliary building, there are no components, unique commodities or bulk commodities associated with the material group earthen structures. The staffs review found that, except for the water stops and certain Teflon materials, there are no other elastomer components in the auxiliary building that are subject to an AMR.

The staff has reviewed Section 2.4-3 of the LRA, the ANO-1 UFSAR, and additional information submitted by the applicant in response to the staff's RAIs. The staff also examined the components and commodities listed in Tables 3.6-4 and 3.6-8 of the LRA to determine if they are the SCs that are subject to an AMR in accordance with 10 CFR 54.21(a)(1). On the basis of the above review, and the scoping inspection of the turbine building, the staff did not identify any omissions by the applicant.

2.4.3.3 Conclusions

On the basis of the review described above and the scope inspection of the turbine building, the staff finds that there is reasonable assurance that the applicant has adequately identified those portions of the auxiliary building and the turbine building that are within the scope of license renewal, and the associated SCs that are subject to an AMR, in accordance with the requirements of 10 CFR 54.4(a) and 54.21(a)(1), respectively.

2.4.4 Intake Structure

In the LRA, Section 2.4.4, "Intake Structure," the applicant describes the intake structure and identifies the structural components of the intake structure that are within the scope and subject to an AMR. The staff reviewed Section 2.4.4 to determine if there is reasonable assurance that the applicant has identified and listed the structural components of the intake structure that are subject to an AMR. The design of the intake structure is described in the ANO-1 UFSAR, Section 5.3.4.

2.4.4.1 Technical Information in the Application

The intake structure (located at the end of the intake canal) houses the circulating water, fire, and service water pumps, motor control centers, and traveling screens. It is constructed primarily of reinforced concrete that is founded on bedrock. The steel trash racks and traveling screens at the entrance of the intake structure protect the circulating water pumps from foreign materials present in the bay water. The intake structure is divided into two sections: the portion of the building area above grade elevation, and the portion of the structure below grade pump bay area that is partially submerged in water. The ANO-1 intake structure is integrally connected to the ANO-2 intake structure with a shear key and additional reinforcing in the slab at the pump level. The intake structure gantry crane is shared between the ANO-1 and ANO-2 intake structures. The gantry crane is supported by steel rail and girders on reinforced concrete piers, and is capable of traversing the entire length of the intake structure. It is normally parked at a safe distance from the intake structure.

The building portion of the intake structure above grade contains pump motors, valve motor actuators, and related equipment. This building area has three predominant elevations, which are El 354', El 366', and El 378'. The heating, ventilation and air conditioning (HVAC) equipment is located in the penthouse at El 378'. The pump motors and valve motor actuators are located at El 366', which is above the plant design flood level of El 361'. They are required to supply water for plant protection (i.e., fire water and service water). The remaining pump motors required for normal plant operation, such as the circulating water and screen wash pumps, are located at El 354'. The system components related to plant protection, which are not adversely affected by flood waters or which would not be required during a flood event (i.e., the intake structure sluice gate actuators), are also located at El 354'.

The below grade portion of the intake structure contains the pump bays for various plant systems. The four circulating water system pump bays take suction directly from Lake Dardanelle. The three service water system pump bays are located directly behind the circulating water pump bays. There are sluice gates in the service water system pump bays that can be aligned so that the fire water and service water pumps can take suction directly from Lake Dardanelle or from the emergency cooling pond as needed.

The portions of the intake structure that provide support to service water system components are designed to seismic Category 1 criteria. The remainder of the intake structure is seismic Category 2 structures. The applicant has determined that the seismic Category 1 structures are within the scope of 10 CFR 54.4(a)(1). However, seismic Category 2 structures are not within the scope of 10 CFR 54.4(a)(2). The applicant listed the structural components and unique commodities of the intake structure in Table 3.6-5 of the LRA. These structural components are within the scope of license renewal because they contribute to at least one of the following intake structure intended functions, as noted in the table:

- provide structural support or functional support to safety-related equipment
- provide shelter or protection to safety-related equipment
- serve as missile (internal or external) barriers
- provide structural or functional support to non-safety-related equipment, failure of which could directly prevent satisfactory accomplishment of required safety-related functions
- provide protection barriers for external flood event

The applicant has determined that these SCs and commodities are subject to an AMR as required by 10 CFR 54.21(a)(1).

2.4.4.2 Staff Evaluation

The staff reviewed Section 2.4.4 of the LRA and the ANO-1 UFSAR to determine if the applicant has adequately implemented its methodologies such that there is reasonable

assurance that the structural components and commodities comprising the intake structure have been properly identified as being within the scope of license renewal and subject to an AMR. After completing the initial review, the staff requested additional information in a letter to the applicant dated
April 18, 2000. The applicant responded to these RAIs in a letter to the NRC dated August 30, 2000.

The intake structure comprises various SCs and commodities that support the SSCs that are within the scope of license renewal. The applicant lists the SCs and commodities in Table 3.6-5 of the LRA that are subject to an AMR. In the table, the applicant combined the structural components and unique commodities of the intake structure in three material groups; steel (including welds), threaded fasteners (including bolts, expansion anchors, and undercut anchors), and concrete (including non-shrink grout, epoxy grout, embedment, and reinforcement). Certain components that are common in other buildings are grouped as the bulk commodities in Table 3.6-8 of the LRA that are reviewed in Section 2.4.7 of this report. There are 18 structural component groupings listed in Table 3.6-5 of the LRA, and 24 bulk commodity groupings listed in Table 3.6-8 of the LRA. Some of the structural components that do not contribute to any of the intended functions of the intake structure are not listed in the tables. SCs and commodities listed in Table 3.6-5 and Table 3.6-8 are subject to an AMR.

In the LRA, Section 2.4.4, the applicant states that the seismic Category 2 portions of the intake structure are not within the scope of 10 CFR 54.4(a)(2). However, some of the seismic Category 2 structures appear to provide functional support to some non-safety-related equipment whose failure could directly prevent satisfactory accomplishment of safety-related functions. The staff requested the applicant to provide additional justification for not including the seismic Category 2 structural components of the intake structure that are within the scope of license renewal.

In its response to this RAI, the applicant states that seismic Category 2 SSCs are those whose failure would not result in the uncontrolled release of radioactivity and would not prevent a safe reactor shutdown, but may interrupt power generation. Section 9.3.2.1 of the ANO-1 UFSAR states that failure of seismic Category 2 equipment in the proximity of the safety-related service water system components will not impact the integrity of the service water system. Therefore, the portions of seismic Category 2 SCs in the intake structure do not meet the criteria of 10 CFR 54.4(a)(2). The staff's review found that the Category 2 SCs in the intake structure do not provide any functional support to non-safety-related equipment whose failure could prevent satisfactory accomplishment of safety-related functions.

The staff did not find any omissions in the SCs identified by the applicant as being subject to an AMR in accordance with 10 CFR 54.21(a).

2.4.4.3 Conclusions

On the basis of the review described above, the staff finds that there is reasonable assurance that the applicant has appropriately identified those portions of the intake structure that are within the scope of license renewal, and the associated SCs that are subject to an AMR, in accordance with the requirements of 10 CFR 54.4(a) and 54.21(a)(1), respectively.

2.4.5 Earthen Embankments

In the LRA, Section 2.4.5, "Earthen Embankments," the applicant describes the earthen embankments at the plant site, and identifies the structures of the earth embankments that are within the scope of license renewal. The structures identified as being within the scope of license renewal are (1) the emergency cooling pond and (2) the intake and discharge canals, which are the seismic Category 1 structures. The design of these structures are shown in the site drawings (figures No. 9-32, 9-33, and 9-35) of the ANO-1 UFSAR.

2.4.5.1 Technical Information in the Application

The earthen embankment structures are partially or totally submerged in Lake Dardanelle. The applicant lists the emergency cooling pond (ECP), intake canal, and discharge canal in Table 3.6-6 of the LRA as the structures of the earthen embankments that are subject to an AMR. The intended function(s) of these components is to provide a heat sink during a DBA or station blackout.

The ECP is a 14-acre kidney-shaped water pond located northwest of the plant. It serves as a heat sink in the unlikely event of a loss of Lake Dardanelle water inventory. The water level of the pond is maintained between 5 and 6 feet by a spillway that discharges to Lake Dardanelle. The emergency cooling pond receives hot discharge from the plant through a 100-ft long weir. The purpose of the weir is to promote a uniform flow distribution in the pond, and to direct the hot water to the surface for maximizing heat rejection. The supply and return lines are at opposite extremes to prevent any hydraulic vortices. The plant intake piping is at the lowest point of the pond. The pond is excavated in impervious clay strata with its bottom at about 4 to 16 feet above rock. The crest voids and the adjacent embankment voids are downstream of the spillway and are pumped with an elastic type of grout to preclude undercutting by water flow over the spillway. The ECP side slopes are protected against wave action with dumped rip-rap. A series of weirs are provided at the channel to the reservoir to control silt settlement.

The intake canal conveys water from Lake Dardanelle to the intake structure that supplies the reservoir water for once-through cooling of ANO-1. The intake canal is approximately 4,000-ft long and the width varies from 80-ft at the mouth to 135-ft at the intake structure with an average depth of 14 feet. The discharge canal returns the used cooling water to the reservoir. The discharge canal is approximately 600-ft long with an average width of 165 feet and depth of 11 feet. Both canals are completely excavated and contain no section formed by dikes or in-fill. Bank slopes of the canals are planted with grass or protected with rip-rap to prevent erosion.

2.4.5.2 Staff Evaluation

The staff reviewed Section 2.4.5 of the LRA, and the drawings in ANO-1 UFSAR to determine if there is reasonable assurance that the components comprising the earthen embankments have been properly identified as being within the scope of license renewal and subject to an AMR.

The applicant identifies and lists the structures of the earth embankments that are subject to an AMR in Table 3.6-6 of the LRA. As shown in the table, the ECP and the intake and discharge canals are listed as the structures subject to an AMR. However, the structures associated with the earth embankments, such as spillway, weir, canal inlet and outlet structures, are not listed in the table as the components subject to an AMR for license renewal. The staff requested additional information regarding the exclusion of earth embankments.

In its response to the staff's RAIs, the applicant states that the spillway and weir are ECP components. They are subject to an AMR along with the overall ECP. The canal inlets and outlets are the components of the intake and discharge canals that are subject to an AMR as part of the intake and discharge canals. The staff's review found that the applicant included these components as being within the scope of license renewal and subject to an AMR, even though they are not individually listed in the table. The staff found no omissions in the SCs of earthen embankments included within the scope of license renewal and subject to an AMR.

2.4.5.3 Conclusions

On the basis of the review described above, the staff finds that there is reasonable assurance that the applicant has adequately identified the structures associated with the earth embankments that are within the scope of license renewal and subject to an AMR, in accordance with the requirements of 10 CFR 54.4(a) and 54.21(a)(1), respectively.

2.4.6 Yard Structures

In the LRA, Section 2.4.6.1, "Aboveground/Underground Yard Structures," the applicant describes the yard structures at the plant site, and identifies the SCs of the yard structures that are within the scope of license renewal. The applicant also identifies the SCs that are subject to an AMR in Table 3.6-7 of the LRA.

2.4.6.1 Technical Information in the Application

As described in Section 2.4.6.1 of the LRA, the following yard structures are within the scope of license renewal and subject to an AMR:

- Q-condensate storage tank foundation
- emergency diesel fuel oil storage tank vault
- bulk fuel oil storage tank foundation
- AAC diesel generator building foundation
- electrical manholes
- borated water storage tank foundation

The Q-condensate storage tank (Q-CST) foundation is a seismic Category 1 structure located at the west side of the ANO-1 reactor building. It is an octagon-shaped reinforced concrete mat foundation supported by concrete piers that are embedded in bedrock. Two valve pits are located partially underground, and on opposite (i.e., north and south) sides of the mat foundation. The south valve pit is for ANO-1, and the north valve pit is for ANO-2. The lower portion of the Q-CST is surrounded by a 5-ft high reinforced concrete wall for missile protection. The applicant determined that the Q-CST foundation is within the scope of 10 CFR 54.4(a)(1).

The emergency diesel fuel oil storage tank vault is a seismic Category 1 structure, which was designed as a reinforced concrete rigid-frame box. The vault is located at the northwest side of the reactor building and contains four diesel fuel storage tanks partitioned into separate rooms.

The foundation of the vault is anchored in rock, and the walls have ventilation openings above the flood elevation. The outside door of the vault is of watertight construction for flood protection. The applicant determined that the emergency diesel fuel oil storage tank vault is within the scope of 10 CFR 54.4(a)(1).

The bulk fuel oil storage tank foundation supports a 180,000-gallon fuel oil storage tank. It is a non-Q, seismic Category 2 reinforced concrete foundation. The applicant determined that the bulk fuel oil storage tank foundation is within the scope of 10 CFR 54.4(a)(2).

The AAC diesel generator building foundation is a seismic Category 2 structure designed to the Uniform Building Code requirements. The AAC diesel generator building is located at the north side berm of the bulk fuel oil storage tank, and is divided into two parts; an electrical equipment area and an engine room. The major components of the AAC diesel generator are located in this building (except the power distribution switchgear). The engine room houses the engine generator set, fuel oil transfer pump, fuel oil day tank, air start system, engine generator control cabinets, HVAC, and the fire protection system. The foundation of the AAC diesel generator building is a reinforced concrete slab founded on grade beams, which are supported by drilled piers (caissons). The AAC system is a non-Q system designed to conform to augmented quality assurance requirements based on NRC Regulatory Guide 1.155, “Station Blackout.” The foundation of the AAC diesel generator building is subject to an AMR because it supports the AAC diesel generator. The applicant determined that the building foundation is within the scope of 10 CFR 54.4(a)(2).

The seismic Category 1 electrical manholes are placed at various locations at the plant site. They are relatively small reinforced concrete structures founded partially underground either on natural soil or on backfill materials. An access-opening in the top slab at grade level is provided. The access-opening is covered with a reinforced concrete or carbon steel cover for missile protection. The foundations of the manholes are completely independent from other structures. The applicant determined that the electrical manholes are within the scope of 10 CFR 54.4(a)(1).

The borated water storage tank (BWST) foundation is the reinforced concrete roof slab of the boron holdup tank vault that is part of the seismic Category 1 auxiliary building. The vault roof requires a 2-ft thick slab to support the BWST, but the vault roof was designed with a 4-ft thick slab to meeting the biological shielding requirements. A small ring wall, filled with oiled sand, was built on the roof slab to separate the tank bottom from the top of the concrete surface. The roof slab has a small slope for the tank drainage purposes. The applicant determined that the BWST foundation is within the scope of 10 CFR 54.4(a)(1).

The yard structures described above are within the scope of license renewal because they perform one or more of the following yard structure intended functions:

- structural support or functional support to safety-related equipment

- shelter or protection to safety-related equipment

- fire-rated barriers to confine or retard a fire from spreading to or from adjacent areas

- missile (internal or external) barriers

- structural or functional support to non-safety-related equipment, failure of which could directly prevent satisfactory accomplishment of required safety-related functions

- protective barriers for internal flood event

The applicant lists 11 structural components, and identifies their intended functions in Table 3.6-7 of the LRA. The 11 structural components are further combined into two groups; steel and concrete. Other structural components that are part of the yard structures, and do not contribute to any of the intended functions of the yard structures, are not included in the table. The steel group includes manhole covers and threaded fasteners. The concrete group includes walls, floor slab, columns, slabs on various foundations, tank vault, drilled piers, manhole covers, and the walls and slabs of the electrical manholes. The structural components listed in the table are subject to an AMR.

2.4.6.2 Staff Evaluation

The staff reviewed Section 2.4.6.1 of the LRA to determine if the applicant has adequately implemented its methodologies such that there is reasonable assurance that the structures and structural components comprising the yard structures have been properly identified as being within the scope of license renewal and subject to an AMR. After completing its initial review, the staff requested additional information regarding yard structures in a letter to the applicant dated April 18, 2000.

In the LRA, Section 2.4.6.1, the applicant describes the aboveground and underground yard structures and trenches. However, there is no supporting information or document that can be used to verify the content of this section. The staff asked that the applicant provide a drawing that shows the location of the yard structures and trenches and highlights the components that are within the scope of license renewal.

In its response to the NRC dated August 30, 2000, the applicant submitted the following site-drawings: Drawings C31(Yard underground utilities), C-2003 (plot plan), and C-2056 (anchor bolt locations of the condensate storage tank). Using these drawings, the applicant highlighted the SCs of the yard structures that are within the scope of license renewal. The staff compared Section 2.4.6.1 and Table 3.6-7 of the LRA with these drawings, to verified that the applicant included all the SCs of the yard structures, that meet the scoping criteria of 10 CFR 54.4(a), as being within the scope of license renewal. As a result of this review, the staff found no omissions by the applicant in scoping the yard structures as defined under 10 CFR 54.4(a). The staff also found no omissions in the SCs identified in Table 3.6-7 of the LRA that are subject to an AMR in accordance with 10 CFR 54.21(a)(1).

2.4.6.3 Conclusions

On the basis of the review described above, the staff concludes that there is reasonable assurance that the applicant has appropriately identified those portions of the yard structures that are within the scope of license renewal, and the associated SCs that are subject to an AMR, in accordance with the requirements of 10 CFR 54.4(a) and 54.21(a)(1), respectively.

2.4.7 Bulk Commodities

In the LRA, Section 2.4.6.2, “Bulk Commodities,” the applicant describes the bulk commodities, and identified the commodity groupings in the buildings and structures that are within the scope of license renewal.

2.4.7.1 Technical Information in the Application

The bulk commodities are the SCs that support or protect various SSCs that are common to two or more buildings or structures. The applicant determines that the bulk commodities that are identified as being within the scope of license renewal are in the reactor building (including reactor building internals), auxiliary building, intake structure, diesel fuel vault, BWST foundation, Q-CST foundation, and pipe trenches. Some of the commodities in the turbine building, such as fire wrap banding, fire damper mountings, fire hose reels, fire wraps, and fire stops, are also included in the scope of license renewal as bulk commodities. The applicant lists the bulk commodities and their associated structures in Table 3.6-8 of the LRA that are within the scope of license renewal because they fulfill one or more of the following intended functions:

- provide structural support and functional support to safety-related equipment
- provide shelter or protection to safety-related equipment (including radiation shielding)
- provide rated fire barriers to confine or retard a fire from spreading to or from adjacent areas
- serve as missile (internal or external) barriers
- provide structural or functional support to non-safety-related equipment, failure of which could directly prevent satisfactory accomplishment of required safety-related functions
- provide protection barrier for internal or external flood events

In the LRA, Table 3.6-8, the applicant combines the bulk commodities into six groups based on the materials of construction. These groups are; steel (including weld), threaded fasteners (including structural bolt, expansion anchor, and undercut anchor), concrete (including non-shrink grout, epoxy grout, embedment, and reinforcement, but not including prestressed concrete), fire barrier, elastomer, and Teflon. No bulk commodities are associated with the material group earthen structures. These bulk commodities are subject to an AMR in accordance with 10 CFR 54.2(a)(1).

2.4.7.2 Staff Evaluation

The staff reviewed Section 2.4.6.2 and Table 3.6-8 of the LRA to determine if there is reasonable assurance that the applicant has appropriately identified and listed the bulk commodities subject to an AMR. The applicant identifies the following bulk commodities and associated structures:

Steel Group

piping and tubing supports	reactor bldg, aux bldg, intake, diesel fuel vault, pipe trenches
pipe whip restraints	reactor bldg, aux bldg
motor-operated valve supports	reactor bldg, aux bldg, intake, diesel fuel vault
hatch frames/covers	aux bldg, intake, Q-CST foundation
conduit supports	reactor bldg, aux bldg, intake, diesel fuel vault, pipe trenches
cable trays and supports	reactor bldg, aux bldg, intake
H+V duct supports	reactor bldg, aux bldg
cabinets, electrical panels and supports	reactor bldg, aux bldg, intake
equipment supports	reactor bldg, aux bldg, intake
hazard barrier curbs	aux bldg, intake
10 CFR 50.48-required banding for fire wraps	reactor bldg, aux bldg, turbine bldg
fire damper mountings and fire hose reels	reactor bldg, aux bldg, turbine bldg, intake, and diesel fuel vault

Threaded Fastener Group:

threaded fasteners on piping and tubing supports, pipe whip restraints, hazard barrier curbs, cabinets, electrical panels and supports, and the supports for MOV, conduit, H+V ducts and equipment	reactor bldg, aux bldg, intake, diesel fuel vault, pipe trenches
pipe lugs, tubing clips and the threaded fasteners for hatch frames/covers	reactor bldg, aux bldg, intake, diesel fuel vault, pipe trenches
threaded fasteners for cable trays and supports	reactor bldg, aux bldg, intake

threaded fasteners for fire damper mountings and fire hose reels	aux bldg, intake, turbine bldg, and diesel fuel vault
<u>Concrete Group</u>	
equipment pads and foundations	reactor bldg, aux bldg, intake
hatch covers and plugs	aux bldg, intake, diesel fuel vault, BWST foundation
Fire Barrier Group	
fire wraps and fire stops	reactor bldg, aux bldg, turbine bldg, diesel fuel vault
Elastomer Group	
water-stops at the construction joints of the exterior concrete walls	reactor bldg, aux bldg, diesel fuel vault, Q-CST foundation
<u>Teflon Group</u>	
piping support restraints	reactor bldg, aux bldg
equipment pad, and foundation plates	reactor bldg and aux bldg

The staff reviewed Table 3.6-8 of the LRA to determine if the applicant has adequately identified the bulk commodities in the structures that are within the scope of license renewal in accordance with 10 CFR 54.4. The staff previously reviewed Table 3.6-8 of the LRA in reviewing the reactor building, reactor building internals, and auxiliary building to verify whether the listed bulk commodities are within these buildings. The staff found that these bulk commodities are part of safety-related SSCs that are common to most nuclear power plants. The staff did not identify any omissions from the bulk commodities identified by the applicant as being subject to an AMR in accordance with the requirements of 10 CFR 54.21(a)(1).

2.4.7.3 Conclusions

On the basis of the review described above, the staff finds that there is reasonable assurance that the applicant has adequately identified the bulk commodities that are within the scope of license renewal and subject to an AMR in accordance with the requirements of 10 CFR 54,4(a) and 54.21(a)(1), respectively.

2.4.8 References for Section 2.4

1. 10 CFR Part 54, “Requirements for Renewal of Operating Licenses for Nuclear Power Plants.”

2. DG-1047, “Standard Review Plan for the Review of License Renewal Application for Nuclear Power Plants,” Working Draft, April 21, 2000.
3. Arkansas Nuclear One - Unit 1, License Renewal Application dated January 31, 2000.
4. ANO-1 Updated Final Safety Analysis Report.

2.5 Electrical and Instrumentation and Controls Systems Scoping and Screening Results

2.5.1 Introduction

The applicant describes it's methodology and process used to identify electrical and instrumentation and controls (EIC) SSCs that are within the scope of license renewal and subject to an AMR in Section 2.1, "Scoping and Screening Methodology," and Section 2.2, "Plant-Level Scoping Results," of the LRA. The list of systems that contain EIC components are documented in Table 2.2-1, which identified the mechanical and electrical systems included within the scope of license renewal. The applicant includes an integrated plant assessment (IPA) that is largely consistent with the guidance recommended in the Nuclear Energy Institute (NEI) document NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 — The License Renewal Rule." The EIC SCs that are subject to an AMR from the SSCs that are within the scope of license renewal are identified in Section 2.5, "Electrical and Instrumentation and Controls System Scoping and Screening Results." The NRC staff reviewed this information to determine whether the applicant has adequately demonstrated that the requirements of 10 CFR 54.4, 10 CFR 54.21(a)(1), and 54.21(a)(2) have been met for electrical SSCs that are within the scope of license renewal and the SCs subject to an AMR.

2.5.2 Scoping of Electrical Systems, Structures, and Components

The applicant's process for identifying electrical components that are subject to an AMR began with a list of all ANO-1 electrical systems. The applicant then performed an assessment to identify and list SSCs that satisfy the criteria under 10 CFR 54.4(a)(1) for safety-related SSCs that are relied upon to remain functional during and following DBEs (as defined in 10 CFR 50.49(b)(1)) to ensure the following capabilities are maintained:

- the integrity of the reactor coolant pressure boundary
- the capability to shut down the reactor and maintain it in a safe shut down condition
- the capability to prevent or mitigate the consequences of accidents that could result in potential off-site exposures comparable to the 10 CFR 100 guidelines

Table 1-2 of the ANO-1 UFSAR, identified the "safety-related" or "Q" systems and components required by the applicant's CLB. The ANO-1 Q-lists include those SSCs relied upon to remain functional during or following DBEs described in ANO-1 UFSAR Chapter 14. The ANO-1 UFSAR Chapter 14 events were based on criteria identical to the scoping criteria specified under 10 CFR 54.4(a)(1).

The scoping criterion of 10 CFR 54.4(a)(2) requires that all non-safety-related SSCs whose failure could prevent satisfactory accomplishment of any of the intended functions of safety-related SSCs be included in the scope of license renewal. The ANO-1 Q-list includes those non-safety-related SSCs whose failure could prevent satisfactory accomplishment of any of the intended functions of safety-related SSCs in accordance with the criteria in 10 CFR 54.4(a)(2).

In addition, 10 CFR 54.4(a)(3) requires that all SSCs relied on in safety analyses or plant evaluations to perform an intended function that demonstrates compliance with Commission regulations for: environmental qualification (10 CFR 50.49), anticipated transients without scram (10 CFR 50.62), and station blackout (10 CFR 50.63), fire protection (10 CFR 50.48), and, pressurized thermal shock (10 CFR 50.61). In summary, the applicant states that the following SSCs have been included within the scope of license renewal:

- components in the EQ program

- components in the diverse reactor over-pressure protection system/diverse SCRAM system (DROPS/DSS) and the DROPS/ATWS mitigation system actuation circuit (AMSAC)

- electrical commodities needed for the AAC diesel generator to perform its intended function

- fire protection equipment necessary to ensure one train of safe shutdown equipment remains free of fire damage, including emergency lighting and selected non-safety-related components

- electrical components necessary to protect against pressurized thermal shock

In the LRA, Table 2.2-1, the applicant identifies the systems, which contain EIC components, that are within the scope of license renewal. The electrical components requiring an AMR are discussed in Section 2.5.3 of the LRA. From the list of SSCs included within the scope of license renewal, the applicant identifies the intended function(s) and eliminates the structures, components, and component types that required moving parts, or a change in configuration or properties that perform those intended functions, as allowed by 10 CFR 54.21(a)(1). Next, the applicant eliminates the structures, components, and component types subject to replacement based on qualified life or specified time period as allowed by 10 CFR 54.21(a)(1)(ii). The remaining electrical components are subject to an AMR. The NRC staff evaluation of the EIC components subject to an AMR is discussed in Section 2.5.3 of this safety evaluation.

After completing its initial review of the ANO-1 LRA, the staff issued RAIs on April 17, April 25, and May 2, 2000. The response to the NRC RAIs was received on July 31, 2000. In its response dated July 31, 2000, the applicant states that NEI 95-10, Appendix B, lists typical structures, components, and commodity groupings that are applicable to an integrated plant assessment. The EIC components and component types identified in NEI 95-10, Appendix B, are representative of the components included within the scope of license renewal as identified in the applicant's response to the RAIs dated July 31, 2000. EIC components that are within the scope of license renewal at ANO-1 include the following:

power supplies # $
circuit breakers * # $
switchgear #
load centers
motor control centers
batteries * # $
cables * # $
connectors * # $

terminal blocks * # $
splices * # $
relays * # $
sensors
electrical bus
insulators
transmitters * # $
meters

diesel generators #
indicators * # $
switches * # $
controllers * #
detectors * #
transformers # $
battery chargers * # $
lights * #
annunciators # $
inverters #
motors * #
solenoid operators * #
alarm units
converters # $
isolators $
signal conditioners # $
recorders #
transducers
motor-generators
heat tracing
electric heaters

* - electrical component types included in fire protection system
# - electrical component types included in the AAC system
$ - electrical component types included in the ATWS system

In the LRA, Section 2.5.3, the applicant also states that the only components subject to an AMR are splices, connectors, terminal blocks, and cables. The “spaces” approach developed for the U.S. Department of Energy by the Sandia National Laboratory was used to perform the AMR for these component types at ANO-1. (Note: EIC components that perform a pressure boundary intended function are considered in the mechanical sections, and structural components such as electrical panels and cabinets are considered in the structural sections.)

2.5.2.1 Environmental Qualification Systems, Structures, and Components

In the LRA, Section 2.5.2.1, “EQ SSCs,” the applicant states that safety-related components that must continue to operate following accidents and high-energy line breaks (HELBs), and that are located in harsh environments resulting from that accident or HELB, are controlled by the environmental qualification (EQ) program. The applicant also states that the EQ Program tracks both components with individual equipment numbers and generic components used throughout the plant (such as cables) and that all long-lived, passive EQ electrical components and commodities located in a harsh environment, which are important to safety, including safety-related and Q-list equipment, non-safety-related equipment whose failure could prevent satisfactory accomplishment of any safety-related function, and the necessary post-accident monitoring equipment is included within the scope of the EQ program and within the scope of license renewal.

In addition, the applicant states that a detailed discussion of the EQ program, and the components covered by the program is contained in Section 4.4 of the LRA. The NRC staff’s evaluation and findings, including a detailed discussion of the EQ Program, and the components covered by the EQ program accordance with 10 CFR 54.21(c), are provided in Section 4.4 of this SER.

2.5.2.2 Anticipated Transient Without Scram Electrical Systems, Structures, and Components

In the LRA, Section 2.5.2.2, “ATWS Electrical SSCs,” the applicant describes the anticipated transient without scram (ATWS) SSCs that are within the scope of license renewal, and identified the electrical SSCs that are subject to an AMR. The staff reviewed this information to determine if there is reasonable assurance that the applicant has identified and listed the SCs associated with ATWS that are subject to an AMR.

2.5.2.2.1 Summary of Technical Information in the Application

In the LRA, Section 2.5.2.2, the applicant states that, in 1990, ANO-1 installed a DROPS/DSS for a diverse reactor trip, and DROPS/AMSAC for a backup actuation of EFW and a diverse main turbine trip. The applicant also states that these systems are in compliance with 10 CFR 50.62, and that the electrical components in the DROPS/DSS and the DROPS/AMSAC are within the scope of license renewal. The AMR includes the cabling associated with field sensors (pressure, flow, and reactor power) that supply input to these systems. The applicant also states that these are small, non-Q, self-contained, microprocessor-based systems with signal isolators connected to the RCS pressure, nuclear instrumentation reactor power, and main feedwater flow signals. Trip relays are installed for interfacing with the plant components. In summary, the applicant states that electronics, in general, are considered active and, therefore, are not subject to an AMR.

2.5.2.2.2 Staff Evaluation

The NRC staff reviewed the scoping results provided in Sections 2.1, 2.2, and 2.5 of the LRA. After the initial review, the NRC staff requested additional information in letters to the applicant dated April 17, 2000, April 25, 2000, and May 2, 2000. The applicant provides Table 2.2-1 of the LRA that contains a list of systems which are in scope of license renewal. However, the applicant did not provide a list of electrical and instrumentation and control component types for the systems identified. The staff requested the applicant provide a list of electrical and instrumentation and control component types that are within the scope of license renewal for the systems identified in Table 2.2-1 and identify in the list the components that are part of ATWS SSCs. The applicant responded to these RAIs in a letter to the NRC dated July 31, 2000. In its response, the applicant provided a list of electrical and instrumentation and control component types that are within the scope of license renewal and identified in the list the components that are specifically part of the ATWS system. The NRC staff reviewed the information in Section 2.5.2.2 of the LRA and additional information provided by the applicant to verify that the applicant identified the ATWS electrical SSCs that are within the scope of license renewal in accordance with 10 CFR 54.4, and did not identify any omissions.

2.5.2.2.3 Conclusions

On the basis of the review described above, the NRC staff finds that there is reasonable assurance that the applicant has identified the ATWS electrical SSCs that are within the scope of license renewal, and the associated SCs that are subject to an AMR, in accordance with the requirements of 10 CFR 54.4(a) and 54.21(a)(1), respectively.

2.5.2.3 Station Blackout Electrical Systems, Structures, and Components

In the LRA, Section 2.5.2.2, “Station Blackout Electrical SSCs,” the applicant describes the Station Blackout (SBO) electrical SSCs that are within the scope of license renewal, and identified the electrical SCs that are subject to an AMR. The staff reviewed this information to determine if there is reasonable assurance that the applicant has identified and listed the SCs associated with SBO that are subject to an AMR.

2.5.2.3.1 Summary of Technical Information in the Application

In the LRA, Section 2.5.2.3, the applicant states that in order to meet the requirements of 10 CFR 50.63, it installed a 4,400-kW diesel generator in a separate structure that is totally independent of the other emergency power sources and their auxiliaries. The system is referred to as the “AAC diesel generator” or as the “station blackout diesel,” and can be used to power the class 1E electrical buses of both ANO units. In summary, the applicant states that the electrical components of the AAC diesel generator that supply the Class 1E buses during a potential station blackout are included within the scope of license renewal. Specific components associated with the AAC diesel generator systems which require an AMR are discussed in Section 2.5.3 of the ANO-1 LRA.

2.5.2.3.2 Staff Evaluation

The NRC staff reviewed the scoping results presented in Sections 2.1, 2.2, and 2.5 of the ANO-1 LRA. After the initial review, the NRC staff requested additional information in letters to the applicant dated April 17, April 25, and May 2, 2000. The applicant provides Table 2.2-1 of the LRA that contains a list of systems which are in scope of license renewal. However, the applicant did not provide a list of electrical and instrumentation and control component types for the systems identified. The staff requested that the applicant provide a list of electrical and instrumentation and control component types that are within the scope of license renewal for the systems identified in Table 2.2-1 and identify in the list the components that are part of station blackout SSCs. The applicant responds to the staff’s RAIs in a letter to the NRC dated July 31, 2000. In its response, the applicant provides a list of electrical and instrumentation and control component types that are within the scope of license renewal, and identifies those components that are specifically part of the station blackout system. The NRC staff reviewed Section 2.5.2.3 of the LRA and the additional information to verify that the applicant identified the SBO electrical SSCs that are within the scope of license renewal in accordance with 10 CFR 54.4(a)(3), and did not identify any omissions.

2.5.2.3.3 Conclusions

On the basis of the review described above, the NRC staff finds that there is reasonable assurance that the applicant has adequately identified the SBO electrical SSCs that are within the scope of license renewal, and the associated SCs that are subject to an AMR, in accordance with the requirements of 10 CFR 54.4(a) and 54.21(a)(1), respectively.

2.5.3 Screening of Electrical Systems, Structures, and Components

In the LRA, Section 2.5.3, the applicant states that as part of the IPA for license renewal, only those SSCs that are long-lived, passive, and within the scope license renewal are subject to an AMR. The applicant also states that components at ANO-1 were categorized as long-lived and passive using NEI 95-10, Appendix B, as a guide. On that basis, the applicant identifies the following electrical component groups as being subject to an AMR at ANO-1: splices, connectors, terminal blocks, and cables. In addition, the applicant further reduced the population of these electrical component groups requiring an AMR by eliminating those piece-parts that are part of a larger complex assembly (e.g., the wiring, terminal blocks, and connectors located internal to a circuit breaker cubicle).

In its response to NRC RAIs dated July 31, 2000, the applicant also states that items physically supporting or protecting electrical equipment that are within the scope of license renewal are discussed in the structural sections of the LRA. For the in-scope battery racks, which are unique to the auxiliary building, refer to the ANO-1 LRA, Sections 2.4.3 and 3.6, and Table 3.6-4. Cabinets, electrical panels, and supports are considered bulk commodities (i.e., common to more than one in-scope structure), and are evaluated in the LRA, Sections 2.4.6.2 and 3.6, and Table 3.6-8.

2.5.3.1 Connectors

2.5.3.1.1 Summary of Technical Information in the Application

In the LRA, Section 2.5.3, the applicant states that connectors are generally considered to be “plug and socket” arrangements that allow easy disconnecting and reconnecting of the electrical components that are long-lived, passive, and subject to an AMR. In addition, the applicant considers cable splices, cable couplers, and insulating tape used in splices as components or sub-components of the connector commodity group that are subject to an AMR.

2.5.3.1.2 Staff Evaluation

The NRC staff reviewed the screening results presented in Section 2.5.3 of the LRA. to verify that the applicant identified all of the electrical components that needed to be included within the connector commodity group, and that are subject to an AMR in accordance with the requirements of 10 CFR 54.21(a)(1). The staff did not identify any omissions.

2.5.3.1.3 Conclusions

On the basis of the review described above, the NRC staff finds that there is reasonable assurance that the applicant has adequately identified electrical connector components that are subject to an AMR consistent with the requirements of 10 CFR 54.21(a)(1).

2.5.3.2 Terminal Blocks

2.5.3.2.1 Summary of Technical Information in the Application

In the LRA, Section 2.5.3, the applicant states that terminal blocks at ANO-1 are molded, solid-section, phenolic blocks capable of withstanding considerable temperature and radiation exposures. The applicant also states that terminal blocks are passive, long-lived electrical components. Therefore, those terminal blocks that are within the scope of license renewal, and that are not piece-parts of larger active assemblies, are subject to an AMR.

2.5.3.2.2 Staff Evaluation

The NRC staff reviewed the screening results relating to terminal blocks presented in Section 2.5.3 of the ANO-1 LRA. The NRC staff reviewed this information to verify that the applicant identified the terminal blocks that are subject to an AMR in accordance with the requirements of 10 CFR 54.21(a)(1), and did not identify any omissions.

2.5.3.2.3 Conclusions

On the basis of the review described above, the NRC staff finds that there is reasonable assurance that the applicant has adequately identified the electrical terminal blocks that are subject to an AMR consistent with the requirements of 10 CFR 54.21(a)(1).

2.5.3.3 Cables

2.5.3.3.1 Summary of Technical Information in the Application

In the LRA, Section 2.5.3, the applicant states that an insulated cable is an assembly of a single electrical conductor (wire) that is covered with insulation, or a combination of conductors that are insulated from one another, and have an overall covering. Cable connections are used to connect the cable conductors to other cables or electrical devices and include connectors, splices, and terminal blocks. Cables in the scope of this review are those that are separate components and not part of some larger complex assembly.

2.5.3.3.2 Staff Evaluation

The NRC staff reviewed the screening results relating to cables presented in Section 2.5.3 of the ANO-1 LRA. After the initial review, the NRC staff requested additional information in letters to the applicant dated April 17, 2000, April 25, 2000, and May 2, 2000. On the basis of the information in 10 CFR 54.21, NEI 95-10, Appendix B, and Section 2.5.3.3 of the LRA, which identifies cables and connectors as being subject to an AMR, more detail with respect to type and categorization of cables in the scope of license renewal is needed for the staff to perform its evaluation. The staff requested the applicant to identify specifically where in the LRA each cable type including connections (e.g., connectors, terminal blocks, and splices) is addressed in the LRA. The applicant responded to these RAIs in a letter to the NRC dated July 31, 2000. In its response, the applicant states that the various types of cables and electrical connection at ANO-1 that are part of in-scope systems are subject to an AMR. The cable types include power cable, instrument cable, communication cable, and uninsulated cable. Connection types include splices, connectors, and terminal blocks. While the LRA did not list the individual cable and connection types, and AMR was performed for all types using the spaces approach from the DOE/Sandia aging management guideline. Table 3.7-1 of the LRA provides the results of this AMR. The NRC staff reviewed the information in Section 2.5.3.3 and additional information provided by the applicant to verify that the applicant identified the cables that are subject to an AMR in accordance with the requirements of 10 CFR 54.21(a)(1), and did not identify any omissions.

2.5.3.3.3 Conclusions

On the basis of the review described above, the NRC staff finds that there is reasonable assurance that the applicant has adequately identified electrical cables that are subject to an AMR consistent with the requirements of 10 CFR 54.21(a)(1).

2.5.3.4 Electrical Bus

2.5.3.4.1 Summary of Technical Information in the Application

In the LRA, Section 2.5.4.1, the applicant states that electrical buses at ANO-1 are not in the scope of license renewal or are not subject to an AMR due to the fact that they are of a larger complex assembly or they are not safety related. The isolated-phase bus that connects the main generator to the main transformers is not safety related. The switchyard bus is likewise not safety-related. In addition, the applicant states that some safety-related 4.16-kV bus is contained within the safety-related 4.16-kV switchgear, however this bus is considered a piece-part of this switchgear. This switchgear is a large complex assembly containing the 4.16-kV bus, breakers, relays, wiring and controls. Because switchgear are considered active components in accordance with the requirements of 10 CFR 54.21(a)(1)(i), this bus is not subject to an AMR.

2.5.3.4.2 Staff Evaluation

The NRC staff reviewed the screening results related to buses presented in Section 2.5.4 of the ANO-1 LRA. After the initial review, the NRC staff requested additional information in letters to the applicant dated April 17, 2000, April 25, 2000, and May 2, 2000, regarding electrical buses under Section 2.5.4.1 of the LRA. Electrical buses were generically excluded from the scope of license renewal based on the characterization that those buses were not safety-related. A component cannot be excluded simply because they are non-safety-related. Any component, including an electrical bus, that is non-safety-related but whose failure could prevent satisfactory accomplishment of the function identified in 10 CFR 54.4(a)(2) or a(3), needs to be included within the scope of license renewal. The staff requested the applicant to provide a justification for excluding electrical buses from the scope of license renewal. The applicant responded to these RAIs in a letter to the NRC dated July 31, 2000. In its response, the applicant states that it has re-reviewed the electrical buses not included within the scope of license renewal and verified that these buses do not meet the criteria under 10 CFR 54.4(a)(1), (a)(2) or (a)(3). The applicant's response resolved the staff's concern. The NRC staff reviewed Section 2.5.4.1of the LRA and the additional information provided by the applicant to verify that there are no buses that are within the scope of license renewal and subject to an AMR in accordance with the requirements of 10 CFR 54.4, and 54.21(a)(1), respectively, and did not identify any buses requiring an AMR.

2.5.3.4.3 Conclusions

On the basis of the review described above, the NRC staff finds that there is reasonable assurance that the applicant has adequately verified that non-safety-related electrical buses are not within the scope of license renewal and the safety-related 4.16-kV bus is not subject to an AMR consistent with the requirements of 10 CFR 54.21(a)(1).

2.5.3.5 Insulators

2.5.3.5.1 Summary of Technical Information in the Application

In the LRA, Section 2.5.4, the applicant states that electrical insulators associated with the ANO-1 switchyard are not within the scope of license renewal since they are not safety-related. Other insulators found in the plant are either not safety-related or are part of a larger complex assembly.

2.5.3.5.2 Staff Evaluation

The NRC staff reviewed the scoping and screening results relating to insulators presented in Section 2.5.4 of the ANO-1 LRA. After the initial review, the NRC staff requested additional information in letters to the applicant dated April 17, April 25, and May 2, 2000, regarding the insulators under Section 2.5.4.2 of the LRA. Insulators were generically excluded from an AMR on the characterization that they were "part of a larger complex assembly or not safety-related." A component can not be excluded from an AMR simply because it is part of a larger complex assembly. If a complex assembly is within the scope of License Renewal Rule, and component within that complex assembly is determined to be passive and long-lived, that component should be subject to an AMR. In addition, any component that is non-safety-related but whose failure could prevent satisfactory accomplishment of the functions identified in 10 CFR 54.4(a)(1), need to be included within the scope of license renewal. If any non-safety-related component within the scope of license renewal perform its intended function(s) without moving parts or without a change in configuration or properties, and is not replaced based on qualified life or specified time period, is subject to an AMR. The staff request the applicant provide a justification for excluding the insulators discussed in Section 2.5.4.2 of the LRA. The applicant responded to these RAIs in a letter to the NRC dated July 31, 2000.

In response to the NRC staff RAIs, the applicant states that "at ANO-1, only the 500 kV system contains insulators that are considered to be separate components." As identified in the RAI response, only the circuit breakers that provide an interface between the 500 kV system and other systems are within the scope of license renewal. None of the insulators in the 500 kV system are in scope of license renewal because they are not safety-related and do not meet the criteria of 10 CFR 54.4(a)(2) and 54.4(a)(3).

Many other components at ANO-1 contain parts that serve as insulating devices. However, all of these components, such as load centers, motor control centers, switchgear, and distribution panels, are active components and thus are not subject to an AMR. The staff reviewed the applicant's response and found that it resolved the staff's concern.

The NRC staff reviewed the information in Section 2.5.4.2 and additional information provided by the applicant and found that, except for insulators in circuit breakers that provide an interface between the 500 kV system and other systems that are within the scope of license renewal, no other insulators in the 500-kV system are within the scope of license renewal. The insulators that are in scope of license renewal are considered part of an active component assembly and do not required an AMR in accordance with 10 CFR 54.21(a)(1).

2.5.3.5.3 Conclusions

On the basis of the review described above, the NRC staff finds that there is reasonable assurance that the applicant has adequately verified that no insulators are subject to an AMR consistent with the requirements of 10 CFR 54.21(a)(1).

2.5.3.6 Transmission Conductor

2.5.3.6.1 Summary of Technical Information in the Application

In the LRA, Section 2.5.4, the applicant states that transmission conductors at ANO-1 do not meet the scoping criteria of 10 CFR 54.4(a), and, therefore, are not in the scope of license renewal, and not subject to an AMR.

2.5.3.6.2 Staff Evaluation

The NRC staff reviewed the scoping and screening results relating to transmission conductors presented in Section 2.5.4 of the ANO-1 LRA. The NRC staff reviewed this information to verify that there are no transmission conductors that are within the scope of license renewal, and therefore, are not subject to an AMR in accordance with the requirements of 10 CFR 54.4 and 54.21(a)(1), respectively, and did not identify any insulators requiring an AMR.

2.5.3.6.3 Conclusions

On the basis of the review described above, the NRC staff finds that there is reasonable assurance that the applicant has adequately verified that no transmission conductors are subject to an AMR consistent with the requirements of 10 CFR 54.21(a)(1).

2.5.4 References for Section 2.5

1. 10 CFR 50.48, “Fire Protection.”
2. 10 CFR 50.49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants.”
3. 10 CFR 50.61, “Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events.”
4. 10 CFR 50.62, “Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants.”
5. 10 CFR 50.63, “Loss of All Alternating Current Power.”
6. 10 CFR Part 54, “Requirements for Renewal of Operating Licenses for Nuclear Power Plants.”
7. DG-1047, “Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants,” Working Draft, April 21, 2000.
8. “Arkansas Nuclear One - Unit 1, License Renewal Application” dated January 31, 2000.
9. “ANO-1 Updated Final Safety Analysis Report.”
10. NEI 95-10, “Industry Guideline for Implementing the Requirements of 10 CFR Part 54–The License Renewal Rule,” Revision 1, January 2000.

3 AGING MANAGEMENT REVIEW RESULTS

3.1 Introduction

In general, the applicant began its aging management review (AMR) by identifying applicable aging effects for the structures and components (SCS) that are subject to an AMR by reviewing the potential aging effects identified in industry literature as defined in the applicant's "tools" for identifying aging effects. The aging effect "tools" are summarized in Appendix C, "Process for Identifying Aging Effects Requiring Aging Management for Non-Class 1 Mechanical Components," of the license renewal application (LRA). From this set of potential aging effects, the component materials, operating environment, and operating stresses were used to determine the applicable aging effects for each component subject to an AMR. These applicable aging effects were then validated by a review of industry and Arkansas Nuclear One, Unit 1 (ANO-1) specific operating experience to ensure that all applicable aging effects were identified for the AMR. The aging effect tools allowed the applicant to identify applicable aging effects irrespective of the specific component type being evaluated. Different component types constructed from the same material, exposed to the same environment, and operated under similar operating stresses will experience similar aging effects. For Class 1 mechanical components, electrical components, and structures, the applicant used NRC generic communication, and industry and site-specific operating experience to identify applicable aging effects. The implementation of this process, and the applicable aging effects identified for the different SCs that are subject to an AMR are presented throughout Section 3.0 of the LRA.

The applicant then reviewed existing programs and activities for the SCs that are subject to an AMR, and identified those programs that can be used to manage the applicable aging effects. They evaluated the effectiveness of the existing programs and activities by reviewing the operating experience of the different applications. The applicant either identified a demonstration of the effectiveness of the different programs and activities consistent with 10 CFR 54.21(a)(3), or developed new programs or activities to manage the remaining applicable aging effects. The applicant provides descriptions of the aging management programs (AMPs) used to manage the effects of aging in Appendix B, "Aging Management Programs and Activities, " of the LRA.

3.2 Summary of Technical Information in the Application

The applicant describes its AMR of the mechanical SCs for license renewal in Exhibit A of the LRA, Section 3.2. "Reactor Coolant Systems," Section 3.3, "Engineered Safeguards," Section 3.4, "Auxiliary Systems," and Section 3.5, " Steam and Power Conversion System." The AMR of structural and electrical SCs is described in Section 3.6, "Structures and Structural Components," and Section 3.7, "Electrical and Instrumentation and Controls." In addition, the applicant describes its "tools" for identifying applicable aging effects in Appendix C, "Process for Identifying Aging Effects Requiring Aging Management for Non-Class 1 Mechanical Components," of the LRA; and the applicant describes its AMPs in Appendix B, "Aging Management Programs and Activities," of the LRA.

In addition, to the information provided in the LRA, the applicant provided system drawings to assist in the review of the LRA. Reviewers had a controlled set of technical specifications (TS) and an Updated Final Safety Analysis Report (UFSAR) available to assist them in their review.

Upon complete its initial review, the NRC staff requested any additional information needed to complete its review of the LRA. Each request for additional information (RAI) was documented in a letter to the applicant. RAI responses are documented in letters from the applicant to the NRC. Requests for additional clarifications were typically initiated in a telephone conference call between the NRC and the applicant. The specific clarifications asked of the applicant during each of the conference calls were documented in a letter to the applicant. The response to each clarification is documented in a series letter from the applicant to the NRC. Each letter is docketed for public access, and concerns of interest identified throughout the staff's evaluation are documented in this safety evaluation report (SER) along with the applicable reference.

3.3 Aging Management Review

The NRC staff's evaluation of the applicant's aging management program and activities began with a review of the AMPs that are common to more than one system that is within the scope of license renewal for ANO-1. The staff then evaluated the applicant's AMR of the specific structures, components, and commodity groups that have been identified as being subject to an AMR in Chapters 2.0 and 3.0 of the LRA, and any additional SSCs identified by the staff during its scoping and screening evaluation, audit, and inspection activities.

As part of this effort, the staff also reviewed the applicant's summary descriptions of the AMPs and the evaluations of the time-limited aging analyses (TLAAs) provided by the applicant in Appendix A, "Safety Analysis Report Supplement," of the LRA, to ensure that they are consistent with the requirements of 10 CFR 54.21(d). The staff identified a number of summary descriptions of AMPs and TLAA evaluations that need additional information to meet the intent of 10 CFR 54.21(d). This was Open Item 3.3-1 and a summary of the information requested by the staff and provided by the applicant can be found in Section 1.4 of this SER. A more detailed discussion of the additional FSAR Supplement information can be found throughout Chapter 3 and Chapter 4 of this SER, as appropriate.

3.3.1 Common Aging Management Programs

This section of the SER contains the staff's evaluation of 11 AMPs that are in Appendix B of the LRA, and are referenced as a part of the AMR for two or more of the systems and/or structures. It should be noted that the staff's conclusions on the evaluations of these 11 common AMPs may be predicated on the assumption that they are implemented in conjunction with other relevant AMPs as discussed in Sections 3.3.2 to 4.8 of this SER for managing the effects of aging for the SCs that are subject to an AMR.

The staff's evaluation of the applicant's AMPs focuses on program elements, rather than the details of specific plant procedures. To determine whether the applicant's AMPs are adequate to manage the effects of aging so that the intended functions will be maintained consistent with the current licensing basis (CLB) for the period of extended operation, the staff used 10 elements to evaluate each program and activity. The 10 elements of an effective AMP were developed as part of the staff's draft standard review plan for license renewal, which was released in 1997. This SER describes the extent to which the ten elements are applicable to a particular program or activity, and evaluates each program and activity against those elements that are determined to be applicable. On the basis of NRC experience with maintenance programs and activities, the staff concluded that conformance with the 10 elements of an AMP, or a combination of AMPs, provides reasonable assurance that an AMP (or combination of programs and activities) is demonstrably effective at managing an applicable aging effect. The following 10 elements of an effective AMP will be considered in evaluating each AMP used by the applicant to manage the applicable aging effects identified within this SER:

1. scope of program
2. preventive actions
3. parameters monitored or inspected
4. detection of aging effects
5. monitoring and trending
6. acceptance criteria
7. corrective actions
8. confirmation process
9. administrative controls
10. operating experience

In the LRA, Appendix B, Section 2.0, the applicant states that the elements involving corrective actions, confirmation processes, and administrative controls for license renewal are in accordance with the site-controlled corrective actions program pursuant to 10 CFR Part 50, Appendix B, and cover all SCs that are subject to an AMR. The staff's evaluation of the applicant's corrective action program is discussed separately, and generically evaluated in Section 3.3.1.2 of the SER.

3.3.1.1 Chemistry Control Program

In the LRA, Section 4.6 of Appendix B, the applicant describes its chemistry control program. The applicant also includes relevant materials from Section 3.2, "Reactor Coolant System," Section 3.3.2, "Aging Effects Requiring Management," for Engineered Safeguards, Section 3.4.2, "Aging Effects Requiring Management," for Auxiliary Systems, Section 3.5.2, "Aging Effects Requiring Management," for Steam and Power Conversion Systems, and Section 3.6, "Structures and Structural Components." These sections address the interaction of primary, secondary, or auxiliary systems chemistry monitoring, diesel fuel monitoring, and service water chemical control with the components in different systems, and describe the resulting aging effects. The overall objective of the chemistry control programs as described in Section 4.6 is to provide procedures for monitoring ANO-1 SCs for corrosion, deposits, structural damage, general cleanliness, appearance, and biological growth.

The staff reviewed the information in Section 4.6 of Appendix B of the LRA to determine whether the applicant has demonstrated that the chemistry control program will adequately manage the applicable aging effects so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation in accordance with 10 CFR 54.21(a)(3).

3.3.1.1.1 Summary of Technical Information in the Application

The following table identifies the chemistry control programs and the corresponding SSCs that are within the scope of the license renewal, and affected by water and diesel fuel oil chemistries. This table also includes the applicable industry codes, standards and guidelines that apply to each program.

Program Name	Industry Guideline, Code, or Standard	Systems and Major Components Included
Primary Chemistry Monitoring Program	EPRI Report TR-105714 “PWR Primary Water Chemistry Guidelines”	Reactor coolant system Borated water storage tanks Spent fuel system
Secondary Chemistry Monitoring Program	EPRI Report TR-102134-R4 (existing program) “PWR Secondary Water Chemistry Guidelines”	Main feedwater system Condensate storage system Steam generators
Auxiliary Systems Chemistry Monitoring	EPRI Report TR-107396 (existing program)“Closed Cooling Water Chemistry Guidelines”	Intermediate cooling water system Chilled water systems emergency diesel generators alternate AC diesel generator
Diesel Fuel Monitoring Program	ASTM D975-1981, “Standard Specification for Diesel Fuel Oils” and VV-F-800D, “Military Specifications”	Bulk fuel oil storage tank Emergency diesel fuel tanks Emergency diesel day tanks Fire pump diesel day tank alternate AC diesel generator day tank
Service Water Chemical Control Program	EPRI Report TR-106229 (existing program) “Service Water System Chemical Addition Guidelines”	Service water system Fire protection system

Table 1: ANO-1 Chemistry Programs with affected systems, major components, and applicable industry codes, standards and guidelines

The applicant concludes that the primary, secondary, auxiliary systems, diesel fuel, and service water programs prevent cracking, loss of material, loss of mechanical closure integrity, and fouling that could cause damage to the affected components. These programs will help maintain component integrity, and allow the associated components to perform its intended function(s) consistent with the CLB for the period of extended operation.

3.3.1.1.2 Staff Evaluation

In accordance with 10 CFR 54.21(a)(3), the staff reviewed the information included in Appendix B of the LRA regarding the applicant’s water and fuel oil chemistry control programs. Specifically, the applicant is required to demonstrate that the effects of aging associated with water and fuel oil chemistries will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation. After completing its initial review, in a letter to the applicant dated June 9, 2000, the staff requested additional information regarding the chemistry control programs. This evaluation also incorporates the information contained in the applicant’s response to the RAIs provided in a letter to the NRC dated September 12, 2000.

The applicant's chemistry monitoring activities are divided into the five AMPs listed in Table 1, above. The applicant describes the chemistry monitoring programs in Appendix B, Section 4.6, of the LRA. The purpose of these chemistry monitoring programs is to maximize long-term availability and operating life of SSCs by minimizing system corrosion, fuel corrosion, and radiation field build-up. In addition, the diesel fuel monitoring program manages the loss of material caused by general corrosion in carbon steel components exposed to fuel oil and manages fouling on the inside surface of the heat exchanger tubes caused by entrained materials in the fuel oil settling out and adhering to the heat exchanger surfaces.

[*Program Scope*] Each program manages aging of the systems and major components listed in Table 1. The applicant identifies the following analyses and activities as applicable to their respective programs:

- Primary Chemistry Monitoring Program: sampling activities and analyses used to mitigate the effects of system corrosion, fuel corrosion, and radiation build-up due to elevated levels of contaminants and oxygen.

- Secondary and Auxiliary Systems Chemistry Monitoring Program: periodic monitoring and control of known detrimental contaminants and oxygen.

- Diesel Fuel Monitoring Program: sampling activities and analyses used to control the presence of water and contamination in the system.

- Service Water Chemistry Monitoring Program: sampling activities and analyses used to mitigate the effects of system corrosion and biological activity.

The staff found the program scope to be acceptable because it is comprehensive by including the systems and major components exposed to the environments covered by the five sub-programs.

[*Preventive/Mitigative Actions*] The preventive or mitigative actions specified for these programs in the LRA are to maintain chemistry parameters within the limits specified by the plant's TS which are based on the EPRI guidelines, other industry standards, and operating experience specific to ANO-1.

In addition, the diesel fuel monitoring program was implemented to prevent the recurrence of corrosion problems experienced at ANO-1. The program provides for the monitoring of the quality of the diesel fuel oil and the addition of biocides and stabilizers, as necessary, to prevent the breakdown of diesel fuel. Monitoring also ensures that contaminants, which could cause fouling, are not present in the fuel oil system. The program provides instructions for routine tank bottom draining to remove sediment and any water accumulation. In addition, diesel tank filtration provides for removal of particulate buildup. The applicant states that the program is performed in accordance with the ASTM D975-1981, other ASTM Standards, and ANO-1 TS 4.6.1, "Diesel Generators."

The staff concluded that these actions are acceptable in preventing or minimizing corrosion, cracking, loss of material, and fouling. In reference to the diesel fuel program, the actions discussed above are adequate to control the presence of water and contamination (including microbiological) in the systems and major components included in the program.

[*Parameters Inspected or Monitored*] The chemistry monitoring parameters are based on TS requirements, EPRI and ASTM guidelines, and ANO-1 operating experience. Table 2 lists the typical parameters monitored for each sub-program:

Program Name	Parameters Monitored
Primary Chemistry Monitoring Program	dissolved oxygen, chloride, fluoride, sulfate, silica, and pH
Secondary Chemistry Monitoring Program	dissolved oxygen, chloride, fluoride, sulfate, and silica
Auxiliary Systems Chemistry Monitoring	pH, iron, copper, hardness, nitrate, and biological count
Diesel Fuel Monitoring Program	water and sediment, particulate, biological count, and sulfur
Service Water Chemical Control Program	corrosion rates, oxidants, and corrosion inhibitor

Table 2: ANO-1 Chemistry Programs with typical monitored parameters

The staff found these parameters acceptable because operating experience and the EPRI guidelines support the monitoring and control of these parameters to mitigate corrosion-related degradations and to ensure water and contamination are not present in the diesel fuel oil system.

[*Detection of Aging Effects*] The chemistry monitoring programs are preventative programs and as such are not credited for detecting aging effects. The staff found this acceptable.

[*Monitoring and Trending*] The ANO-1 chemistry programs consist of sampling parameters, locations, allowable values with specific guidance when parameter values are exceeded, and frequencies. The LRA outlined the following frequencies for each of the five sub-programs:

- Primary Chemistry Monitoring Program: daily, weekly, monthly, quarterly, or as required, depending on the plant operating conditions, the parameter involved, and the results of previous analyses.

- Secondary Chemistry Monitoring Program: continuous, daily, weekly, or as required, depending on the plant operating conditions, the parameter involved, and the results of previous analyses.

- Auxiliary Systems Chemistry Monitoring Program: weekly, monthly, or quarterly depending on the plant operating conditions, the parameter involved, and the results of previous analyses.

- Diesel Fuel Monitoring Program: monthly and quarterly. In addition, each new shipment of diesel fuel is sampled prior to unloading into the bulk fuel oil storage tank.

- Service Water Chemistry Monitoring Program: daily, twice per week, weekly, or as required, depending on the plant operating conditions, the parameter involved, and the results of previous analyses.

From analysis of the various measurements made over a period of time, trends in water and fuel oil chemistry may be established. Deviations from the established parameter ranges, detected through trending, help diagnose the environment which may cause failure of a component. The staff concludes that trending of the sampling data can provide early indication of chemistry deviations, allowing for timely corrective action.

[*Acceptance Criteria*] Acceptance criteria for the chemistry monitoring programs are part of the CLB and are established based on TS requirements, EPRI or ASTM guidelines, ANO-specific experience, and NRC Generic Letter 89-13 in the case of the service water chemistry control program. In addition, these criteria are defined by the sampling parameter, the sampling location, and the plant operating conditions. The staff found that the acceptance criteria are acceptable because they have low thresholds to allow for the early detection and correction of water chemistry deviations.

[*Operating Experience*] Operating experience with the systems covered by the chemistry programs has demonstrated the effectiveness of the program. In addition, the program is subject to continuous oversight to incorporate relevant industry and ANO-specific experience. There has been no significant chemistry-related degradation of any primary, secondary or auxiliary system component according to the applicant. However, the staff noted that there is chemistry-related degradation of the steam generators. The steam generator issues, including chemistry degradation, are discussed in the steam generator integrity AMP. As a result of operating experience, the applicant's corrective action program facilitated the development of the diesel fuel monitoring program and effective changes in the service water chemistry control program.

On the basis of the incorporation of EPRI and ASTM guidelines, acceptable operating experience to date, and the effective use of the applicant's corrective action program to address problems, the staff found that the chemistry programs provide reasonable assurance that the aging effects will be managed to allow components within scope to perform their intended functions for the period of extended operation.

3.3.1.1.3 Conclusions

On the basis of this review, the staff finds that there is reasonable assurance that the chemistry monitoring programs will adequately manage the effects of aging associated with loss of material, loss of mechanical closure integrity, cracking and fouling so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

3.3.1.2 Quality Assurance Program

In accordance with 10 CFR 54.21(a)(3), an applicant is required to demonstrate that the effects of aging on SCs that are subject to an AMR will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation. Consistent with this approach, Section A.2, “Quality Assurance for Aging Management,” of the NRC’s “Draft Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants” states that an applicant’s AMPs should contain the elements of corrective actions, confirmation processes, and administrative controls in order to ensure proper aging management.

3.3.1.2.1 Summary of Technical Information in the Application

In the LRA, Section 3.1.2, “Quality Assurance,” the applicant states that information regarding Quality Assurance is contained in Section 2.0, “Program and Activity Attributes,” of Appendix B of the LRA. This section of the LRA provides a description of two major renewal activities:

- identification of plant-specific programs and activities that will manage the effects of aging

- demonstration of the aging management programs and activities

The ANO-1 programs and activities that are credited with managing the effects of aging can be divided into new and existing programs and activities. The applicant uses a series of specific attributes to describe these programs and activities. Specifically, the LRA, Section 2.0 of Appendix B defines the following attributes:

[Purpose] A clear statement of the reason why the program or activity exists for ANO-1 license renewal.

[Scope] A description of the ANO-1 SCs that are within the scope of license renewal and subject to an AMR that are encompassed by the program or activity.

[Aging Effects] A description of the aging effects requiring management, or the relevant physical conditions to be monitored for the SCs subject to an AMR.

[Method] A description of the types of actions, or techniques used to identify or manage the effects of aging or relevant conditions (e.g., visual examination of the component).

[Sample Size] For new programs or activities, a sample can be identified from the total population of affected SCs for inspection or monitoring. If a sample is chosen for inspection or monitoring, a description of the sample is provided.

[Industry Codes or Standards] A description of an industry code (e.g., ASME Section XI, or IEEE) or an industry standard (e.g., ASTM or NRC-approved B&WOG report) that guides or governs the program or activity. This attribute may not be applicable to some programs and activities.

[Frequency] A description of the frequency of action that is established for detection of aging effects or relevant physical conditions.

[Acceptance Criteria or Standard] Descriptions of the acceptance criteria or standards applicable to the relevant conditions to be monitored or the chosen examination methods.

[Timing of New Program or Activity] An identification of the specific timing for the implementation or modification of a new or modified program or activity.

[Regulatory Basis] An identification of any regulatory basis for existing programs and activities, such as the TS. This attribute may not be applicable to some programs and activities.

[Operating Experience and Demonstration] A demonstration that an existing program or activity can adequately manage the effects of aging, as well as any operating experience that is relevant to the demonstration.

[Demonstration] A demonstration that a new program or activity can adequately manage the effects of aging.

In addition to these attributes, corrective actions and administrative controls are common to all programs and activities that are used to manage the effects of aging. The applicant's quality assurance program also applies to all ANO-1 safety-related SCs. All corrective actions and administrative (document) control apply to AMPs and activities for non-safety-related SCs that are subject to an AMR.

The attributes that the applicant selects to describe its AMPs are founded on the attributes defined in NEI 95-10, Revision 0, Sections 4.2 and 4.3, and the NRC's Draft Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants.

Summary descriptions of new and existing programs and activities are contained in the applicant's UFSAR Supplement, which is provided as Appendix A of the LRA. In addition, Appendix B of the LRA contains a detailed description of the applicant's programs and activities that are credited for managing the effects of aging.

3.3.1.2.2 Staff Evaluation

In reviewing the LRA, the NRC staff evaluated the process described by the applicant to demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation. Specifically, the staff reviewed the AMR and the AMPs to verify that the applicant's processes adequately address the elements of corrective actions, confirmation processes, and administrative controls in order to ensure proper management of applicable aging effects.

The NRC draft standard review plan (SRP), Section A.2, "Quality Assurance for Aging Management Programs," states that 10 CFR Part 50, Appendix B (Appendix B) requirements, which apply to safety-related SCS, are adequate to address the corrective actions, confirmation processes, and administrative control elements of an AMP for license renewal. An applicant may expand the scope of its Appendix B program to include non-safety-related SCS, or may choose an alternative approach to address corrective actions, confirmation processes, and administrative controls.

The LRA, Appendix B, Section 2.0, states that the scope of Appendix B SCs has been expanded to cover those non-safety-related SCs that are within the scope of license renewal. In a letter dated June 9, 2000, the staff requested additional information to determine how the applicant addressed the quality assurance aspects of the draft SRP for non-safety-related SCs. In its response to the NRC dated September 12, 2000, the applicant states that the corrective action program and the administrative (document) controls, which govern the applicable site procedures, are in accordance with the site-controlled quality assurance program pursuant to 10 CFR Part 50, Appendix B. These programs apply to the non-safety-related SCs that are within the scope of license renewal.

The staff also determined that the description in the LRA, Appendix B, Section 2.0, does not specifically include the confirmation process as an attribute of each AMP governed by the ANO-1 quality assurance program. In a letter dated June 9, 2000, the NRC staff requested additional information to determine how the applicant was addressing the confirmation process attribute for each applicable AMPs. In its response dated September 12, 2000, the applicant stated that the confirmation process is part of the ANO-1 corrective action program and is in accordance with the site-controlled quality assurance program pursuant to 10 CFR Part 50, Appendix B. Thus the confirmation process applies to all of the AMPs and activities described in Appendix B of the ANO-1 LRA, as well as to all of the SCs that are within the scope of license renewal and subject to an AMR.

In the LRA, Appendix A, “Safety Analysis Report Supplement,” the applicant did not provide a summary description for the quality assurance AMP. Therefore, the staff requested that the applicant provide a summary description that adequately describes the corrective action program, which includes the confirmation process, and administrative control program in accordance with 10 CFR Part 50, Appendix B, as it applies to license renewal in accordance with 10 CFR 54.21(d). This was FSAR Item 3.3.1.2.3 of Open Item 3.3-1. In a letter to the NRC dated March 14, 2001, the applicant provided a revised FSAR Supplement that contained a summary description of the program and activities applicable to the corrective action program. The staff found the applicant’s summary description of the corrective action program acceptable, and considers FSAR Item 3.3.1.2.3 of Open Item 3.3-1 resolved.

3.3.1.2.3 Conclusions

On the basis of the review described above, the NRC staff finds that there is reasonable assurance that the applicant’s AMPs and activities contain the necessary aspects of quality assurance, including the elements of corrective actions, confirmation processes, and administrative controls, in order to ensure proper management of applicable aging effects. Further, the staff finds that the applicant’s quality assurance description in the LRA, as supplemented by its response (dated September 12, 2000) to the NRC staff’s RAI, meets the requirements of 10 CFR 54.29, and is consistent with the guidance in the applicable sections of the draft SRP and, therefore, is acceptable.

3.3.1.3 Structures and System Walkdowns Program

In the LRA, Appendix B, Section 4.13, the applicant describes its structures and system walkdowns AMP, which it referred to as the Maintenance Rule program. The Maintenance Rule program assesses the overall conditions of buildings and structures and identifies any ongoing degradation through a visual inspection process. The staff reviewed the LRA to determine whether the applicant demonstrates that the aging effects of the SCs that are within the scope of license renewal will be adequately managed by this inspection program during the period of extended operation as required by 10 CFR 54.21(a)(3).

3.3.1.3.1 Summary of Technical Information in the Application

In the LRA, Sections 3.4, 3.5, and 3.6, and Appendix B, Section 4.13, the applicant describes the Maintenance Rule program. The Maintenance Rule program is responsible for the management of the following applicable aging effects: loss of material, cracking, change in material properties, and loss of mechanical closure integrity. The applicant identifies the aging effects for steel components and commodities, concrete components and commodities, prestressed concrete components, and threaded fasteners for the following structures or component groups:

- reactor building
- auxiliary building
- intake structure
- yard structures
- bulk commodities

The applicant also states that the inspection of coated surfaces of the SCs, which are within the scope of license renewal and subject to an AMR, is also part of the Maintenance Rule program. Additional guidance for coatings inspections will be incorporated into the existing Maintenance Rule program guidelines during a future revision of the applicant’s System Engineering Desk Guide, which must take place before the end of the initial 40-year license term for ANO-1.

Each structure or component is visually inspected from the interior and exterior if accessible at a frequency that varies depending on the structure or component being inspected. The applicant states that the acceptance criteria for this program are, “no unacceptable visual indications of cracking, loss of material, or change of material properties of structures or components.”

The Maintenance Rule program is an existing program. This program has identified unacceptable indications of aging, which has resulted in implementation of corrective actions. Since the Maintenance Rule program incorporates proven monitoring techniques, acceptance criteria, corrective actions, and administrative controls from existing programs and procedures, the applicant states that the Maintenance Rule program will adequately manage the applicable aging effects so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

3.3.1.3.2 Staff Evaluation

In accordance with 10 CFR 54.21(a)(3), the staff reviewed the information included in the sections listed above of the LRA, regarding the applicant's demonstration of the Maintenance Rule program to ensure that the effects of aging discussed above will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation for all SCs that are subject to an AMR with the Maintenance Rule as an AMP.

[*Program Scope*] The applicant lists the structures, components, and commodities covered by the Maintenance Rule AMP in Sections 3.4, 3.5, and 3.6 of the LRA. The staff found that the scope of the structural monitoring program is acceptable since it includes a walkdown inspection of all SCs that are within the scope of license renewal.

[*Preventive or Mitigative Actions*] The applicant did not identify any specific preventive or mitigative activities, nor did the staff identify a need for such action.

[*Parameters Inspected or Monitored*] The Maintenance Rule program requires a visual inspection of SCs. In the LRA, the applicant did not identify a procedure for inspecting normally inaccessible areas. In a letter to the NRC dated September 7, 2000, the applicant states that the Maintenance Rule program provides for walkdowns of accessible areas and normally inaccessible areas that become accessible with changing plant conditions. In addition, the applicant states that if the inspection results for accessible SCs indicate that potential degradation may be occurring in an inaccessible area, then an evaluation of the inaccessible area will be performed. The staff found this approach reasonable. In the letter to the NRC dated September 7, 2000, the applicant states that the Maintenance Rule AMP applies only to external surfaces. The staff found the parameters monitored, such as cracking and spalling of concrete and corrosion of steel, acceptable because they are directly related to the degradation of civil SCS, and visual inspections are effective and adequate to detect such conditions.

[*Detection of Aging Effects*] The aging effects that are managed by the Maintenance Rule program are identified through visual inspections. In the letter to the NRC dated September 7, 2000, the applicant states that checklists with various indications of potential degradation are provided to the personnel performing the structural walkdowns. Any deficiencies identified during the walkdowns are then documented on program specific report forms. In the letter to the NRC dated September 7, 2000, the applicant states that the structural monitoring conducted under the Maintenance Rule program is performed by engineers qualified in accordance with the ANO Engineering Support Personnel (ESP) Program, which is an accredited Institute of Nuclear Power Operations (INPO) training program. In addition, the engineer responsible for the evaluation of any findings identified during the walkdowns is also trained in accordance with the ESP Program.

With respect to an inspection frequency, the applicant did not specify a minimum walkdown frequency; however, in the letter to the NRC dated September 7, 2000, the applicant states the current periodic walkdown frequency for in-scope structures is once per five years. In addition, the applicant states that the frequency of structural walkdowns will be adjusted as necessary, based on reviewing general plant monitoring activities and procedures, ongoing inspection results, and industry experience. The applicant's operating experience to date supports the continuation of a 5-year frequency for inspections. Furthermore, the staff found that the 5-year frequency is consistent with industry experience and is, therefore, acceptable.

[*Monitoring and Trending*] The applicant did not identify any monitoring and trending activities in its description of the Maintenance Rule program in the LRA; however, in the letter to the NRC dated September 7, 2000, the applicant states that deficiencies identified by this program are documented so that the results can be trended. This documentation includes a written description and may also include measurements to determine the severity of the deterioration, photographs, and sketches. In addition, in the letter to the NRC dated September 7, 2000, the applicant states that the technical requirements for performing structural condition monitoring are described in an ANO engineering standard. The staff found that the monitoring and trending activities as described by the applicant are adequate to ensure that corrective actions will be taken prior to loss of function.

[*Acceptance Criteria*] The applicant identifies general acceptance criteria in its description of the Maintenance Rule program in the LRA. In the letter to the NRC dated September 7, 2000, the applicant provides the following additional information relating to the Maintenance Rule program acceptance criteria:

- Concrete is inspected for spalling (>1" in depth), cracking (> 1/16" in width), exposed rebar, water in-leakage, chemical leaching, peeling paint, and discoloration.

- Masonry walls are inspected for cracks, deteriorated penetrations, and missing or broken blocks.

- Structural steel is inspected for flaking rust, widespread corrosion (> 1/32" in depth), deteriorated coatings, beam/column deflection, loose or missing fasteners or support items, support misalignment, degradation of close tolerance machined or sliding surfaces (i.e., Teflon), missing grout beneath base plates, and pitting (> 1/32" in depth).

In addition, in the letter to the NRC dated September 7, 2000, the applicant states that each structural component or commodity is categorized as either “acceptable” if there are no deficiencies, “acceptable with deficiencies” if the components or commodities are degraded but still capable of performing their intended function, or “unacceptable” if the components or commodities are damaged and unable to perform their intended function. In the letter to the NRC dated September 7, 2000, the applicant states that the Maintenance Rule program is based in part on NEI 96-03, “Industry Guideline for Monitoring the Condition of Structures at Nuclear Power Plants.” However, the staff has not accepted the NEI 96-03 guideline for use in license renewal (letter from Thomas T. Martin, NRC, to Thomas E. Tipton, NEI, dated October 1, 1996). Other guidance documents for the Maintenance Rule AMP include NUREG-1526, “Lessons Learned from Early Implementation of the Maintenance Rule at Nine Nuclear Power Plants” and ACI-349.3R-96, “Evaluation of Existing Nuclear Safety-Related Concrete Structures.” Although the staff has not accepted NEI 96-03, the staff found that the acceptance criteria specified by the applicant are adequate to ensure that the structure and component intended function(s) are maintained under all CLB design conditions during the period of extended operation.

[*Operating Experience*] The Maintenance Rule program is an existing program; however, the applicant did not provide a description of the findings of the Maintenance Rule baseline inspection and subsequent Maintenance Rule inspection activities. In the letter to the NRC dated September 7, 2000, the applicant states that the findings from the initial baseline inspection included “some areas with exposed rebar which were slightly rusted, minor water in-leakage, and numerous concrete surface cracks, which did not exceed the acceptance criteria.” In addition, the applicant states that none of these findings resulted in the loss of the structure or component intended function(s). The staff found that the applicant’s operating experience has demonstrated that the Maintenance Rule program has effectively maintained the integrity of the SCS, and that the effects of aging will be adequately managed so that the structure and component intended function(s) will be maintained during the period of extended operation.

In the LRA, Appendix A, “Safety Analysis Report Supplement,” the applicant did not adequately explain that the inspection activities of the Maintenance Rule program are limited to the outer surfaces of the SCs include within the scope of the program in its summary description of the Maintenance Rule program. Therefore, the staff requested that the summary description be revised to adequately describes the Maintenance Rule program, as it applies to license renewal in accordance with 10 CFR 54.21(d). This was FSAR Item 3.3.1.3.3 of Open Item 3.3-1.

In its revised summary description of Section 16.2.13 of the FSAR Supplement, the applicant provides a summary description of the Maintenance Rule program that clarifies that the scope of this program is limited to the exterior surfaces of the SCs included within the program. The staff finds the revised summary description of the Maintenance Rule program as submitted in a letter to the NRC dated March 14, 2001, acceptable, and considers FSAR Item 3.3.1.3.3 of Open Item 3.3-1 resolved.

3.3.1.3.3 Conclusions

On the basis of the review described above, the staff finds that the applicant has demonstrated that the Maintenance Rule program can manage applicable aging effects of loss of material, cracking, change in material properties, and loss of mechanical closure integrity so that there is reasonable assurance that the commodities and components covered by this inspection program will perform its intended functions in accordance with the CLB for the period of extended operation.

3.3.1.4 Other Common AMPs

3.3.1.4.1 Buried Pipe Inspection Program

The applicant describes its buried pipe inspection program in Section 3.1 of Appendix B and section 3.4, “Auxiliary Systems,” of the LRA. In addition, the applicant supplements the information on this program in its letter to the NRC dated September 12, 2000. These sections address the interaction of the buried pipe inspection with the components in the auxiliary system and describe the resulting aging effects.

The staff reviewed the applicant’s description of the program in Section 3.1 of Appendix B in the LRA, and the additional information provided in the September 12, 2000, letter to determine whether the applicant has demonstrated that it will adequately manage the effects of aging caused by the loss of material to buried piping in the plant during the period of extended operation as required by 10 CFR 54.21(a)(3).

3.3.1.4.1.1 Technical Information in the Application

The applicant will use the newly developed buried piping inspection program to manage the loss of material due to external surface corrosion of buried carbon steel piping, pumps, strainers and valves in the fuel oil, and service water systems. The aging effect on the surfaces of the pipe results from a loss of the protective coating.

The applicant identifies the following systems that are affected by the buried pipe inspection program and that are within the scope of the LRA:

- service water system
- fuel oil system

3.3.1.4.1.2 Staff Evaluation

In accordance with 10 CFR 54.21(a)(3), the staff reviewed the information included in the sections listed above of the LRA, and the additional information provided in the September 12, 2000, letter regarding the applicant's demonstration of the buried pipe inspection program to ensure that the effects of aging due to corrosion of the external surfaces of piping caused by loss of the protective coating will be adequately managed so that intended functions will be maintained consistent with the CLB for the period of extended operation for all components in the service water and fuel oil systems.

[*Program Scope*] In the LRA, Section 3.1 of Appendix B, the applicant states that the buried pipe inspection program includes the safety-related portions of underground carbon steel piping of the fuel oil and service water systems. Since all of the buried piping in these two systems is included in this AMP, the staff found the scope of the program adequate.

[*Preventive/Mitigative Actions*] There are no preventive or mitigative actions associated with the inspection program, nor did the staff identify a need for such actions.

[*Parameters Monitored/Inspected*] In the LRA, Section 3.1 of Appendix B, the applicant states that the underground carbon steel piping will be inspected for damaged protective coating and corrosion on the external surface of the piping when it is uncovered. Such inspection would detect loss of material due to excessive corrosion and increased susceptibility to loss of material as indicated by damaged coating in the vicinity of the area that is inspected.

[*Detection of Aging Effects*] In the LRA, Section 3.1 of Appendix B, the applicant states that inspection will be performed when underground piping is uncovered during plant maintenance or modification activities. These are not scheduled inspections. In the letter to the NRC dated September 12, 2000, the applicant states that excavations of buried piping occur only when repairs or other maintenance activities on these systems are required, and that to excavate solely for the purpose of inspection may increase the possibility of damaging the protective coatings. The applicant expects portions of these piping systems to be randomly excavated every five to ten years based on previous operating and maintenance experience. The staff found this approach reasonable and therefore acceptable.

[*Monitoring/Trending*] In the LRA, Section 3.1 of Appendix B, the applicant states that this inspection program will be initiated prior to the end of the initial 40-year license term. Additionally, the applicant states that if defective protective coatings or loss of material is observed, then the frequency of inspections will be evaluated. In the letter to the NRC dated September 12, 2000, the applicant deferred to their corrective action program to determine the need for and scope of additional examinations when defective coatings or loss of material is found during one of the random excavations. There are no trending activities associated with the random visual examinations performed in this AMP, nor did the staff identify a need for such action.

[*Acceptance Criteria*] In the LRA, Section 3.1 of Appendix B, the applicant states that the acceptance criteria will be defined in plant procedures. In the letter to the NRC dated September 12, 2000, the applicant states that visually inspecting and confirming that the protective coating and wrapping are intact will ensure that no external corrosion is occurring. Although no specific acceptance criteria have been defined, the staff agrees that, provided the coatings/wrappings remain undamaged, no external surface corrosion should occur.

[*Operating Experience*] The applicant states that the buried pipe inspection program will be effective in the future for managing the effects of aging since it incorporates proven monitoring techniques, acceptance criteria, corrective actions, and administrative controls from existing programs and procedures. The staff found this acceptable.

After its initial review, the staff requested that the applicant add the fire protection systems buried pipe to the FSAR summary description of the buried pipe inspection program. This was FSAR Item 3.3.1.4.1.3 of Open Item 3.3-1. However, upon completing a review of the LRA, the applicant’s responses to the staff’s request for additional information, the programs credited with managing the aging of fire protection systems buried piping, and the site-specific operating experience, the staff verified that monitoring the frequency of initiation and the ability of the jockey pump to maintain system pressure to be an effective means of monitoring fire protection system leakage due to loss of material and, therefore, the buried pipe inspection program is not needed to manage the applicable aging effects of the FP system buried pipe. Upon reconsideration of this issue, the staff found that no change to Section 16.1.1 of the FSAR Supplemented, as submitted with the LRA, was needed.

3.3.1.4.1.3 Conclusions

On the basis of the review described above, the staff finds that the buried pipe inspection program can adequately manage the loss of material from buried piping so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

3.3.1.4.2 Heat Exchanger Monitoring Program

The applicant describes its heat exchanger monitoring program in Section 3.3 of Appendix B of the LRA. In addition, the applicant supplements the information in the LRA relating to this program in a letter to the NRC dated September 12, 2000. The applicant credits this program with managing the effects of aging for heat exchangers in several systems. The staff reviewed this section of the application and the supplemental information to determine whether the applicant has demonstrated that the effects of aging on heat exchanger functions will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

3.3.1.4.2.1 Technical Information in the Application

The applicant identifies the following systems that are affected by the heat exchanger monitoring program and that are within the scope of the LRA:

- low pressure injection system
- chilled water system
- service water system
- emergency feedwater system

The applicant uses the heat exchanger monitoring program to manage the following:

- loss of material for the heat exchangers in the low pressure injection system

- loss of material for the heat exchangers (evaporators and condensers), in the chilled water system

- loss of material for the heat exchangers in the service water system

- loss of material for the heat exchangers in the emergency feedwater system

3.3.1.4.2.2 Staff Evaluation

In accordance with 10 CFR 54.21(a)(3), the staff reviewed the information included in Section 3.3 of Appendix B of the LRA regarding the applicant's demonstration of the heat exchanger monitoring program to ensure that the loss of material, as discussed above, will be adequately managed so that intended functions will be maintained consistent with the CLB for the period of extended operation.

[*Program Scope*] In the LRA, Tables 3.3-2, 3.4-9, 3.4-10, and 3.5-3, the applicant includes heat exchangers within the low pressure injection, chilled water and service water, and emergency feedwater systems subject to the aging effects of loss of material, cracking and fouling. All heat exchangers within these systems are included within the scope of this program. Therefore, the staff found the scope of the program adequate.

[*Preventive/Mitigative Actions*] This is a condition monitoring program. As such, there are no preventive or mitigative actions, nor did the staff identify a need for such action.

[*Parameters Monitored*] In the LRA, Section 3.3 of Appendix B, the licensee states that it will perform nondestructive examinations such as eddy current inspections or visual inspections periodically. In the letter to the NRC dated September 12, 2000, the applicant clarifies that eddy current inspections are intended to detect loss of material and cracking in heat exchanger tubing. In addition, the applicant states that the internal visual inspections of heat exchanger heads, covers and tube sheets would be performed when accessible. The staff found that the parameters that are monitored are capable of demonstrating that the pressure boundary function of the applicable components will be maintained.

[*Detection of Aging Effects*] Eddy current examinations are volumetric methods accepted by industry to be effective for detecting age-related degradation in the applicant's heat exchanger tubing. In addition, periodic visual inspection of component interior surfaces will detect a significant loss of material or cracking and, therefore, provide reasonable assurance that the structural integrity of these components is maintained for the period of extended operation.

[*Monitoring and Trending*] In response to staff's request for additional information on monitoring and trending activities dated September 12, 2000, the applicant states that eddy current testing would be performed on a sampling of heat exchanger tubing at a minimum of once every ten years. If these examinations show degraded conditions, the sampling and frequency would then be increased. No trending of the eddy current examinations is planned because of the acceptance criterion discussed below. The visual inspections will be performed at the same frequency of the eddy current examinations. Trending of visual inspections is not applicable. The staff found this frequency to be reasonable.

[*Acceptance Criteria*] The applicant provides additional information concerning acceptance criteria for this AMP in the letter to the NRC dated September 12, 2000. If eddy current examinations reveal a 60% through-wall loss of material, then the tubes will be plugged or evaluated to justify continued service. This criterion is based upon industry standards and has previously been found acceptable by the staff in NUREG-1723. The applicant also states that during visual inspections, any evidence of degradation that could lead to loss of function will be evaluated in accordance with site corrective action procedures. These criteria are acceptable to the staff.

[*Operating Experience*] The heat exchanger monitoring program is new, therefore, in the September 12, 2000, letter, the applicant states that no operating experience is available. However, the eddy current examinations and visual inspections are consistent with industry standards. The applicant indicates that if initial or periodic examinations reveal the need to expand the sample size or increase the frequency of these activities, such actions would occur. The staff found this acceptable, and that operating experience with this AMP will be accrued over the period of extended operation.

After its initial review, the staff requested that the applicant clarify, in its summary description of the heat exchanger monitoring program, that fouling on the service water side of the decay heat removal heat exchangers is managed through two AMPs, the heat exchanger monitoring program and the service water integrity program. This was FSAR Item 3.3.1.4.2.3 of Open Item 3.3-1. However, upon completing a review of the LRA, the applicant's responses to the staff's request for additional information, and the applicable AMPs, the staff verified that fouling will be adequately managed by programs, other than the heat exchanger monitoring program (i.e., the Service Water Integrity Program or system surveillance testing). The staff found that no change to Section 16.1.3 of the FSAR Supplement, as submitted with the LRA, was needed.

3.3.1.4.2.3 Conclusions

On the basis of the review described above, the staff finds that the applicant has demonstrated that the heat exchanger monitoring program, in conjunction with the service water integrity program, can adequately manage the effects of aging associated with loss of material, and cracking so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

3.3.1.4.3 Wall Thinning Inspection Program

In the LRA, Section 3.7 of Appendix B, the applicant describes the wall thinning inspection program. The applicant credits this program with managing the effects of aging for loss of material due to corrosion of the internal surfaces of carbon steel piping and components. The staff reviewed this section of the application to determine whether the applicant has demonstrated that the effects of aging of the carbon steel piping and components will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

3.3.1.4.3.1 Technical Information in the Application

The applicant identifies the following systems that are affected by the wall thinning inspection program, and that are within the scope of the LRA:

- emergency feedwater system (EFW)
- chemical addition system (CAS)
- main steam system (MSS)
- reactor building isolation system (RBIS)

The applicant states that the wall thinning inspections will be performed to ensure that the wall thickness is above the minimum required value in order to avoid leaks or failures under normal conditions and postulated transient and accident conditions, including seismic events.

3.3.1.4.3.2 Staff Evaluation

In accordance with 10 CFR 54.21(a)(3), the staff reviewed the information included in Section 3.7 of Appendix B of the LRA regarding the applicant's demonstration of the wall thinning inspection program to ensure that the effects of aging, as discussed above, will be adequately managed so that the intended functions will be maintained consistent with the CLB for the period of extended operation.

[*Program Scope*]   The staff found the scope of this inspection program to be adequate because it includes all of the applicable components and systems of the EFW, CAS, MSS, and RBIS. The applicant clarifies, in a letter to the NRC dated September 12, 2000, that the only components in the RBIS to which this program is applied are the carbon steel piping and valves associated with reactor building penetration numbers 51 and 59. The staff found the scope of this program adequate for managing loss of material in the limited areas for which it is intended.

[*Preventive/Mitigative Actions*] There are no preventive or mitigative actions, nor did the staff identify a need for such action.

[*Parameters Monitored*] In the letter to the NRC dated September 12, 2000, the applicant indicates that industry-accepted methods such as ultrasonic testing will be employed to assess wall thickness. The staff determined that ultrasonic testing is capable of measuring the remaining wall thickness for the piping and valves within the scope of this program and, therefore, found the examination parameters acceptable.

[*Detection of Aging Effects*] The applicant states that the wall thinning inspection will be initiated prior to the end of the initial 40-year license term for ANO-1. Furthermore, the applicant states that the inspections will be performed periodically at a frequency to be determined prior to the end of the initial 40-year license term for ANO-1. The frequency of inspections will depend upon the results of previous inspections, the calculated rate of material loss, and industry and plant operating experience. The staff found the basis on which the applicant determines the inspection frequency reasonable.

[*Monitoring and Trending*] The applicant identifies the loss of material due to corrosion and performs trending for those locations based on subsequent inspection results. The staff found this aspect of the program acceptable because trending of the inspection results will enhance the applicant's ability to detect aging effects before there is a loss of intended function.

[*Acceptance Criteria*] The applicant states that wall thickness measurements that are found to be, and projected to remain, greater than the minimum wall thickness values for specific components' design code(s) will be acceptable. Therefore, any wall thickness values that are projected to fall below the minimum allowable, as determined by the applicable design code, will be found unacceptable and corrective measures implemented. The staff concludes that this acceptance criterion is adequate to demonstrate that a loss of material due to wall thinning will be managed for the period of extended operation.

[*Operating Experience*] The wall thinning inspection program is new, therefore, in the letter to the NRC dated September 12, 2000, the applicant states that no operating experience is available. However, the ultrasonic wall thickness examinations are consistent with industry standards. The applicant has indicated that if initial or periodic examinations reveal the need to expand the sample size or increase the frequency of these activities, such actions would occur. The staff found this acceptable. The operating experience associated with this AMP will be accrued over the period of extended operation.

After its initial review, the staff requested that the applicant state in its summary description of the wall thinning inspection program that the only components in the RBIS to which the wall thinning inspection program is applied are the carbon steel components that are associated with reactor building penetration numbers 51 and 59. This was FSAR Item 3.3.1.4.3.3 of Open Item 3.3-1. However, after a review of the LRA, the applicant's responses to the staff's request for additional information, and the applicable AMPs, the staff verified that the wall thinning inspection program was not limited to the chilled water system components of penetrations 51 and 59. Other reactor building isolation system carbon steel components credit the Wall Thinning Inspection Program. These other penetrations are correctly listed in the program description in Appendix B of the LRA (Section 3.7) and in the FSAR Supplement as submitted with the LRA. Upon reconsideration of this issue, the staff found that no change to Section 16.1.7 of the FSAR Supplement, as submitted with the LRA, is needed.

3.3.1.4.3.3 Conclusions

On the basis of the review described above, the staff finds that the applicant has demonstrated that the wall thinning inspection program can adequately manage the effects of aging associated with loss of material of the internal surfaces of carbon steel piping and components so that the intended function will be maintained consistent with the CLB for the period of extended operation.

3.3.1.4.4 Boric Acid Corrosion Prevention Program

In the LRA, Section 4.5, “Boric Acid Corrosion Prevention,” of Appendix B, the applicant states that the boric acid corrosion prevention program will be used to manage the loss of material and the loss of mechanical closure integrity that may be caused by exposure to borated water for the external surfaces of piping, valves, tanks and bolting made of carbon steel so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

3.3.1.4.4.1 Technical Information in the Application

The applicant credits the boric acid corrosion prevention program with mitigation of the aging effects (loss of material) associated with general corrosion of components from exposure to concentrated boric acid. The program consists of visual inspections that are performed during plant cooldowns and heatups. Additional inspections may be performed based on the nature of the reactor coolant system (RCS) leakage detected during normal operations.

In the LRA, Sections 3.2, 3.3, 3.4 and 3.6, the applicant states that this program is used to manage the loss of material from exposure to concentrated boric acid for all the systems and components located in the reactor and auxiliary buildings, including the RCS components, mechanical components, and structural components. In the LRA, Section 4.6, the applicant describes the boric acid corrosion prevention program for managing the aging effects associated with corrosion of components from exposure to concentrated boric acid.

3.3.1.4.4.2 Staff Evaluation

In accordance with 10 CFR 54.21(a)(3), the staff reviewed the information on the boric acid corrosion prevention program contained in Section 4.5 of Appendix B of the LRA regarding the applicant’s demonstration that it will adequately manage the effects of aging from corrosion so that the intended function of the affected components will be maintained consistent with the CLB for the period of extended operation. The applicant provided additional information regarding this program in a letter dated September 12, 2000.

[*Program Scope*] The staff found the scope of the boric acid corrosion prevention program acceptable because it includes the SCs exposed to borated water systems in the reactor and auxiliary buildings. The program also includes an evaluation of all components potentially affected by borated water leakage.

[*Preventive/Mitigative Actions*] As part of this program, the applicant performs visual inspections to identify pressure boundary leakage of borated water that may be in contact with carbon steel components. Active leaks are repaired, redirected or evaluated for continued service prior to plant start-up. Plant procedures provide guidance for system walkdown requirements and corrective measures that must be taken when leakage is identified. Therefore, the staff concludes that the program will be effective for mitigating boric acid corrosion on exposed surfaces of the RCS and other structures or components containing, or exposed to, borated water.

[*Parameters Monitored*] The staff found the parameters monitored to be acceptable because any leakage, discoloration or accumulation of residue on the surface of components, insulation or the surrounding floor areas provide sufficient indication of borated water leakage.

[*Detection of Aging Effects*] The inspections for borated water leakage are performed during each plant cool-down and heat-up cycle. Additional inspections may be performed based on RCS leakage detected during normal operation. The frequency of these inspections ensures that leakage is detected prior to an extensive corrosion attack that may occur if concentrated boric acid remains in contact with carbon steel components for prolonged periods. The staff concludes that detection of significant external loss of material or loss of mechanical closure integrity will occur prior to compromising the intended function of these components. This conclusion is supported by both ANO-specific and industry-wide operating experience to date.

[*Monitoring and Trending*] There are no monitoring and trending aspects to the boric acid corrosion prevention program, nor did the staff identify a need for such action.

[*Acceptance Criteria*] The applicant states that each identified leak of borated water and any staining or buildup of boric acid crystals are evaluated to ensure that no components are damaged to the extent that they would be unable to fulfill their intended safety function. The staff found the acceptance criteria specified by the applicant to be appropriate to ensure that the intended functions of the RCS and other structures or components containing, or exposed to, borated water.

[*Operating Experience*] The applicant states that operating experience indicates that the boric acid corrosion prevention program has been successful in ensuring the proper identification, evaluation, and repair of borated water leakage in contact with carbon steel components. In the letter to the NRC dated September 12, 2000, the applicant states that no significant age-related degradation was identified during the review of operating experience. The staff concludes that the applicant has demonstrated that the boric acid corrosion prevention program has been effective in preventing damage to components due to exposure of concentrated boric acid.

3.3.1.4.4.3 Conclusions

On the basis of the review described above, the staff finds that the applicant has demonstrated that the boric acid corrosion prevention program can be used to manage the loss of material damage due to leakage from the borated water systems so that the intended function will be maintained consistent with the CLB for the period of extended operation.

3.3.1.4.5 Flow-Accelerated Corrosion Prevention Program

In the LRA, Section 3.5, "Steam and Power Conversion," and in Section 4.9 of Appendix B, the applicant describes the flow-accelerated corrosion prevention program. These sections address flow-accelerated corrosion of the components in the steam and power conversion system and describe the prevention program. The overall objective of the flow-accelerated corrosion prevention program is to provide a programmatic approach for predicting, identifying, inspecting, and managing loss of material for components that are adversely affected by flow-accelerated corrosion.

The staff reviewed the applicant's description of the program in the LRA to determine whether the applicant has demonstrated that it will adequately manage the effects of aging caused by the metal loss by flow-accelerated corrosion from components made of carbon steel (which occurs only under certain conditions of flow, chemistry, geometry, and material) in the plant during the period of extended operation as required by 10 CFR 54.21(a)(3).

3.3.1.4.5.1 Technical Information in the Application

The applicant identifies the following systems that contain components, which are affected by the flow-accelerated corrosion and are within the scope of license renewal:

- main steam system
- main feedwater system

3.3.1.4.5.2 Staff Evaluation

In accordance with 10 CFR 54.21(a)(3), the staff reviewed the information in Appendix B, Section 4.9, "Flow Accelerated Corrosion Prevention" and Section 3.5, "Steam and Power Conversion" of the LRA regarding the applicant's demonstration of the flow-accelerated corrosion program to ensure that the effects of aging due to flow-accelerated corrosion will be adequately managed, so that the intended functions of the components, included in the LRA, will be maintained consistent with the CLB for the period of extended operation. The staff's evaluation is provided below.

[*Program Scope*] The staff found the scope of the inspection program adequate in that it includes all the components affected by flow-accelerated corrosion in the main steam system (MSS), and the main feedwater system (MFW). The components to be inspected are determined by analytical methods, previous examination results, plant-specific engineering judgement, and industry experience.

[*Preventive/Mitigative Actions*] The staff found that the preventive or mitigative actions are acceptable because the program utilizes a combination of computer codes, previous examination results, and industry experience to prevent flow-accelerated corrosion from affecting the components' intended functions.

[*Parameters Monitored*] The staff found that the inspection parameters are acceptable because in the components affected by flow-accelerated corrosion, wall thickness is measured by nondestructive examination and analytical methods that adequately quantify the amount of wall thinning.

[*Detection of Aging Effects*] The staff concludes that the detection of aging effects before there is a loss of intended function can be reasonably predicted by the monitoring program because the frequency of inspections is based on results of previous inspections and predictive methods. Satisfactory operating experience to date also supports this conclusion.

[*Monitoring and Trending*] The monitoring and trending is based on analytical methods that are sufficient to provide predictability of the extent of degradation so timely corrective or mitigative actions are possible. The staff found these methods reasonable.

[*Acceptance Criteria*] The applicant states that the acceptance criteria for the flow-accelerated corrosion prevention program are located in site procedures. Any measured wall thickness below, or projected to be below, 70 percent of nominal wall at the next refueling outage is evaluated to determine if additional areas need to be examined. Any component with a measured wall thickness below, or projected to be below, the ASME B31.1 minimum wall will be replaced, unless a local wall thinning evaluation can show acceptability for continued service. The staff found that the acceptance criteria based on the minimum allowable wall thickness are acceptable.

[*Operating Experience*] The applicant's operating experience has demonstrated that the flow-accelerated corrosion inspection program is an effective program for managing the effects of aging. The program, by incorporating proven monitoring techniques, acceptance criteria, corrective actions, and administrative controls, accurately predicts aging effects due to flow-accelerated corrosion.

3.3.1.4.5.3 Conclusions

On the basis of the review described above, the staff finds that there is reasonable assurance that the flow-accelerated corrosion prevention program can be used to adequately manage the loss of material due to flow-accelerated corrosion in the pipes and components of the steam and power conversion system so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

3.3.1.4.6 Leakage Detection In Reactor Building Program

In the LRA, Section 3.2, "Reactor Coolant System" and Section 3.5, "Steam and Power Conversion," as well as in Section 4.12 of Appendix B of the LRA, the applicant describes the leakage detection in reactor building program. The purpose of leakage detection in the reactor building, with respect to license renewal, is to monitor for leakage in order to manage the consequences of cracking, loss of material, and loss of mechanical closure integrity.

3.3.1.4.6.1 Technical Information in the Application

The applicant identifies the following aging effects for the systems/components listed below that are affected by the leakage detection in the reactor building program and that are within the scope of the LRA:

Reactor Coolant System

- cracking at welded joints of the stainless steel piping/stainless steel clad carbon steel piping in primary water

- cracking at welded joints of the stainless steel clad carbon steel branch connections with alloy 82/182 weld build-up in primary water

Main Steam

- loss of material, loss of mechanical integrity, and cracking of the carbon steel piping/tubing/valves in treated water

- loss of material, loss of mechanical integrity, and cracking of the stainless steel piping/tubing/valves in treated water

Main Feedwater

- loss of material and cracking of the carbon steel piping/tubing/valves in treated water

3.3.1.4.6.2 Staff Evaluation

In accordance with 10 CFR 54.21(a)(3), the staff reviewed the information included in the sections listed above regarding the applicant's demonstration of the leakage detection in reactor building program, to ensure that the effects of aging will be adequately managed, so that the intended functions will be maintained consistent with the CLB for the period of extended operation.

[*Program Scope*] For the reactor building, leak detection is mainly focused on the RCS, therefore, the scope of the program will cover the reactor vessel. However, the procedures also have the potential to detect leaks from other systems in the reactor building, such as the main steam and main feedwater systems. The program provides for the following:

- determination of the principle location of leakage

- examination requirements and procedures for locating small leaks

- engineering evaluations and corrective actions to ensure that boric acid corrosion does not lead to degradation of the leakage source or adjacent structures or components which could cause a loss of intended function of the structure or component

Thus, the staff found the scope of leakage detection in reactor building program acceptable.

[*Preventive/Mitigative Actions*] There are no preventive or mitigative actions taken as part of this program, nor did the staff identify a need for such action.

[*Parameters Inspected or Monitored*] The applicant describes how it monitors RCS inventory in the ANO-1 TS 3.1.6 and Table 4.1-2, and in the ANO-1 site procedures. Leakage detection in the reactor building is accomplished by three different means. These include the RCS inventory balance, reactor building sump monitoring, and building atmospheric monitoring for radioactivity. The staff found the parameters monitored acceptable because these parameters are in accordance with the applicant's TS, which the staff previously approved.

[*Detection of Aging Effects*] Potential indicators of RCS leakage are monitored continuously throughout each shift. The RCS leak rate determination is performed as required by ANO-1 TS 3.1.6 and Table 4.1-2 and ANO-1 site procedures. Determination of leaks in excess of the acceptance criteria will lead to the identification of the responsible degraded or failed component, so that the applicant will undertake further programmatic action, such as repair and replacement, as necessary, to manage these aging effects. The staff found the frequency of monitoring acceptable because it is consistent with the applicant's TS, industry practice, and staff expectations.

[*Monitoring and Trending*] As mentioned above, the RCS inventory balance, the reactor building sump, and the atmosphere in the reactor building will all be monitored to detect leaks from the RCS. If leaks from the RCS are detected, the applicant will undertake further programmatic actions, such as repair and replacement, as necessary, to manage the effects of aging that caused the leak.

The reactor coolant balance is defined in ANO-1 TS Table 4.1-2 and requires the leak rate to be determined periodically. RCS allowable leakage during power operation is limited, as specified in ANO-1 TS 3.1.6. For the building sump, the fill rate is trended. In conjunction with the RCS leak rate, it is used as an indication of the source of leakage into the sump (i.e., leakage inside/outside the building, RCS leakage or non-RCS leakage such as from the main feedwater or main steam systems, etc.). The reactor building radioactive monitoring program consists of the reactor building atmosphere particulate detector and the reactor building atmosphere gaseous detector. The readings from these instruments are logged and trended. The staff found these aspects acceptable in that the trending prompts other corrective action be taken before there is a loss of intended function.

[*Acceptance Criteria*] The acceptance criteria for the RCS allowable leakage limit is specified in ANO-1 TS 3.1.6. Identified non-RCS leakage is evaluated on a case-by case basis. The staff found the acceptance criteria acceptable because they are consistent with the applicant's TS.

[*Operating Experience*] The applicant indicates that a review of the ANO-1 operating experience confirms that these activities are effective in detecting leakage in the reactor building. In 1989 a "non-isolable" leak on an RCS drain line was detected when RCS leakage increased from the previous day's leak rate by approximately 0.2 gpm. This example demonstrated that the program is effective in detecting small amounts of RCS leakage. Therefore, the staff found that the leaks caused by cracking, loss of material, or loss of mechanical closure integrity would be detected by this program. The staff found that the applicant's operating experience supports the attributes of this program.

3.3.1.4.6.3 Conclusions

On the basis of the review described above, the staff finds that the applicant has demonstrated that the reactor building leak detection program can continue to be used to effectively detect leaks from the RCS so that the aged components can be identified, and either repaired or replaced so that the intended function(s) can be maintained consistent with the CLB for the period of extended operation.

3.3.1.4.7 Oil Analysis Program

The applicant describes its AMR of the oil analysis program in Section 4.14, “Oil Analysis” of Appendix B and Sections 3.3, “Engineering Safeguards,” 3.4, “Auxiliary Systems,” and 3.5, “Steam and Power Conversion Systems,” of the LRA. These sections address the aging effects caused by loss of material and cracking in lubricating oil environments and describe the prevention program.

3.3.1.4.7.1 Technical Information in the Application

The applicant identifies the following systems in the LRA that contain components, which may display corrosion degradation in a lubricating oil environment and require management of the resulting aging effects.

- low-pressure injection/decay heat removal system
- high-pressure injection/makeup and purification system
- reactor building spray system
- fire protection system
- emergency diesel generator system
- alternate AC diesel generator system
- chilled water system
- control room ventilation system
- emergency feedwater system

These systems contain the components which include the compressor in control room ventilation system and different lubricating oil coolers, heat exchangers, pipes, valves and filters. These components are made of carbon steel, stainless steel, cast iron, brass, copper, bronze, admiralty metal and copper-nickel alloy. In the lubricating oil environment, these components could experience loss of material and cracking, which have to be managed.

3.3.1.4.7.2 Staff Evaluation

[*Program Scope*] The applicant identifies all of the components in the systems, containing lubricating oil, in Section 4.14 of the LRA. The staff found the scope of this program to be adequate.

[*Preventive/Mitigative Actions*] The applicant performs periodic sampling of lubrication oil. Analyses of these samples are intended to maintain the quality of lubricant in the affected components. By analyzing for anomalous particulate and moisture, and ensuring that beneficial additives remain present in lubricating oils, the staff concludes that aging effects will be mitigated for those components exposed to an oil environment.

[*Parameters Monitored*] The applicant’s oil analysis program is aimed at detecting water and particulate that might be present in the oil through routine sampling and analyses. A particle count and moisture check are performed for those components that receive regular oil changes. For components that do not get scheduled oil changes, viscosity and neutralization number are the parameters that are determined. In the letter to the NRC dated September 12, 2000, the applicant specifies various methods for monitoring these parameters including visual inspection, titration, distillation, particle count, spectral analyses and the standard method for checking kinematic viscosity. The staff found that the parameters monitored are sufficient to provide indications of environmental conditions or worn components that may lead to the loss of intended function.

[*Detection of Aging Effects*] This AMP is a preventive program, and as such is not aimed at, or credited for, detecting aging effects. The staff found this acceptable because there are other AMPs that satisfy this attribute.

[*Monitoring and Trending*] The frequency of sampling is based upon equipment manufacturer recommendations and standard industry practice for each component. The staff found this approach reasonable.

[*Acceptance Criteria*] The criteria for various tests performed in this AMP are contained in site procedures, and are based on national standards, where applicable. These include SAE749D Class 6 and SAE Class 3 for particle concentration limits, the CRC Handbook of Lubrication limits water to be less than a trace level (0.1percent). The viscosity of oils is maintained within 10 percent of the base viscosity and metallic particle limits, as determined through spectral analyses, conform to the equipment technical manuals. The staff found that adequate acceptance criteria exist to allow the applicant to determine if potential degradation, which may compromise the intended function of components exposed to lubricating oils, is occurring.

[*Operating Experience*] In the letter to the NRC dated September 12, 2000, the applicant states that oil analysis results have been collected since 1990 for components that are within the scope of license renewal. The results indicate that lubricating oils are being maintained free of excess water, contamination that would degrade the oil pH level is not occurring, and that proper additives remain present to neutralize any acids that may form during component operation. This historical data indicates that the applicant has maintained the quality of lubricating oils, thereby mitigating the aging effects that could compromise the intended functions of applicable components. The staff found that the applicant's operating experience supports the attributes of its program.

3.3.1.4.7.3 Conclusions

On the basis of the review described above, the staff finds that the applicant has demonstrated that aging effects associated with lubricating oil can be managed so that there is reasonable assurance that the affected system will perform its intended function(s) in accordance with the CLB for the period of extended operation.

3.3.1.4.8 Reactor Building Leak Rate Testing Program

In the LRA, Section 4.16, "Reactor Building Leak Rate Testing," of Appendix B, the applicant states that the reactor building leak rate testing program provides reasonable assurance that leakage from the reactor building will not exceed required maximum values for reactor building leakage.

3.3.1.4.8.1 Technical Information in the Application

The applicant states that the reactor building leak rate testing consists of Type A, Type B, and Type C testing. For the purposes of AMPs for license renewal, only Type A and Type C tests are considered.

Type A testing measures the primary reactor building overall integrated leakage rate. This is also known as integrated leak rate testing. Type C testing measures primary reactor building isolation valve leakage rates, whether the valves are manual or automatic. This is known as local leak rate testing.

The following systems credit the reactor building leak rate testing program:

- reactor building isolation system
- high- and low-pressure injection systems
- fire protection system
- auxiliary building sump and reactor building drain system
- chilled water system

3.3.1.4.8.2 Staff Evaluation

The staff reviewed the information included in Section 4.16 of Appendix B of the LRA to ensure that the reactor building leak rate testing program will detect the loss of material and cracking of the systems within scope for the period of extended operation.

[*Program Scope*] In this AMP, the applicant includes all cast iron and carbon steel pumps, piping, and valves within the reactor building isolation system, high pressure and low pressure injection systems, fire protection system, auxiliary building sump system, and chilled water system. In addition, the scope of the reactor building leak rate testing program includes the reactor building penetrations. The staff found the scope of the reactor building leak rate testing program adequate for the components of the systems listed above.

[*Preventive/Mitigative Actions*] There are no preventive or mitigative actions taken as part of this program, nor did the staff identify a need for such action.

[*Parameters Monitored*] A description of the parameters monitored is provided in the letter to the NRC dated September 12, 2000. Leakage through containment penetrations, access-openings, and the containment structure is the parameter monitored by this program. The applicant states that the tests identify loss of material and cracking by measuring leakage on individual components after pressurizing the penetration to the specified test pressure. Because these tests are limited to the piping and valves associated with reactor building penetrations, the staff found that these local leak tests apply only to the components of the affected systems located between the inboard and outboard containment isolation valves.

[*Detection of Aging Effects*] In the letter to the NRC dated September 12, 2000, the applicant states that this testing program would not detect the loss of material or cracking before the pressure boundary is compromised. However, the applicant further states that the reactor building leak rate testing program is used in conjunction with other programs to manage the effects of aging of those particular components. Therefore, the staff found this program acceptable, since it is used in conjunction with other programs, to effectively manage the effects of aging on components.

[*Monitoring and Trending*] The applicant indicates that integrated leak rate tests are performed once every ten years as long as the calculated leakage remains less than the maximum allowable leakage rate. The applicant states that local leak rate tests are based on past local leak rate test results, service conditions, design, safety impact, previous failure causes, and common mode detection. Local leak rate tests are performed when any adjustment or maintenance on an isolation barrier is performed that can affect sealing characteristics. The staff found this approach reasonable.

[*Acceptance Criteria*] The applicant states that the acceptability of testing results is defined in ANO-1 TS 6.8.4. The reactor building leak rate testing program is an NRC-approved program that is part of the ANO-1 CLB and is implemented in accordance with ANO-1 TS 6.8.4. The staff found this acceptable.

[*Operating Experience*] The applicant states in Sections 4.16.1 and 4.16.2, Appendix B of the LRA, that operating experience indicates the program is effective in detecting unacceptable leakage through the containment pressure boundary. In the letter to the NRC dated September 12, 2000, the applicant states that, historically, leakage rates have been well within the maximum allowable as described in the TS. If maintenance is required on a valve because of leakage in excess of acceptable limits, an additional leak rate test is performed to confirm the adequacy of the maintenance. The staff found that the operating experience to date supports the attributes of this program.

3.3.1.4.8.3 Conclusions

On the basis of the reviewed described above, the staff finds that the specified leak rate testing, in conjunction with other credited programs, provides reasonable assurance that the applicable aging effects will be managed so that containment isolation components located between the inboard and outboard containment isolation valves, in the spent fuel system, fire protection system, auxiliary building sump and reactor building drain system, and chilled water system will continue to perform its intended function(s) consistent with the CLB for the period of extended operation.

3.3.1.4.9 Inservice Inspection Plan

In the LRA, Section 4.3 of Appendix B, the applicant described the inservice inspection plan. The applicant credits the examinations performed under the ASME Code, Section XI, inservice inspection program with managing the effects of aging for Class 1, 2, 3, and MC pressure-retaining components and their supports during the period of extended operation. The staff has reviewed the section of the application to determine whether the applicant has demonstrated that the effects of aging will be adequately managed by the inservice inspection plan during the period of extended operation as required by 10 CFR 54.21(a)(3).

3.3.1.4.9.1 Technical Information in the Application

In the LRA, Section 4.3, the applicant states that its intent is to meet the requirements of the latest edition and addenda to the ASME Code, Section XI, that are incorporated by reference in 10 CFR 50.55a(b) for inservice inspection. These requirements are subject to the conditions specified in 10 CFR 50.55a, to the extent practical within the limitations of design, geometry, and materials of construction of the component or the support.

The applicant cites several sections of the LRA which identifies the applicable components subject to an AMR and the applicable aging effects of those components for the inservice inspection program. The applicant also identifies the various editions of the ASME Section XI Code for Subsections IWB, IWC, IWD, IWE, IWF, IWL and augmented Inspections, under the ANO-1 Inservice Inspection Plan, that will manage these aging effects for the period of extended operation.

3.3.1.4.9.2 Staff Evaluation

ASME Inservice Inspection Program, Subsections IWB, IWC, and IWD

In the LRA, Sections 4.3.1, 4.3.2, and 4.3.3 of Appendix B, the applicant identifies cracking, loss of material, and loss of mechanical closure integrity as the component aging effects requiring management. The applicant states that the ANO-1 inservice inspection plan addresses Subsections IWB, IWC, and IWD Inspections complying to the ASME Code, Section XI, 1992 Edition with portions of the 1993 Addenda, manage the above-mentioned aging effects for the period of extended operation.

The purpose of the ASME Section XI, Subsections IWB, IWC, and IWD Inspections for the ANO-1 program is to identify and correct degradation of ASME Class 1, 2, and 3 pressure retaining components and their integral attachments, in accordance with 10 CFR 50.55(a) and ANO-1 TS 4.0.5. to maintain their intended function consistent with the CLB for the period of extended operation.

[*Program Scope*] The scope of the program is identified specifically for each component and for applicable component features, as described in Table 3.2-1 of the LRA. Items listed in this table are consistent with the items in ASME Section XI, Tables IWB-2500-1, IWC-2500-1, and IWD-2500-1 with the exception of RCS piping and approved alternatives. These exceptions are based upon risk-informed methodology, and are discussed in Section 4.3.1 of the LRA. The applicant has implemented a risk-informed methodology to select RCS piping welds for inspection in lieu of the requirements specified in the 1992 Edition of the ASME Section XI, Table IWB-2500-1, Examination Category B-J. The staff has previously approved this methodology for ANO-1 and, therefore, finds it acceptable for the period of extended operation, as well.

[*Preventive/Mitigative Actions*] There are no preventive or mitigative actions associated with the inservice inspection program, nor did the staff identify a need for such action.

[*Parameters Monitored*] The parameters monitored are specified in the ASME Code, Section XI, for each type of examination required. The applicant performs three different types of examinations such as, volumetric, surface, and visual examinations. Volumetric examinations consist of radiographic, ultrasonic or eddy current examinations performed to locate surface and subsurface flaws. Surface examinations use magnetic particle or dye penetrate testing to locate surface flaws. Visual examinations are conducted to assess the condition of the surface, locate evidence of leakage or detect discontinuities and imperfections to determine mechanical and structural condition of the component. The staff accepts the parameters being monitored during inservice inspection of Class 1, 2, and 3 components in managing the age-related degradation.

[*Detection of Aging Effects*] The volumetric, surface, and visual examination of components along with system pressure tests are used to detect the following aging effects: cracking, loss of mechanical closure integrity at bolted connections, and loss of material. The staff accepts the nondestructive examination methods prescribed by the Code for each class of component to be reliable and effective in detecting age-related degradation of components that are within the scope of license renewal.

[*Monitoring and Trending*] Frequency of inspection and the inspection method are specified by the Code. Indications found by nondestructive examinations are evaluated in accordance with the Code and if allowed to remain, will require monitoring and will be used for comparison with future inservice examination results. This provides for trending of the aging effect and establishes a baseline for the degradation process and the extent of degradation with time. The staff accepts this methodology to undertake further programmatic actions, such as repair and replacement, as necessary, to manage these aging effects.

[*Acceptance Criteria*] Flaws detected by nondestructive examination during inservice inspection are evaluated in accordance with the acceptance standards established in the ASME Code, Section XI, for the examination categories listed in Table 3.2-1 of the application. The staff accepts the flaw evaluation methodology of the Code as the industry standard and, therefore, the management of aging effects based on the Code criteria.

[*Operating Experience*] The operating experience with the inservice inspection program indicates that it has been successful in identifying and leading to correction of aging effects as expected of this program. Therefore, the program is effective in the management of age-related degradation with inservice inspection.

ASME Inservice Inspection Program, Subsection IWE

Section 4.3.4 of Appendix B of the LRA identifies loss of material as the aging effect requiring management. The LRA identifies ASME Code, Section XI, 1992 Edition with the 1993 Addenda as the guidance used to manage loss of material for the period of extended operation. However, in a letter to the NRC dated September 7, 2000, the applicant states that the correct Code addenda is the 1992 Addenda. The purpose of ASME Code, Section XI, Subsection IWE, is to identify and correct degradation of Class MC pressure retaining components and their integral attachments, and of metallic shell and penetration liners of Class CC pressure retaining components and their integral attachments, in accordance with 10 CFR 50.55a.

[*Program Scope*] The program scope, as initially described in the LRA, was not acceptable to the staff because it did not address all of the components required by 10 CFR 50.55a and ASME Code, Section XI, Subsection IWE. In the letter to the NRC dated September 7, 2000, the applicant clarified that they are committed to the full scope and requirements of 10 CFR 50.55a and Subsection IWE. The staff found the revised scope, as stated in the letter, acceptable.

[*Preventative/Mitigative Actions*] The applicant did not identify any preventative actions for this AMP, nor did the staff identify a need for such action.

[*Parameters Monitored/Inspected*] As described in the LRA, the parameters monitored were not acceptable to the staff. In the letter to the NRC dated September 7, 2000, the applicant commits to following the requirements for monitoring the parameters as stated in ASME Code, Section XI, Subsection IWE. The applicant identified the following inspection activities: volumetric (radiographic, ultrasonic), surface (magnetic particle, dye penetrant), and visual (VT-3 and VT-1) examinations. With the applicant's commitments discussed in the letter, the staff found the monitored parameters and inspections acceptable.

[*Detection of Aging Effects*] The volumetric, surface, and visual examinations of components as specified in ASME Code, Section XI, Subsection IWE, detect aging effects such as loss of material. The staff accepts the methods prescribed by the Code for each class of component to be reliable and effective in detecting age-related degradation of components that are within the scope of license renewal.

[*Monitoring and Trending*] The frequency of inspection and the inspection methods are specified by the ASME Code. Indications found during these inspections are evaluated in accordance with the Code and, if allowed to remain, will require monitoring that will be used for comparison with future inservice results. This monitoring provides for trending of the aging effect, establishes a baseline for the degradation process, and the extent of degradation. The staff accepts this AMP's methodology for managing the applicable aging effects.

[*Acceptance Criteria*] The applicant indicates that detected flaws are compared to the acceptance criteria in the ASME Code, Section XI, Subsection IWE. The staff accepts the Code as the industry standard to manage aging effects based on the Code criteria.

[*Operating Experience*] Since ASME Code, Section XI, Subsection IWE, has been recently implemented by the industry, the operating experience is limited. The staff agrees that this program provides reasonable assurance (for the SCs subject to an AMR) that the applicable aging effects will be adequately managed for the period of extended operation.

ASME Inservice Inspection Program, Subsection IWF

In the LRA, Section 4.3.5 of Appendix B, the applicant identifies the loss of material, cracking, and changes in material properties as the aging effects requiring management. The applicant identifies ASME Code, Section XI, 1992 Edition with the 1993 Addenda, as the guidance used to manage the above-mentioned aging effects for the period of extended operation. However, in the letter to the NRC dated September 7, 2000, the applicant states that the correct ASME Code addenda is the 1992 Addenda. In addition, the applicant documented its intent to also use the NRC-approved alternative provided by the staff in a letter dated December 16, 1996. The purpose of the ASME Code, Section XI, Subsection IWF, is to identify and correct degradation of ASME Class 1,2,3 or MC component supports in accordance with 10 CFR 50.55a and ANO-1 TS 4.0.5.

[*Program Scope*] The applicant's program for IWF is in accordance with ASME Code, Section XI, Subsection IWF, and covers the component supports for ASME Class 1, 2, 3, or MC components. The staff found the program scope acceptable.

[*Preventative/Mitigative Actions*] The applicant did not identify any preventative actions, nor did the staff identify a need for such action.

[*Parameters Monitored/Inspected*] As described in the LRA, the parameters monitored for Subsection IWF were not acceptable to the staff. In the letter to the NRC dated September 7, 2000, the applicant committed to follow the requirements for monitoring the parameters as stated in ASME Code, Section XI, Subsections IWF. The applicant identified visual (VT-3) examinations for general mechanical and structural condition as further defined by Table IWF-2500-1 as the inspection activity. Upon including the applicant's commitments discussed in the letter, the staff found the monitored parameters and inspections acceptable.

[*Detection of Aging Effects*] The visual examinations of components as specified in ASME Code, Section XI, Subsection IWF, detect aging effects such as loss of material, cracking, and change in material properties. The staff accepts the methods prescribed by the Code for each class of component to be reliable and effective in detecting age-related degradation of components that are within the scope of license renewal.

[*Monitoring and Trending*] No trending activities are identified, nor are they required by IWF.

[*Acceptance Criteria*] The applicant indicates that detected flaws are compared to the acceptance criteria in the ASME Code, Section XI, Subsection IWF-3400. The staff accepts the Code as the industry standard to manage aging effects based on the Code criteria.

[*Operating Experience*] ASME Code, Section XI, Subsection IWF, has been implemented by the industry for a number of years, and no significant shortcomings have been identified. IWF will be revised by ASME and endorsed by the NRC, as deemed appropriate based on future operating experience. The staff agrees that continued implementation of IWF can provide assurance that aging of the in-scope component supports will be adequately managed for the period of extended operation.

ASME Inservice Inspection Program, Subsection IWL

In the LRA, Section 4.3.6 of Appendix B, the applicant identifies the loss of material for tendon anchorage as the aging effect requiring management. The LRA identifies ASME Code, Section XI, 1992 Edition with the 1993 Addenda as the guidance used to manage the loss of material for the period of extended operation. However, in the letter to the NRC dated September 7, 2000, the applicant states that the correct Code addendum is the 1992 Addenda. In addition, the applicant indicates that the NRC-approved alternative will be used for IWL inspections consistent with 10 CFR 50.55(a).

The purpose of ASME Code, Section XI, Subsection IWL, is to provide instructions and documentation requirements for assessing the quality and structural performance of the reactor building's post-tensioning systems and concrete surfaces.

[*Program Scope*] The program scope as stated in the LRA was not acceptable to the staff because it did not appear to cover all components included within its CLB as defined by 10 CFR 50.55a and ASME Code, Section XI, Subsection IWL. In the letter to the NRC dated September 7, 2000, the applicant committed to the full scope and requirements of 10 CFR 50.55a, Subsection IWL and the NRC-approved alternative. On the basis of the applicant's commitments, the staff found the program scope acceptable.

[*Preventative/Mitigative Actions*] The applicant did not identify any preventative actions for this AMP, nor did the staff identify a need for such action.

[*Parameters Monitored/Inspected*] In the LRA, the applicant identifies parameters that were not acceptable to the staff. In the letter to the NRC dated September 7, 2000, the applicant commits to following the requirements for monitoring the parameters as stated in ASME Code Section XI, Subsection IWL. The applicant identifies tendon surveillance and concrete surface examinations as the inspected parameters. On the basis of the applicant's commitments, the staff found the inspections acceptable.

[*Detection of Aging Effects*] The method of detecting aging effects for IWL as described in the LRA was not acceptable to the staff. In the letter to the NRC dated September 7, 2000, the applicant committed to the full scope and requirements of ASME Code, Section XI, Subsection IWL, and 10 CFR 50.55a, including the limitations under 10 CFR 50.55a(a) and 50.55a(b)(2)(ix) for this AMP. The staff accepts the methods prescribed by the Code for each class of component to be reliable and effective in detecting age-related degradation of components that are within the scope of license renewal.

[*Monitoring and Trending*] In the LRA, the applicant did not identify any trending activities. Trending of prestress losses between scheduled surveillances is required by 10 CFR 50.55a. In the letter to the NRC dated September 7, 2000, the applicant verifies that this AMP meets the full scope and requirements of ASME Section XI, Subsection IWL in accordance with 10 CFR 50.55a. On the basis of this AMP meeting the requirements for IWL and 10 CFR 50.55a with respect to monitoring and trending, the staff accepts this methodology to manage aging effects in this AMP.

[*Acceptance Criteria*] The acceptance standards for IWL are specified in Subsection IWL-3000. The staff accepts the Code as the industry standard to manage aging effects based on the Code criteria.

[*Operating Experience*] ASME Code, Section XI, Subsection IWL, has been recently implemented by the industry, therefore the operating experience is limited. In 1993, during the 20^{th} year in-service inspection, ANO-1 detected water in one tendon sheath; corrosion on one shim; and slightly low ultimate strength in one tendon. The shim was replaced. Tests showed that the low tendon strength was an original condition. Tendons are experiencing normal relaxation. The staff concludes that IWL provides reasonable assurance that aging of the applicable SCs will be adequately managed for the period of extended operation.

ASME Section XI, Augmented Inspections

In the LRA, Section 4.3.7, the applicant identifies a program of augmented inspections to manage aging effects such as loss of material and cracking for components that are beyond the scope of ASME Code, Section XI.

[*Program Scope*] In the LRA, Table 3.4-4, the applicant shows that the ASME Section XI, Augmented inspections are used to manage loss of material, cracking, and loss of mechanical closure integrity for stainless steel, brass, bronze and admiralty piping and valves in the auxiliary building sump and reactor building drains system. In the letter to the NRC dated September 12, 2000, the applicant clarifies that the augmented inspection program is not used to manage loss of mechanical closure integrity. However, local leak rate testing, reactor building leak rate testing and the boric acid corrosion prevention program are used to manage the loss of mechanical closure integrity for the carbon steel fasteners of the stainless steel and carbon steel reactor building isolation valves shown in Table 3.4-4 of the LRA. For the auxiliary building sump and reactor building drains system, the stainless steel piping and valves associated with penetration 68, the augmented inspection program manages loss of material and cracking. The augmented inspection program manages loss of material and cracking for the stainless steel decay heat pump room drain valves. In Table 3.4-4 the activity for the brass, bronze and admiralty valves should be preventive maintenance. This is consistent with the discussion of preventive maintenance in Appendix B, Section 4.1.5, of the LRA. No other component in the auxiliary building sump and reactor building drain system credits the augmented inspection program for aging management. The staff found the scope of the program to be sufficient.

[*Preventive/Mitigative Actions*] There are no preventive or mitigative actions associated with the inservice inspection program, nor did the staff identify a need for such action.

[*Parameters Monitored*] In the letter to the NRC dated September 12, 2000, the applicant states the following: “The welds of the piping wetted by the reactor building sump water will be subjected to volumetric examination. Specific welds in stainless piping of the main steam system will be subjected to volumetric examination. At least a one-time inspection of the penetration 68 piping and components will be accomplished. The method will be volumetric examination. At least a one-time inspection of the decay heat pump room drain valves will be performed using a volumetric method. Inspections of penetrations 10, 47, 58, and 64 will be performed using a volumetric method. These methods have proven effective in the industry for identifying cracking and loss of material.”

The staff agrees that a volumetric examination method is effective in detecting loss of material and cracking in the subject components.

[*Detection of Aging Effects*] The volumetric, surface, and visual examination of components along with system pressure tests are used to detect applicable aging effects (i.e., cracking, loss of mechanical closure integrity at bolted connections, and loss of material). The staff accepts the nondestructive examination methods in the inspection plan for augmented inspection to be reliable and effective in detecting age-related degradation of subject components.

[*Monitoring and Trending*] The frequency of inspections and the inspection methods are specified by the applicant's augmented inspection program outlined in the inservice inspection plan. The program will have provisions for the frequency of initial and successive inspections of applicable components in the auxiliary building sump and reactor building drain system. The applicant's program conforming to ASME Code requirements, provides a reasonable assurance for managing age-related degradation prior to impacting the intended safety functions of the applicable components.

[*Acceptance Criteria*] Detected degradation will be evaluated by comparing examination results to appropriate acceptance standards in ASME Section XI that have been established by industry consensus. The staff found the acceptance criteria set forth in the Code to be satisfactory for managing the effects of aging of these components.

[*Operating Experience*] In the letter to the NRC dated September 12, 2000, the applicant states that in reviewing the operating experience, no age-related failures were found for the applicable valves and piping in the auxiliary building sump and reactor building drains system. The augmented inspection for this system is a new program and has not yet been performed. Thus, no assessment of the effectiveness of the program can yet be made based on program experience. The applicant also stated that, in general, augmented inspections use the same nondestructive examination methods that are used for Section XI inspections on Class 1, 2, 3 SCs and, thus, these methods have been proven effective in the industry for detecting the applicable aging effects.

The staff concludes that the applicant's augmented inspection program will satisfactorily detect and manage the applicable aging effects of the SCs.

3.3.1.4.9.3 Conclusions

On the basis of the review described above, the staff finds that the applicant has demonstrated that the inservice inspection plan can be used to adequately manage the effects of aging associated with the Class 1, 2, 3, and MC components and component supports, and those components associated with the Section XI augmented inspections so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

3.3.1.5 References for Section 3.3.1

1. NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 — The License Renewal Rule," Revision 1, January 2000.
2. Working Draft, "NRC Generic Aging Lessons Learned Draft Report (GALL)," August 2000.
3. DG-1047, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," Working Draft, April 21, 2000.
4. 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."
5. "Arkansas Nuclear One - Unit 1, License Renewal Application," January 31, 2000.
6. NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment."

7. ASME Boiler and Pressure Vessel Code, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” 1992 Addenda.
8. 10 CFR 50.55a, “Codes and Standards.”
9. EPRI Report TR-105714, “PWR Primary Water Chemistry Guidelines.”
10. EPRI Report TR-102135-R4, “PWR Secondary Water Chemistry Guidelines.”
11. EPRI Report TR-107396, “Closed Cooling Water Chemistry Guidelines.”
12. EPRI Report TR-106229, “Service Water System Chemical Addition Guidelines.”
13. ASTM D975-1981, “Standard Specification for Diesel Fuel Oils.”
14. Letter from T. Martin (NRC) to T. Tipton (NEI), October 1, 1996.
15. NUREG-1526, “Lessons Learned from Early Implementation of the Maintenance Rule at Nine Nuclear Power Plants.”

3.3.2 Reactor Coolant System

The reactor coolant system (RCS) mechanical components that require an AMR as part of the LRA include the following: Babcock and Wilcox (B&W) designed vessels (i.e., the reactor vessel, the control rod drive mechanism pressure boundary, the pressurizer, and the once-through steam generators (OTSG), reactor vessel internals (specifically the plenum assembly, the core support barrel assembly, the lower internals assembly, the thermal shield, and the thermal shield restraint), reactor coolant pumps, and all Class 1 piping, valves, and bolting.

In the LRA, Section 3.2, the applicant describes the AMR of the RCS mechanical components. The applicant describes the applicable AMPs in Appendices A and B of the LRA.

In addition, the applicant describes the TLAAs applicable to RCS components in the following sections of the LRA:

LRA Section	Subject
4.1	Identification of Time-Limited Aging Analysis
4.2	Reactor Vessel Neutron Embrittlement
4.3	Metal Fatigue
4.8	Other Time-Limited Aging Analyses
4.8.1	Reactor Vessel Underclad Cracking
4.8.2	Reactor Vessel Incore Instrumentation Nozzles – Flow-Induced Vibration Endurance Limit
4.8.3	Leak Before Break
4.8.4	Reactor Coolant Pump Motor Flywheels

The staff reviewed the above sections of the LRA to determine whether the applicant has provided adequate information to meet the requirements stated in 10 CFR 54.21(a)(3) for managing the effects of aging of the RCS for license renewal.

The applicant's AMPs for the RCS are described in a series of Babcock and Wilcox Owners Group (B&WOG) topical reports. These reports are:

BAW-2243A	Demonstration of the Management of Aging Effects for the Reactor Coolant System Piping
BAW-2244A	Demonstration of the Management of Aging Effects for the Pressurizer
BAW-2248A	Demonstration of the Management of Aging Effects for the Reactor Vessel Internals
BAW-2251A	Demonstration of the Management of Aging Effects for the Reactor Vessel

The staff previously approved all of the B&WOG topical reports, having determined that they presented adequate information to meet the requirements of 10 CFR 54.21(a)(3) for managing the effects of aging on the RCS.

An applicant may incorporate NRC-approved reports by reference if the conditions of approval in the SER of the specific report are met. In the LRA, Section 2.3.1.2, the applicant uses the following process to meet the conditions stated in the SERs for incorporating approved B&WOG topical reports by reference:

- Comparison of the component intended functions for the RCS components under review. The ANO-1-specific component screening review first identifies the component intended functions and then compares these intended functions to those identified in the generic B&WOG topical reports. Differences were noted by the staff, and justifications were provided by the applicant.

C Identification of the items that are subject to an AMR. ANO-1 drawings and pertinent design and field change data are reviewed. The process establishes the full extent to which the scope of the generic B&WOG topical reports bound the ANO-1 RCS components.

C Identification of the applicable aging effects. An independent assessment of the applicable aging effects is performed by reviewing plant operating environment, operating stresses, and plant-specific operating experience. This reveals potential aging effects not identified in the generic B&WOG topical reports. Aging effects for items that are determined to be subject to an AMR, and that are not identified in the generic B&WOG topical reports, are evaluated.

The applicant states that Sections 2.3.1.3 through 2.3.1.9 of the LRA describe the RCS mechanical components that require an AMR. In the LRA, Section 3.2.1, the applicant also lists the B&WOG topical reports that are applicable to the ANO-1 RCS. The applicant incorporates these topical reports by reference.

3.3.2.1 RCS Piping and Letdown Coolers

RCS piping subject to an AMR includes portions of the American Society of Mechanical Engineers (ASME) Class 1 RCS pressure boundary that are connected to the following components: the reactor vessel, the OTSG (primary side), the pressurizer, and the reactor coolant pump. Other systems that are attached to the Class 1 piping and that contains ASME Class 1 components, include the core flood system, the makeup/high pressure injection system, and the decay heat/low pressure injection system. In addition, vents, drains, and instrument lines contain ASME Class 1 components. This RCS piping includes piping (e.g., fittings, branch connections, safe ends, and thermal sleeves), valve bodies (pressure-retaining parts of RCS isolation/boundary valves), bolted closures, and bolted connections.

In addition, two RCS letdown coolers, located inside the reactor building, are heliflow shell and tube heat exchangers with spiral Type 304 stainless steel tubes and manifolds, carbon steel casing shells, and carbon steel casing end plates. During normal operation, the letdown coolers cool the letdown flow from the RCS to prevent damage to the ion exchange resins of the purification system. The primary water enters the tubes at approximately 555°F and is cooled to approximately 120°F by intermediate cooling water (treated water) flowing through the shell.

3.3.2.1.1 Technical Information in the Application

The applicant reviews the current design and operation of the ANO-1 RCS piping using the process described in Sections 2.3.1.2 of the LRA and confirms that the ANO-1 ASME Class 1 piping is bounded by the description of Class 1 piping contained in BAW-2243A with regard to materials of construction and operating environment. Components within the Class 1 inservice inspection (ISI) boundary (i.e., subject to ASME Section XI, Subsection IWB boundary) were designed in accordance with the USAS B31.7 Class 1 standard. RCS items that are not within the scope of BAW-2243A, and subject to an AMR, include the resistance temperature element (RTE) thermowells, the instrumentation tubing, and the letdown coolers.

The fast response RTE connections include a thermowell mounted within the mounting boss. The thermowell, which is constructed from Type 304 austenitic stainless steel, was omitted from the scope of BAW-2243A. In addition, the evaluation boundary in BAW-2243A did not include non Class 1 instrumentation tubing, that connect the second isolation valve to the instrumentation. These items are constructed from austenitic stainless steel and are part of the RCS pressure boundary at ANO-1. The operating environment for the RTE thermowell and the instrumentation tubing is the same as for the RCS piping, which includes the primary coolant that is periodically monitored by the primary water chemistry monitoring program.

The tube side of the two letdown coolers was designed in accordance with ASME Section III, Class C, and the shell side was designed in accordance with ASME Section VIII.   The operating environments for the letdown coolers include primary water on the tube side and treated water from the intermediate cooling water system on the shell side.

In accordance with the requirements for the Class 1 components of ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, RCS piping is designed to accommodate service loading levels A through D. Operations under level A and level B service conditions contribute to normal aging stresses in the piping. The applicant states that ANO-1 has not been subjected to a level C or D event.

Effects of Aging

The aging effects applicable to the RCS piping are cracking, loss of fracture toughness, loss of material, and loss of bolting preload and are consistent with those described in BAW-2243A. The topical report identified a reduction in fracture toughness due to thermal embrittlement for RCS valves fabricated from cast austenitic stainless steel (CASS) as an applicable aging effect. In the LRA, Section 3.2.2, the applicant evaluates the materials of construction and the temperature environment of these valves.

On the basis of this evaluation, the applicant states that there is no Class 1 piping fabricated from CASS at ANO-1. However, Class 1 CASS valves were identified and are exposed to elevated temperature during power operation. The applicant has identified a threshold temperature of 482^{o}F (250^{o}C) below which reduction of fracture toughness is not an applicable aging effect. On the basis of this screening criterion, the applicant identifies three Class 1 valves fabricated from CASS, which are exposed to operating temperatures above the fracture toughness threshold value. These (two) valves are 2½-inch valves in the letdown line, and one 2½-inch pressurizer spray line block valve.

The instrumentation tubing and the RTE thermowells are fabricated from austenitic stainless steel and are subjected to the same environment as the other stainless steel piping in the RCS. Aging effects associated with these components are consistent with aging effects identified for stainless steel piping.

Aging effects for the letdown coolers include the following: cooler tube cracking as a result of fatigue and stress corrosion cracking on the tube surfaces exposed to intermediate cooling water, loss of material of tubes due to vibration, loss of material from the interior of the shell by corrosion, loss of material of the external surface of the heat exchanger by boric acid wastage, and loss of mechanical closure integrity.

Survey of Industry and ANO-1 Operating Experience

The applicant does not describe any additional industry experience other than that included in BAW-2243A applicable to RCS piping. However, the applicant has reviewed site-specific operating experience to validate the applicable aging effects for the RCS piping. This review included the station information management system, the condition reporting system, and the licensing event database.

The following five leaks associated with RCS small-bore piping were identified: a 1½-inch drain off reactor coolant pump (RCP) 32B in 1989-90, a leak at a high pressure injection/makeup (HPI/MU) vent line in 1989, pinhole leaks in a coupling adjacent to a decay heat valve in 1993, a core flood tank drain in 1996, and a circumferential crack on an HPI/MU drain in 1998. The applicant states that all the leaks and cracks were caused by vibrational fatigue due to design problems.

Loss of material caused by boric acid wastage was identified on the discharge cold leg HPI nozzle region. Operating experience also indicated that several elements of bolting maintenance practices at ANO-1 needed further improvement, such as personnel training, installation and maintenance procedures, plant-specific bolting degradation history, and corrective measures. The applicant states that appropriate actions have been taken to improve these elements of bolting maintenance at ANO-1.

Aging Management Programs

In the LRA, Section 3.2.2, the applicant identifies the following AMPs for the RCS piping and letdown coolers:

- boric acid corrosion prevention program
- primary water chemistry monitoring program
- ASME Section XI Inservice Inspection Program, Subsection IWB (as supplemented by Code Case N-560 for Examination Category B-J welds)
- leakage detection in reactor building
- Alloy 600 AMP

- bolting and torquing activities for small-bore piping and small-bore nozzles inspections

The applicant concludes that these programs would manage the effects of aging in such a way that the intended function of the components of the RCS piping and letdown coolers would be maintained consistent with the CLB under all design loading conditions for the period of extended operation.

3.3.2.1.2 Staff Evaluation

In accordance with 10 CFR 54.21(a)(3), the staff reviewed the information from Section 2.3.1.3 (Table 2.3-2), Section 3.2.2 (including Table 3.2-1), Appendix B (Sections 4.1, 4.3, 4.4, 4.5, 4.6, 4.12) of the LRA, and the staff's final SER on topical report BAW-2243A, regarding the applicant's demonstration that the effects of aging will be adequately managed so that the intended functions would be maintained consistent with the CLB for the period of extended operation for the RCS piping and letdown coolers.

There are several action items in the staff's final SER on BAW-2243A. The action items, applicant's responses, and staff's evaluations are provided in the following paragraphs.

Action Items from Previous Staff Final SER for BAW-2243A

The applicant resolved the seven "Renewal Applicant Action Items" from BAW-2243A in the following manner:

- Item 1: When incorporating BAW-2243A in its renewal application, the license renewal applicant is to verify that its plant is bounded by the topical report.

  Response: The applicant participated in the development of BAW-2243A by providing ANO-1-specific design and operational information. The applicant has reviewed the current design and operation of the ANO-1 RCS piping using the process described in Section 2.3.1.2 of the LRA, and confirms that the ANO-1 RCS piping is bounded by the description contained in BAW-2243A. The staff found this response to be acceptable.

- Item 2: The renewal applicant is to commit to programs described as necessary in the report to manage the effects of aging during the period of extended operation on the functionality of the RCS piping components.

  Response: The applicant considers the programs described in Section 3.2.2, and Appendices A and B of the LRA to be commitments. The staff found this response to be acceptable.

- Item 3: A summary description of these programs is to be provided in the license renewal SAR supplement in accordance with 10 CFR 54.21(d).

  Response: Summary descriptions of these programs are contained in Appendix A of the LRA. The staff found this response to be acceptable.

- Item 4: Any deviations from the AMPs described within this report as necessary to manage the effects of aging during the period of extended operation to maintain the functionality of RCS piping components or other information presented in the report, such as materials of construction and edition of the ASME Section XI Code (including mandatory appendices), will have to be identified by the renewal applicant and evaluated on a plant-specific basis in accordance with 10 CFR 54.21(a)(3).

  Response: In the LRA, the applicant states that there are no deviations from the AMPs described in BAW-2243A, or other information presented in the report. However, the NRC staff found that this may not be the case, and in a letter to the applicant dated May 5, 2000, the staff requested additional information to clarify the discrepancies. The staff reviewed the applicant’s response, and concluded that there were still five open items that needed to be addressed. By letter to the NRC dated October 3, 2000, the applicant provides a revised response to the staff’s RAI, which addresses the historical results of small-bore piping and Alloy 600 piping and nozzles inspections. Also, by letter dated November 2, 2000, the applicant provides additional responses addressing all five open items. The specific concerns, and the applicant’s resolution to these concerns are discussed in more detail in the following paragraphs of this section of the staff’s SER.

- Item 5: The B&WOG defers to the renewal applicant referencing this topical report for the development of the details of the inspection of the Alloy 82/182-clad hot-leg segment and plant selection of that inspection, and the sample inspection of small-bore RCS piping. The applicant must provide details of these two augmented inspection programs in its renewal application for staff review and approval.

  Response: In the LRA, the applicant states that descriptions of the applicable AMP are provided in Appendix B of the LRA. However, during the staff’s initial review the applicant had not demonstrated how the small-bore piping and small-bore nozzle inspection program, and the Alloy 600 AMP will bound the required (two), one-time augmented inspection programs suggested by the staff in Section 4.4 of the final SER for BAW-2243A. The two augmented inspection programs are among the five open items indicated in the response to Action Item 4, and the applicant satisfactorily responded to these two open items in a letter to the NRC dated November 2, 2000. The staff found these responses to be acceptable.

- Item 6: Since B&WOG elected to exclude TLAAs applicable to the RCS piping components from the scope of the topical report and indicated that they will be resolved on a plant-specific basis, any renewal applicant referencing this report will have to evaluate TLAAs applicable to the RCS piping components in its renewal application in accordance with the requirements of 10 CFR 54.21(c).

  Response: Evaluations of ANO-1-specific TLAAs are provided in Sections 4.3 and 4.8 of the LRA. The staff found this response to be acceptable.

- Item 7: Since the staff has made no finding relative to whether BAW-2243 constitutes the complete list of RCS piping components subject to an AMR or the adequacy of the scoping methodology, the individual applicant must identify and list the SCs subject to an AMR and describe a methodology for developing this list as part of the LRA.

Response: A list of RCS components that are subject to an AMR is provided in Section 2.3 of the LRA. The methodology for developing and maintaining this list of components is consistent with the guidance contained in NEI 95-10, Revision 0. The staff found this response to be acceptable.

3.3.2.1.2.1 Effects of Aging

Two 2½-inch valves in the letdown line and one 2½-inch pressurizer spray line block valve are fabricated from CASS and are exposed to operating temperatures above the threshold that makes them susceptible to a reduction in fracture toughness due to thermal embrittlement. The applicant states that the saturated lower bound fracture toughness of these valve bodies is approximately equal to the fracture toughness of austenitic weldments in pipes that were fabricated using the submerged arc welding process. Therefore, the applicant concludes that volumetric inspections of stainless piping welded joints for small-bore piping in the letdown piping or the pressurizer spray piping will bound the subject valves since valve bodies have thicker walls and lower stresses than adjacent piping.

The following five leaks associated with RCS small-bore piping were identified at ANO-1:

- 1½-inch “drain off RCP 32B in 1989–90
- HPI/MU vent line in 1989
- coupling adjacent to a decay heat valve in 1993
- core flood tank drain in 1996
- HPI/MU drain in 1998

The applicant states that these leaks were caused by vibrational fatigue resulting from design deficiencies, which have since been corrected.

Both letdown coolers are heliflow, shell and tube, heat exchangers with spiral Type 304 stainless steel tubes and manifolds, carbon steel casing shells, and carbon steel casing end plates. In the LRA, Table 3.2-1, the aging effects associated with stainless steel components are cracking, loss of material, and loss of mechanical closure integrity. The aging effect associated with carbon steel components is loss of material.

The staff found the aging effects identified by the applicant for the RCS piping and letdown coolers to be consistent with published literature and industry experience and, therefore, acceptable.

3.3.2.1.2.2 Aging Management Programs

The staff evaluation of the applicant’s AMPs focused on the program elements rather than details of specific plant procedures. The staff’s approach to evaluating each program and activity used to manage the applicable aging effects is described in Section 3.3.1 of this SER.

The staff’s review of AMPs that apply to the RCS piping and letdown coolers may be found in Section 3.3.2.4 of this SER. These programs include primary water chemistry monitoring program, ASME Section XI inservice inspection program, subsection IWB, leakage detection in reactor building, and bolting and torquing activities. The Alloy 600 AMP is evaluated in Section

3.3.2.3 of this SER. The small-bore piping and small-bore nozzle inspection program is evaluated below.

In the LRA, the applicant identifies Section XI ISI, Examination Category B-P and leakage detection in reactor building as the AMPs for the letdown coolers. In response to Generic Letter (GL) 88-05, the applicant has a boric acid corrosion prevention program to locate coolant leakage and/or evidence of boric acid wastage on carbon steel components and bolts. In the LRA, Table 3.2-1, the applicant does not include this AMP for managing the applicable aging effects of the letdown coolers whose casing shell, end plates, and some bolting are made of carbon steel. However, in a letter to the NRC dated September 6, 2000, the applicant confirmed that the boric acid corrosion prevention program is also applicable to letdown coolers for managing the loss of material for these components.

In the LRA, Section 3.2.2, the applicant discusses the aging effects caused by treated water supplied by the intermediate cooling water system on the letdown cooler tube surfaces, but does not identify the auxiliary systems chemistry monitoring program as an applicable AMP for letdown coolers. In a letter to the NRC dated November 2, 2000, the applicant included the auxiliary systems chemistry monitoring program as an applicable AMP for managing fatigue and stress corrosion cracking on the letdown cooler stainless steel tube surfaces exposed to the treated water. The staff’s evaluation of the auxiliary systems chemistry monitoring program may be found in Section 3.3.2.7 of this SER.

BAW-2243A, Table 4-1, identifies the RCS piping component groupings, and AMPs that manage the applicable aging effects. The AMPs applicable to RCS piping have been approved in the staff’s final SER for BAW-2243A. Therefore, the staff found that these AMPs provide an acceptable demonstration that the aging effects of RCS piping components, within the scope of the topical report, will be adequately managed. A comparison between Table 4-1 of BAW-2243A and Table 3.2-1 of the LRA indicates that the applicant has either committed to all AMPs that are identified in Table 4-1of BAW-2243A or has satisfactorily addressed the alternate AMPs as discussed in the following paragraphs.

The applicant credits the examinations performed under the ASME Section XI, ISI program with managing the effects of aging for Class 1 components during the period of extended operation. For piping above 1 inch NPS, examination categories B-P for all pressure retaining components, B-F for welds between dissimilar materials, and B-J for welds between similar materials are considered appropriate for managing the applicable aging effects.

In a letter to the NRC dated September 6, 2000, the applicant states that Code Case N-560 includes provisions for the Examination Category B-F. However, there is no mention of such provision found in the Code Case. Also, the applicant implies that the NRC, in a letter to the applicant entitled “Risk-Informed Alternative to Certain Requirements of ASME Code Section XI, Table IWB-2500-1 at Arkansas Nuclear One, Unit 1 (TAC MA2023),” dated August 25, 1999, included the dissimilar metal piping welds that are inspected at ANO-1 under Code Case N-560. The NRC letter did not discuss the dissimilar metal piping welds that are inspected at ANO-1. Further, the staff specifically indicated that “[t]his authorization does not constitute an NRC approval of Code Case N-560 for generic use.” In addition, Examination Category B-J requires sample weld inspections while Examination Category B-F requires all welds to be inspected in each inspection cycle.

In a follow-up letter to the NRC dated October 10, 2000, the applicant states that the 1989 Addenda of the ASME Section XI Code moved the dissimilar welds in the piping (i.e., safe-end to pipe welds) from Examination Category B-F to Examination Category B-J, and left the dissimilar metal piping butt welds and socket welds (i.e., nozzle to safe-end welds) in Examination Category B-F. Therefore, at ANO-1, the nozzle to safe-end welds are inspected in accordance with the requirements of Examination Category B-F of ASME Section XI, 1992 Edition with portions of the 1993 Addenda. The safe-end to pipe welds are inspected in accordance with the requirements of the risk-informed Code Case N-560, which is documented in the applicant's response to RAI # 17, which resulted from review of the ANO-1 risk-informed inservice inspection pilot application (Letter from J.D. Vandergrift to the USNRC, dated May 17, 1999). The staff found this explanation to be acceptable.

Pipe sizes above 4 inch NPS are subject to both volumetric and surface examinations in addition to leakage detection, while piping >1-inch and < 4-inch NPS is subjected to only surface examination and leakage detection. Piping #1-inch NPS is subjected to leakage detection only. Operating experience has shown that cracking originates from the inside surface of the piping and therefore, any through-wall crack in piping <4 inch NPS may be left unnoticed until the piping fails under a design basis loading. Therefore, the staff concluded in Section 4.4.2 of the final SER for BAW-2243A that one time volumetric inspection of small-bore piping is a requirement for license renewal for B&W plants. The staff also suggested that the applicant develop a program to manage cracking of 1 inch NPS piping and less. Since the subject piping is exempt from surface and volumetric examination in accordance with ASME Section XI inspections, the staff questioned why 1-inch NPS piping and less is not included in the risk-informed selection of small-bore piping welds.

In a letter to the NRC dated September 6, 2000, the applicant states that the small-bore piping and small-bore nozzle inspection program requires use of risk-informed volumetric examination and therefore, one-time volumetric inspection of small-bore piping with >1 inch and <4 inch NPS is not necessary at ANO-1. However, the small-bore piping and small-bore nozzle inspection program at ANO-1 does not include piping #1 inch NPS in its risk-based selection process. The staff requested further clarification on the small-bore piping and small-bore nozzle inspection AMP given in Appendix B, Section 4.3.8 of the LRA.

In a letter to the NRC dated November 2, 2000, the applicant states that none of the sample welds selected under the small-bore piping and small-bore nozzles inspection AMP are socket welds. The risk-informed selection provides a representative sample of piping that has the same material and environment combination, and the same identified aging effects as the 1-inch and smaller piping. In addition, the risk (based on probability and consequences) of failure of the 1-inch and smaller piping is bounded by the risk of failures at locations selected for inspection in the small-bore piping program. Consistent with the ASME Section XI Code, 1-inch and smaller piping is exempt from surface and volumetric examinations since the consequences of leakage from these smaller pipe sizes are less than the consequences of leakage from the larger piping. Operating experience has confirmed that leakage from 1-inch and smaller piping is readily detected and corrected prior to loss of the system function. Therefore, the staff found that when combined with the ANO-1 small-bore piping and small-bore nozzles inspection program, the existing ASME Section XI visual inspection of the 1-inch and smaller piping provides the necessary management to address potential aging effects prior to loss of function.

To manage potential cracking by PWSCC of Alloy 600 and Alloy 82/182 locations including the hot-leg flow meter element in the RCS piping, the applicant has implemented the Alloy-600 AMP at ANO-1. The program was developed on the basis of recommendations provided by the B&WOG, as a result of NRC's issuance of Information Notice (IN) 90-10. The scope of the program includes the flow meter section of the hot-leg. The staff notes that the sample selection for this AMP includes monitoring the most susceptible locations (for ANO-1, these locations are identified to be piping components in the pressurizer) to bound the Alloy 600 items and Alloy 82/182 weld locations.

In a letter to the NRC dated September 6, 2000, the applicant states that in March 2000, cracks were discovered in a number of ANO-1 Alloy 600 RCS hot-leg level instrumentation nozzles during a visual inspection. These nozzles were field-installed in 1986, and were of a different design than other RCS nozzles. The nozzle assemblies were two piece designs with a corrosion barrier sleeve welded on the inside and outside diameters of the hot-leg pipe and a nozzle inserted in the sleeve welded to the outside diameter of the hot-leg pipe. Unlike other RCS piping nozzles, these field-installed nozzles were not stress relieved. The cracking occurred in the weld material and not in the nozzle or sleeve. A root cause evaluation determined the failure mode to be cracking caused by PWSCC. The design of the nozzles resulted in high thermal and residual stresses at the root of the Alloy 182 weld (a weld material similar to Alloy 600) that connects the Alloy 600 nozzle to the ferritic piping. The high stresses led to the cracking of the welds. These specific welds were not directly evaluated in the Alloy 600 susceptibility model as operating experience indicated that nozzles and sleeves are more susceptible to PWSCC than the Alloy 82/182 welds that attach the Alloy 600 nozzles and sleeves to ferritic steel. In addition, the residual stresses due to field fabrication were not accounted for when conducting the susceptibility ranking. Proper consideration of residual stresses due to field installation at the hot-leg level tap locations would have resulted in the highest susceptibility rating for these items. These subject hot-leg level taps are the only Alloy 600 RCS piping penetrations that have not been stress relieved.

The Alloy 600 susceptibility model did not indicate a high susceptibility to cracking at these weld locations since they were not specifically modeled. Apparently, the thermal stresses imposed by the sleeves caused bending stresses in the welds that promoted PWSCC. A new design was developed to eliminate the high thermal stresses associated with welding. Nozzle replacements have been performed on all but one of these nozzles using Alloy 152 (a weld material similar to Alloy 690) and Alloy 690 nozzle material, which has been demonstrated to be resistant to PWSCC. The repaired hot-leg level taps have a low susceptibility to cracking since the repairs were made using materials that are resistant to PWSCC. An NRC-approved temporary weld repair was performed on the one remaining nozzle that will be replaced during refueling outage 1R16. This is acceptable to the staff.

In response to GL 85-20, for managing the potential cracking of the thermal sleeves, each plant examined the HPI connection thermal sleeves and replaced those susceptible to cracking with newly-designed thermal sleeves. The status of the HPI/MU branch connections at ANO-1 is described in Section 4.3.4.4 of the LRA. The applicant indicates that ASME Section XI, Examination Category B-J, as modified by Code Case N-560, is an augmented examination of thermal sleeves that will manage this aging effect. The applicant has repaired nozzles with loose or damaged thermal sleeves. Any increase in the size of the gap between the thermal sleeve and the safe end has been corrected. In addition, the applicant developed procedures for maintaining adequate minimum flow, instituted an augmented inspection program for the nozzles, and performed stress analyses with the methodology and scope of inspection recommended by the B&WOG safe-end task force. The applicant indicates that ultrasonic testing of the knuckle region of the HPI nozzles every fifth refueling cycle, and radiography of the thermal sleeves will continue through the period of extended operation. In its final SER for BAW-2243A, the staff agreed that the augmented inspection using volumetric methods is necessary to manage potential cracking of the HPI thermal sleeves during the period of extended operation. Flaws in weld or base metal, which cannot be accepted based on geometry screening or the fracture analysis methods of the ASME Code, are corrected by repair or replacement.

Valve bodies pressure retaining welds and valve bodies are inspected under examination categories B-M-1 and B-M-2, respectively. These inspections include volumetric examination of welds and visual VT-3 examination of internal surfaces. As indicated in Table 3.2-1 of the LRA, the effects of thermal embrittlement on CASS valve components of the RCS at ANO-1 are managed by ASME Section XI, Examination Category B-P. Both BAW-2243A and its final SER indicate that Examination Categories B-M-1 for pressure retaining welds in valve bodies and B-M-2 for valve bodies with the flaw evaluation procedure specified in IWB-3640, are required to manage both cracking and loss of fracture toughness in CASS items due to thermal aging. In a letter to the NRC dated September 6, 2000, the applicant states that the following three ANO-1 ASME Class 1 valves fabricated from CASS are susceptible to a reduction in fracture toughness by thermal embrittlement: two 2½-inch valves in the letdown line and the 2½-inch pressurizer spray line block valve. These valves do not receive a surface examination in accordance with Examination Category B-M-1 since these valve bodies do not contain welded joints. In addition, since these valves are <4 inch NPS, they are not required to be inspected in accordance with Examination Category B-M-2. Therefore, no ASME Class 1 CASS valves at ANO-1 are inspected in accordance with Examination Categories B-M-1 or B-M-2, and the CASS evaluation procedure reported in BAW-2243A does not apply.

The staff accepts the applicant's position on these ASME Section XI examination categories, because the staff position on CASS components in the NRC letter dated May 19, 2000, concludes that the valve bodies with NPS less than 4 inches are adequately covered by existing inspection requirements in Section XI of the ASME Code. Screening for susceptibility to thermal aging is not required and the current ASME Code inspection requirements are sufficient. This NRC position is supported by a bounding fracture analysis finding that valves within this range do not require additional inspection or evaluation to demonstrate that the material has adequate toughness, even for severe thermal embrittlement conditions.

For pressure retaining bolting, examination Category B-G-2 includes visual VT-1 examination of all bolts, studs, and nuts. Examination Category B-P is also used for the valves and bolting to detect system leakage.

The staff found that the applicant has appropriately identified the AMPs applicable to RCS piping and letdown coolers, and these AMPs will be adequate to detect aging effects in RCS piping and letdown coolers such that if unacceptable degradation is detected, the applicant will undertake further programmatic actions, including repair and replacement, as necessary, to manage the effects of aging.

Small Bore Piping and Small Bore Nozzles Inspections

Small bore piping is defined by ASME Code as piping less than 4 inch NPS, which includes pipes, fittings, and branch connections fabricated from stainless steel, Alloy 600, and clad carbon steel. The current ASME Section XI inspection requirements for managing cracking of small-bore piping welds do not require volumetric inspection for piping between 1 and 4-inch NPS, and do not require either surface or volumetric inspections for piping #1 inch NPS. The staff concluded, therefore, that the current inspection programs do not fully assure the management of potential weld cracking for the period of extended operation. Further, in many instances, small-bore piping cannot be isolated from the RCS. Specifically, the staff, in its final SER for BAW-2243A, suggested that a one-time volumetric inspection be performed for selected components between 1 inch and 4 inch NPS, and a program be established to manage cracking of 1 inch NPS and less. Any unacceptable indication of cracking of piping welds requires that an engineering analysis be performed to determine proper corrective action.

The applicant states that the existing small-bore piping and small-bore nozzle inspections at ANO-1 involve volumetric examination of the most risk-significant welds and, therefore, satisfy the one-time volumetric inspection for piping between 1 inch and 4 inch NPS, as required by the staff. The program uses ASME Code Case N-560, in lieu of the requirements specified in the 1992 Edition of ASME Section XI, Table IWB-2500-1, Examination Category B-J. In a letter to the NRC dated September 6, 2000, the applicant demonstrates that, combined with the ANO-1 Small Bore Piping and Small Bore Nozzles Inspection Program, the existing ASME Section XI visual inspection of the 1-inch and smaller piping provides the necessary management to address potential aging effects prior to loss of function.

[*Program Scope*] The program includes ISI to monitor the effects of cracking on the intended function of small-bore piping (i.e., piping less than 4 inch NPS) of the RCS and connected lines.

[*Preventive/Mitigative Actions*] There are no preventive/mitigative actions associated with this program, nor did the staff identify a need for such.

[*Parameters Monitored*] The applicant uses a risk-informed method to select RCS piping welds for inspection based on ASME Code Case N-560. This method consists of two essential elements: (1) a degradation mechanism evaluation to assess the failure potential of the piping system under consideration, and (2) a consequence evaluation to assess the impact on plant safety in the event of a piping failure. A sample population of welds in the following Class 1 small-bore piping is selected. Currently, the applicant selected the following sample items: 1½-inch pressurizer spray line, 2½-inch makeup and purification lines, 2½-inch letdown line, and 1½-inch cold leg suction drain line. The program description includes volumetric examination of inspection locations selected on the basis of detailed evaluations of material susceptibility, operating environment, stress, and risk.

[*Detection of Aging Effects*] The applicant performs volumetric examination of selected weld locations to detect aging effects (i.e., cracking or indications, loss of material), before there is a loss of intended function.

[*Monitoring and Trending*] The applicant periodically monitors weld degradation. The inspection frequency is established in accordance with ASME Code Case N-560.

[*Acceptance Criteria*] If flaws or indications exceed the acceptance standards of ASME Paragraph IWB-3400, they are evaluated in accordance with IWB-3132, and additional examinations are performed in accordance with IWB-2430.

[*Operating Experience*] The risk-informed method for selecting welds for inspection incorporates the elements necessary to manage cracking of small-bore piping and small-bore nozzles during the period of extended operation. ANO-1 experienced cracked welds in an RCS drain line in 1989 and the root cause of the cracking was determined to be a weld defect that propagated by vibrational fatigue. Another incident involved vibration induced socket weld failure in a vent and drain line due to piping vibration. The applicant took appropriate steps to isolate several socket welds from high vibration loads. The applicant indicates that failure of Class 1 small-bore pipe has been rare at ANO-1.

On the basis of the elements described in the LRA, the staff found the small-bore piping and small-bore nozzles inspections program to be acceptable for managing the applicable aging effects for the period of extended operation.

3.3.2.1.3 Conclusions

The staff has reviewed the information included in Section 3.2.2, "RCS Piping and Letdown Coolers," and Appendices A and B of the LRA and additional information provided by the applicant in response to the staff RAIs. The staff concludes that there is reasonable assurance that the applicant has demonstrated that the effects of aging associated with the RCS piping and letdown coolers will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

3.3.2.2 Pressurizer

The B&WOG topical report BAW-2244A, "Demonstration of the Management of Aging Effects for the Pressurizer," describes the pressurizer as a vertical cylindrical vessel connected to the reactor vessel outlet piping by the surge line piping. The pressurizer vessel is protected from thermal effects by a thermal sleeve in the surge line nozzle, and by a surge diffuser located above the surge pipe entrance to the vessel. The electrically-heated pressurizer establishes and maintains the reactor coolant pressure within the prescribed limits, and provides a surge chamber and a water reservoir to accommodate reactor coolant volume changes during operation.

3.3.2.2.1 Technical Information in the Application

In the LRA, Section 2.3.1.4, the applicant states that the current design and operation of the ANO-1 pressurizer have been reviewed and compared with those specified in BAW-2244A. On the basis of this review, the applicant identifies the following pressurizer SCs at ANO-1 that are subject to an AMR. The applicant lists the following SCs of the pressurizer in Section 3.2.3 of the LRA that were subject to an AMR:

- pressurizer vessel
- nozzles
- other pressure-retaining items

C bolted closures
C integral attachments
C immersion heaters
C

Intended Functions

As described in BAW-2244A, Section 3, the only intended function of the B&W pressurizer at ANO-1 is to maintain the integrity of the RCS pressure boundary.

Effects of Aging

In the LRA, Section 3.2.3, the applicant states that the operating environment of the ANO-1 pressurizer is consistent with that described in Section 3 of BAW-2244A. The ANO-1 primary water chemistry monitoring program includes specifications to periodically monitor the primary coolant parameters. Limitations are established on dissolved oxygen, halides, and other impurities. Corrective actions are taken in the event that the primary coolant parameters are out of specification.

In accordance with the requirements for the Class 1 components of ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, the pressurizer is designed to accommodate all service loading levels (i.e., A through D). Operation under level A and B service conditions contribute to the normal aging stresses for the pressurizer. Since ANO-1 has not been subjected to level C and D events, the applicant states that BAW-2244A bounds the ANO-1 pressurizer with respect to the qualitative assessment of stress.

The following aging effects are listed in Table 3.2-1 of the LRA:

C Aging of the pressurizer vessel, which is made from stainless steel clad carbon steel, may cause cracking at welded joints, and cracking of the stainless steel cladding on the vessel in primary water.

C Aging of full-penetration welded nozzles, NPS > 1-inch, made from stainless steel clad carbon steel, may cause cracking at the welds in primary water.

C Aging of safe-ends of full-penetration nozzles, NPS > 1-inch, made from Alloy 600, may cause cracking of the welds in primary water.

C Aging of pressure-retaining partial-penetration welds, NPS # 1-inch, made from Alloy 600, may cause cracking at or near the weld in primary water.

C Aging of dissimilar metal welds, NPS > 1-inch, made from Alloy 82/182, may cause cracking of the weld in primary water.

C Aging of immersion heaters (sheaths, end plugs, and welds) and the weld that connects the heaters to the diaphragm plate, made from stainless steel, may cause cracking in primary water.

- Aging of integral attachments, made from carbon steel, may cause weld cracking in the reactor building environment.

- Aging of manway studs, > 2-inches in diameter, made from low-alloy steel, may cause cracking, loss of material, and loss of mechanical closure integrity, from boric acid corrosion.

- Aging of heater bundle studs, # 2-inches in diameter, made from low-alloy steel, may cause cracking, loss of material, and loss of mechanical closure integrity, from boric acid corrosion.

Survey of Industry Operating Experience

With respect to industry experience with B&W pressurizer, the applicant references Section 3 of BAW-2244A that gives a detailed description of pressurizer aging effects, and lists possible and observed degradation for pressurizer components that are within the scope of license renewal. The applicant describes aging effects in terms of component, material, environment, and stress level.

To validate its choice of applicable aging effects for ANO-1, the applicant conducts a search for instances of aging at ANO-1 using the station information management system, condition reporting system, and the licensing event data base. The applicant concludes that all aging effects were bounded by those given in BAW-2244A.

Aging Management Programs

In the LRA, Section 3.2.3, the applicant identifies the following AMPs that are used to manage the effects of aging:

- boric acid corrosion prevention program
- primary water chemistry monitoring program
- ASME Section XI Inservice Inspection Program, Subsection IWB (as supplemented by Code Case N-560 for Examination Category B-J welds)
- reactor building leakage detection
- Alloy 600 AMP
- bolting and torquing activities
- small-bore piping and small-bore nozzle inspections
- pressurizer examinations (new program for license renewal, described in Appendix B of the LRA)

The applicant concludes that these programs would manage aging effects in such a way that the intended function of the components of the pressurizer would be maintained consistent with the CLB, under all design loading conditions during the period of extended operation.

3.3.2.2.2 Staff Evaluation

In accordance with 10 CFR 54.21(a)(3) the staff reviewed the information included in Sections 2.3.1.4 (Table 2.3-3), 3.2.3 (Table 3.2-1), Appendix B Sections 3.4, 4.1, 4.3, 4.4, 4.5, 4.6, 4.12

of the LRA, and the final SER for BAW-2244A, regarding the applicant's demonstration that the effects of aging will be adequately managed so that the intended functions will be adequately maintained consistent with the CLB for the period of extended operation for the pressurizer.

From Section 3.3.2.3.1 of this SER, the staff found that the only intended function for the pressurizer is to maintain the integrity of the RCS pressure boundary.

There are several Renewal Applicant Action Items in the final SER for BAW-2244A . The Action Items, the applicant's responses, and the staff's evaluations are given below.

Action Items from Previous Staff Evaluation of BAW-2244A

As discussed below, the staff found that four of the six responses provided by the applicant regarding the Renewal Application Action Items from this report resolve the applicable action items. The two exceptions are Action Items 4 and 5.

- Item 1: When incorporating BAW-2244A in its renewal application, the renewal applicant is to verify that its plant is bounded by the topical report.

  Response: The applicant participated in developing BAW-2244A by providing ANO-1-specific design and operational data. The applicant states in Section 3.2.3 of the LRA that ANO-1 is bounded by the topical report. The staff found this response to be acceptable.

- Item 2: The renewal applicant is to commit to programs identified as necessary in the topical report to manage the effects of aging on the functionality of the pressurizer.

  Response: Program descriptions in Table 3.2-1 of the LRA are considered by the applicant to be commitments to manage aging effects in the pressurizer. A new program commitment, "Pressurizer Examinations," is described in Section 3.4 of Appendix B of the LRA in which the integrity of vessel cladding is evaluated. This program is in response to an Open Item that was included in the SER for BAW-2244A, in which cladding failure evaluation was required by the staff.

  In a letter to the applicant dated May 5, 2000, the staff requested confirmation from the applicant that the tripod legs, the surge nozzle to stainless steel safe-end joint, and stainless steel nozzle forgings are included along with the components managed by the pressurizer examination program defined in Section 3.4, Appendix B, of the LRA. They were not specifically mentioned in Table 3.2-1 (pp. 3-29 and 3-30 of the LRA) since this table is general in its format and does not specifically identify components. In its response, the applicant states that the first two components will be managed by the pressurizer examination program. With respect to the stainless steel nozzle forgings, the applicant states that no pressurizer nozzle forgings are made from stainless steel. The staff found these responses to be acceptable.

- Item 3: A summary description of these programs is to be provided in the license renewal FSAR supplement in accordance with 10 CFR 54.21(d).

Response: A summary of the required existing AMPs is given in Table 3.2-1 of the LRA. The commitment to the new "Pressurizer Examinations" program is included in Section 16.1.4 of the FSAR supplement.

C Item 4: Any deviations from the AMPs described in the topical report as necessary to manage the effects of aging during the period of extended operation, to maintain the functionality of the pressurizer, or other information presented in the report, such as materials of construction and edition of the ASME Code Section XI (including mandatory appendices) must be identified by the renewal applicant and evaluated on a plant-specific basis in accordance with 10 CFR 54.21(a)(3).

Response: No deviations from the AMPs described in BAW-2244A have been identified by the applicant. However, the staff found that some deviations exist since new program activities, not addressed in the topical report, are to be implemented by the applicant. In addition, although the applicant has appropriately identified all AMPs applicable to the pressurizer, the aging effects and associated AMPs for managing these aging effects related to each pressurizer component or grouping items listed in Table 3.2-1 of the LRA are not complete when compared to those included in Table 4-1 of BAW-2244A. However, by letter dated November 2, 2000, the applicant has satisfactorily addressed the remaining AMPs applicable to the pressurizer components. The staff found this response to be acceptable, and evaluates the AMPs in the following paragraphs of this section of the SER.

C Item 5: Since the B&WOG defers the development of details of the additional sample volumetric inspection program of small-bore nozzles and safe ends to the renewal applicant referencing this topical report, the renewal applicant must provide details of the additional sample inspection program in its renewal application for staff review and approval.

Response: The applicant describes the small-bore piping inspection program in Appendix B, Section 4.3.8 of the LRA. Staff evaluation of the program is given in Section 3.3.2.3.2.2 of this SER. The staff initially found that the applicant had not committed to performing the one time volumetric inspection of small-bore piping and small-bore nozzles for piping >1-inch and <4-inch NPS and piping # 1-inch NPS as discussed in Section 4.6 of BAW-2244A. However, by letter dated November 2, 2000, the applicant satisfactorily addressed the staff's concerns relating to this AMP. The staff found this response to be acceptable, and discusses the concerns in more detail in the following paragraphs of this section of the SER.

C Item 6: Since the B&WOG elected to exclude TLAAs applicable to the pressurizer from the scope of the topical report and stated that they will be resolved on a plant-specific basis, any license renewal applicant referencing this report must evaluate the TLAAs applicable to the pressurizer in its renewal application, in accordance with the requirements of 10 CFR 54.21(c).

Response: The TLAAs pertinent to ANO-1 are given in Table 4.1-1 of the LRA. They include the metal fatigue and analytical evaluation of flaw analyses. These are common to the reactor vessel TLAAs, and are evaluated by the staff in Section 4 of this SER.

3.3.2.2.2.1 Effects of Aging

The aging effects identified by the applicant as being applicable to the pressurizer are listed in Table 3.2-1 of the LRA, and are summarized in Section 3.3.2.3.1 of this SER. The list of aging effects given in Table 3.2-1 of the LRA is not bounded by the aging effects given in Table 4.1 of BAW-2244A. Specifically, loss of material on the external surfaces of several pressurizer components (e.g., vessel, nozzles, piping) exposed to boric acid leakage is not identified by the applicant in its LRA. In addition, the applicant had not explained why this aging effect is not applicable to these pressurizer components at ANO-1.

In a letter to the NRC dated September 6, 2000, the applicant states that the first row of Table 3.2-1 applies to all RCS components, including the pressurizer, and its components. The first row of the table identifies loss of material due to exposure to boric acid leakage as an applicable aging effect. Therefore the loss of material due to exposure to boric acid leakage applies to the pressurizer components at ANO-1. This response is acceptable to the staff.

3.3.2.2.2.2 Aging Management Programs

The staff's evaluation of the ANO-1 AMPs focused on the program elements rather than details of plant procedures. To determine whether the applicant's AMPs are adequate to manage the effects of aging so that the intended functions will be maintained consistent with the CLB for the period of extended operation, the staff evaluated the following The staff evaluation of the applicant's AMPs focused on the program elements rather than details of specific plant procedures. The staff's approach to evaluating each program and activity used to manage the applicable aging effects is described in Section 3.3.1 of this SER.

As mentioned in Section 3.3.2.3.1 of this SER, the following existing and new AMPs for the pressurizer are applicable to the period of extended operation:

- boric acid corrosion prevention program
- primary water chemistry monitoring program
- ASME Section XI Inservice Inspection Program, Subsection IWB (as supplemented by Code Case N-560 for Examination Category B-J welds)
- leakage detection in reactor building
- Alloy 600 AMP
- bolting and torquing activities
- small-bore piping and small-bore nozzles inspections
- pressurizer examinations (new program to check for stainless steel cladding failure)

All of these AMPs, with the exception of the Alloy 600 AMP, the small-bore piping and small-bore nozzle inspections program, and the pressurizer examinations program, have been reviewed in Section 3.3.2.4 of this SER for the reactor vessel. These AMPs are equally applicable to the pressurizer. The staff found that these programs will be adequate to detect aging effects in RCS components such that if unacceptable degradation is detected, the applicant will undertake further programmatic actions, including repair and replacement, as necessary, to manage the effects of aging.

Table 4-1 of BAW-2244A presents pressurizer component groupings and AMPs that manage the applicable aging effects. All of the AMPs applicable to RCS piping have been approved in the staff's final SER for BAW-2244A. Therefore, the staff found that these AMPs provide an acceptable demonstration that the effects of aging of pressurizer components, within the scope of the topical report, will be adequately managed. A comparison by the staff of Table 3.2-1 of the LRA and Table 4-1 of BAW-2244A did identify a discrepancy between Table 4-1 of BAW-2244A and Table 3.2-1 of the LRA.

In a letter to the NRC dated September 6, 2000, the applicant correctly identifies the boric acid corrosion prevention program for the external surfaces of the pressurizer to manage the aging effect (i.e., loss of material) due to corrosion. This is acceptable to the staff.

In the LRA, the applicant does not identify the vent and sampling nozzles, thermowell, level sensing nozzles and sampling nozzles as being inspected in accordance with ASME Section XI, Examination Category B-E for pressurizer retaining partial penetration welds. In its response to this concern dated September 6, 2000, the applicant states that the 1993 Addenda eliminated ASME Section XI, Examination Category B-E and the visual examinations previously performed under Examination Category B-E are now covered under Examination Category B-P in accordance with the 1993 Addenda. Therefore, the subject nozzles are currently examined by Examination Category B-P. This is acceptable to the staff.

The applicant also states that the small-bore nozzles are listed under "pressure retaining partial penetration welds, NPS less than or equal to 1-inch" and are subject to small-bore piping and small-bore nozzles inspection AMPs. The staff found this acceptable.

In the LRA, Table 3.2-1, the applicant identifies that Examination Category B-G-1 is applicable to bolting # 2-inch diameter. The staff questioned this categorization in its letter to the applicant dated May 5, 2000. In a letter to the NRC dated September 6, 2000, the applicant corrects the administrative error, and stated that the Examination Category B-G-2 is applicable to pressurizer bolting # 2-inch diameter. This is acceptable to the staff.

Review of the Alloy 600 AMP, the small-bore piping and small-bore nozzle inspections program, and the pressurizer examination program are provided in the following paragraphs:

Alloy 600 Aging Management Program

Section 4.1 in Appendix B of the LRA describes the ANO-1 Alloy 600 AMP. The program is designed to manage primary water stress corrosion cracking (PWSCC) of Alloy 600 and Alloy 82/182 locations for the period of extended operation.

[*Program Scope*] The program includes inspection of the components most susceptible to PWSCC. The scope does not cover Alloy 600 OTSG tubes, which are addressed separately in the steam generator integrity program.

AMPs for preventing or mitigating cracking in pressurizer Alloy 600 and Alloy 82/182 PWSCC are listed in Table 3.2-1 of the application as: (a) primary water chemistry monitoring, (b) ASME Section XI testing, and (c) leakage detection in reactor building. If cracking is detected, the applicant will undertake further programmatic actions, including repair or replacement, as necessary, to manage these effects.

[*Preventative/Mitigative Actions*] There are no preventive or mitigative actions with this program beyond standard repair and replacement plans should PWSCC occur, and the staff did not identify a need for such preventive or mitigative actions.

[*Parameters Inspected or Monitored*] The applicant has implemented an augmented inspection program for Alloy 600 nozzles attached to the pressurizer which includes supplementary examination according to Code Case N-560 for the Examination Category B-J welds. Visual inspection (VT-2) of each nozzle from the vessel exterior will be performed during each refueling outage. In addition, the repaired Alloy 600 level sensing nozzle in ANO-1, the ferritic steel in the nozzle-bore, is periodically examined using ultrasonic testing. There will also be VT-2 visual inspection of the pressurizer partial-penetration welded nozzles from the exterior of the vessel during refueling outages.

Visual inspection from the exterior of the pressurizer vessel is stated by the applicant to be justified on the basis that PWSCC of Alloy 600 components will cause longitudinal cracks that will leak but not compromise the structural integrity of the pressurizer.

The applicant also plans to volumetrically inspect the dissimilar metal welded joint in the pressurizer spray nozzle at the Alloy 600 safe-end to clad carbon steel. In addition, the welded joint that connects the spray nozzle safe-end to the stainless steel spray line is volumetrically inspected each interval in accordance with ASME Section XI.

The applicant plans to continue the monitoring program at Alloy 600 and Alloy 82/182 locations that are deemed to be the most susceptible to cracking during the period of extended operation. The procedure to determine these locations is as follows:

- Identify all Alloy 600 items and Alloy 82/182 weld metal at ANO-1.

- Select a PWSCC susceptibility model. The model will be similar to a model chosen for the CRDM nozzle PWSCC inspection and repair strategic evaluation that was applied to CRDM penetrations

- Select a reference Alloy 600 item for calculation of relative time to crack initiation. The actual item selected is the ANO-1 pressurizer instrumentation nozzle that leaked in 1990.

- Evaluate the differences in operating parameters between the reference Alloy 600 part and the remaining Alloy 600 and Alloy 82/182 welds. The specific material and operational parameters that were compared include maximum inside surface stress, operating temperature, microstructure, surface condition, and water chemistry.

- Calculate the relative susceptibility factor for the Alloy 600 and Alloy 82/182 welds relative to the time of crack initiation for the reference Alloy 600 item. The procedure for computing the susceptibility factor is given in BAW-2301, “Degradation of Control Rod Drive Mechanism and Other Vessel Closure Head Penetrations.”

From this study, the applicant determined the following three most susceptible groupings were:

- pressurizer sample nozzles
- pressurizer level tap nozzles
- thermowell nozzles
- pressurizer vent nozzle
- 4-inch NPS Alloy 600 safe-end that connects the stainless steel spray line to the stainless steel-clad carbon steel spray nozzle

The applicant identifies that the CRDM penetrations at ANO-1 were not among the top three groupings with respect to PWSCC. However, the applicant plans to continue an AMP for these penetrations in the CRDM nozzle and other vessel closure penetration inspection program, as described in Section 3.3.2.4.2.2 of this SER. The staff found that the actual cracking of an ANO-1 pressurizer level nozzle was stated in the LRA to be within a factor of two of the calculated time. The applicant notes the uncertainty, but states that, on a relative basis, the calculations define components that are most susceptible to cracking.

In a letter to the applicant dated May 5, 2000, the staff requested that the applicant provide detailed information on the applicability of an EPRI model for determining the relative susceptibility of pressurizer components to PWSCC. In a letter to the NRC dated September 6, 2000, the applicant states that the model for PWSCC in Alloy 600 and Alloy 82/182 was originally developed to determine susceptibility to cracking in the CRDM nozzles and other vessel penetration nozzles. In applying this model to the ANO-1 pressurizer, the applicant states that the new reference for the model was the cracked ANO-1 pressurizer nozzle, since this was most appropriate for this plant, rather than other cracked components from other B&W plants. Using the new reference model, the applicant states that other ANO-1 pressurizer components were ranked in terms of their susceptibility to cracking. The applicant defends the applicability of the original EPRI model for assessing the cracking of ANO-1 pressurizer components on the basis that the adapted model did not require any specific adjustments when applied to ANO-1. The applicant also identifies that NRC approval for the applicability of the EPRI model to pressurizer components is justified since the staff has previously approved the use of the EPRI model for PWSCC evaluations through the SER for Oconee, as described in Section 3.4.3.3 of NUREG-1723.

The staff found that the model provides useful information on cracking behavior of components but the LRA does not state whether the calculated failure time for the pressurizer level nozzle was greater or less than the actual cracking time. If the calculated value is smaller, it would indicate that the model is conservative. However, for ranking purposes the model is acceptable.

In a letter to the applicant dated May 5, 2000, the staff questioned if the cracking model predicts whether or not any of the Alloy 600 or Alloy 82/182 pressurizer components will have 75 percent through-wall cracks during the period of extended operation. In response, the applicant states that the cracking model does not calculate times for 75 percent through-wall cracking, but only the relative susceptibility for PWSCC in Alloy 600 and Alloy 82/182 components. The three most susceptible groupings of components were given in the applicant's response, and only two of these groupings undergo volumetric testing. In a letter to the NRC dated November 2, 2000, the applicant further clarifies that the nozzle region of the ferritic pressurizer shell metal adjacent to the repaired level-sensing nozzle, which is a Group 1 item, is volumetrically inspected periodically to ensure the integrity of the exposed ferritic steel. The Group 3 spray nozzle safe-end is inspected each interval in accordance with Examination Category B-F. Specifically, the dissimilar metal weld that connects the ferritic spray nozzle to the Alloy 600 safe-end receives volumetric and surface examination each inspection interval.

[*Detection of Aging Effects*] The applicant has implemented a comprehensive nondestructive evaluation program to detect cracking of Alloy 600 and Alloy 82/182 welds. These are complemented by a calculation model used to predict the most useful test locations. Some validation of the model was obtained when it confirmed that an actual failure in a pressurizer level nozzle was calculated to be among the most likely failure locations.

[*Monitoring and Trending*] The above-described program that is used to monitor crack susceptible locations provides reasonable assurance that cracking in the most susceptible locations can be detected. Furthermore, the growth of any detected crack may be trended using additional nondestructive testing techniques.

[*Acceptance Criteria*] Acceptance Criteria are given in ASME Code Section XI, 1992 Edition, including Mandatory Appendices VII and VIII.

[*Operating Experience*] The 1990 ANO-1 pressurizer Alloy 600 level sensing nozzle, mentioned above, developed a through-wall longitudinal crack near the heat-affected zone, on the inside diameter of the vessel. The failure was caused by PWSCC. The crack was identified by audible indications and dye-penetrant testing. Several other through-wall longitudinal cracks have also been found in Alloy 600 nozzles. The stresses that caused the cracks are stated to be the high hoop stresses caused by weld shrinkage during nozzle installation. The applicant states that the cracks are small, and are expected to remain small because the stress will decrease as the cracks grow.

The augmented inspections, described above, on the three most susceptible locations will be continued by the applicant over the period of extended operation. The staff found that the Alloy 600 AMP will satisfactorily manage the aging of affected Alloy 600 pressurizer components over the period of extended operation.

Small Bore Piping and Small Bore Nozzles Inspections

This program is described in Section 4.3.8 in Appendix B of the LRA, and identifies aging effects in small-bore piping and nozzles.

[*Program Scope*] The small-bore piping and nozzle inspections verify that service-induced weld cracking is not occurring in the small RCS piping. The small-bore piping and nozzles that fall within the scope of this program are defined as RCS piping and nozzles less than 4 inch NPS that do not receive volumetric inspection in accordance with ASME Code Section XI.

[*Preventative/Mitigative Actions*] There are no preventive/mitigative actions associated with this program, nor did the staff identify a need for such.

[*Parameters Inspected or Monitored*] The applicant has elected to implement a risk-informed method to select RCS piping welds for inspection in lieu of the requirements specified in the 1992 Edition of ASME Section XI, Table IWB-2500-1, Examination Category B-J. This approach is based on Code Case N-560, and consists of: selection of a degradation mechanism evaluation to assess failure potential for the piping under consideration, and use of a consequence evaluation to assess impact on plant safety in the event of piping failure. The results of the two programs are coupled to determine the risk significance of piping sections in the system. The most risk-significant sections of piping are selected for weld inspection. Class 1 small-bore piping sections selected for volumetric examinations of their welds include the 1½-inch pressurizer spray line, 2½-inch makeup and purification lines, 2½-inch letdown line, and 1½-inch cold leg suction drain line.

[*Detection of Aging Effects*] Cracking of the welds will be detected by volumetric examination.

[*Monitoring and Trending*] Any flaw growth in welds will be trended by ongoing inspections. The frequency of these inspections is specified in Code Case N-560. If the flaws reach an unacceptable size, the applicant will undertake further programmatic actions, such as repair and replacement, as necessary, to manage the effects of aging.

[*Acceptance Criteria*] The acceptance criteria are provided in ASME Section XI IWB-3400 and IWB-3132, as provided in Code Case N-560.

[*Operating Experience*] The small-bore piping and small-bore nozzles inspection program was used at ANO-1 to investigate cracking in RCS drain lines following the discovery of a crack in a drain line weld in 1989. The root cause of the crack was stated to be vibrational fatigue crack propagation that was initiated in a weld defect. Vibrational fatigue failures in socket welds have also been detected at ANO-1 on small-bore vents and drains under 2 inches NPS. In locations of high stress, several socket welds at ANO-1 were reinforced. The applicant cites its risk-informed methodology for managing cracking of small-bore piping and nozzles. The most susceptible welds are calculated by considering regions of maximum stress and lower bound material properties, and only these welds are inspected according to NRC-approved versions of ASME Code Section XI, as supplemented by Code Case N-560.

The staff found that continued implementation of this program provides reasonable assurance that the aging effects will be managed so that the applicable components will continue to perform their intended functions consistent with the CLB for the period of extended operation.

Pressurizer Examinations

Section 3.2 of the LRA and BAW-2244A identify the aging effects that will require new or additional inspections for license renewal. These aging effects include cracking of pressurizer cladding, including items attached to the cladding (e.g., tripod legs), which may result in cracking or loss of underlying ferritic steel; cracking of the structural welds that connect the heater sheaths to the diaphragm plates; and cracking of small-bore safe-ends.

Management of aging effects for pressurizer Alloy 600 small-bore nozzles are given in the Alloy 600 AMP reviewed in this section of the SER. Small bore safe-end aging is managed through the small-bore piping and small-bore nozzles inspection program, and is also reviewed in this

Section of the SER. The pressurizer cladding examination and the pressurizer heater penetration weld examination are described below.

Pressurizer Cladding Examination

[*Program Scope*] The pressurizer cladding examination assesses the cracking of cladding by thermal fatigue, which may propagate to the underlying ferritic steel of the pressurizer cladding and attachment welds to the cladding of the pressurizer.

[*Prevention/Mitigating Actions*] The applicant does not specify any preventative or mitigating actions. However, the staff found that the primary water chemistry monitoring program is an effective preventative action to avoid cracking of the stainless steel cladding.

[*Parameters Inspected or Monitored*] In order to provide assurance that cracking of the cladding has not penetrated into the underlying base metal of the pressurizer, the applicant intends to perform volumetric examination of pressurizer items that are most susceptible to thermal fatigue. The applicant identifies the items with the highest fatigue cumulative usage factors and this includes the circumferential weld that connects the shell to the lower head, and the full-penetration weld that connects the pressurizer surge nozzle to the lower head.

In accordance with ASME Section XI, Examination Category B-B, volumetric examination of the circumferential shell-to-head weld is performed each inspection interval. In addition, 1-foot of longitudinal weld adjacent to the heater belt forging is volumetrically examined. The weld between the surge nozzle and the lower head is volumetrically examined each inspection interval in accordance with Examination Category B-D. The applicant intends to continue these examinations through the period of extended operation to manage any cracking of cladding that may extend into the base metal at the locations most susceptible to thermal fatigue. The applicant states that the first inspections will be completed before the current licensing period ends.

[*Detection of Aging Effects*] The detection of cracking in the two pressurizer items as specified above will be achieved through periodic volumetric inspection procedures from ASME Section XI as specified by the applicant.

[*Monitoring and Trending*] Although the applicant does not specify monitoring and trending activities, the staff expects that any cracking detected in the volumetric inspection program will be trended, using data from the periodic inspections, in order to determine the extent and depth of cracking in the cladding.

[*Acceptance Criteria*] The acceptance criteria, as specified by the applicant, are those for volumetric examinations in accordance with ASME Section XI, IWB-3510 and IWB-3512.

[*Operating Experience*] The applicant cites Haddam Neck as the only plant in which pressurizer cladding cracking was detected. This plant, however, was not built by B&W. The applicant states that because of differences in design, fabrication, and operation, cracking of cladding in ANO-1 is not expected. Nevertheless, the comprehensive cladding inspection program for ANO-1 has been formulated. This commitment to a pressurizer cladding inspection program closes an open item that was identified in the SER to BAW-2244A.

In conclusion, the staff found the ANO-1 volumetric inspection program to detect cracking in pressurizer cladding to be acceptable. The staff also finds the ANO-1 inspection program to be more comprehensive than the minimum inspection program it had recommended during the review of the topical report, which required only a one time inspection of the cladding for cracks.

Pressurizer Heater Bundle Penetration Welds Examination

[*Program Scope*] The pressurizer heater bundle penetration welds examination assesses the cracking at the heater bundle penetration welds, which may lead to reactor coolant leakage. The program covers the heater-to-diaphragm plate penetration welds inside the pressurizer. The pressurizer contains three heater bundles.

[Preventative/Mitigative Actions] The applicant does not specify any preventative or mitigative actions. However, the staff found that the primary water chemistry monitoring program will be an effective preventative action to minimize cracking in the heater bundle welds.

[*Parameters Inspected or Monitored*] For the first heater bundle to be replaced, the applicant intends to perform a surface examination of sixteen peripheral welds. A visual examination (VT-3 or equivalent) of the remaining welds of the heater bundle will also be performed. In addition, the applicant will inspect exterior portions of the heater bundle each outage in accordance with Section XI Examination Category B-P. In accordance with IWA-5242, as modified by Code Case N-533 for bolted connections, the applicant will remove the insulation surrounding the penetrations and perform a VT-1 visual inspection. This addresses Open Item 2 in the SER for BAW-2244A.

The heater bundle penetration weld examination is a one-time inspection to be performed at ANO-1 or Oconee Nuclear Station. If the results of the inspection are not acceptable, then the applicant states that the results may be used as baseline information for inspections of the remaining ANO-1 heater bundles.

[*Detection of Aging Effects*] The effects of aging will be detected by the implementation of the above-mentioned ISI programs.

[*Monitoring and Trending*] Although the applicant does not specify monitoring and trending activities, any cracking detected in the inspection program will be trended, using data from the periodic inspections, in order to determine the extent and depth of cracking in the heater bundle with time.

[*Acceptance Criteria*] The acceptance criteria for the surface examinations and visual examination (VT-3) will be those specified by ASME Section XI. For the heater bundle, the exterior portions will be examined under Examination Category B-P. In accordance with IWA-5242, as modified by Code Case N-533 for bolted connections, the applicant will remove the insulation surrounding the penetrations and perform a VT-1 visual inspection.

[*Operating Experience*] The applicant states that no stainless steel heater sheath-to-diaphragm plate penetration weld cracking has occurred to date in B&W plants. Cracking has been observed in non-B&W plants on similar heater penetrations but these failures occurred on more susceptible Alloy 600 penetrations through hemispherical heads. Therefore, the applicant does not expect to observe cracking in pressurizer heaters during the period of extended operation. If a heater bundle is removed for replacement, the applicant will carry out surface examination of the 16 peripheral heater penetrations and VT-3 or equivalent examinations of the remaining welds of the heater bundle to determine if cracking of these welds is an applicable aging effect.

In conclusion, the staff found that the pressurizer heater bundle penetration weld examination program is sufficiently comprehensive to detect aging and cracking in the welds. This provides reasonable assurance that the aging effects associated with stainless steel heater penetration welds will be managed such that the applicable components will continue to perform their intended functions during the period of extended operation.

3.3.2.2.3 Conclusions

The staff has reviewed the information included in Section 3.2.3, “Pressurizer,” and Appendices A and B of the LRA and additional information provided by the applicant in response to the staff RAIs. On the basis of this review, the staff concludes that the applicant has demonstrated that the effects of aging associated with the pressurizer can be adequately managed such that there is reasonable assurance that this system will perform its intended function(s) consistent with the CLB during the period of extended operation.

3.3.2.3 Reactor Vessel

The reactor vessel (RV) consists of the cylindrical vessel shell, lower vessel head, closure head, nozzles, interior attachments, and all associated pressure retaining bolting. Coolant enters the inlet nozzles, passes down the annulus between the thermal shield and vessel inside wall, reverses at the lower head, passes up through the core, turns around through the plenum assembly, and leaves the reactor vessel through the outlet nozzles.

3.3.2.3.1 Technical Information in the Application

In the LRA, Section 3.2.1.5, the applicant states that the components of the RV subject to an AMR include the shell and closure head, the nozzles, interior attachments, and bolted closures, and that the reactor vessel is bounded by BAW-2251A, “Demonstration of the Management of Aging Effects for the Reactor Vessel.”

In BAW-2251A, the owners group concludes that components not subject to an AMR include the closure O-ring and fasteners, monitoring pipes, lifting lugs, flow stabilizers, bolted reactor vessel attachments, and the seal ledge.

Intended Functions

As described in BAW-2251A, the intended functions of the reactor vessel include:

- maintaining the integrity of the reactor pressure boundary

- providing structural support for the vessel internals and core

Effects of Aging

The operating environment of the ANO-1 reactor vessel is consistent with that described in Section 3 of BAW-2251A. The ANO-1's primary water chemistry monitoring program includes specifications to periodically monitor the primary coolant. Limitations are established on dissolved oxygen, halides, and other impurities. Corrective actions are taken in the event that the primary coolant parameters are out of specification.

In accordance with the requirements for the Class 1 components of ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, the reactor vessel is designed to accommodate service loadings (i.e.,A through D). Operation under level A and B service conditions contributes to the normal aging stresses for the reactor vessel. Since ANO-1 has not been subjected to level C and D events, BAW-2251A is considered by the applicant to bound the ANO-1 vessel with respect to the qualitative assessment of stress.

In the LRA, Table 3.2-1, the applicant identifies the following RV aging effects:

- Aging of the reactor vessel and closure head, which are made from stainless steel clad low-alloy steel, may cause cracking at welded joints, reduction of fracture toughness, and cracking of SA508, Class 2, forgings due to intergranular separation.

- Aging of reactor vessel nozzles, made from stainless steel clad low-alloy steel, containing full-penetration welds, may cause cracking at the welds, and cracking at the nozzle inside radius.

- Aging of reactor vessel nozzles, made from Alloy 600. These nozzles contain partial-penetration welds, and include control rod drive mechanism (CRDM) and incore instrumentation nozzles. Aging may cause cracking at or near the welds.

ANO-1 is participating in the ongoing Materials Reliability Project (MRP) for monitoring stress corrosion induced cracking of Inconel type materials such as CRDM nozzles that contain partial penetration welds. This ongoing effort is used to assess the need for inspecting these materials on the basis of modeling results and industry experience.

Survey of Industry Operating Experience

To validate its choice of applicable aging effects for ANO-1, the applicant conducted a search for instances of aging at ANO-1 using the station information management system, the condition reporting system, and the licensing event data base. From this review, the applicant concludes that all aging effects were bounded by those given in BAW-2251A.

Aging Management Programs

The AMPs that will be continued through the period of extended operations include:

- primary water chemistry monitoring program
- CRDM nozzle and other vessel closure penetrations inspection program
- ASME Section XI Inservice Inspection Program, Subsection IWB
- leakage detection in reactor building

- reactor vessel integrity program
- boric acid corrosion prevention program
- Alloy 600 AMP
- bolting and torquing activities

The applicant concludes that these programs would manage aging effects in such a way that the intended function(s) of the components of the RV would be maintained consistent with the CLB, under all design loading conditions during the period of extended operation.

3.3.2.3.2 Staff Evaluation

In accordance with 10 CFR 54.21(a)(3) the staff reviewed the information included in the LRA Sections 2.3.1.5 (Table 2.3-4), 3.2.4 (Table 3.2-1), Appendix B Sections 4.1, 4.3, 4.4, 4.5, 4.6, 4.7, 4.12, 4.18, and the final SER for BAW-2251A, regarding the applicant's demonstration that the effects of aging will be adequately managed so that the intended functions will be adequately managed consistent with the CLB for the period of extended operation for the reactor vessel.

There were several action items in the final SER for BAW-2251A. The action items, the licensee's responses, and the staff's evaluations are given below.

Action Items from Previous Staff Evaluation of BAW-2251A

As discussed below, the staff found that most of the licensee's responses to the Renewal Application Action Items from BAW-2251A resolve the action items. Exceptions are described below.

- Item 1: The license renewal applicant is to verify that its plant is bounded by the topical report. Further, the renewal applicant is to commit to programs described as necessary in the topical report to mange the effects of aging during the period of extended operation on the functionality of the reactor vessel components. Applicants for license renewal will be responsible for describing any such commitments and identifying how such commitments will be controlled. Any deviations from the AMP within this topical report described as necessary to manage the effects of aging during the period of extended operation and to maintain the functionality of the reactor vessel components or other information presented in the report, such as materials of construction, will have to be identified by the renewal applicant and evaluated on a plant-specific basis in accordance with 10 CFR 54.21(a)(3) and (c)(1).

  Response: The licensee participated in the development of BAW-2251A by providing ANO-1 design and operating information. The applicant has reviewed the current design and operation of the ANO-1 pressure vessel and confirms that the RV is bounded by the description contained in BAW-2251A.

  The staff found this response to be acceptable, with the exception of the leakage monitoring pipes. This is discussed more fully in Item 3, below.

- Item 2: A summary description of the programs and evaluation of TLAAs is to be provided in their license renewal FSAR supplement in accordance with 10 CFR 54.21(d).

  Response: Changes in the reactor vessel programs and TLAA evaluations have been included in the FSAR supplement in Appendix A of the LRA. The staff evaluated the summary description of the FSAR supplement, and found the applicant's response to be acceptable.

- Item 3: Since the staff has not made any finding on whether the B&WOG topical report provides the complete list of reactor vessel components subject to an AMR or whether the scoping methodology is adequate, individual plant applicants will need to provide a comprehensive list of SCs that are subject to an AMR and the methodology for developing this list as part of their LRAs. Any components determined by the applicant to be subject to an AMR for license renewal, but not within the scope of the topical report, are required to be addressed in the LRA.

  Response: The applicant has reviewed the current design and operation of ANO-1 reactor vessel to confirm that the reactor vessel is bounded by the description contained in BAW-2251A. No additional RV items were identified as subject to an AMR.

  The staff considers that the monitoring pipes from the closure head should have an AMR (see letter from C.I. Grimes to D.J. Firth, dated October 27, 1999). This letter describes the staff's reassessment of the conclusion that the innermost O-ring seal is the first pressure boundary. This earlier conclusion is inconsistent with the staff's position contained in license renewal issue number 98-0012, "Consumables" that was issued to Nuclear Energy Institute (NEI) on April 20, 1999. In this position the staff stated that packing, gaskets, seals, and O-rings are not typically required by the CLB to fulfill the functions of 10 CFR 54.4(a)(1)(I). The position also stated that according to ASME, Section III, NB-2121, and ND-2121, packing, gaskets, seals, and O-rings are not relied upon for a pressure retaining function in components for which these Code design practices apply. Inasmuch as these Code design practices do not apply to the O-ring in the closure head, and since the sealing surface of the vessel flange is the pressure boundary, the staff determines that the O-rings are not within the scope of 10 CFR Part 54. However, because the leakage monitoring pipes penetrate the sealing surfaces of the vessel flanges, they should be treated as part of the reactor coolant system pressure boundary and, therefore, are within the scope of 10 CFR Part 54 (also see Item 14).

  In a letter to the NRC dated August 24, 2000, the applicant states that the monitoring pipes do not directly support the RCS pressure boundary and, therefore, are not within the scope of license renewal. The applicant also states that if the inner reactor vessel closure flange O-ring fails, and RCS fluid is introduced into the monitoring pipes, leak flow would be limited because the ½-inch diameter orifice holes in the flange are smaller than the inside diameters of these pipes. In addition, any leakage through the monitoring pipes, during normal operation is estimated by the applicant to be within the makeup system capacity of the RCS. The staff found this response to be acceptable.

- Item 4: The B&WOG has determined that the lower CRDM service support structure, including the weld that connects the lower CRDM service support skirt to the reactor vessel closure head, and the reactor vessel support skirt to the transition forging, are subject to an AMR for license renewal. However, the B&WOG has decided to exclude them from the scope of the topical report. Thus, a renewal applicant needs to address them in the LRA.

  Response: The reactor vessel support skirt (including the attachment weld that connects the reactor vessel support skirt to the transition forging) and control rod drive service structure, including the attachment weld to the reactor vessel, are described in the application, along with the applicable aging effects and AMPs that manage the applicable aging effects.

  The staff confirmed that aging of these components is discussed in Section 3.6.1.4 of the LRA and, therefore, finds this response to be acceptable.

- Item 5: The LRA for ANO-1 needs to evaluate the thermal fatigue evaluation of the reactor vessel studs on a plant-specific basis.

  Response: The applicant states that a fatigue evaluation of the vessel studs is not applicable to ANO-1.

  The staff found this response to be acceptable because of the much lower usage factor for the studs at ANO-1, compared to those at Oconee, for which there was a much more detailed and conservative evaluation of stud fatigue.

- Item 6: A license renewal applicant needs to address the plant-specific methodology and instrumentation used to assess the number of operational transients in its renewal application for staff review. The staff review will also include the number of operating cycles applicable to the reactor vessel studs.

  Response: The ANO-1 program that monitors operational transients is included in the LRA and is evaluated in Section 4.3.5 of this SER.

- Item 7: The B&WOG identifies flaw growth acceptance in accordance with the ASME Section XI ISI program as a TLAA, but indicates that the flaw growth acceptance evaluation is plant-specific, is not within the scope of the report, and will be resolved on a plant specific basis. Thus, a license renewal applicant needs to address it in the renewal application.

  Response: The ANO-1 program to manage analytic evaluation of flaws is addressed in the application.

  The staff found this response to be acceptable, and a review of the flaw growth analyses is given in Section 4.8.1.3 of this SER.

- Item 8: Alloy 600 components in the reactor vessel such as the CRDM housing and other penetrations may be subject to crack initiation and growth. The B&WOG originally proposed to use the ASME Section XI program, supplemented by leak detection and surveillance of boric acid, to manage cracking of Alloy 600 components. In an April 1997, response to the staff's RAI concerning Generic Letter 97-01, "Stress Corrosion Cracking of Control Rod Drive Mechanisms and Other Vessel Head Penetrations," the B&WOG stated: "Each participating plant will address additional requirements for RV head penetrations, including closure head penetrations less than 2-inch NPS (i.e., thermocouple nozzles at TMI-1 and ONS-2)." Thus, a license renewal applicant referencing the topical report will need to submit its plant-specific program to manage cracking of Alloy 600 components in the reactor vessel in its renewal application for staff review.

Response: In a letter to the NRC dated August 24, 2000, the applicant states that it will continue the augmented visual inspections for the three most susceptible Alloy 600 and Alloy 82/182 groupings during the period of extended operation. Any additional inspection locations will be based on a "qualitative assessment of risk" as prescribed by Code Case N-560. In this Code Case, the probability of piping failure is estimated by developing a risk matrix that ranks possible failures in terms of the degradation mechanism and the severity of the consequence. The staff found this acceptable.

C Item 9: During the review of the topical report, the staff had a question regarding the need to update the reactor vessel fracture toughness estimates with new data as it became available. In its August 11, 1997, RAI response, the B&WOG states:, "Each license renewal applicant will define a process to ensure that the TLAA evaluations reported in BAW-2251 are tracked such that the TLAA remains valid through the period of extended operation. The process will be defined on a plant-specific basis at the time of the LRA." Thus, a LRA needs to describe such a process in its application for staff review. If new information affects the conclusions of the topical report for the applicant's plant, the applicant needs to update its TLAA evaluation as appropriate and provide the updated evaluations in its renewal application for staff review.

Response: The ANO-1 reactor vessel integrity program is described in Appendix A of the LRA.

Section 16.2.18 of Appendix A of the LRA specifies the master integrated reactor vessel surveillance program as being the means for addressing the effects of radiation on vessel properties. Per Section 4.18.1 of Appendix B of the LRA, the applicant intends to continue monitoring the data obtained from the fracture toughness specimens in the B&WOG surveillance program. Should the results project that the fracture toughness of the ANO-1 vessel will fall outside the screening guidelines, the applicant will consider a flux reduction program to ensure that the vessel remains within the guidelines during the period of extended operation. If this cannot be achieved, the applicant will submit a safety analysis to determine what actions will be necessary to prevent failure of the reactor vessel if continued operation beyond the screening criteria is allowed. The staff found this response to be acceptable.

C Item 10: In its August 11, 1997, RAI response, the B&WOG indicated that Oconee Unit 2 and TMI Unit 1 will provide updated predictions of RT_{PTS} for welds WF-25 and SA-1526, respectively, when the plant-specific application for license renewal is submitted.

For plants with an RT_{PTS} value for 48 EFPY exceeding the corresponding PTS screening criteria, a license renewal applicant must address the 10 CFR 50.61(b)(3) requirements by developing, and requesting staff approval for reasonable practicable flux reduction programs to avoid exceeding the PTS criterion.

Response: The applicant states that this action item is not applicable to ANO-1.

The staff agrees that the development of a flux reduction program is not applicable to ANO-1 since the applicant has recalculated RT_{PTS} for 48 EFPY and has shown that it remains below the PTS screening criterion for the period of extended operation (see Section 4.2.1.3 of this SER).

- Item 11: If an applicant has installed flow stabilizers using Alloy 600 and/or 82/182 weld material, the applicant must include the flow stabilizers in its Alloy 600 AMP. Alloy 600 and Alloy 82/182 weld materials are susceptible to cracking in primary water environments.

  Response: The applicant states that the ANO-1 flow stabilizers are made from austenitic stainless steel and attached to the cladding using stainless steel weldments.

  The staff found this response to be acceptable since austenitic stainless steels have not been identified as being prone to cracking under the current reactor vessel service conditions.

- Item 12: Embrittlement of the reactor vessel will be managed to ensure intended functions for the reactor vessel for 60 years. For the staff to determine if the plant could be operated for 60 years, an applicant must show that an operating window will be available between the pressure-temperature limits and the net positive suction curves for the RCPs for 60 years. Otherwise, the applicant will propose aging management activities to minimize the extent of embrittlement, or other alternatives, to permit safe plant operation for 60 years. Should the applicant show that the reactor could only be operated for a period less than 60 years, the duration of the renewed license, if granted, would be limited to that time period.

  Response: The applicant states that pressure-temperature limits were developed in accordance with the requirements of ASME Section XI, Appendix G, as modified by Code Case N-588 for circumferential flaws in welds and by Code Case N-640 for the use of the K_{IC}, fracture toughness curve. It is further stated that the operating window at 48 EFPY exceeds the current pressure-temperature operating window, which has been approved by the NRC for 32 EFPY. The increased operating window is said to be attributable to the use of the two Code Cases.

  The staff found this response to be acceptable. A staff review of the pressure-temperature limit curves is given in Section 3.3.2.4.2.2 of this SER.

- Item 13: The neutron fluence must be experimentally monitored by ex-vessel or in-vessel dosimetry, and if modifications to the design and operation of the plant change either the neutron energy spectrum, gamma heating or the reactor inlet temperature, the licensee must notify the NRC and propose a program to determine the impact of the modifications.

Response: The applicant states that reactor vessel fluence monitoring is addressed in the ANO-1 Reactor Vessel Integrity Program that is described in Appendix B of the application. The staff found this response to be acceptable. A review of the vessel fluence monitoring program is given in Section 3.3.2.4.2.2 of this SER.

- Item 14: During review of the ONS-1 LRA the staff raised a question regarding the aging management of the reactor vessel monitoring pipes. The issue was not addressed in the topical report. Since this is a requirement in 10 CFR 54.21(a)(3), the applicant must provide details of its AMP for the reactor vessel monitoring pipes.

  Response: The applicant provided appropriate responses to the staff's RAI, which is discussed in the applicant's response to Action Item 3 of this SER.

3.3.2.3.2.1 Effects of Aging

The applicant identifies the following reactor vessel aging effects:

- cracking at welded joints, reduction of fracture toughness due to neutron irradiation embrittlement, and cracking of SA508, Class 2, forgings due to intergranular separation for the reactor vessel which is made of stainless steel clad low-alloy steel

- cracking at the full-penetration welds, and cracking at the nozzle inside radius for the reactor vessel nozzles which are also made from stainless steel clad low-alloy steel

- cracking of partial-penetration welds in the CRDM and incore instrumentation nozzles made from Alloy 600

On the basis of the published literature and industry experiences, the applicant has identified the applicable aging effects for the reactor vessel.

3.3.2.3.2.2 Aging Management Programs

The staff evaluation of the applicant's AMPs focused on the program elements rather than details of specific plant procedures. The staff's approach to evaluating each program and activity used to manage the applicable aging effects is described in Section 3.3.1 of this SER.

As mentioned above in Section 3.3.2.4.1 of this SER, the following existing AMPs will be continued during the period of extended operations:

- primary water chemistry monitoring program
- CRDM nozzle and other vessel closure penetrations inspection program
- ASME Section XI Inservice Inspection Program, Subsection IWB
- leakage detection in reactor building
- reactor vessel integrity program
- boric acid corrosion prevention program

- Alloy 600 AMP
- bolting and torque activities

The primary water chemistry monitoring program, ASME Section XI Inservice Inspection Program, Subsection IWB, leakage detection in reactor building, and boric acid corrosion prevention programs are discussed in Section 3.3.1 (Common Aging Management Programs) of this SER. The Alloy 600 AMP at ANO-1 does not include RV components in its risk-based selection . The remaining AMPs, including the evaluation of the Alloy 600 aging management specifically for RV components, are evaluated below:

The staff's evaluation of the primary water chemistry monitoring program, ASME Section XI Inservice Inspection Program, Subsection IWB, leakage detection in reactor building, and boric acid corrosion prevention program are provided in Section 3.3.1 of the staff's evaluation, "Common Aging Management Programs."

CRDM Nozzle and Other Vessel Closure Penetrations Inspection Program

On April 1, 1997, the NRC issued Generic Letter (GL) 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations," to holders of operating licenses for pressurized water reactors. In the GL, the staff requested that addressees provide a description of their programs for inspecting the control rod drive mechanism (CRDM) penetration nozzles and other vessel head penetration (VHP) nozzles at their facilities. The staff encouraged the PWR-industry to coordinate its responses to GL 97-01 with the B&WOG, the Combustion Engineering Owners Group (CEOG), and the Westinghouse Owners Group (WOG), as well as the Nuclear Energy Institute (NEI) and the EPRI in order to provide an industry-wide integrated response to GL 97-01. The coordinated response to GL 97-01 for B&W designed reactors was provided by the B&WOG in Topical Report BAW-2301, "B&WOG Integrated Response to Generic Letter 97-01, Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations" (July 1997). In this report, the B&WOG provided a description of its engineering model for ranking the susceptibility of VHP nozzles in B&W designed reactors to develop and grow primary water stress corrosion cracking (PWSCC) type cracks similar to those detected in foreign reactor, Bugey Unit 3, and D.C. Cook Unit 2. In the B&WOG integrated program, only the VHP nozzles at the most susceptible B&WOG member plants are scheduled for voluntary volumetric inspections. Of these, the VHP nozzles at Oconee Unit 2 and Crystal River Unit 3 have been identified as the most susceptible nozzles of the B&WOG-member facilities. Duke Power Company has completed inspections of the VHP nozzles at Oconee Unit 2. Although ANO-1 is a member facility in the B&WOG, the applicant did not volunteer to conduct volumetric inspections of the plant's CRDM nozzles during the current license period because the nozzles are not among those identified as being the most susceptible in the member facilities.

Appendix B, Section 4.7 of the LRA describes the applicant's basis for monitoring and evaluating the unit's CRDM penetration nozzles during the period of extended operation, consistent with Topical Report BAW-2301. The regulatory basis for the program is provided in the applicant's responses to GL 97-01, dated July 29, 1997 and February 24, 1999. The design of the reactor vessel head at ANO-1 does not include thermocouple nozzles (the design of the vessel heads at Oconee Unit 1 and Three Mile Island Unit 1 includes thermocouple nozzles). Therefore, aging of thermocouple nozzles is not an issue for the ANO-1 facility.

[*Program Scope*] The program includes ISI, based on the guidelines of GL 97-01, to detect cracks or coolant leakage.

[*Preventative/Mitigative Actions*] Programs to prevent or mitigate degradation or failure of the CRDM, and other closure penetrations are listed in Table 3.2-1 of the LRA as: (a) Primary Chemistry Monitoring, (b) ASME Section XI ISI-IWB Examinations, Categories B-E and B-P, (c) Leakage Detection in Reactor Building, (d) CRDM Nozzle and Other Closure Penetration Inspection (CRDM penetration only), and (e) Alloy 600 Aging Management (incore nozzles).

[*Parameters Inspected or Monitored*] Parameters are specific to the AMPs cited above.

[*Detection of Aging Effects*] The aging effect for the CRDM nozzles at ANO-1 is primary water stress corrosion cracking of Alloy 600 nozzles that are welded to the primary pressure boundary vessel using partial penetration welds. Although circumferential cracks may be possible, past experience indicates that they are very unlikely. Operating experience has shown that axial cracking will occur before circumferential cracks, and no circumferential cracks have been identified to date, although, one crack did have a component at approximately 95 degrees. The applicant intends to monitor the inspection programs at other B&W plants in order to determine whether similar inspections will be initiated at ANO-1. The applicant's programs that will be implemented as part of the CRDM nozzle and other vessel closure penetration programs are listed above, and the specific aspects of each will be utilized to detect any cracking.

[*Monitoring and Trending*] As a member of the B&WOG, the applicant is following the monitoring and trending programs implemented at Crystal River Unit-3 and Oconee under the auspices of B&WOG and NEI. The staff approved the NEI/B&WOG program for assessing the potential for PWSCC to occur in VHP of B&W-designed plants. The current NEI/B&WOG program requires re-inspection for two of twelve Oconee Unit 2 CRDM nozzles from the top of the head, and an inspection of all CRDM penetrations at Crystal River Unit-3.

According to the applicant, re-examination in 1996 showed no change in the indications. Other plants in Europe were found to have similar cracking phenomena, as described in Generic Letter 97-01. The Crystal River inspection is planned for 2001.

[*Acceptance Criteria*] The applicant states that, from an analysis used to determine the most susceptible nozzles for inspection (described in BAW-2301), the ANO-1 CRDM nozzles were the least susceptible to cracking, with a relative predicted time to failure well in excess of 48 EFPY. Because of this, the applicant currently does not plan to implement an applicant-specific inspection program, but will continue to monitor the results of the NEI/B&WOG effort to determine whether a future program will become necessary. If such a program is implemented in the future, the applicant will analyze and evaluate axial flaws using NUMARC acceptance criteria, which were approved by the NRC on November 9, 1993. Circumferential flaws, should they occur, will be analyzed and addressed with the NRC on a case-by-case basis. The staff found this to be a formal commitment by the applicant that needs to be added to the FSAR Supplement to help meet the goals of the above-described inspection programs.

[*Operating Experience*] A full inspection of Oconee Unit 2 from beneath the reactor vessel head was performed in 1994. In addition, a re-inspection was completed on two Oconee Unit 2 CRDM nozzles in 1996. Results were submitted to the NRC. A subsequent re-inspection at

Oconee Unit 2 was completed in 1999. The 1994 inspection revealed a small number of nozzles with crack-like indications that were insignificant in depth. Re-inspection in 1999 showed no change from earlier inspections.

On the basis of the program elements described in the LRA, the staff found the CRDM nozzle and other vessel head penetrations inspection program to be acceptable for the period of extended operation.

Reactor Vessel Integrity Program

For the reactor vessel, Section 3.2.4 and Table 3.2-1 of the LRA identify reduction in fracture toughness resulting from neutron irradiation as an aging effect requiring management for the period of extended operation. The ANO-1 reactor vessel integrity program is designed to manage this aging effect.

The reactor vessel integrity program consists of the following five interrelated programs:

- master integrated reactor vessel surveillance program
- cavity dosimetry program
- fluence and uncertainty calculations
- pressure/temperature limit curves, and
- monitoring effective full power years

Through the reactor vessel integrity program, the applicant intends to comply with the requirements of 10 CFR 50.60, Appendices G and H, and 10 CFR 50.61. The five interrelated programs are reviewed separately in the following paragraphs.

Master Integrated Reactor Vessel Surveillance Program (MIRVSP)

Criteria for the first 40 years are specified in 10 CFR Part 50, Appendix H, “Reactor Vessel Materials Surveillance Program,” for monitoring changes in the fracture toughness of ferritic materials in the reactor beltline region to neutron irradiation, and thermal environments. Appendix H requires that the surveillance program design, and withdrawal schedule meet the requirements of American Society for Testing and Materials (ASTM) E-185, “Standard Practice for Conducting Surveillance Tests for Light Water Cooled Nuclear Power Vessels.”

Regulatory Guide (RG) 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials,” describes general procedures acceptable to the NRC staff for calculating the effects of neutron irradiation embrittlement of the low-alloy steels used for light-water-cooled reactor vessels. Surveillance data from the Appendix H program are used in RG 1.99, Revision 2 calculations, if applicable.

The staff found that the MIRVSP met the requirements of Appendix H with respect to the design, and withdrawal schedule for an integrated surveillance program (Appendix H, paragraph lll.C). Specifically, in a letter dated July 11, 1997, the staff approved the withdrawal schedules and accepted the technical basis for an integrated surveillance program with regards to design and operating conditions by approving BAW-1543, Revision 4, including Supplements 1 and 2, “Master Integrated Reactor Vessel Surveillance Program,” to demonstrate continuous management of aging effects for all plants included in BAW-1543, Revision 4, Supplement 2, which includes ANO-1. However, staff approval in the 1997 letter was for a 40-year license term.

[*Program Scope*] The scope of the MIRVSP program, approved by the staff, consists of three elements:

- plant-specific capsules
- supplementary weld metal surveillance capsules (SUPCAPS)
- high-fluence supplementary weld metal surveillance capsules (HUPCAPS)

Each licensee participating in the integrated surveillance program has provided at least six plant-specific capsules to the program. There are six “SUPCAPS” with the target fluence varying from 6.1×10^{18} n/cm^2 to 1.6×10^{19} n/cm^2. There are eight HUPCAPS with the target fluence varying between 1.3×10^{19} n/cm^2 and 2.4×10^{19} n/cm^2. The applicant states that “[t]he specimens are irradiated in capsules located near the reactor vessel inside wall, thus enabling the reactor vessel materials to become irradiated to beyond anticipated license renewal levels” (p. B-85 LRA).

[*Preventative/Mitigative Actions*] There are no preventive and mitigative actions associated with the MIRVSP, nor did the staff identify a need for such actions.

[*Parameters to be Monitored*] The parameter to be monitored is the increase in 30 ft-lb temperature from unirradiated and irradiated specimens at Davis-Besse and Crystal River-3. (In addition, Charpy USE is monitored, refer to Section 4.2.2 of the LRA.) The tests are in accordance with the applicable ASTM standards identified in Section 5.0 of BAW-1543A, Revision 2. Since the specimens are irradiated in locations near the reactor vessel inside wall, they should become irradiated to levels beyond the anticipated license renewal fluence levels. The applicant states, however, that the MIRVSP schedule may be altered due to unscheduled down times at the host plants. In addition, some capsules may receive additional irradiation to fully satisfy license renewal fluence requirements.

The staff found that since the applicant is relying on the MIRVSP to provide materials irradiation data on reactor vessel embrittlement to cover the period of extended operation, it should justify this approach by comparing the maximum fluence from the MIRVSP to that expected for ANO-1 after 48 EFPY. In a letter to the NRC dated September 6, 2000, the applicant states that the maximum 48 EFPY fluence for the ANO-1 vessel is 1.5×10^{19} n/cm^2, as specified in Table 4-1 of BAW-2251A. The applicant also states that the expected target fluences of various capsules in the MIRVSP program are equal to, or greater than, the ANO-1 maximum fluence value. As discussed in the program scope, the surveillance program will provide data at neutron fluences between 6.1×10^{18} n/cm^2 to 2.4×10^{19} n/cm^2. Therefore, the program will provide data exceeding the neutron fluence at the end of the license renewal period (i.e., 48 EFPY).

The staff also noted that the LRA does not indicate how the neutron energy spectrum, gamma heating, and coolant temperature test parameters, being monitored in the MIRVSP, are applicable to ANO-1. In its response, the applicant references Section 4 of BAW-2251A in which these parameters are specified as being mandatory requirements for a valid surveillance program. The ANO-1 reactor vessel integrity program described in Section 4.18 of Appendix B of the LRA will ensure that the neutron energy spectrum, gamma heating, and reactor inlet temperature are monitored such that the MIRVSP capsule data continues to be applicable to the requirements specified in Section 4 of BAW-2251A regarding fluence, neutron energy spectrum, gamma heating, and vessel inlet temperature.

[*Detection of Aging Effects*] The aging of the affected components will be detected by quantifying the change in temperature at 30 ft-lb of energy between unirradiated and irradiated specimens.

[*Monitoring and Trending*] Monitoring and trending is performed by periodic measurement of fracture toughness values from the specimens that are periodically removed from participating plants in the MIRVSP program. Should the fracture toughness value fall below applicable acceptance criteria, the applicant will undertake further programmatic actions, such as repair or replacement, as necessary, to manage these aging effects.

[*Acceptance Criteria*] The acceptance criteria for fracture toughness are that the RT_{PTS} value for each reactor vessel material shall remain below the screening criteria of 270°F for plates, and axial welds and below 300°F for circumferential welds. The requirement also includes a Charpy upper shelf energy (USE) greater than 50 ft-lbs. For the materials whose Charpy USE fall below 50 ft-lb, there are provisions in Appendix G of 10 CFR Part 50. Specifically, the applicant must demonstrate that, during the period of extended operation, the Charpy USE has a margin of safety against fracture equivalent to that specified in Section XI of the ASME Boiler and Pressure Vessel Code.

[*Operating Experience*] Operating experience concerning loss in reactor vessel fracture toughness is presented in Section 3.5.3 of BAW-2251A and was reviewed by the applicant using the ANO-1 data base. No instances of reactor vessel component failure from losses in fracture toughness were found.

The staff found the MIRVSP for ANO-1 acceptable since the applicant is a participant in an ongoing B&WOG surveillance program that will directly measure the increase in 30 ft-lb transition temperature as a function of neutron irradiation. The data will be applied to the ANO-1 reactor vessel, and the applicant will assure that the fracture toughness values meet the requirements of 10 CFR Part 50 or the applicable sections of the ASME Boiler and Pressure Vessel Code, as described above under "Acceptance Criteria."

Cavity Dosimetry Program

[*Program Scope*] The purpose of the cavity dosimetry program is to verify the accuracy of the fluence calculations and to determine uncertainty values. Within the scope of the program, the applicant has installed cavity dosimetry. The method involves installing a range of dosimeters in the cavity region outside the reactor vessel. The staff found that the scope of the program is sufficient to verify the accuracy of fluence calculations and to determine fluence uncertainty values.

[*Parameters Measured*] The parameter to be determined is the fast fluence value for irradiated specimens to verify the accuracy of fluence calculations and to determine uncertainty. The fluences are obtained from the data obtained from the dosimeters.

[*Detection of Aging Effects*] No aging effects are specifically monitored, per se, in this effort.

[*Monitoring and Trending*] The program monitors and trends the accumulation of neutron fluence from the dosimeter data.

[*Acceptance Criteria*] The data are used in conjunction with the fluence and uncertainly calculations.

This program is consistent with the requirements of 10 CFR 50.60, Appendix H, "Reactor Vessel Materials Surveillance Program Requirements and 10 CFR 50.61" and, therefore, acceptable to the staff.

Fluence and Uncertainty Calculations

[*Program Scope*] The purpose and scope of the reactor vessel fluence and uncertainty calculations are to provide accurate predictions of the actual reactor vessel neutron fast fluence value for use in the development of the pressure/temperature limit curves and pressurized thermal shock calculations.

The cavity dosimetry program yields irradiated dosimeters whose activation values are analyzed in terms of ANO-1 specific geometry models (i.e., fuel, reactor vessel, capsule holders, concrete structures), macroscopic cross-sections, cycle specific sources using the DORT and GIP computer codes, and reference microscopic cross-sections using BUGLE 93.

[*Acceptance Criteria*] Based on the calculations, the reactor vessel fluence uncertainty values are to be within the NRC-suggested ±20 percent. In a letter dated October 3, 2000, the applicant clarifies the method used to determine the fluence at the inner diameter of the reactor vessel. The fluence was not determined by using the technique reviewed and approved by the staff in BAW-2241A. The generic reactor vessel aging management report BAW-2251, was completed prior to finalization of the BAW-2241A approach. Fluence evaluations were subsequently completed for ANO-1 using the method described in BAW-2241A. The 48 EFPY fluence estimates reported in BAW-2251 for the most limiting locations within the beltline region are conservative with respect to the fluence estimates obtained using the BAW-2241A method.

Since the calculations are consistent with requirements of 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," the staff found the program acceptable for the period of extended operation.

Pressure/Temperature Limit Curves

[*Program Scope*] The purpose and scope of this program is to establish pressure/temperature limit curves to establish the normal operating, inservice leak test, and hydrostatic test limits for the RCS, as applicable to the ANO-1 pressure vessel. The curves are used to limit operations based on the material properties of the vessel caused by neutron irradiation.

[*Parameters Monitored*] Pressure/temperature limit curves are generated assuming that a 1/4T surface flaw exists, and using the fracture mechanics methodology in ASME Section XI,

Appendix G. The pressure-temperature curves are determined by using bounding input heatup and cooldown transients.

[*Acceptance Criteria*] The pressure/temperature limit curves are valid for a specified number of effective full power years (EFPY). The curves must be updated before this time period is exceeded.

Since this procedure meets the requirements of 10 CFR Part 50, Appendix G, the staff found it acceptable for the period of extended operation.

Monitoring Effective Full-Power Years

[*Program Scope*] The purpose and scope of this program are to accurately monitor and tabulate the accumulated operating time experienced by the reactor vessel. The EFPY data are used to ensure that the power history is within ±0.3 effective full-power days (EFPD) of the plant computer generated value, and to determine the period of time for which the pressure-temperature limit curves are applicable.

[*Parameters Monitored*] The program monitors and tabulates the accumulated operating time experienced by the ANO-1 reactor vessel. The EFPY of plant operation are based on reactor incore power calculations obtained from the plant's Nuclear Applications Software program. Site reactor engineers determine EFPY values by comparing burnup estimated from incore instrumentation to the thermal power calculated burnup.

[*Acceptance Criteria*] For a given fuel cycle, the calculation based on power history must be within ±0.3 EFPD of the plant computer generated value. The program is acceptable to the staff because it is consistent with the requirements of 10 CFR 50.60, Appendices G and H, ANO-1 TS 3.1.2, and BAW-10046, Revision 1, which was reviewed and approved by the staff in a letter dated June 22, 1977.

Alloy 600 Aging Management Program

The Alloy 600 AMP, as described in the LRA, does not specifically include inspection of any reactor vessel component. This is because none of the vessel components were in the top three groupings of Alloy 600 and Alloy 82/182 components that were determined to be most susceptible to cracking. Only components in the top three most-susceptible groupings, which are from the pressurizer, are to be evaluated. If future analyses show that reactor vessel components fall into the top three grouping, then the applicant will include them in the inspection program. The staff found this approach to be acceptable. Alloy 600 reactor vessel components that are susceptible to cracking are managed through the CRDM, and other vessel closure penetrations inspection program, as described above in this section. Additional discussion of the Alloy 600 AMP is included in Section 3.3.2.3.2.2 of this SER.

Bolting and Torquing Activities

This program is designed to prevent degradation of bolting and, if degradation exists, to identify and correct the affected area.

[*Program Scope*] The scope of the bolting and torquing activities program includes inspections and preparation of mating surfaces in accordance with generic industry guidelines. The program covers pressure boundary bolting applications associated with components subject to an AMR. The applicant includes bolted flange connections for vessels (i.e., manways and inspection ports), flanged joints in piping, body-to-body joints in valves, and pressure-retaining bolting associated with pumps, valves, and miscellaneous process components.

[*Preventative/Mitigative Actions*] By conducting the bolting and torquing activities program, the possibility of cracking, loss of material, and loss of mechanical closure integrity will be reduced. If such degradation is detected, the applicant will undertake further programmatic actions, such as repair and replacement, as necessary, to manage these aging effects.

[*Parameters Inspected or Monitored*] An ANO-1 site procedure gives guidance regarding inspections and preparation of mating surfaces, threaded fasteners, and bolted joints. Instructions are provided for proper tightening of fasteners and the use of wrenching devices.

[*Detection of Aging Effects*] Examination of threaded fasteners includes visual observations of male and female threads for major defects (nicks, burrs, evidence of galling, etc.).

[*Monitoring and Trending*] Monitoring of flaw growth will make it possible to quantify the extent of bolting degradation so that the applicant may undertake further programmatic action, such as repair and replacement, as necessary, to manage the aging effect.

[*Acceptance Criteria*] Acceptance criteria are given in ANO-1 site procedures. Typical criteria are that mating surfaces are smooth and free of major defects. Other criteria include proper and adequate thread engagement, no loose fasteners, and use of appropriate torque values. The basis for the bolting and torquing activities program is ANO letter dated August 12, 1982, (J.R. Marshall (AP&L) to J.T. Collins (NRC), and the initial AP&L Response to IE Bulletin 82-02, “Degradation of Threaded fasteners in the Reactor Pressure Boundary of PWR Plants.”

[*Operating Experience*] Procedures for bolting and torquing activities at ANO-1 are based on generic industry guidance. The guidance was based on industry experience that has proven effective in maintaining the integrity of bolted closures.

On the basis of the above review, the staff found that the applicant’s program on bolting and torquing activities, which is part of the CLB, will continue to be adequate to assure that threaded joints will perform their intended functions during the period of extended operation.

After its initial review, the staff requested that the FSAR Supplement summary description for the CRDM nozzle and other vessel closure penetrations inspection activities include a statement that the applicant will analyze and evaluate axial flaws using NUMARC acceptance criteria, and will analyze and address circumferential flaws with the NRC on a case-by-case basis. This was FSAR Item 3.3.2.4.3 under Open Item 3.3-1.

In its revised summary description of Section 16.2.7 of the FSAR Supplement, the applicant states that if an inspection program is determined to be necessary for the CRDM nozzle and other vessel closure penetrations, the applicant will analyze and evaluate axial flaws using NUMARC acceptance criteria, and address circumferential flaws with the NRC on a case-by-case basis. The staff finds the revised summary description as submitted by the applicant in a letter to the NRC dated March 14, 2001, acceptable.

3.3.2.3.3 Conclusions

The staff has reviewed the information included in Section 3.3.2.4 “Reactor Vessel” and Appendices A and B of the LRA, and additional information provided by the applicant in response to the staff’s RAIs. On the basis of this review, the staff concluded that the applicant has demonstrated that the effects of aging associated with the reactor vessel can be adequately managed such that there is reasonable assurance that the reactor vessel will continue to perform its intended function consistent with the CLB for the period of extended operation.

3.3.2.4 Reactor Vessel Internals

The reactor vessel internals (RVI) consist of two major structural subassemblies located within the reactor vessel: the plenum assembly (PA) and the core support assembly (CSA). The CSA is further subdivided into three principal subassemblies: the core support shield assembly (CSS), the core barrel assembly (CBA), and the lower internals assembly (LIA). The reactor vessel internals are not integrally attached to (i.e., not welded to) the reactor vessel and can be removed during refueling outages when necessary. The control rod assemblies (CRA), the fuel assemblies (FA), and the incore monitoring system (IMS) are not considered part of the reactor vessel internals. There are no pressure-retaining or pressure boundary welds within the scope of the RVI.

3.3.2.4.1 Technical Information in the Application

The applicant reviewed the current design and operation of the ANO-1 reactor vessel internals using the process described in Sections 2.3.1.1 and 2.3.1.2 of the LRA. On the basis of this review, the applicant determined that all RVI components in the plenum assembly and the core support assembly at ANO-1 are bounded by the description contained in B&W Topical Report BAW-2248A, except for the following:

- one of the control rod drive mechanisms (CRDMs) was removed and the control rod guide assembly (CRGA) in the plenum was modified to accept the reactor vessel level monitoring probe fabricated from Type 304L austenitic stainless steel

- portions of the surveillance specimen holder tubes and associated supports fabricated from austenitic stainless steel remain bolted to the core support shield, although all the specimens have been removed

- the thermal shield, and thermal shield upper restraint, which are all fabricated from austenitic stainless steel

Therefore, the applicant includes the reactor vessel level monitoring system (RVLMS) probe supports, surveillance specimen holder tubes (SSHT), and thermal shield and thermal shield upper restraint in the scope of license renewal and subjected them to AMR for ANO-1.

The applicant identifies eight intended functions applicable to the ANO-1 RVI. The first five items are those listed in BAW-2248A and the remaining three items are specific to ANO-1, as follows:

Bounded by BAW-2248A:

- provide support and orientation of the reactor core (i.e., the fuel assemblies)
- provide support, orientation, guidance and protection of the control rod assemblies
- provide a passageway for the distribution of the reactor coolant flow to the reactor core
- provide a passageway for support, guidance, and protection for the in-core instrumentation
- provide a secondary core support for limiting the downward displacement of the core support structure in the event of a postulated failure of the core barrel

ANO-1-specific:

- support the reactor vessel level monitoring probe
- provide gamma and neutron shielding of the reactor pressure vessel
- provide support for the surveillance specimen assemblies in the annulus between the thermal shield and the reactor vessel wall

Effects of Aging

The operating environment, or chemistry of the fluid in contact with the ANO-1 reactor vessel internals, is maintained in accordance with the primary water chemistry monitoring program. In accordance with ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NG, “Core Support Structures,” the vessel internals are designed to accommodate service loadings (i.e., levels A through D). Operation under level A and B service conditions contributes to the normal aging stresses for the RVI. The applicant states that the ANO-1 RCS has not been subjected to a level C or D event.

All RVI components are fabricated from stainless steel, except the fuel assembly support pad dowels for both upper and lower grid assemblies and the barrel-to-former bolt dowels which are made from NiCr alloy (Inconel X750). Miscellaneous locking device parts in the vent valve body are modified with NiCrFe alloy (Inconel 600) and/or NiCr alloy (Inconel 718).

In the LRA, Table 3.2-1, the applicant identifies the following applicable aging effects for the components subject to an AMR:

- RVLMS probes are consistent with other stainless steel items in the plenum assembly and are susceptible to cracking and loss of mechanical closure integrity caused by stress relaxation of the RVLMS brazement guide assembly j-bolts and nut

- thermal shield that surrounds the core barrel and shield upper restraint are consistent with other stainless steel items in the core barrel assembly and are susceptible to cracking and reduction in fracture toughness

- upper SSHT assembly and some of the brackets and bolts are susceptible to cracking and loss of mechanical closure integrity by stress relaxation of the bolting

Section 3.1 of BAW-2248A states that changes in the dimensions of the RVI components from void swelling are not significant aging effects . The basis for this conclusion is the lack of evidence of void swelling under pressurized water reactor conditions. However, EPRI TR-107521 cites several sources with conflicting results. One source predicts swelling as great as 14 percent for PWR baffle-former assemblies over a 40-year plant lifetime, whereas results from another source indicate that swelling would be less than 3 percent for the most highly irradiated sections of the internals at 60 years. The issue of concern to the staff is the impact of change of dimensions due to void swelling on the ability of the RVI to perform their intended functions. The specific impacts of concern are constriction of critical coolant paths, interference with control rod insertion, and excessive baffle bolt loading.

Survey of Industry and ANO-1 Operating Experience

To validate its determination of applicable aging effects for all the RVI components discussed above, the applicant surveyed industry experience, NRC generic communications, and its own operating history.

BAW-2248 discussed NRC IN 91-05 relating to the cracking in Alloy A-286 bolts used in reactor coolant pumps and the B&W-designed RVI. In the LRA, Section 3.2.5, the applicant also identifies NRC IN 98-11, which discusses the cracking of RVI baffle former bolts in foreign plants, and the current and planned activities of the B&WOG to address the potential for cracking of the baffle bolts. On the basis of the applicant’s review of ANO-1 operating data using the station information management system, condition reporting system, and licensee event database, cracking of the thermal shield bolting and core barrel bolting fabricated from Alloy A-286 was identified as an issue. These failures were attributed to inter-granular stress corrosion cracking (IGSCC), and were not detected by visual examinations. Another failure associated with the surveillance specimen holder tubes was due to a design flaw and was considered not to be caused by aging-related degradation. The applicant identifies no additional aging effects beyond those discussed above.

Aging Management Programs

In the LRA, Section 3.2.5, the applicant identifies the following AMPs for the RVI:

- ASME Section XI Inservice Inspection Program, Subsection IWB
- primary chemistry monitoring program
- reactor vessel internals AMP (RVIAMP)

The applicant concludes that these programs would manage aging effects in such a way that the intended function of the components of the RVI would be maintained consistent with the CLB, under all design loading conditions, for the period of extended operation.

3.3.2.4.2 Staff Evaluation

In accordance with 10 CFR 54.21(a)(3), the staff reviewed the information included in Sections 2.3.1.6 (Table 2.3-5), 3.2.5 (including Table 3.2-1), Appendix B (Sections 3.5, 4.3.1, 4.6.1) of the ANO-1 LRA, and the staff's final SER on the topical report BAW-2248, regarding the applicant's demonstration that the effects of aging will be adequately managed so that the intended function would be maintained consistent with the CLB for the period of extended operation for the RVI.

There are 12 action items in the staff's final SER on BAW-2248. The action items, licensee's responses, and staff's evaluations are given below.

Action Items from Previous Staff Evaluation of BAW-2248

As discussed below, the staff found that the applicant's responses (Table 2.3-5 of the LRA) to the Renewal Applicant Action Items from this report resolve the 12 action items from BAW-2248.

- Item 1: The license renewal applicant is to verify that the critical parameters for the plant are bounded by the topical report. Further, the renewal applicant is to commit to programs described as necessary in the topical report to manage the effects of aging during the period of extended operation on the functionality of the reactor vessel internals components. The applicant for license renewal will be responsible for describing any such commitments and proposing the appropriate regulatory controls. Any deviations from the AMPs within the topical report described as necessary to manage the effects of aging during the period of extended operation and to maintain the functionality of the reactor vessel internal components or other information presented in the report, such as materials of construction, will have to be identified by the renewal applicant and evaluated on a plant-specific basis in accordance with 10 CFR 54.21(a)(3) and (c)(1).

  Response: The applicant participated in the development of BAW-2248 by providing ANO-1-specific design and operational information. The applicant has reviewed the current design and operation of the ANO-1 reactor vessel internals, and has determined that the internals are bounded by the descriptions contained in BAW-2248 with the exception of the RVLMS probe supports, SSHTs and supports that are bolted to the core support shield, and the thermal shield and thermal shield upper restraint, which are within the scope of license renewal for ANO-1 (see Section 2.3.1.6 of the LRA). The ANO-1 AMPs for the RVI are described in Appendix B of the LRA. The NRC staff evaluation of the AMPs for these three components is provided in Section 3.3.2.5.2.2 of this SER.

- Item 2: A summary description of the programs and evaluation of TLAAs is to be provided in the license renewal FSAR supplement in accordance with 10 CFR 54.21(d).

  Response: Appendix A of the LRA contains new and revised FSAR sections to provide a summary description of the programs and evaluation of TLAAs, in accordance with 10 CFR 54.21(d). These are evaluated by the staff throughout this SER, when applicable.

- Item 3: The license renewal applicant must identify whether an intended function of the RVI is to provide shielding for the reactor pressure vessel (RPV). If not an intended function, the license renewal applicant should provide justification for that conclusion. Should a license renewal applicant determine that the RVI's intended function is to provide shielding for the RPV, then the items that support this intended function, such as the thermal shield and thermal shield upper restraint assemblies, must be identified and reviewed in accordance with 10 CFR 54.21(a)(3).

  Response: The applicant has included gamma and neutron shielding as one of the RVI intended functions (see Table 3.2-1 of the LRA). Consideration of aging effects for the components supporting this intended function, specifically the thermal shield and the thermal shield upper restraint, and aging management are described in the application. The NRC staff evaluation of the AMP for components supporting this intended function is provided in Section 3.3.2.5.2.2 of this SER. The staff found this to be acceptable.

- Item 4: The applicant must commit to participate in the B&WOG RVIAMP, and any other industry programs as appropriate, to continue the investigation of potential aging effects for RVI components, and to establish monitoring and inspection programs for RVI components. The applicant shall provide the NRC with either annual reports or periodic updates (after completion of significant milestones) on the status of the RVIAMP commencing within one year of the issuance of the renewed license.

  Response: The applicant states that they will participate in the B&WOG RVIAMP and other industry programs, as appropriate, to continue the investigation of potential aging effects for RVI components, and to establish monitoring and inspection programs for RVI components. The applicant will provide periodic updates (after completion of significant milestones) to the NRC on the status of the ANO-1 RVI inspection program commencing within one year of the issuance of the renewed license. The staff found this to be acceptable.

- Item 5: The applicant must describe plans for augmented inspection of RVI components for management of SCC/IASCC and loss of fracture toughness (neutron embrittlement) of the RVI components. This description should specify the sample size, the examination method, acceptance criteria and timing of the inspection, or the process to be used to specify these items.

  Response: The applicant provides a description of programs in the ANO-1 RVIAMP in Appendix B of the LRA. In a letter to the NRC dated September 6, 2000, the applicant states that the ANO-1 inspection is planned for the 2nd or 3rd period of the fifth inspection interval. Sample size, appropriate inspection methods (e.g., visual or volumetric), and acceptance criteria will be determined based on the results of the B&WOG Reactor Vessel Internals AMP and other industry programs investigating the aging effects of the reactor vessel internals. The NRC staff evaluation of this program is given in Section 3.3.2.5.2.2 of this SER.

- Item 6: According to the B&WOG, one of its objectives in BAW-2248 states, "it is intended that NRC review and approval of this report will allow that no further review of the matters described herein will be needed when the report is incorporated by reference in a plant specific renewal license application." The license renewal applicant must address the baffle-former bolt cracking issues addressed in Section 3.3.2 of this SE pertaining to Refs. 4 and 5, with regard to the industry ITG project, initiated after April 23, 1998, to address generic RVI materials issues. The B&WOG indicates this industry effort resulted in subsequent changes in the B&WOG RVI AMP. The ITG is currently addressing the issues of cracking of baffle bolts. The B&WOG indicates that the changes in the AMP now require the applicants to be responsible for using the industry ITG project developed information to determine the necessary steps (e.g., inspection, operability determinations, and replacements) for the management of the applicable baffle bolt aging effects.

Response: The applicant provides a description of programs in the ANO-1 RVIAMP in Appendix B of the LRA. In a letter to the NRC dated August 24, 2000, the applicant states that it will use the tools provided by the ITG and the B&WOG to determine the necessary steps (e.g., inspections, operability determinations, and replacements) to manage the applicable aging effects in baffle-former bolts at ANO-1. The ANO-1 RVIAMP includes commitments to participate in B&WOG and applicable ITG programs. The NRC staff evaluation of this program is given in Section 3.3.2.5.2.2 of this SER.

- Item 7: The applicant must describe plans for augmented inspection of RVI components for management of loss of fracture toughness by thermal aging embrittlement of the RVI components. This description should specify the sample size, the examination method, acceptance criteria and timing of the inspection, or the process to be used to specify these items.

  Response: The applicant provides a description of programs in the ANO-1 RVIAMP in Appendix B of the LRA. In a letter to the NRC dated August 24, 2000, the applicant states that it will use the EPRI ITG and B&WOG programs to quantify the susceptibility of the CASS and martensitic components to reduction of fracture toughness by thermal and irradiation embrittlement. Where reduction of fracture toughness is identified as a concern for the period of extended operation, the sample size for augmented inspection will be determined by considering both the susceptibility and consequences of failure of the affected parts. The analytical evaluations will be completed before the inspection method can be selected. The inspection is planned for the 2nd or 3rd period of the fifth inspection interval at ANO-1. The NRC staff evaluation of this program is given in Section 3.3.2.5.2.2 of this SER.

- Item 8: The applicant must describe plans for management of stress relaxation for bolted closures of the RVI. This description should specify the critical locations, monitoring and inspection techniques, and timing of the inspection, or the process to be used to specify these items.

  Response: The applicant provides a description of programs in the ANO-1 RVIAMP in Appendix B of the LRA. In a letter to the NRC dated August 24, 2000, the applicant states that it is committed to participate in both ITG and B&WOG RVIAMP industry programs. In addition, the inspection at ANO-1, planned for the end of the fifth interval, will focus on the effect of stress relaxation for critical bolted connections at ANO-1. The NRC staff's evaluation of this program is given in Section 3.3.2.5.2.2 of this SER.

- Item 9: The applicant must address aging management of void swelling. An adequate AMP would include participation in industry program(s) to address the significance of void swelling (either individually or through an owners' or industry group), a commitment to develop a sufficient inspection program (including the basis, methods, locations to be examined, timing, frequency and acceptance criteria) for management of the issue based upon the results of the industry programs, and a commitment to implement the inspection program prior to the end of the current license period.

  Response: The applicant provides a description of programs in the ANO-1 RVIAMP in Appendix B of the LRA. In a letter to the NRC dated August 24, 2000, the applicant states that it is participating in industry and B&WOG programs that will assess the applicability of void swelling to the B&W-designed internals. The inspection program will be developed prior to the end of the current license period. However, actual inspections, if required, would be performed in the 2nd or 3rd period of the fifth inspection interval at ANO-1. The applicant has identified dimensional changes due to void swelling for plates, forgings, welds, core barrel bolts, and thermal shield in the reactor vessel internal as part of the RVIAMP. The program elements for this aging effect in the RVIAMP have not been developed yet. The applicant is committed to develop these program elements for the void swelling as part of the development of the RVIAMP. The NRC staff evaluation of this program is given in Section 3.3.2.5.2.2 of this SER.

- Item 10: If flaws have been detected in the reactor vessel internals, a TLAA plant-specific evaluation must be performed to determine the flaw growth acceptance in accordance with the ASME B&PV Code, Section XI, inservice inspection requirements.

  Response: The applicant has stated that no flaws requiring analytical evaluation have been found in the inspections of the RVI, therefore, no flaw growth evaluations need to be performed for ANO-1. The staff found this to be acceptable.

- Item 11: The applicant must address the plant-specific plans to continue monitoring and tracking design transient occurrences.

  Response: The ANO-1 transient cycle logging program described in Section 4.3.5 of the LRA provides monitoring and tracking of the design transient occurrences. The NRC staff evaluation of this program is provided in Section 4.3 of this SER.

- Item 12: Plant-specific analysis is required to demonstrate that, under loss-of-coolant-accident and seismic loading, the internals have adequate ductility to absorb local strain at the regions of maximum stress intensity and that irradiation accumulated at the expiration of the renewal license will not adversely affect deformation limits. The RVIAMP must develop data to demonstrate that the internals will meet the deformation limits at the expiration of the renewal license.

  Response: The applicant will develop the necessary data, and perform the necessary analyses, to demonstrate that the reactor vessel internals will have sufficient ductility to absorb local strain in the regions of high stress intensity, and will meet the deformation limits at the end of the period of extended operation. At present, the applicant plans to complete the evaluation prior to the end of the current term of operation. The development of data and analysis will be performed as part of the applicant's RVIAMP, which is discussed in Section 3.3.2.5.2.2 of this SER.

3.3.2.4.2.1 Effects of Aging

In the LRA, Section 3.2.5, the applicant identifies the following aging effects as applicable to ANO-1 reactor vessel internals:

- loss of material

- cracking of base metal, welds, and bolting

- reduction of fracture toughness of base metal, welds, and bolting by irradiation embrittlement

- reduction of fracture toughness of cast austenitic stainless steel (CASS) items due to thermal aging embrittlement

- dimensional changes by void swelling

- loss of mechanical closure integrity (by stress relaxation and cracking)

The staff reviewed the information in the LRA, and determined that the applicant has identified the aging effects that are applicable to ANO-1 reactor vessel internals, on the basis of the description of the environment, the materials of construction, the ANO-1 operating experience, and the applicant's review of industry experience.

3.3.2.4.2.2 Aging Management Programs

The staff's evaluation of the applicant's AMPs focused on the program elements rather than details of specific plant procedures. To determine whether the applicant's AMPs are adequate to manage the effects of aging so that the intended functions will be maintained consistent with the CLB for the period of extended operation, the staff evaluated the following ten elements:

- scope of program
- preventive actions
- parameters monitored or inspected
- detection of aging effects
- monitoring and trending
- acceptance criteria
- corrective actions
- confirmation process
- administrative controls
- operating experience

In Appendix B of the LRA, Section 2.0, of the applicant states that corrective actions, confirmatory process, and administrative controls for license renewal are implemented in accordance with the site controlled corrective actions program (i.e., the Entergy Quality Assurance Program) pursuant to 10 CFR Part 50, Appendix B, for all SCs that are subject to an AMR. The staff's evaluation of the applicant's corrective action program is presented separately in Section 3.3.1.2 of this SER. Therefore, the corrective actions, confirmation process, and administrative controls will not be discussed further in this section.

The applicant identifies the following AMPs used to manage the effects of aging for the reactor vessel internals:

- ASME Section XI Inservice Inspection Program, Subsection IWB
- primary chemistry monitoring program
- reactor vessel internals AMP (RVIAMP)

ASME Section XI Inservice Inspection Program, IWB Inspections (Examination Category B-N-3)

In the LRA, Table 3.2-1, the applicant states that this program is used to manage loss of material, cracking (due to IASCC and SCC), reduction of fracture toughness (due to thermal embrittlement and neutron irradiation embrittlement), loss of closure integrity, and dimensional changes due to void swelling of the RVI components. Examination Category B-N-3 in Section XI of the ASME B&PV Code (1992 Edition with portions of the 1993 Addenda for pressure testing) requires a VT-3 examination of portions of the removable core support structures. IWA-2213 of Section XI of the ASME Code states that VT-3 examinations are conducted to determine the general mechanical and structural condition of components and their supports, and to detect discontinuities and imperfections, such as loss of integrity at bolted or welded connections, loose or missing parts, debris, erosion, wear, or corrosion. This existing program is adequate to manage the loss of material because the VT-3 examination is capable of detecting loss of material. However, for the other aging effects, the VT-3 examination may not be sufficiently sensitive to the degradation expected from the aging effects. The RVIAMP will determine whether VT-3 examination is adequate. Details of the review for this program are found in Section 3.3.1 of this SER.

Primary Water Chemistry Monitoring Program

The staff's evaluation of this AMP is provided in Section 3.3.1 of this SER.

ANO-1 Reactor Vessel Internals Aging Management Program (RVIAMP)

Examination Category B-N-3 of the ASME Section XI ISI program, Subsection IWB, requires a visual VT-3 inspection of removable core support structures. This inspection program may not be adequate to detect aging effects for certain reactor vessel internal components since their locations are not easily accessible using current technology. The purpose of this program is to determine whether visual inspections (VT-1 type) and nondestructive examinations of the ANO-1 reactor vessel internals are necessary during the period of extended operation. The major activities associated with this program include participation in industry activities, reporting results to the NRC, and performance of inspections.

The scope of this inspection program covers cracking due to irradiation-assisted stress corrosion cracking (IASCC) and SCC, reduction of fracture toughness due to irradiation embrittlement and thermal embrittlement, dimensional changes due to void swelling, and loss of closure integrity due to stress relaxation. The inspection applies to the RVI stainless steel items including the following:

- items comprised of plates, forgings, welds, core barrel bolts, and thermal shield bolts

- baffle bolts

- items fabricated from CASS and martensitic steel which include control rod guide tube (CRGT) spacers, vent valve bodies, vent valve retaining rings and incore guide tube assembly spiders. The vent valve retaining rings, fabricated from martensitic stainless steel, are also included in this inspection

Within this program, the applicant has committed to participate in the B&WOG RVIAMP and other industry programs, as appropriate. Reports will be provided to the NRC on a periodic basis after completion of significant milestones, commencing within one year after the issuance of the renewed license. A final report will also be submitted by the applicant at, or about, the end of the initial 40-year operating license term. The final report will summarize the understanding of aging effects applicable to the reactor vessel internals and will contain the applicant's inspection plan including methods for each inspection. On the basis of the information developed from the RVIAMP, the applicant will implement an AMP for the RVI. Should data or evaluations indicate that this inspection can be modified or eliminated, the applicant will provide plant-specific justification to demonstrate the basis for the modification or elimination.

For RVI components, including plates, forgings, welds, thermal shield and bolting other than baffle bolting, the applicant identifies cracking due to stress corrosion and irradiation-assisted stress corrosion, reduction of fracture toughness due to irradiation embrittlement, dimensional changes due to void swelling, and loss of bolted closure integrity due to stress relaxation as applicable aging effects. The applicant currently plans to perform a visual inspection of the plates, forgings, welds, core barrel bolts, and thermal shield bolts, visual inspection may not provide access to critical areas on bolts. Activities are in progress to develop and qualify the inspection method. The sample size for the inspection will be determined as part of the development of the inspection program. Critical crack size will be determined by analysis and acceptance criteria will be developed prior to the inspection.

For the baffle bolts, the applicant identifies cracking due to IASCC, reduction of fracture toughness due to irradiation embrittlement, and dimensional changes due to void swelling as applicable aging effects. The applicant currently plans to perform a volumetric inspection of the baffle bolts. Activities are in progress to develop and qualify the inspection method. The sample size for the inspection will be determined as part of the development of the inspection program. The number of baffle bolts needed to be intact, and its locations will be determined by analysis and acceptance criteria for dimensional changes due to void swelling will be developed prior to the inspection.

For the CASS and martensitic steel components, the applicant identifies reduction of fracture toughness by thermal embrittlement and irradiation embrittlement as applicable aging effects. Since the reduction of fracture toughness cannot be measured through traditional in-situ examination techniques, an analytical approach to assess the effects of reduction of fracture toughness on the applicable reactor vessel internal items will be used. The specific inspection method will depend on the results of this analysis. The sample size for the inspection will be determined as part of the development of the inspection program. The critical crack size given the service loading conditions and the service-degraded material properties will be determined by analysis. Acceptance criteria for all aging effects will be determined prior to the inspection.

With regard to inspection frequency, in a letter to the NRC dated August 24, 2000, the applicant states that it plans to perform the inspection of the ANO-1 internals in the second or third period of the fifth inspection interval. At present, the inspection at ANO-1 is planned after the first inspection is completed at Oconee. The need for subsequent inspections at ANO-1 will be based on the first inspection results both at Oconee and ANO-1.

The applicant proposes to begin inspection under the RVIAMP during the 20-year period of extended operation, and the inspection will be performed once during the 20-year extension period. The need for subsequent inspections will be based on the results from this inspection.

This inspection program is acceptable since pertinent components will be examined using appropriate inspection methods, in a time frame that will provide useful results throughout the period of extended operation.

The staff concludes that the ANO-1 RVIAMP can provide sufficient data and analyses to demonstrate adequate management of aging effects for the RVI such that their intended function will be maintained consistent with the CLB for the period of extended operation, in accordance with 10 CFR 54.21(a)(3).

3.3.2.4.3 Conclusions

On the basis of the review described above, the staff finds that the applicant has demonstrated that the effects of aging associated with the reactor vessel internals will be adequately managed so that there is reasonable assurance that the intended function will be maintained with the CLB for the period of extended operation.

3.3.2.5 Once-Through Steam Generators

ANO-1 has two OTSG. Each is a vertical, straight-tube, once-through, counterflow, shell-and-tube heat exchanger with shell-side boiling. The OTSG consists of upper and lower hemispherical heads welded to tubesheets that are separated by a seven-course shell assembly. Over 15,000 straight Alloy 600 tubes are held in alignment by fifteen tube support plates.

Primary coolant from the reactor enters the steam generator through a single inlet nozzle in the top of the upper head. Coolant flows downward through the straight parallel tubes, is cooled by the secondary coolant on the shell side, and then exits through two outlet nozzles in the lower head. Secondary coolant enters through a ring of ports that penetrate the shell approximately midway up the shell assembly. The feedwater travels downward through an annulus between the lower baffle and the shell. Near the lower tubesheet, the feedwater turns inward and then flows upward around the tubes, and through the tube support plates. As the feedwater absorbs heat from the primary coolant, it boils and then becomes superheated. The dry steam exits the steam generator through two steam outlet nozzles just above the feedwater inlet ports.

3.3.2.5.1 Technical Information in the Application

The OTSGs were designed as Class A vessels in accordance with ASME Section III, 1965 Edition, with Addenda through Summer of 1967. In the LRA, Section 2.3.1.7, the applicant identifies OTSG components that are subject to an AMR. They include the following: hemispherical heads, secondary shell, tubes, plugs, mechanical sleeves, tubesheets, primary nozzles, primary manway and inspection port assemblies, main and emergency feedwater nozzles, main and emergency feedwater header and riser piping, steam outlet nozzles, instrument nozzles, temperature sensing connections, drain nozzles, secondary manway and inspection port covers, associated pressure retaining bolting, and integral attachments inspected in accordance with ASME Section XI, Subsections IWB and IWC. ASME Class 1 RCS piping attached to the primary OTSGs' nozzles, including the welded joints, is addressed in Section 2.3.1.3 of the LRA. Secondary piping attached to the OTSGs' nozzles, including the main and emergency feedwater headers and riser piping, is addressed in Sections 2.3.4.2 and 2.3.4.3 of the LRA, respectively. The OTSG supports are addressed in Section 2.4.2 of the LRA.

Intended Functions

The applicant identifies the following intended functions to be applicable to the ANO-1 OTSGs:

- maintain primary pressure boundary.

- maintain secondary pressure boundary.

- provide heat transfer from the primary fluid to the secondary fluid.

- provide reactor building isolation.

Effects of Aging

Items fabricated from low-alloy steel included the hemispherical heads, tubesheets, transition ring, and pressure retaining bolting. Items fabricated from carbon steel include primary inlet and exit nozzles, secondary shell, secondary outlet nozzles, main and auxiliary feedwater headers and riser piping, primary and secondary manway covers, primary and secondary inspection port covers, secondary vent nozzles, drain nozzles, level sensing nozzles, and main and emergency feedwater nozzles. Items fabricated from Alloy 600 include the primary drain nozzles, nozzle dam support rings, tubes, and secondary temperature sensing connections. The plugs and sleeves installed inside the tubes are made of Alloy 600 or Alloy 690, and are less susceptible to loss of material.

As discussed In the LRA, Section 3.2.6 and Table 3.2-1,the applicant divides the aging effects into two separate groups of OTSG components to facilitate the aging analysis. These include components within the primary pressure boundary inspected in accordance with ASME Section XI, Subsection IWB, and components within the secondary pressure boundary inspected in accordance with ASME Section XI, Subsection IWC. In accordance with the requirements for the components of ASME Boiler and Pressure Vessel Code, Section III, the OTSG is designed to accommodate service loads (i.e., levels A through D). Operation under level A and B service conditions contribute to the normal aging stresses for the OTSG. The applicant states that the ANO-1 RCS had not been subjected to a level C or D event.

The applicant identifies the following aging effects applicable to OTSG components:

- loss of material
- cracking
- mechanical distortion of tubes
- loss of mechanical closure integrity

The applicant states that loss of material due to intergranular attack (IGA), pitting, wear, fretting, and erosion/corrosion, and wastage are applicable aging effect of steam generator tubes and other internal components. The external surfaces of the pressure boundary components are subject to loss of material due to boric acid wastage. Cracking at welded joints is an aging effect requiring management for clad low-alloy steel heads, clad low-alloy tubesheets, clad carbon steel nozzle forgings, and other secondary pressure boundary components. In addition, cracking caused by primary water stress corrosion cracking has been identified as an applicable aging effect for Alloy 600 and Alloy 690 tubes, plugs, sleeves, and drain nozzles. Mechanical distortion may occur in the steam generator tubes as a result of denting. Finally, stress relaxation and corresponding loss of preload of bolted connections and adjacent flange surfaces may lead to localized leakage of reactor coolant and loss of mechanical closure integrity due to boric acid wastage.

Survey of Industry and ANO-1 Operating Experience

The applicant reviewed NRC generic communications, licensee event reports from nuclear power plants other than ANO-1, NRC NUREGs, and its own operating experience using the station information management system, condition reporting system, and licensee event database, to validate the identified aging effects requiring management for the OTSGs. No additional aging effects were identified from this review.

Aging Management Programs

In the LRA, Section 3.2.6 and Table 3.2-1, the applicant identifies the following AMPs for the OTSGs:

- boric acid corrosion prevention program
- primary water chemistry monitoring program
- secondary water chemistry monitoring program
- ASME Section XI Inservice Inspection Program, Subsections IWB and IWC
- leakage detection in reactor building
- steam generator integrity program
- Alloy 600 AMP
- bolting and torquing activities

The applicant concludes that these programs would manage aging effects in such a way that the intended function(s) of the components of the OTSGs would be maintained consistent with the CLB, under all design loading conditions during the period of extended operation.

3.3.2.5.2 Staff Evaluation

In accordance with 10 CFR 54.21(a)(3), the staff reviewed the information included in Sections 2.3.1.7, 3.2.6 (including Table 3.2-1), Appendix B (Sections 4.1, 4.3.1 and 4.3.2, 4.4, 4.5, 4.6.1 and 4.6.2, 4.12, and 4.20) of the ANO-1 LRA, regarding the applicant's demonstration that effects of aging will be adequately managed so that the intended function would be maintained consistent with the CLB for the period of extended operation for the OTSGs.

3.3.2.5.2.1 Effects of Aging

The aging effects identified by the applicant as applicable to ANO-1 OTSGs include the following:

- loss of material in both primary and secondary components,
- cracking in both primary and secondary components,
- mechanical distortion of tubes, and
- loss of mechanical closure integrity in primary and secondary components

Three other effects of aging that could impact the intended functions of the steam generators are cracking or rupture of tubes (due to fatigue), outside-diameter stress corrosion cracking (ODSCC), and flow-accelerated corrosion (FAC).

Flow of secondary fluid can cause high-frequency vibration and/or fluid elastic instability conditions of tubes and interaction with the tube support structures. Past operating experience with tube failures caused by high-frequency vibration was noted at Oconee, leading to forced outages in 1994.

A recent B&W owners group report, prepared by Framatome Technologies Inc. (Report No. 77-5003013-00, 2/99) on the OTSG internals, has indicated that FAC can occur if there is a significant blockage of flow due to fouling. During the last chemical cleaning of the OTSG at Oconee 2, higher than expected corrosion rates were noted. Although the corrosion allowance has not exceeded the design threshold, an increase in the allowance for the tube support plate land is necessary should a second cleaning be required.

The installation of sleeves in SG tubes causes a mechanical distortion of the tube at the expansion joint of the sleeve. The increased stress in the tube makes it susceptible to circumferential cracking at this location.

In a letter to the NRC dated September 6, 2000, the applicant states that cracking of OTSG tubes may be caused by any of the specific mechanisms of fatigue, PWSCC, intergranular stress-corrosion cracking, or ODSCC. Tube locations in the lane and wedge region that may be susceptible to high cycle fatigue have received preventive sleeving at ANO-1. The sleeving was performed to preclude future tube failures in this region of the OTSG. ODSCC has been observed in most PWR steam generators in the United States and abroad that contain Alloy

600 tubes. This mechanism has primarily been observed in recirculating steam generators at or near the tube support plates, at the top of the tubesheet, and in the freespan regions, but has not been as prevalent in OTSG tubes.

The applicant also states that the aging effects associated with OTSGs are identified in Section 3.2.6 of the LRA, but did not identify all applicable AMPs for each aging effect as requested in an RAI. In addition, the applicant did not describe the changes, if any, in the SG tube integrity program at ANO-1 in order to address tube degradation identified in NRC IN 97-49 and 97-88. The NRC IN 97-49 addresses degradation at dented locations, the expansion transition region, freespan locations, sleeved regions, and sludge pile region found during OTSG tube inspections. The NRC IN 97-88 discusses the potential difficulties experienced by the applicant in qualifying and applying eddy current depth-sizing techniques. Although the applicant has not specifically addressed these aging effects in its LRA, the steam generator tube integrity program discussed in Section 3.3.2.6.2.2 is broad enough to address these aging effects.

Loss of material is identified as an applicable aging effect in Section 3.2.6 of the ANO-1 LRA for the OTSG tubes, plugs and sleeves. FAC is an aging mechanism that may result in loss of material at the carbon steel tube support plates, however, the tube support plates are not subject to an AMR. The Framatome Technologies Inc. (FTI) report deals specifically with OTSG internals, which are fabricated from carbon steel, and there is no discussion in the FTI report that identifies loss of Alloy 600 tube material adjacent to the tube support plate due to FAC as an aging issue. In the unlikely event that FAC of the tube support plates results in loss of material of the Alloy 600 tubes due to wear in regions adjacent to the tube support plates, tube integrity would be assured through tube inspections performed in accordance with the ANO-1 steam generator integrity program.

The staff agrees that the applicant identifies the aging effects that are applicable to ANO-1 OTSGs, on the basis of the description of the environment, the materials of construction, the ANO-1 operating experience, and the applicant's review of industry performance experience.

3.3.2.5.2.2 Aging Management Programs

The staff evaluation of the applicant's AMPs focused on the program elements rather than details of specific plant procedures. The staff's approach to evaluating each program and activity used to manage the applicable aging effects is described in Section 3.3.1 of this SER.

As mentioned above in Section 3.3.2.6.1 of this SER, the following existing AMPs will be continued during the period of extended operations:

- boric acid corrosion prevention program
- primary water chemistry monitoring program
- secondary water chemistry monitoring program
- ASME Section XI Inservice Inspection Program, Subsections IWB and IWC
- leakage detection in reactor building
- steam generator integrity program
- Alloy 600 AMP
- bolting and torque activities

The staff's evaluation of the primary water chemistry monitoring program, ASME Section XI Inservice Inspection Program, Subsections IWB and IWC, and leakage detection in reactor building AMP may be found in Section 3.3.1 of this SER. The Alloy 600 and bolting and torquing activities AMPs are discussed in Sections 3.3.2.3 and 3.3.2.4, of this SER respectively. The staff's evaluation of the steam generator integrity AMP is discussed below.

The staff found that these programs will be adequate to detect aging effects in OTSG components such that if unacceptable degradation is detected, the applicant will undertake further programmatic actions, including repair and replacement, as necessary, to manage the effects of aging.

Steam Generator Integrity Program

The steam generator integrity program at ANO-1 ensures that steam generator integrity is maintained under normal operating, transient, and postulated accident conditions. The program is structured to meet the NEI 97-06, "Steam Generator Program Guidelines," which references several EPRI directive and non-directive guidelines. These EPRI directives include steam generator examination and tube integrity assessment, both primary and secondary water chemistry, primary-to-secondary leakage, in-situ pressure testing, tube plug assessment, and tube sleeving assessment. The program provides for comprehensive examinations of SG tubes, sleeves, and plugs to identify and repair degraded conditions before the degradation exceeds allowable limits. The secondary side internal components are examined in accordance with the program guidelines given in NEI 97-06.

In response to NRC staff concerns with the scope of the steam generator tube integrity program, the applicant states that the aging effects, and the applicable AMPs given in Section 3.2.6 of the LRA include those addressed by both IN 97-49 and IN 97-88. Also, there is no need for upgrading its steam generator tube integrity program as a result of these information notices, since this program is contained in the ANO-1 TS, and is based on the applicant's references to these subject information notices. By letter to the NRC dated November 2, 2000, the applicant states again that the ANO-1 steam generator tube integrity program is broad enough to adequately address the tube degradation identified and that no changes to the program were needed to address these two information notices.

[*Program Scope*] The steam generator integrity program, in combination with ANO-1 TS 4.18, provides for comprehensive examinations of the steam generator internals, tubing, and associated repairs, such as plugs and sleeves. In a letter to the NRC dated August 24, 2000, the applicant clarifies that the steam generator internals include tube support plates, tie rods, and internal baffles. The applicant states that it performs inspections of its steam generators each refueling outage. The number of steam generator tubes, sleeves, and plugs to be examined each outage meets, at a minimum, its TS requirements. The secondary side internal components are examined in accordance with NEI 97-06 guidelines.

[*Preventive Actions*] The applicant does not discuss preventive actions for the steam generator integrity program, however, the primary and secondary water chemistry monitoring programs are listed as AMPs for the OTSG. The staff concludes that the primary and secondary water chemistry monitoring programs are necessary and effective for reducing the corrosive effects of the operating environment.

[*Parameters Monitored*] The applicant applies nondestructive test techniques, primarily eddy current testing (ECT) to detect cracking of the steam generator tubes, sleeves, and plugs. In its LRA, the applicant follows EPRI's "PWR Steam Generator Examination Guidelines" with respect to ECT. The guidelines provide, among other things, criteria for qualification of personnel, specific techniques, and the associated acquisition and analysis of data (including the procedure, probe selection, analysis protocol, and reporting criteria). Following the EPRI guidelines, the applicant performs the appropriate type of eddy current test technique depending on the region of the steam generator (e.g., top of the tubesheet, freespan, U-bends, sleeves) and the mode of degradation.

The ECT methods used at ANO-1 are in accordance with ASME Section XI and the EPRI, "PWR Steam Generator Examination Guidelines," EPRI NP-6201, Revision 5, Appendix H. The ECT inspections include a full-length bobbin coil examination to identify areas of potential degradation, where it is qualified for use. For other regions of the steam generator where the bobbin coil is not qualified for detection of certain degradation, such as the roll expansion in the tubesheet region and in sleeves, an eddy current technique with a better probability of detection is employed (e.g., rotating pancake coil probe). Tubes with defects as defined in the TS are plugged or sleeved. The current inspection methods, subsequent evaluation procedures, and qualification requirements meet CLB requirements.

[*Detection of Aging Effects*] The applicant uses the ECT data to identify indications on tubes, sleeves, or plugs if any loss of material or cracking exists. Corrective actions will be taken before there is any loss of intended function. Secondary side internals of the steam generator are assessed during each outage, and visual inspections are periodically performed. When necessary, leak testing may be performed to locate a primary-to-secondary leak, and tube pulls can be performed to further evaluate flaw morphology and to assess structural and leakage integrity. The TS defines the maximum interval between inspections. On the basis of operational assessments performed in accordance with NEI 97-06, more frequent inspections than required by the TS may be performed to ensure that the degraded tubes are removed from service before required safety margins are no longer satisfied.

[*Monitoring and Trending*] The applicant monitors tube degradation during each inspection as part of its commitment to NEI 97-06. The condition monitoring program uses inspection results to ensure that steam generator tube integrity has been maintained over the past operating cycle. The post outage evaluation or operational assessment assures that the safety margins will be maintained until the next inspection.

[*Acceptance Criteria*] The ANO-1 TS require ECT, or other equivalent techniques, that are capable of detecting defects with a penetration of 20 percent or more of the minimum allowable, as-manufactured tube wall thickness. Tubes with indications greater than, or equal to, 20 percent of the nominal wall thickness are considered "degraded." Tubes with indications greater than the plugging or repair limits are considered "defective" and must be plugged or repaired. Tubes or sleeves with indications greater than, or equal to, 40 percent of the original nominal tube thickness is considered defective. In general, ECT techniques are not capable of reliably depth sizing indications. Accordingly, tubes or sleeves with eddy current indications are plugged/repaired upon detection.

In its response to staff RAIs, the applicant also indicates that the acceptance criteria are consistent with NEI 97-06 guidelines and ASME code requirements as documented in site procedures. The staff will review these procedures for verification that the acceptance criteria are consistent with the industry guidance and ASME code during the aging management review inspections.

[*Operating Experience*] The B&WOG collects operating experience data on OTSGs and includes it in the plant-to-plant trending report. Steam generator inspection and testing activities in plants with OTSGs have proven effective in identifying indications of aging effects due to cracking and loss of material. Appropriate corrective actions have been taken to mitigate any problems found during their operation. Thus, the staff found that the steam generator integrity program is effective in managing aging effects in steam generators, specifically for tubes, plugs, sleeves, and internals.

In conclusion, the staff found that the combination of AMPs discussed for the OTSGs provides reasonable assurance that the aging effects for steam generators will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

3.3.2.5.3 Conclusions

On the basis of the review described above, the NRC staff finds that the applicant has demonstrated that the effects of aging associated with the OTSGs will be adequately managed such that there is reasonable assurance that this system will perform its intended function in accordance with the CLB during the period of extended operation.

3.3.2.6 Reactor Coolant Pumps

Reactor coolant pumps (RCPs) propel the reactor coolant through the reactor core, RCS piping, and steam generators. The four RCPs installed at ANO-1 are Byron-Jackson pumps and were designed, fabricated, tested, and inspected as Class A components in accordance with ASME Section III, 1968 Edition. The applicant states that the ANO-1 RCPs are not code stamped.

3.3.2.6.1 Technical Information in the Application

In the LRA, Section 3.2.7, the applicant states that the intended function of the RCPs for license renewal is to maintain the RCS pressure boundary. The RCP components that support this intended function and are subject to an AMR include the pump casing, pump cover, integral seal injection heat exchangers, and pressure-retaining bolting. Non-Class 1 piping, instrumentation, and other components attached to the RCPs are addressed in Section 2.3.2 of the LRA. Class 1 piping connected to the pump, including the welded joints is discussed in Section 2.3.1.3 of the LRA.

The upper and lower halves of the pump casing are cast austenitic stainless steel (CASS). The pump cover is also made out of CASS and serves as a housing for the mechanical seal, radial bearing, thermal barrier, and recirculating impeller. The cover is clamped between the carbon steel driver mount and the stainless steel pump casing. Bolting is used to secure the cover to the casing and includes cover-to-case studs and nuts, which are fabricated from low-alloy steel. Bolting used to secure the seal housing and/or seal glands to the cover includes studs and nuts, and are less than 2 inches in diameter and fabricated from low-alloy steel.

Effects of Aging

The materials and operating environment of the RCPs, including bolted closures and connections are similar to the RCS piping reviewed in Section 3.3.2.2 of this SER. The seal water heat exchanger in each of the RCPs is a double coil tube-in-tube design. The inner tube carries primary water and the outer tube carries treated water from the intermediate cooling water system.

The applicant identifies the following aging effects for the components of the four RCPs that are subject to an AMR:

Pump casing:	cracking at welded joints reduction of fracture toughness
Pump cover:	cracking by fatigue reduction of fracture toughness
Seal water heat exchangers:	cracking loss of material of the inner tubes
Bolted closures and connections:	cracking of the bolting material loss of mechanical closure integrity loss of material

Survey of Industry and ANO-1 Operating Experience

To validate the identified aging effects requiring AMPs, the applicant reviewed industry operating experience, that included NRC generic communications, licensee event reports, reports from nuclear power plants including ANO-1, and NRC NUREGs; a site specific review that included a search for instances of RCP aging using the station information management system, condition reporting system, and the licensing database. The applicant found no other aging effects that are applicable to its RCPs, and indicated that its findings are validated by results presented in the PWR RCS license renewal industry report.

Aging Management Programs

In the LRA, Section 3.2.7, the applicant identifies the following AMPs for the RCPs:

- ASME Section XI Inservice Inspection Program, Subsection IWB (Supplemented by Code Case N-481 to manage reduction of fracture toughness of pump casings including an augmented inspection of an RCP [visual inspection of the pressure retaining surfaces, including the cover, prior to entering the period of extended operation]. The flaw tolerance evaluation complies with Code Case N-481 and is a TLAA discussed in Section 4.3.6 of the LRA.)

- primary water chemistry monitoring program

- auxiliary system water chemistry monitoring program
- leakage detection in reactor building
- boric acid corrosion prevention program
- bolting and torquing activities

The applicant concludes that these programs would manage the applicable aging effects so that the intended function(s) of the components of the RCPs would be maintained consistent with the CLB, under all design loading conditions for the period of extended operation.

3.3.2.6.2 Staff Evaluation

In accordance with 10 CFR 54.21(a)(3), the staff reviewed the information included in Sections 3.2.7 (including Table 3.2-1), and Appendix B (Sections 4.3, 4.4, 4.5, 4.6, 4.12) of the ANO-1 LRA, regarding the applicant’s demonstration that the effects of aging will be adequately managed so that the intended function would be maintained consistent with the CLB for the period of extended operation for the RCPs.

3.3.2.6.2.1 Effects of Aging

The applicant states that the applicable aging effects include the following:

- cracking at welded joints and reduction of fracture toughness of the CASS pump casing
- cracking by fatigue and reduction of fracture toughness of the CASS pump cover
- cracking and loss of material of the inner tubes of the seal water heat exchangers
- cracking of the bolting material (low-alloy steel), loss of mechanical closure integrity
- loss of material of the bolted closures and connections

The aging effects of the pressure retaining bolting are the same as those pertinent to the RCS piping bolted closures and connections evaluated in Section 3.2.2 of the LRA.

On the basis of the description of the RCP internal and external environments, materials used in the fabrication of various RCP components, the ANO-1 experience, and the applicant’s survey of industry experience, the NRC staff concluded that the applicant identified the aging effects that are applicable for the RCPs.

3.3.2.6.2.2 Aging Management Programs

The staff evaluation of the applicant’s AMPs focused on the program elements rather than details of specific plant procedures. The staff’s approach to evaluating each program and activity used to manage the applicable aging effects is described in Section 3.3.1 of this SER.

Seal Water Heat Exchangers

The aging effects applicable to seal water heat exchangers are cracking and loss of material of the inner tube, which carries the primary reactor water. The AMPs, identified by the applicant in Table 3.2-1 of the LRA, include primary water chemistry monitoring, Section XI ISI, and leakage detection in the reactor building. Since the inner tube of the heat exchangers maintains the reactor coolant pressure boundary and the outside surface of the tube is exposed to treated water from the intermediate cooling water system, controlling the quality of the treated water is important for managing effects of aging on the inner tube's outside surfaces. Although, Section 3.2.7 of the LRA identifies the auxiliary systems chemistry monitoring program as one of the AMPs applicable to RCPs, the applicant did not include monitoring of treated water as one of the AMPs for the cracking and loss of material aging effects for the inner tube. However, in a letter to the NRC dated September 6, 2000, the applicant agrees to include the auxiliary system chemistry monitoring program as an applicable AMP for the seal water heat exchanger.

As discussed in Section 3.3.2.7.1 of this SER, the following existing AMPs will be continued during the period of extended operations:

- ASME Section XI Inservice Inspection Program, Subsection IWB (as supplemented by Code Case N-481)
- primary water chemistry monitoring program
- auxiliary systems chemistry monitoring program
- leakage detection in reactor building
- boric acid corrosion prevention program
- bolting and torquing activities

The staff's review of several AMPs that apply to the RCPs may be found in Sections 3.3.1 and 3.3.2.4 of this SER. These programs include the Primary Water Chemistry Monitoring Program, ASME Section XI inservice Inspection Program, Subsection IWB, Leakage Detection in Reactor Building, and Bolting and Torquing Activities. The staff's evaluation of the Auxiliary Systems Chemistry Monitoring Program is also discussed in Section 3.3.1 of this SER. The staff's evaluation of the ASME Section XI ISI Program, Subsection IWB, for pumps is discussed below.

In the LRA, Table 3.2-1, the applicant identifies the AMPs used to manage the effects of aging associated with various RCP components.

ASME Section XI ISI Program, Subsection IWB

The effects of thermal embrittlement on the RCP casing and cover fabricated from CASS are managed, both during the current license term, and for the extended period of operation by elements of the plant ISI program. The ISI program includes the applicable requirements of the

ASME Section XI, Subsection IWB. Specifically, cracking of welded joints in the pump casings and cracking of base material in pump casings are managed under the requirements of Examination Categories B-L-1 and B-L-2, respectively. Examination Category B-P manages the pressure retaining boundary of the pump casing by implementing visual VT-2 examinations for system leakage during each refueling outage and during the hydrostatic test conducted at the end of each test interval.

The applicant adopted Code Case N-481, “Alternative Examination Requirements for Cast Austenitic Pump Casings,” in lieu of the volumetric examinations specified in Examination Category B-L-1, Item B12.10. The applicant indicates that a fatigue crack growth calculation was performed, as discussed in Section 4.3.6 of the LRA, and accepted by the NRC during a relief request for successive inspection of the RCP weld flaws. The calculation includes the heatup and cooldown cycles applicable for 60-years of operation and therefore, the flaw growth evaluation for the pump casing is acceptable for the period of extended operation.

Thermal aging embrittlement on CASS components can reduce the fracture toughness of the materials over time. The reduction in fracture toughness is dependent on the type of casting process (static or centrifugal casting), the chemistry of the material, with particular attention to the molybdenum content and those chemical elements that induce the formation of delta ferrite, and the length of time at a temperature conducive to the embrittlement process. Static castings are more susceptible than centrifugal casting; high molybdenum-content castings are more susceptible than low-molybdenum-content castings; and high delta ferrite casting. Operating temperatures on the order of 600^{o}F increase the rate of aging embrittlement relative to operating temperatures on the order of 550^{o}F. The ANO-1 RCP CASS material is statically cast CF8M, as identified in Section 4.8.3 of the LRA.

In the LRA, Section 4.8.3, the applicant discusses the leak-before-break (LBB) analysis results applicable to RCP inlet and exit nozzles using fatigue flaw growth and qualitative assessment of thermal aging of CASS materials. The flaw stability analysis was performed using the lower-bound CASS fracture toughness curves obtained from the ANL report (NUREG/CR-6177). The most limiting material and location used in the analysis were determined to be the base metal material of the 28-inch cold leg pipe. Both the suction and discharge nozzles of the RCP casing are attached to the 28-inch cold leg pipes. The applicant indicates that these nozzles have geometry and loading similar to the limiting location used for the LBB analysis. Thus, the discharge and suction nozzles of the RCP casings were evaluated using lower-bound CASS fracture toughness properties. The applicant concludes that stable crack growth is applicable to the RCP nozzles and that the analysis remains valid for the period of extended operation. Further studies were performed on the aging of the shielded metal arc welding (SMAW) weldment that connects the stainless steel transition pieces to the RCP nozzles, using aged stainless steel weld material given in NUREG/CR-6428. The applicant states that the flaw stability analysis using thermally aged lower-bound CASS material bounds the J-R curve for aged stainless steel weld material. Therefore, reduction of fracture toughness of the RCP nozzles was determined to be acceptable for the period of extended operation.

The applicant states that Code Case N-481 is used to manage reduction of fracture toughness of the pump casings, and includes an augmented inspection of a RCP. This inspection will be performed before entering the period of extended operation, and will include visual inspection of the pressure retaining surfaces, including the cover. The staff found this to be acceptable.

3.3.2.6.3 Conclusions

The staff has reviewed the information in Section 3.2.7 “Reactor Coolant Pumps” and Appendices A and B of the LRA, and additional information provided by the applicant in response to the staff RAIs. The staff concludes that the applicant has demonstrated that the effects of aging associated with the RCPs will be adequately managed such that there is reasonable assurance so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

3.3.2.7 Control Rod Drive Mechanism Pressure Boundary

In the LRA, Section 2.3.1.9, the applicant describes the control rod drive mechanism (CRDM) pressure boundary components. The CRDM pressure boundary includes the CRDM motor tube assemblies, the closure insert assemblies, and the vent assemblies. During normal operation, the CRDM motor tube assemblies are filled with borated reactor coolant at system operating pressure. Thermal barriers in the lower motor tube mechanism, and the CRDM cooling system maintain the temperatures in the housings below system temperature. The CRDM motor tube assemblies are constructed from Alloy 82/182 clad low-alloy steel. The CRDMs are active components, and are not subject to an AMR.

3.3.2.7.1 Technical Information in the Application

In the LRA, Section 3.2.8, of the applicant identifies the CRDM motor tube assembly; the CRDM closure insert and vent assemblies; and RVLMS adapter flange assembly, as being subject to an AMR. The control rod drive tube motor housings are designed to accommodate service loadings (i.e.,levels A through D). Operation under level A and level B service conditions contribute to the normal aging stresses for the control drive tube motor housings. ANO-1 has only been subjected to level A and B service loadings. The applicant states that the control rod drive tube motor housings are fabricated from austenitic and martensitic stainless steels with the exception of the center section of the type B drive which is a low-alloy steel forging clad with Alloy 82/182. The closure insert and vent assemblies, and the RVLMS adapter flange assembly are made from austenitic stainless steel.

Intended Functions

The intended function of the CRDM motor tube assemblies, closure insert assemblies, and vent assemblies is to provide the reactor coolant pressure boundary around the CRDMs.

Effects of Aging

The applicant identifies the following applicable aging effects that require an AMR: cracking at welded joints, and loss of mechanical closure integrity for the control rod drive tube motor housing; loss of mechanical closure integrity for the closure insert and vent assemblies; and cracking at welded joints for the RVLMS adapter flange assembly.

Aging Management Programs

The applicant states that the following existing AMPs are used to manage the effects of aging of the CRDM pressure boundary components during the period of extended operation:

- ASME Section XI Inservice Inspection Program, Subsection IWB
- leakage detection in reactor building
- primary water chemistry monitoring program
- bolting and torquing activities

The applicant concludes that these programs would manage aging effects in such a way that the intended function of the CRDM pressure boundary would be maintained consistent with the CLB, under all design loading conditions for the period of extended operation.

3.3.2.7.2 Staff Evaluation

In accordance with 10 CFR 54.21(a)(3) the staff reviewed the information included in Sections 2.3.1.9, 3.2.8 (Table 3.2-1), and Appendix B Sections 4.3, 4.4, 4.6, and 4.12 of the LRA regarding the applicant's demonstration that the effects of aging will be adequately managed so that the intended functions will be adequately maintained consistent with the CLB for the period of extended operation.

3.3.2.7.2.1 Effects of Aging

The applicant identifies the following applicable aging effects requiring an AMR:

- cracking at welded joints, and loss of mechanical closure integrity for the control rod drive tube motor housing

- loss of mechanical closure integrity for the closure insert and vent assemblies

- cracking at welded joints for the RVLMS adapter flange assembly

In order to validate the completeness of these aging effects, the applicant reviewed industry operating experience including NRC generic communications, licensee event reports from other nuclear power plants, and NRC NUREG reports. From this review, the applicant did not identify any additional aging effects requiring an AMR. These assemblies, during operation, are exposed to reactor coolant whose chemistry is controlled by the primary water chemistry monitoring program. The staff found this list of aging effects to be acceptable for the CRDM pressure boundary components.

3.3.2.7.2.2 Aging Management Programs

In the LRA, Table 3.2-1, the applicant lists the following existing AMPs that will be used to manage the CRDM pressure boundary components for the period of extended operation:

- leakage detection in reactor building
- ASME Section XI Inservice Inspection Program, Subsection IWB

- primary water chemistry monitoring program
- bolting and torquing activities

The staff has reviewed the leakage detection in reactor building program, the primary water chemistry monitoring program, and the bolting and torquing activities program in Section 3.3.2.4 of this SER, and finds them to be adequate to manage the effects of aging of the RCS pressure boundary components so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

In a letter the applicant dated May 5, 2000, the staff asked whether any CRDM components were subject to an AMP to evaluate the potential for boric acid corrosion due to leakage of coolant from the RCS. In its response dated September 6, 2000, the applicant states that CRDM pressure boundary components are made from either stainless steel or nickel-plated low-alloy steel. Thus, the boric acid corrosion program is not applicable since it applies only to ferritic steels.

The staff also expressed concern that localized concentration of contaminants in stagnant regions of the CRDM pressure boundary could cause cracking. In its response, the applicant describes the various ANO-1 programs that are used to control the primary water chemistry to plant standards in order to avoid corrosion problems. The possibility of PWSCC in the CRDM motor tube housings caused by contaminant buildup due to relatively stagnant water conditions was not specifically addressed. However, the applicant pointed out that PWSCC is unlikely in the ANO-1 CRDM because of low oxygen levels, low temperatures, and corrosion-resistant materials. The applicant added that, apart from Palisades where cracks were observed in the vicinity of the “J” welds, which attach the seal housing tube to the autoclave flange (see Palisades inspection report 50-255/99012 (DRP)), no other incident of this type of failure has been observed at any other B&W plants. If such failures do occur, the applicant states that they will be detected by ASME Code Section XI ISI activities. It should be noted that these inspection activities, which are described in Appendix B, Section 4.7 of the LRA (Control Rod Drive Mechanism Nozzle and Other Vessel Closure Penetration Inspection Program) will be conducted at other B&W plants (not at ANO-1). According to BAW-2301, ANO-1 is one of the least susceptible plants with respect to PWSCC, therefore, it was not selected for this inspection activity. The staff found that the applicant’s plan to detect cracking via the aforementioned ISI program is adequate to mitigate the effects of PWSCC in the CRDM pressure boundary components should it occur.

The ASME Section XI IWB program was also reviewed by the staff in relation to the reactor vessel AMP, in Section 3.3.2.4 of this SER, and found to be acceptable. In the LRA, Table 3.2-1, the applicant credits the following Examination Categories for the different CRDM pressure boundary components:

- B-O and B-P for the CRDM motor tube housing
- B-G-2 and B-P for the closure insert and vent assembly and associated bolting
- B-G-2 for RVLMS adapter flange/closure assembly

Table IWB-2500-1 of ASME Code Section XI, shows that the Examination Category B-O applies to welds in control rod housings, and includes volumetric or surface examination. The Examination Category B-P applies to all reactor vessel pressure retaining boundary components, among others, and involves a VT-2 examination. The Examination Category B-G-2 applies to bolts, studs, and nuts for control rod drive housings, among other components, and involves a VT-1 examination.

The staff found that the ASME Code Section XI examinations are acceptable and will be capable of managing the effects of aging in the CRDM pressure boundary components consistent with the CLB for the period of extended operation.

3.3.2.7.3 Conclusions

On the basis of the review described above, the NRC staff finds that the applicant has demonstrated that the effects of aging associated with the control rod drive mechanism pressure boundary will be adequately managed such that there is reasonable assurance that the CRDM pressure boundary will perform its intended function in accordance with the CLB during the period of extended operation.

3.3.2.8 References for Section 3.3.2

1. 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."
2. DG-1047, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," Working Draft, April 21, 2000.
3. "Arkansas Nuclear One - Unit 1, License Renewal Application," January 31, 2000.
4. BAW-2243A, "Demonstration of the Management of Aging Effects for the Reactor Coolant System Piping," The B&W Owners Group Generic License Renewal Program, June 1996.
5. Jimmy D. Vandergrift (Entergy) letter to U.S. Nuclear Regulatory Commission Document Control Desk. "Arkansas Nuclear One–Unit 1 Additional Information in Support of Risk-Informed Inservice Inspection Pilot Application."
6. BAW-2244A, "Demonstration of the Management of Aging Effects for the Pressurizer," The B&WOG Generic License Renewal Program, December 1997.
7. BAW-2251A, "Demonstration of the Management of Aging Effects for the Reactor Vessel," The B&WOG Generic License Renewal Program, June 1996.
8. NRC Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.
9. NRC GL 92–01, Revision 1, Supplement 1, "Reactor Vessel Structural Integrity," May 18, 1995.
10. 10 CFR 50.55a, "Codes and Standards."
11. 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Light Water Nuclear Power Reactors for Normal Operation."
12. 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events."
13. 10 CFR 50.Appendix G, "Fracture Toughness Requirements."
14. 10 CFR 50.Appendix H, "Reactor Vessel Material Surveillance Program Requirements."
15. ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, 1995 Edition through 1996 Addenda.
16. BAW-2248A, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," The B&WOG Generic License Renewal Program, December 1999.

THIS PAGE IS INTENTIONALLY LEFT BLANK

3.3.3 Engineered Safeguards Systems

In the LRA, Section 2.3.2, the applicant describes the SCs of the engineered safeguards systems that are within the scope of license renewal and subject to an AMR. The engineering safeguards systems include the following eight systems: core flood, low pressure injection/decay heat, high pressure injection/makeup and purification, reactor building spray, reactor building cooling and purge, sodium hydroxide, reactor building isolation, and hydrogen control systems. The NRC staff reviewed the information in Section 3.3 of the LRA, and Section 6.0 of the UFSAR to determine whether the applicant provided sufficient information to demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

3.3.3.1 Technical Information in the Application

Core Flood

The safety function of the core flood (CF) system is to provide core cooling after intermediate and large-break loss-of-coolant accidents (LOCAs). The functions of the CF system that are within the scope of license renewal are to maintain RCS pressure boundary, and to prevent or mitigate the consequences of accidents that could potentially exceed offsite exposure limits comparable to 10 CFR Part 100 guidelines. The intended function included in the AMR was to maintain component pressure boundary.

In the LRA, Section 2.3.2.1, the applicant identifies the SCs of the CF system that are within the scope of license renewal and subject to an AMR. Table 3.3-1 of the LRA lists individual components of the system, including the tanks, piping, bolting, external valve parts, tubing, valves and orifices. Stainless steel components are identified as being subject to cracking from the internal environment of borated water, whereas carbon steel components are subject to loss of material and loss of mechanical closure integrity from external exposure to borated water due to system leakage. Cracking of stainless steel is to be managed by existing programs, including the primary and secondary water chemistry monitoring programs. Loss of material from external and internal surfaces is to be managed by existing programs, including the American Society of Mechanical Engineers (ASME) Section XI ISI-IWC pressure tests, boric acid corrosion prevention program, Maintenance Rule program, and the plant operating procedures for monitoring levels in the core flood tanks.

In the LRA, Appendix B, the applicant describes the existing programs for managing the effects of aging. The programs for monitoring primary and secondary chemistry are described in Sections 4.6.1 and 4.6.2 of Appendix B to the LRA. These programs specifically mention the borated water storage tanks and the inclusion of piping of the CF system if such piping is part of the primary coolant system. Appendix B to the LRA, Section 4.5, describes the boric acid corrosion prevention program, and references the applicant's response to Generic Issue 88-05, “Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants.” In the LRA, Section 3.3.3, the applicant states that the scope of the cited boric acid program includes the CF system. Activities under the Maintenance Rule are described in Section 4.13 of Appendix B. Inspections under IWC of ASME Section XI are discussed in Section 4.3.2 of Appendix B, with a specific statement that these inspections apply to the CF system.

In the LRA, Section 3.3.4, the applicant lists the NRC bulletins and information notices applicable to aging effects within the engineered safeguards systems, and states that they performed a site-specific operating history review to validate the accuracy and completeness of the list of aging mechanisms in Table 3.3-1. As a result of this review, the applicant did not identify any aging effects beyond those identified in NRC bulletins and information notices. The staff submitted an RAI for circumstances associated with the replacement of CF piping (and also the LPI/DH and HPI piping). The applicant explained that stress corrosion cracking of this piping occurred due to sodium thiosulphate contamination. In accordance with NRC bulletins and information notices, the applicant replaced the piping, and implemented a water quality monitoring program. The applicant states that these measures, to date, have prevented any recurrence of the problem.

Low Pressure Injection/Decay Heat

The low pressure injection/decay heat (LPI/DH) system is a dual-purpose system, which also operates as the DH system to remove decay heat from the core, and sensible heat from the RCS during the latter stages of cooldown. The LPI system injects borated water into the reactor vessel to cool the core in the event of a LOCA. The functions of the LPI/DH system that are within the scope of license renewal are to maintain the RCS pressure boundary, and to prevent or mitigate the consequences of accidents that could potentially exceed offsite exposure limits comparable to 10 CFR Part 100 guidelines. The intended functions that were included in the AMR for the LPI/DH system are to maintain component pressure boundary, and to transfer heat for heat exchangers. The components of the LPI/DH system that the applicant considers to be within the scope of license renewal and subject to an AMR are described in Section 2.3.2.2 of the LRA.

In the LRA, Table 3.3-2, the applicant lists component groupings of the LPI/DH system that provide a pressure boundary and heat transfer function including the piping, tubing, valves, flow elements, separators, heaters, pumps, bolting, external valve parts, heat exchangers, the BWST, and appurtenances wetted by sump water. Stainless steel components are identified as subject to aging from cracking and loss of material from a borated water internal environment (and, in some cases, exposure to raw water in sump areas of containment). Carbon steel components are subject to aging from loss of material on external surfaces from borated or raw water caused by system leakage. Internal and external loss of material from corrosion are listed as aging effects for the carbon steel BWST from borated water and the external-ambient environment. The applicant indicates that cracking of stainless steel is to be managed by primary water chemistry monitoring program, and by ASME Section XI and augmented inspection programs; loss of material on internal surfaces is to be managed by existing programs of ASME Section XI ISI-IWC pressure tests, boric acid corrosion prevention program, the Maintenance Rule, the reactor building leak rate testing program, and the reactor building sump closeout inspection; and BWST loss of material will be managed by preventive maintenance activities.

The stainless steel materials of heat exchangers in the LPI/DH systems are subject to the same aging effects as is its associated piping, which includes cracking and loss of material. These aging effects are managed to ensure that pressure boundary functions remain intact using the same AMPs as is used to manage aging the associated stainless steel piping. In addition, the heat exchangers are subject to fouling that can affect the heat transfer function. The effects of fouling are managed by programs for heat exchanger monitoring, ASME Section XI ISI-IWC pressure tests, the oil analysis program (for lube oil coolers to ensure that the oil remains free of water or other contaminants), and service water integrity monitoring.

The existing AMPs are described in Appendix B of the LRA. The program for monitoring primary water chemistry is described in Section 4.6.1 of Appendix B in the LRA, and includes the borated water storage tank. Section 4.5 describes the boric acid corrosion prevention program, and references the applicant's response to Generic Issue 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants." In the LRA, Section 3.3.3, the applicant states that the scope of this program includes the LPI/DH system. Activities under the Maintenance Rule are described in Section 4.13 of Appendix B, and the LRA states that the activities under the Maintenance Rule will prevent aging effects not otherwise addressed by other AMPs. Inspections under IWC of ASME Section XI are discussed in Section 4.3.2 of Appendix B where the applicant states that these inspections apply to the LPI/DH system. The oil analysis program is described in Section 4.14 of Appendix B to the LRA. A new activity for heat exchanger monitoring for the aging effects of cracking and loss of material is described in Section 3.3 of Appendix B.

In the LRA, Section 3.3.4, the applicant lists the NRC bulletins and information notices applicable to aging effects within the engineering safeguards systems, and states that a review of site-specific operating experience was performed to validate the accuracy and completeness of the list of aging mechanisms in Table 3.3-2. This review by the applicant did not reveal any aging effects beyond those identified in NRC bulletins and information notices.

High Pressure Injection/Makeup and Purification

The safety function of the high-pressure injection/makeup and purification system (HPI/MUP) system is to provide high-pressure injection of borated water into the RCS during emergency conditions. This system is normally operated as part of the MUP system, which performs various functions in support of the RCS during normal operations. The safety function of the HPI/MUP system that are within the scope of the license renewal based on the requirements of 10 CFR 54.4(a) is to maintain RCS system pressure boundary, and to prevent or mitigate the consequences of an accident. The component-level intended function of the HPI/MUP system is to maintain pressure boundary. The HPI/MUP components that the applicant considers to be within the scope of license renewal and subject to an AMR are described in Section 2.3.2.3 of the LRA.

In the LRA, Table 3.3-3, the applicant lists individual components of the HPI/MUP system that provide a pressure boundary function, including piping, valves, flow elements, separators, filters, pump casings, bolting, external valve parts, tanks, and heat exchangers (lube oil coolers). Stainless steel components are subject to cracking from the internal environment of borated water. Carbon steel and cast iron components are subject to loss of material from external exposure to borated water and lubricating oil due to leakage. Cracking of stainless steel is to be managed by existing programs that monitor primary water chemistry. Loss of material of internal surfaces is to be managed by existing programs of ASME Section XI ISI-IWC pressure tests, the boric acid corrosion prevention program, the Maintenance Rule, the oil analysis program, and the reactor building leak rate testing program.

Some valves in the HPI lube oil system are made of brass or bronze. The applicant states that these materials are subject to loss of material, and are managed by the lube oil program.

The stainless steel materials of heat exchangers (lube oil coolers) in the HPI/MUP systems are subject to the same aging effect as is its associated piping. The aging effect of concern is cracking, which is managed by the oil analysis program, that ensures that the oil remains free of water or contaminants. In addition, the heat exchangers are subject to fouling that can affect its heat transfer function. The effects of fouling are managed by programs for heat exchanger monitoring and oil analysis (in the case of lube oil coolers).

The applicant describes the existing programs for managing aging effects in Appendix B to the LRA. The programs for monitoring primary water chemistry are described in Section 4.6.1 of Appendix B to the LRA, with specific mention of borated water storage tanks. Section 4.5, describes the ANO-1 boric acid corrosion prevention program, and references the applicant's response to Generic Issue 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants." In the LRA, Section 3.3.3, the applicant states that the scope of this program includes the HPI/MUP system. Activities under the Maintenance Rule are described in Section 4.13 of Appendix B. The applicant states that these activities will prevent aging effects not otherwise addressed by other AMPs. Inspections under IWC of ASME Section XI are discussed in Section 4.3.2 of Appendix B, which contains a statement that these inspections apply to the HPI/MUP system. The oil analysis program is described in Section 4.14 of Appendix B to the LRA. A new activity for heat exchanger monitoring is described in Section 3.3 of Appendix B. Section 4.16 of Appendix B describes the existing reactor building leak rate testing program.

In the LRA, Section 3.3.4, the applicant lists the NRC bulletins and information notices applicable to aging effects in the engineering safeguards systems, and states that a review of site-specific operating experience was performed to validate the accuracy and completeness of the aging mechanisms listed in Table 3.3-3.

Reactor Building Spray

The safety function of the reactor building spray (RBS) system is to reduce the reactor building pressure following accidents that pressurize the reactor building. This reduction in pressure reduces the driving force for leakage of radioactive materials from the reactor building following a LOCA. The RBS system also reduces the concentration of fission products in the reactor building atmosphere following a LOCA. Maintaining RBS system pressure boundary integrity is the intended function that needs to be considered during the AMR. In the LRA, Section 2.3.2.4, the applicant identifies the components of the RBS system that are within the scope of license renewal and subject to an AMR.

In the LRA, Table 3.3.4, the applicant lists the individual components of the SCs that are subject to an AMR, including bolting, external valve parts, piping, tubing, valves, separators, pump casings, and heat exchangers (lube oil coolers). The applicant identifies carbon steel components as being subject to loss of material and loss of mechanical closure integrity due to boric acid leakage on external surfaces; stainless steel components as being subject to cracking from internal exposure to borated water; and heat exchangers as being subject to fouling.

The applicant states that cracking of stainless steel components is managed by monitoring primary and secondary water chemistries. Loss of material to the external surfaces of carbon steel components is managed by ASME Section XI ISI-IWC pressure tests, and the boric acid corrosion prevention program; and fouling is managed by oil analyses. The applicant describes each of these AMPs in Appendix B to the LRA.

The applicant describes the programs for monitoring primary and secondary water chemistries in Sections 4.6.1 and 4.6.2 of Appendix B to the LRA. None of the chemistry monitoring programs described in Appendix B are listed as specifically applicable to the RBS system. The staff has, however, identified the borated water storage tank, which is monitored for primary water chemistry, as the source of water for the RBS system. Section 4.5 of Appendix B describes the ANO-1 boric acid corrosion prevention program and references the applicant's response to Generic Issue 88-05, “Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components at PWR Plants.” In the LRA, Section 3.3.3, the applicant states that the scope of this program includes the RBS system. Inspections under IWC of ASME Section XI are discussed in Section 4.3.2 of Appendix B, with a specific statement that these inspections apply to selected components of the RBS system.

In the LRA, Section 3.3.4, the applicant lists the NRC bulletins and information notices applicable to aging effects in the engineered safeguards systems, and states that a review was performed based on site-specific operating experience to validate the accuracy and completeness of the list of aging mechanisms listed in Table 3.3-4. As a result of this review, the applicant did not identify any aging effects beyond those identified in NRC bulletins and information notices.

Reactor Building Cooling and Purge

The safety function of the reactor building cooling and purge (RBCP) system is to reduce the post-accident pressure and temperature in the reactor building, and to provide mixing of the reactor building atmosphere following a loss-of-coolant accident. During normal plant operation the system must maintain the reactor building temperature below the maximum allowed for the equipment, and below the safety analysis initial temperature assumptions. The reactor building purge has no safety function; however, the penetrations in this system must maintain reactor building integrity under accident conditions. The intended function subject to an AMR is to maintain system pressure boundary integrity. For the heat exchangers, heat transfer is an intended passive function subject to an AMR. In the LRA, Section 2.3.2.5, the applicant identifies the components of the RBCP system that it considers to be within the scope of license renewal and subject to an AMR.

In Table 3.3.5 of the LRA, the applicant lists component groups of the RBCP system, including ducts, dampers, piping, valves, fan and cooler housings, and heat exchangers. Carbon steel components are identified as being subject to loss of material and loss of mechanical closure integrity due to wetting of surfaces by condensation from a gas-air environment. External surfaces are subject to loss of material by wetting associated with an external-ambient environment. Components of heat exchangers made of stainless steel and 90/10 material are identified as not being affected by either the internal gas-air or external-ambient environments. The heat exchanger materials are subject to fouling from the internal gas-air environment due to wetting by condensation.

Losses of material and loss of mechanical closure integrity for internal surfaces of carbon steel components are managed by preventive maintenance and other activities under the Maintenance Rule. Aging effects for external surfaces and for fouling of heat exchangers are managed by the Maintenance Rule.

The AMPs for managing aging effects are described in Appendix B to the LRA. Activities under preventive maintenance programs are described in Section 4.15 of Appendix B. Preventive maintenance activities described in this section include those for cleaning and inspection of the reactor building ventilation cooling coils. Section 4.15 also states that there will be new preventative maintenance activities that will address aging effects prior to the end of the initial 40-year license term of ANO-1.

Activities under the Maintenance Rule are described in Section 4.13 Appendix B. These activities consist of periodic system and structural walkdowns for visual examinations for evidence of cracking and loss of material.

In the LRA, Section 3.3.4, the applicant lists the NRC bulletins and information notices applicable to aging effects in the engineered safeguards systems, and states that a review was performed at ANO-1 based on site experience to validate the accuracy and completeness of the list of aging mechanisms identified in Table 3.3-5. This review by the applicant did not identify any aging effects beyond those identified in NRC bulletins and information notices.

Sodium Hydroxide

The safety function of the sodium hydroxide (SH) system is to provide a solution of sodium hydroxide (NaOH) to the ECCS suction headers. The increased pH improves iodine absorption and retention in the water, thereby minimizing the gaseous iodine, and the offsite dose following a LOCA. The sodium hydroxide system is described in the UFSAR, Section 6.2. In the LRA, Section 2.3.2.6, the applicant identifies the components of the SH system that it considers to be within the scope of license renewal and subject to an AMR.

In the LRA, Table 3.3.6, the applicant lists component groups of the SH system subject to an AMR, including piping, valves, bolting, external valve parts, and the SH storage tank. Carbon steel components are identified as being subject to external loss of material and loss of mechanical closure integrity due to leakage of SH solutions onto external surfaces. With the exception of the SH tank, there are no carbon steel surfaces listed as exposed to aging effects from the internal environment. The internal surface of the carbon tank is listed as exposed to loss of material from the internal environment of sodium hydroxide solutions, whereas the external surface of the tank is also exposed to the external-ambient environment with the aging effect being loss of material. Stainless steel components are identified as being subject to loss of material and cracking in a SH environment.

Cracking of stainless steel components is managed by the NaOH tank level monitoring and ASME Section XI IWD pressure tests. The ASME Section XI pressure tests are also listed as the program activity for managing internal loss of material. Loss of material from external surfaces of carbon steel components (bolting and external valve parts) is managed by ASME Section XI IWD pressure tests and the NaOH tank level monitoring program. Wall thinning inspections and the Maintenance Rule are also listed for managing loss of material from the NaOH tank.

The existing programs for managing aging effects are described in Appendix B to the LRA. Inspections under IWD of ASME Section XI are discussed in Section 4.3.3 of Appendix B to the LRA, which contains a statement that these inspections apply to selected components of the SH system. Section 3.7 of Appendix B describe wall thinning inspections of the sodium hydroxide tank. Section 4.21.7 of Appendix B describes the NaOH tank level monitoring program for managing loss of material and loss of mechanical closure integrity.

In the LRA, Section 3.3.4, the applicant lists the NRC bulletins and information notices applicable to aging effects in the engineering safeguards systems, and states that a review of site-specific operating experience was performed to validate the accuracy and completeness of the list of aging mechanisms identified in Table 3.3-6. As a result of this review, the applicant did not identify any aging effects beyond those identified in NRC bulletins and information notices.

Reactor Building Isolation

The safety function of the reactor building isolation (RBI) system is to allow passage of fluids, materials, personnel, electrical signals, and electrical power across the reactor building boundary. In addition, the RBI system seals penetrations that are not required for operation to provide a fission product barrier between the inside of the reactor building, and the outside environment. This capability is also required for the non-safety-related systems that penetrate the reactor building. Therefore, the penetrations have a reactor building isolation function in addition to their system function. The reactor building isolation system is described in the UFSAR, Section 5.2.5. In the LRA, Section 2.3.2.7, the applicant identifies the components of the RBI system that it considers as being within the scope of license renewal and subject to an AMR. The intended safety function that is subject to an AMR is system pressure boundary integrity.

In the LRA, Table 3.3.7, the applicant lists the component groups of the RBI system subject to an AMR, including piping, valves, and bolting. Internal surfaces of carbon steel piping and valves are subject to loss of material from an environment of treated water. Those components that are exposed to a gas-nitrogen environment are listed as having no applicable aging effects, and, therefore, do not require an AMR. External carbon steel surfaces of piping and valves that are exposed to gas-air environments are being subjected to loss of material. Moisture and condensation from the air environment would be the most likely cause for this loss of material. Carbon steel bolting is identified as being subject to external loss of material and loss of mechanical closure integrity due to leakage of borated water.

Stainless steel components are identified as being subject to cracking due to internal exposure to treated water and borated water. Cracking of stainless steel is to be managed by multiple AMPs, including ASME Section XI IWC inspections, augmented inspections, reactor building leak rate testing, primary and secondary chemistry monitoring, and auxiliary systems chemistry monitoring. External surface of stainless steel components is exposed to external-ambient environments, and does not experience aging.

Loss of material of carbon steel components is to be managed by multiple AMPs including ASME Section XI IWC inspections, the boric acid corrosion program, the reactor building leak rate testing program, wall thinning inspections, the Maintenance Rule activities, secondary chemistry monitoring, and auxiliary systems chemistry monitoring.

The AMPs used to manage aging are described in Appendix B to the LRA. Inspections under IWC of ASME Section XI are discussed in Section 4.3.2 of Appendix B of the LRA with a specific statement that these inspections apply to selected components of the RBI system. In the LRA, Section 3.7 of Appendix B, the applicant describes wall thinning inspections of specific RBI system carbon steel components. The boric acid corrosion prevention program is described in Section 4.5 of Appendix B, but has no specific reference to RBI system components. Appendix B, Section 4.3.7, describes augmented inspection activities that will be added for the purpose of license renewal. These augmented inspections include a number of inspections of reactor building penetrations.

The program of reactor building leak testing is described in Section 4.16 of Appendix B, with the integrated leak test being relevant to the integrity of the reactor building penetrations. Activities under the Maintenance Rule are discussed in Section 4.13 of Appendix B. Appendix B also describes the various programs in effect for water chemistry monitoring, including those for primary chemistry (Section 4.6.1), secondary chemistry (Section 4.6.2), and auxiliary systems chemistry (Section 4.6.3).

In the LRA, Section 3.3.4, the applicant lists NRC bulletins and information notices applicable to aging effects in the engineered safeguards systems, and states that a review of site-specific operating experience was performed to validate the accuracy and completeness of the list of aging mechanisms in Table 3.3-7. As a result of this review, the applicant did not identify any aging effects beyond those identified in NRC bulletins and information notices.

Hydrogen Control

The safety function of the hydrogen control (HC) system is to provide a direct reading of the hydrogen concentration in the reactor building using the hydrogen analyzer system and to reduce the hydrogen concentration following a LOCA using the hydrogen recombiners. The HC system is described in the UFSAR, Section 6.6. In the LRA, Section 2.3.2.8, the applicant identifies the components of the HC system that it considers as being within the scope of license renewal and subject to an AMR.

In the LRA, Table 3.3.8, the applicant lists the component groups that are subject to an AMR, including piping, valves, recombiners, heat exchangers, and sample stations. The internal surfaces are exposed to a gas-air environment for which there are no applicable aging effects for either ferritic or stainless steel components. Carbon steel components are identified as being subject to loss of material when exposed to external-ambient environments. Stainless steel and Incoloy-800 components are also exposed to a gas air environment or external-ambient environment for which there are no applicable aging effects. The external surface of the heat exchangers tubes are identified as being subject to fouling.

Loss of material from carbon steel components is managed by Maintenance Rule activities. The existing Maintenance Rule program for managing the applicable aging effects of the HC system is described in Section 4.13 of Appendix B to the LRA. Fouling is managed by preventative maintenance activities which is described in Section 4.15 of Appendix B.

In the LRA, Section 3.3.4, the applicant lists the NRC bulletins and information notices applicable to aging effects in the engineering safeguards systems, and states that a review of site-specific operating experience was performed to validate the accuracy and completeness of the list of aging mechanisms in Table 3.3-8. This review did not identify any aging effects beyond those identified in the cited NRC bulletins and information notices.

3.3.3.2 Staff Evaluation

The staff's evaluation consisted of a review of the effects of aging for the SCs of the eight engineered safeguards systems, and of the AMPs identified by the applicant to address these aging effects for the period of extended operations. The objective was to determine if there is a reasonable assurance that the applicant has demonstrated that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation in accordance with 10 CFR 54.21(a)(3). The review was, in part, based on NRC aging management research, and the results from a previous AMR for a plant of similar B&W design. Consistent with the SRP for license renewal, Section 3.0, the proposed AMPs were reviewed from the standpoint of the 10 elements of an acceptable AMP. They were discussed in the ANO-1 LRA and further explained in the applicant's July 31, 2000, response.

3.3.3.2.1 Effects of Aging

The staff reviewed the information in the LRA for the eight engineered safeguards systems to evaluate if the applicant has adequately identified the effects of aging for the range of materials and environments that apply to these systems. The NRC staff's evaluation of the applicable aging effects is discussed below on a system-by-system basis.

Core Flood

The materials used in the core flood system are listed as stainless steel and carbon steel. Internal surfaces of carbon steel components are protected from corrosive effects of boric acid by stainless steel clad. External surfaces of the carbon steel components are subject to loss of material caused by boric acid leakage onto the exterior surface of these components. The applicable aging effect for stainless steel components is cracking in the presence of boric acid. On the basis of the descriptions of the CF system in terms of the relevant materials and the internal and external operating environments, the staff determined that the applicant has included the effects of aging consistent with published literature and industry experience.

In a letter to the applicant dated June 1, 2000, the NRC staff requested additional information relating to the effects of boric acid on stress corrosion cracking, and the extent of contaminants such as chlorides and sulfates as contributing causes to cracking and loss of material in the CF system (and also the LPI/DH and HPI/MUP systems). The applicant noted that cracking and loss of material are identified as applicable aging effects which include cracking from stress corrosion and exposure to contaminants. The applicant also stated that water chemistry programs and activities that are used to manage cracking will continue during the period of extended operation. The staff found this response acceptable.

The applicant states that they performed a review of operating experience for the CF system that includes a review of industry operating experience using INPO's Nuclear Plant Reliability Data System (NPRDS), a review of generic communication including NRC bulletins and information notices, and a site-specific operating history review, and did not identify any aging effects other than cracking and loss of material in the ANO-1 CF system. The staff is not aware of any industry operating experience that would indicate any aging effects beyond those listed in Table 3.3-1 of the LRA.

On the basis of the descriptions of the CF system in terms of the relevant materials, and the internal and external operating environments, the staff determined that the applicant has identified the applicable aging effects consistent with published literature and industry experience.

Low Pressure Injection/Decay Heat

The materials used in the low pressure injection/decay heat system are listed as stainless steel and carbon steel. Carbon steel is limited to the borated water storage tank (BWST), bolting, certain parts of pumps and valves, and appurtenances located in the containment sump. External surfaces of the carbon steel components are subject to loss of material only if leakage of boric acid exposes the surfaces to corrosive attack. The aging effect for stainless steel components is cracking in the presence of boric acid. On the basis of the descriptions of the LPI/DH system in terms of the relevant materials and the internal and external operating environments, the staff determined that the applicant has identified aging effects consistent with published literature and industry experience.

The applicant states that they performed a review of operating experience for the LPI/DH system that includes a review of industry operating experience using NPRDS, a review of generic communication including NRC bulletins and information notices, and a site-specific operating history review, and did not identify any occurrences of cracking or loss of material in ANO-1 LPI/DH system. The staff is not aware of any operating experience that would indicate any aging effects beyond those listed in Table 3.3-2 of the LRA.

In a letter to the applicant dated June 1, 2000, the staff requested additional information regarding aging effects of the borated water storage tank carbon steel components that are coated to protect the inner surface of the tank from loss of material due to corrosion. The applicant states that the coating material is Plastite. The applicant also informed the staff that the preventative maintenance program is used to visually inspecting the Plastite coating on the interior of the BWST during each refueling outage, and the inspection activities will be enhanced to specifically require inspection for coating integrity and no loss of material of the BWST carbon steel surface. The staff found the applicant's response acceptable.

The staff determined that the LRA failed to identify loss of material as an applicable aging effect for stainless steel piping, fittings and flow orifices in the LPI/DH systems as well as the core flood, HPI/makeup and purification, reactor building spray and reactor building isolation for borated water environments. In a letter to NRC dated September 12, 2000, the applicant acknowledged errors in compiling results of the AMR, and verified that loss of material was included in its AMR, and is managed by the primary chemistry monitoring. With these changes, the staff determined that the oversight in the LRA was adequately addressed.

The staff was concerned that external surfaces of the LPI/DH piping could be exposed to a range of external environments other than the controlled air environments identified in the LRA. The staff requested additional information to address piping buried in concrete or exposed to external weather environments such as acid rain, ground water, heat and humidity, and to determine if cracking and/or loss of material should have been considered as applicable aging effects. In a letter to the NRC dated September 12, 2000, the applicant states that the BWST and associated carbon steel piping are in contact with concrete and oiled sand environments, however, these environments will not result in any additional loss of material and cracking that are not already addressed in the LRA. The staff reviewed this information and determined that the LRA adequately identified the aging effects for the external environments.

The staff also requested additional information relating to the intended function of the heat exchangers (lube oil coolers) in the LPI/DH system, as well as the HPI/MUP system, and the reactor building spray system. The applicant acknowledged errors in compiling results of the AMR and verified that the AMR considered both the pressure boundary and heat transfer function of the heat exchangers. In a letter to the NRC dated September 12, 2000, the applicant states that the aging effects of cracking and loss of material along with the identified AMPs were found applicable and implemented for the heat transfer function. The staff found this response acceptable.

On the basis of the descriptions of the LPI/DH system in terms of the relevant materials, and the internal and external operating environments, the staff determined that the applicant has identified the applicable aging effects consistent with published literature and industry experience.

High Pressure Injection/Makeup and Purification

The materials used in the HPI/MUP are primarily stainless steel and carbon steel. Brass and bronze are used in valves and cast iron is used as part of valves and filters. Carbon steel is used in bolting, and certain parts of pumps, valves, filters and tanks. The internal surfaces are exposed to lube oil, and are not subject to aging. External surfaces of the carbon steel components are subject to loss of material from boric acid leakage. The applicable aging effect for stainless steel components is cracking in a boric acid internal environment. On the basis of the descriptions of the HPI/MUP system in terms of the relevant materials, and the internal and external operating environments, the staff determined that the applicant has identified the applicable aging effects consistent with published literature and industry experience.

The applicant states that they performed a review of operating experience for the HPI/MU system that includes a review of industry operating experience using NPRDS, a review of generic communication including NRC bulletins and information notices, and a site-specific operating history review, and did not identify any aging effects other than cracking and loss of material in ANO-1 HPI/MU system. The staff is not aware of any operating experience that would indicate any aging effects beyond those listed in Table 3.3-3 of the LRA.

In a letter to the applicant dated June 1, 2000, staff requested additional information on the potential of brass/bronze valves in the HPI/MUP systems to be exposed to environments (such as lube oil leakage and boric acid leakage) other than external-ambient that could result in loss of material. In its response to the NRC dated September 12, 2000, the applicant verifies that these components are only exposed to an external-ambient environment, and provides additional justification on the basis of a review of industry and site operating experience, which showed a lack of incidents of exposure to other environments. The staff found the applicant's response acceptable.

The staff also requested additional information regarding the exposure of HPI gear drive reservoirs and associated pumps to potentially corrosive environments. This was determined not to be a concern because normal plant operations make it highly unlikely to experience a corrosive environment. The staff found the applicant's response acceptable.

On the basis of the descriptions of the HPI/MU system in terms of the relevant materials, and the internal and external operating environments, the staff determined that the applicant has identified the applicable aging effects consistent with published literature and industry experience.

Reactor Building Spray

The materials used in the reactor building spray system are listed as stainless steel and carbon steel. External surfaces of the carbon steel components (bolting and valve parts) of this system are subject to loss of material only if exposed to boric acid leakage. Internal surfaces are of stainless steel and are not subject to loss of material from exposure to boric acid.

The applicable aging effect for stainless steel components (such as piping) exposed to boric acid is cracking. The staff noted that much of the piping in the spray system would be exposed to borated water only in the event that the system is activated to mitigate a severe accident. As such, aging effects of cracking would be limited to those parts of the system that are exposed to borated water during periodic testing of the RBS pumps. These internal surfaces will be wetted during the brief periods of testing, and will continue to be wetted by any undrained water left in the system after testing. The staff therefore concurs with the applicant that cracking of these stainless steel components requires aging management. In a letter to the applicant dated June 1, 2000, the staff also requested additional information regarding the potential loss of material associated with alternate wetting and drying of the stainless steel material. In its response dated September 12, 2000, the applicant verifies that it did identify loss of material as an applicable aging effect in its AMR of the RBS system, and determined that the loss of material will be managed by the same AMPs used to manage cracking. The applicant states that loss of material was not added to Table 3.3-4 because of administrative error. The staff found this response acceptable.

Stainless steel materials are also used for heat exchangers and are subject to fouling. Fouling would impact the heat transfer function of the heat exchangers.

The applicant states that they performed a review of operating experience for the RBS system that includes a review of industry operating experience using NPRDS, a review of generic communication including NRC bulletins and information notices, and a site-specific operating history review, and did not identify aging effects other than cracking, loss of material, and heat exchanger fouling in the RBS system. The staff is not aware of any operating experience that would indicate any aging effects beyond those listed in Table 3.3-4 of the LRA.

In a letter to the applicant dated June 1, 2000, the staff requested additional information concerning alternate wetting of RBS piping surfaces with boric acid solutions and subsequent drying (conditions during surveillance testing) were the subject of an RAI. Such conditions could concentrate boric acid and create environments conducive to loss of material and cracking. The applicant states that the chemistry control programs were identified in the LRA as the method to manage the aging effect of cracking, but noted that Table 3.3-4 should be modified to list these same programs for managing the aging effect of loss of material. The staff determined that the AMR as modified will be adequate to address the aging effects of concern.

The staff also requested additional justification for not including loss of material as an applicable aging effect for bolting in an external-ambient environment in the RBS and ESF systems. The applicant's response referred to a detailed discussion of normal plant environments in Section 9.3.2 of Appendix C of the LRA, and by indicating that loss of material for such bolting is managed for potential exposure to borated water leakage. On the basis of this additional information, the staff determined that the applicant has addressed loss of material for bolting in the RBS and ESF systems from boric acid leakage.

On the basis of the descriptions of the RBS system in terms of the relevant materials, and the internal and external operating environments, the staff determined that the applicant has identified the applicable aging effects consistent with published literature and industry experience.

Reactor Building Cooling and Purge

The materials used in the reactor building cooling and purge system are listed as stainless steel, carbon steel, and 90/10 CuNi alloy. External surfaces of these materials are exposed to the external-ambient environment of the reactor building. Internal surfaces are exposed to gas-air environments along with wetting associated with condensation. The staff determined that loss of material from exposure to boric acid leakage is an applicable aging effect for components of the RBCP system.

In the LRA, Table 3.3-5, the applicant does not identify any aging effects for stainless steel and 90/10 CuNi component that are exposed to the gas-air or the external-ambient environments. Given the corrosion resistance of these materials, the staff found this conclusion to be acceptable.

Carbon steel components are listed in Table 3.3-5 as being subject to the aging effect of loss of material, and AMPs are identified. Because the surfaces of these components are exposed to the external-ambient conditions of the reactor building, and a gas-air environment (e.g., inner surface of ducting), the staff concurs with the applicant that wetting of surfaces by condensation necessitates programs to manage the effects of aging associated with loss of material.

The applicant states that they performed a review of operating experience for the RBCP system that includes a review of industry operating experience using NPRDS, a review of generic communication including NRC bulletins and information notices, and a site-specific operating history review, and did not identify aging effects other than loss of material and heat exchanger fouling in ANO-1 RBCP system. The staff is not aware of any operating experience that would indicate any aging effects beyond those listed in Table 3.3-5 of the LRA.

In a letter to the applicant dated June 1, 2000, the staff requested additional information relating to loss of material of copper nickel 90/10 alloy (tubes) exposed to gas air environments. The concern was condensation due to contact with moist air. In its response to the NRC dated September 12, 2000, the applicant states that the materials and components of concern are heat exchanger tubes exposed to air that has been filtered and cooled by pre-filters and chilled water coils, thereby removing particulate and moisture. A review of operating experience by the applicant did not reveal degradation of the CuNi components in the environment of concern. The staff determined that there is an adequate basis for not including loss of material as an aging effect for the 90/10 components.

On the basis of the descriptions of the RBCP system in terms of the relevant materials and the internal and external operating environments, the staff found that the applicant has identified the applicable aging effects consistent with published literature and industry experience.

Sodium Hydroxide

In the LRA, Table 3.3-6, the applicant lists the materials used in the sodium hydroxide system are stainless steel and carbon steel. The carbon steel components (bolting, external valve parts, and the SH tank) of this system are subject to loss of material from internal exposure to sodium hydroxide, and loss of material from exposure to an external-ambient environment or from leakage of sodium hydroxide solutions. The aging effect for stainless steel components is cracking in the presence of sodium hydroxide solutions containing impurities.

The applicant states that they performed a review of operating experience for the SH system that includes a review of industry operating experience using NPRDS, a review of generic communication including NRC bulletins and information notices, and a site-specific operating history review, and did not identify aging effects other than cracking and loss of material in the SH system. The staff is not aware of any operating experience that would indicate any aging effects beyond those listed in Table 3.3-6 of the LRA.

In a letter to the applicant dated June 1, 2000, the staff requested additional information relating to the levels of impurities in the sodium hydroxide, and the potential for cracking of the stainless steel and carbon steel components of the SH system. Additional information was requested on levels of chlorides and other impurities, comparisons of ANO-1 levels to industry levels, and information on temperatures in the sodium hydroxide storage tank. In its response dated September 12, 2000, and October 3, 2000, the applicant states that operator logs show maximum temperatures of 95EF or lower, which is below the threshold temperatures for stress corrosion cracking in stainless and carbon steels. On the basis of this information the staff agrees that cracking is not a potential aging effect for the materials of the SH system.

On the basis of the descriptions of the SH system in terms of the relevant materials and the internal and external operating environments, the staff found that the applicant has identified the applicable aging effects consistent with published literature and industry experience.

Reactor Building Isolation

In the LRA, Table 3.3-7, the applicant lists the materials used in the RBI system as carbon steel and stainless steel. Surfaces of the carbon steel components (piping, valves and bolting) of this system are listed as being subject to loss of material from internal contact with treated water and external boric acid leakage. External surfaces of carbon steel components can also experience loss of material from moisture and condensation when exposed to ambient gas-air environments. The aging effect for internal surfaces of stainless steel components (piping and valves) is identified as cracking in the presence of treated and borated water environments containing small concentrations of impurities such as chlorides. No aging effects for external surfaces of stainless steel components exposed to external-ambient environment are identified.

The applicant states that they performed a review of operating experience for the RBI system that includes a review of industry operating experience using NPRDS, a review of generic communication including NRC bulletins and information notices, and a site-specific operating history review, and did not identify aging effects other than cracking and loss of material in the RBI system. The staff is not aware of any operating experience that would indicate any aging effects beyond those listed in Table 3.3-7 of the LRA.

On the basis of the descriptions of the RBI system in terms of the relevant materials and the internal and external operating environments, the staff found that the applicant has identified the applicable aging effects consistent with published literature and industry experience.

Hydrogen Control

In the LRA, Table 3.3-8, the applicant lists the materials used in the HC system as carbon steel, stainless steel, and Incoloy-800. External surfaces of some components of this system are exposed to the external-ambient environment of the reactor building. The aging effect for carbon steel is loss of material to the external surfaces. No aging effects were identified for the external surface of Incoloy-800 or stainless steels components. The applicant does not identify any applicable aging effect associated with internal gas-air environment for any of the HC system components.

In Table 3.3-8, the applicant also identifies external fouling of stainless steel heat exchanger tubes as an applicable aging effect that can impact the heat transfer function of these components. The environment of the outside surfaces of tubes is identified as gas-air. The inside surfaces are also exposed to a gas-air environment that is not identified as having any applicable aging effects.

The applicant states that they performed a review of operating experience for the HC system that includes a review of industry operating experience using NPRDS, a review of generic communication including NRC bulletins and information notices, and a site-specific operating history review, and did not identify aging effects other than cracking, loss of material, and heat exchanger fouling in the HC system. The staff is not aware of any operating experience that would indicate any aging effects beyond those listed in Table 3.3-8 of the LRA.

On the basis of the descriptions of the HC system in terms of the relevant materials and the internal and external operating environments, the staff found that the applicant has identified the applicable aging effects consistent with published literature and industry experience.

3.3.3.2.2 Aging Management Programs

The staff reviewed the applicant's AMP for the SCs of the engineered safeguards systems. The evaluation of the AMPs focused on the program elements rather than on plant specific procedures for implementing these programs. The objective was to determine if the proposed ANO-1 AMPs will manage the effects of aging so that the intended functions are maintained consistent with the CLB for the period of extended operation. The staff evaluated the following ten elements:

- scope of the program
- preventative measures
- parameters monitored or inspected
- detection of aging effects
- monitoring and trending
- acceptance criteria
- corrective actions
- confirmation processes
- administrative controls
- operating experience

The applicant indicates that corrective actions, the confirmation process, and administrative controls for license renewal are implemented in accordance with site-controlled corrective action programs pursuant to 10 CFR Part 50, Appendix B, and covers all SCs that are subject to an AMR. The staff's evaluation of the applicant's corrective action program is discussed separately in Section 3.3.1.2 of this SER. Thus, these three elements are not discussed further in this section.

In most cases the AMPs for the engineered safeguards systems were programs common to many systems at ANO-1. These common programs are reviewed in Section 3.3.1 of this SER. The following paragraphs will focus on issues unique to the specific systems addressed below, and will summarize the staff's evaluation of the engineered safeguards AMPs.

The NRC staff requested additional information common to multiple engineered safeguards AMPs. These RAIs are discussed in the following paragraphs.

In the LRA, Section 3.3.3, the applicant states that ASME Section XI is applicable to all the engineered safeguards systems with the exception of the RBCP system and the HC system, whereas Tables 3.3-1, 2, 3 and 4 do not list Section XI as an AMP for certain components (e.g.,BWST and LPI/DHR piping). In a letter to the applicant dated June 1, 2000, the NRC staff requested a justification for the exclusion of Section XI examinations from the table in question. In its response to the NRC dated September 12, 2000, the applicant states that all components of a system coming under Section XI are not inspected. Furthermore, in those cases, the applicant states that they identify other aging management activities rather than take credit for Section XI inspections. The staff found this response acceptable.

In addition, the staff requested additional information on the levels of chlorides and sulfates in engineered safeguards systems with particular concerns for stagnant portions of the systems, and to determine if stated limits on impurities are consistently met in these parts of the systems. In its initial response to the NRC dated June 1, 2000, the applicant states that EPRI guidelines and industry standards are currently followed at ANO-1, and will continue to be followed during the period of extended operations. The applicant also provided additional clarification in a letter to NRC dated October 3, 2000. This response included quantitative values for chloride (3.1 ppm) and sulfur (2 ppm) in the sodium hydroxide. These values of impurities are not considered aggressive to carbon steel. The applicant also indicates that the carbon steel used in the SH system have relatively low tensile strengths (30 to 50 ksi). Industry data does not indicate a problem with stress corrosion cracking in low-strength carbon steels. The staff found the applicant's response acceptable.

In the LRA, Section 3.3.3, the applicant identifies the heat exchanger monitoring as an AMP for several of the engineered safeguards systems (RBCP, LPI/DHR, and HPI/MUP), whereas Tables 3.3-2, 3.3-3, and 3.3-5 do not list this monitoring program for a number of specific components. The NRC staff requested additional clarification on the applicability of heat exchanger monitoring program, and in its response, the applicant clarifies which programs are actually credited for managing aging. For example, the service water integrity program is used to manage fouling of the DH coolers, rather than the heat exchanger monitoring program. The staff found the applicant's response acceptable.

In the LRA, Section 4.4 of Appendix B, "Bolt and Torquing Activities," the applicant addresses aging management of carbon steel bolting, and indicates that this program applies to all engineered safeguards systems under the topic of bolting and torquing. The staff noted that this program was not included in Tables 3.3-1, -2, -3, -4, -5 and -6, and requested additional clarification or justification for this apparent omission. In its response, the applicant states that the bolting and torquing activities are credited in the LRA only for the RCS, whereas the LRA takes credit for several other activities as described in Appendix B to the LRA for managing aging effects in the bolting of the engineered safeguards systems. The staff found the applicant's response acceptable.

Core Flood System

In the LRA, Table 3.3-1, the applicant lists the following AMPs for the CF system:

- core flood tank monitoring program
- ASME Section XI IWC (inservice inspections and pressure tests)
- boric acid corrosion prevention program
- Maintenance Rule
- primary chemistry monitoring program
- secondary chemistry monitoring program

The applicant states that the list of AMPs will manage the applicable aging effects so that the intended function of the components of the CF system will be maintained consistent with the CLB under all design loading conditions for the period of extended operation. The staff reviewed the cited programs for their appropriateness and potential effectiveness for managing the aging effects of concern.

The applicant uses the core flood tank monitoring program for managing the loss of material and loss of mechanical closure integrity for bolting and for external corrosion of the core flood tank caused by component/system leakage. The following is the staff's review the core flood tank monitoring program as described in Appendix B of the LRA:

[*Program Scope*] Because the applicant includes carbon steel components in both core flood tanks in the program, the staff found the scope of the program adequate.

[*Preventive/Mitigative Actions*] There is no preventative or mitigative action associated with this program nor does the staff see a need for such action.

[*Parameters Monitored*] The applicant states that core flood tank levels and pressures are monitored per ANO-1 operating procedures once per shift during plant operation, and that alarms activate if pressure or level moves outside the acceptable range. A decrease in the levels or pressures could possibly indicate leakage and exposure of components to boric acid corrosion. The core flood tank monitoring program is expected to provide reasonable assurance that significant boric acid leakage will be detected in a timely manner.

[*Detection of Aging Effects*] The core flood tank monitoring program can be used to detect the potential exposure of other CF components to boric acid corrosion. Indications of leakage will then result in other inspections to establish the source of leakage, and the extent of aging associated with the leakage. The staff found that the core flood monitoring program can be used as an effective supplement to the boric acid corrosion prevention program for the timely detection of loss of material.

[*Monitoring and Trending*] The applicant records the levels and pressures in the core flood tanks once per shift. In addition, the level is continuously monitored by alarms as part of the control room instrumentation. No other monitoring or trending activities are associated with this program. The staff found the monitoring activities to be acceptable.

[*Acceptance Criteria*] The applicant states that any unacceptable drop in core flood tank pressures or level is defined by existing site procedures.

[*Operating Experience*] The applicant states that operating experience with the core flood tank monitoring program has proven successful in identifying small amounts of leakage. The staff found that this activity provides reasonable assurance that boric acid leakage and the associated effect of material loss will be detected on a timely basis.

The staff found that the core flood tank monitoring program provides reasonable assurance that, in combination with the boric acid corrosion prevention program, the applicable aging effects can be managed for the period of extended operation.

The applicant uses ASME Section XI IWC inspection activities (inservice inspections and pressure tests) as a means to manage the loss of material, and loss of mechanical closure integrity. The staff determined that this program includes surface and volumetric inspections and leakage testing for managing loss of mechanical closure integrity. The program focuses on monitoring for boric acid leakage, and will be used to complement the boric acid corrosion prevention program as a means to manage loss of material. The ASME inspections will also serve to manage cracking of stainless and carbon steel components. ASME Section XI IWC inspection activities apply to several systems that are within the scope of license renewal, and are evaluated in Section 3.3.1 of this SER.

The applicant uses the boric acid corrosion prevention program as a means to manage the loss of material from the external surface of the core flood tank, from exposure to boric acid leakage. A description of the boric acid corrosion prevention program is given in Section 4.5 of Appendix B of the LRA. ANO-1 site-specific operating experience is used to demonstrate the effectiveness of the program for managing loss of material. The boric acid corrosion prevention program is a common AMP that is used by more than one system at ANO-1, and is reviewed in Section 3.3.1 of this SER.

The applicant uses the Maintenance Rule activities as a means to manage the loss of material and loss of mechanical closure integrity. A description of this program is given in Section 4.13 of Appendix B to the LRA. The staff considers this monitoring program to be a common AMP used by more than one system at ANO-1. The Maintenance Rule activities are also AMPs that are common to more than one system at ANO-1, and are reviewed in Section 3.2 of this SER.

The applicant uses the primary and secondary chemistry monitoring programs as means to manage cracking of piping, tubes, valves, orifices and the cladding of the core flood tanks. A description of these programs is given in Sections 4.6.1 and 4.6.2 of Appendix B to the LRA. The staff considers these programs to be common AMPs at ANO-1, and are evaluated in Section 3.3.1 of this SER.

On the basis of the review described above, the staff found that the AMPs identified by the applicant for the CF system will manage the applicable aging effects so that there is reasonable assurance that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

Low Pressure Injection/Decay Heat

In the LRA, Table 3.3-2, the applicant lists the following AMPs:

- ASME Section XI IWC pressure tests
- boric acid corrosion prevention program
- Maintenance Rule program
- primary chemistry monitoring program
- ASME Section XI ISI-IWC Inspections
- augmented inspections
- heat exchanger monitoring program
- service water integrity program
- preventative maintenance
- oil analysis program
- reactor building leak rate testing program
- reactor building sump closeout inspection

The applicant states that the list of AMPs will manage the effects of aging so that the intended function(s) of the components of the LPI/DH system will be maintained consistent with the CLB, under all design loading conditions for the period of extended operation. The staff evaluated these programs for their appropriateness and effectiveness for managing the aging effects of concern.

The boric acid corrosion prevention, Maintenance Rule, and primary and secondary chemistry monitoring programs were evaluated by the staff as part of CF system review. Because the LPI/DH system has the same materials and environments as the CF system, the previous staff reviews of these AMPs apply to both systems and need not be repeated. Specifically, for stainless steel components, these four programs will manage the loss of material, cracking, and heat exchanger fouling; and for carbon steel components, these programs will manage loss of material and loss of mechanical closure integrity.

The applicant uses ASME Section XI IWC program inspections as a means to manage external loss of material and loss of mechanical closure integrity for carbon steel bolting and external valve parts. The applicant also uses the program for managing internal and external cracking of stainless and carbon steel piping, valves, and appurtenances (wetted by sump water) exposed to borated and raw water environments. Section XI IWC inspections include nondestructive examinations by volumetric, surface, and visual methods performed on a periodic basis. A description of the program is given in Section 4.3.2 of Appendix B to the LRA. The staff considers the ASME Section XI IWC inspection program to be common to several systems at ANO-1. AMPs that are common to more than one system are reviewed in Section 3.3.1 of this SER.

The applicant uses the heat exchanger monitoring program as a means to manage loss of material from the inner surface of the decay heat cooler tubes exposed to borated water. A description of the program is given in Section 3.3 of Appendix B in the LRA. The staff reviewed the program description, and has determined that the program is designed to address structural integrity concerns by use of visual and nondestructive examination methods. Such methods are considered by the staff to be appropriate for managing the aging effect of loss of material. However, in a letter to the applicant dated June 1, 2000, the staff requested additional information regarding the use of the heat exchanger monitoring program for managing fouling. In its response, the applicant states that Table 3.2-2 was incorrect in crediting the heat exchanger monitoring program, and ASME Section XI ISI-IWC for managing fouling of the decay heat coolers. Rather the service water program, which includes heat transfer testing, is the appropriate program to credit for managing fouling of the decay heat coolers. The staff agrees with the applicant's response. The heat exchanger monitoring program is described in Section 3.3 of Appendix B to the LRA. The staff considered this program to be a common AMP, and evaluated the heat exchanger monitoring program in Section 3.3.1 of this SER.

The applicant uses the service water integrity program as a means to manage fouling (and its impact on the heat transfer function) of the DH coolers. A discussion of this program as implemented at ANO-1 is given in the LRA, Section 4.19 of Appendix B. The following is the staff's evaluation of the heat exchanger monitoring program:

[*Program Scope*] The service water integrity program includes a range of activities to manage loss of material, cracking, loss of flow capacity, and loss of heat transfer capacity of components that are part of, or supported by the service water system. This includes all heat exchangers that are supplied with cooling water by the service water system. The staff found the scope of the service water integrity program to be adequate to manage aging of the DH coolers in the LPI/DH system.

[*Preventive/Mitigative Actions*] System cleaning by mechanical and chemical methods is performed as necessary in accordance with the recommendations of GL 89-13 to ensure that flow rates are maintained above required values. Chemistry control consistent with GL 89-13 provides treatment of the service water system with biocides to minimize macro biological fouling. The staff found these measures to be acceptable to minimize the adverse effects of fouling for heat exchangers.

[*Parameters Monitored*] The applicant includes tests for flow rates and heat transfer rates to monitor the performance and possible functional degradation of heat exchangers. Both types of tests will detect decreased flow and heat transfer rates so that cleaning is initiated to remove fouling before such fouling reaches unacceptable levels. The service water integrity program provides reasonable assurance that there is no significant loss of heat transfer function.

[*Detection of Aging Effects*] The service water integrity program detects the aging effects of fouling by monitoring heat exchangers at regular intervals for evidence of decreased flow rates and degraded heat transfer rates. The staff found that these activities will be adequate to detect fouling that can degrade the heat transfer function of the monitored components.

[*Monitoring and Trending*] The applicant performs the activities of the service water integrity program during every refueling outage in accordance with commitments made in response to GL 89-13. The frequencies of specific activities are adjusted based on results of the testing and inspections. The staff found the activities associated with the applicant's commitments in response to GL 89-13 will be adequate to monitor and trend the effects of aging in components of the service water system.

[*Acceptance Criteria*] Acceptance criteria have been developed by the applicant and are covered by site specific procedures consistent with the guidance in GL 89-13.

[*Operating Experience*] The applicant states that the service water integrity program has been in effect at ANO-1 since 1980 and has served to ensure the operability of the system. Experience has shown that activities of testing, monitoring, chemistry controls, and preventative maintenance are effective in managing the effects of aging within the system. The staff found that these activities provide reasonable assurance that the aging effects, including fouling of heat exchangers, will be effectively managed during the period of extended operation.

The staff found that the service water integrity program provides reasonable assurance that the applicable effects will be managed for the period of extended operation.

The applicant uses the oil analysis program as a means to manage the loss of material and fouling of lube oil coolers. A discussion of this program as implemented at ANO-1 is given in the LRA, Section 4.14 of Appendix B. The staff evaluated the program description and determined that this program is designed to manage the loss of material. However, in a letter to the applicant dated June 1, 2000, the staff requested additional information regarding the use of the oil analysis program for managing fouling. In the case of the LPI/DHR lube oil coolers, the applicant justifies the use of the oil analysis program, with reference to site-specific and industry operating experience as being an adequate program for managing fouling.

Fouling of the service water side is addressed by heat transfer testing performed as part of the service water integrity program. Fouling of heat exchanger surfaces will gradually decrease the flow and heat transfer rates. Such decreases will be detected in a timely manner because the measurements are taken during each refueling outage. The staff considers the oil analysis program to be a common AMP used by more than one system at ANO-1. The AMPs that are common to more than one system are evaluated in Section 3.3.1 of this SER.

The applicant uses the reactor building leak rate testing program as a means to manage loss of material for bolting and external valve parts caused by boric acid leakage. A description of this program is given in the LRA, Section 4.16 of Appendix B. The staff considers the reactor building leak rate testing program to be a common AMP. The staff's evalaution of this program is provided in Section 3.3.1 of this SER.

The applicant uses the reactor building sump closeout inspection program as a means to manage loss of material and cracking for piping, valves, and appurtenances (wetted by sump water) because of external exposure to borated and raw water environments. A description of this program is given in the LRA, Section 4.17 of Appendix B. The following is the staff's evaluation of the reactor building sump closeout inspection program:

[*Program Scope*] The applicant includes inspection activities for all equipment and structures within and surrounding the reactor building sump for loss of material (carbon steel components) and cracking (stainless steel components). The staff found the scope of the program adequate to address the aging effects of concern.

[*Preventive/Mitigative Actions*] There is no preventative or mitigative actions associated with this program nor does the staff see a need for such actions.

[*Parameters Monitored*] The applicant states that the inspections are visual in nature and address a range of concerns, including the presence of foreign material, loose or missing bolts, tears in sump screens, and other evidence of significant structural distress, in addition to general corrosion and cracking. The staff found that the visual inspections of the reactor building sump closeout inspection program monitors the conditions that relate to the aging effects of concern.

[*Detection of Aging Effects*] The reactor building sump closeout inspection program is used to detect applicable aging effects. The staff found that the reactor building sump closeout inspection program, which is performed at the end of each refueling outage, provides reasonable assurance that significant degradation of sump components will be detected in a timely manner.

[*Monitoring and Trending*] The applicant does not identify any trending activities, but does describe operating experience in sufficient detail to indicate that the applicant has been monitoring the condition of components in the reactor building sump. The staff found the ANO-1 level of monitoring by the applicant to be adequate to identify trends of aging effects.

[*Acceptance Criteria*] The applicant does not cite any formal acceptance criteria, but describes conditions such as loose or missing bolting, and excessive pitting or corrosion of piping external piping surfaces that would require corrective actions. The staff found the acceptance criteria acceptable.

[*Operating Experience*] The applicant states that inspections of components in the reactor sump have not yet identified any significant aging. There have been service induced deficiencies of carbon or stainless steel valve parts. External surfaces of other components show little or no evidence of corrosion, pitting or cracking. The staff found that the reactor building sump closeout inspection program provides reasonable assurance that the loss of material and cracking can be detected in a timely manner.

The staff found that the reactor building sump closeout inspection program provides reasonable assurance that the applicable aging effects of LPI/DH system can be managed for the period of extended operation.

In a letter to the applicant dated June 1, 2000, the NRC staff requested additional information regarding the AMPs for the LPI/DH system. The staff noted that the internal surface of the carbon steel BWST is coated to mitigate the loss of material. The staff requested additional information regarding inspection of the tank interior surface with particular attention to the integrity of the coating material that was identified as Plastite. The applicant identifies that the preventive maintenance program includes an inspection of the tank internal surface during each refueling outage. However, the inspection methods were not provided in sufficient detail to allow the staff to determine the ability of the inspections to adequately manage aging effects. In a letter to the NRC dated October 3, 2000, the applicant documents the additional detail provided to the NRC staff in a conference call. Initially the inspection of the BWST was performed with cameras, and is currently being visually inspected to the maximum extent practical from the manway. The staff found the additional information acceptable.

The staff also noted that for other storage tanks (e.g., demineralized water tank), the applicant lists other AMPs (e.g., ASME Section XI, Maintenance Rule and level monitoring) in addition to the preventive maintenance program. These programs are not listed for the BWST. In a letter to the NRC dated September 12, 2000, the applicant states that the BWST is not within the scope of ASME Section XI. Credit was not taken for these other programs because a one-time inspection (visual and ultrasonic) of the internal and external surfaces of the BWST verified no loss of material from the tank wall. The staff found this response acceptable.

On the basis of the review described above, the staff found that the AMP identified by the applicant for LPI/DH system can manage the applicable aging effects so that there is reasonable assurance that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

High Pressure Injection/Makeup and Purification

In the LRA, Table 3.3-3, the applicant lists the following AMPs for the high pressure injection/makeup and purification (HPI/MUP) system:

- ASME Section XI ISI-IWC inspections (pressure tests)

- boric acid corrosion prevention program
- Maintenance Rule
- primary chemistry monitoring program
- oil analysis program
- reactor building leak rate testing program

The applicant states that the AMPs will manage the effects of aging so that the intended function(s) of the HPI/MUP system will be maintained consistent with the CLB, under all design loading conditions, for the period of extended operation.

The staff reviewed the AMPs for their appropriateness and effectiveness for managing the aging effects of concern as part of its review of the CF system and LPI/DH system. The HPI/MUP system has the same materials (stainless and carbon steels), aging effects (loss of material, cracking, loss of mechanical closure integrity, and fouling) and environments (borated water, external-ambient and lube oil) as the CF and LPI/DH systems. Therefore, the previous evaluations performed by the staff for AMPs apply to the HPI/MUP system, and are not repeated here.

There were no additional RAIs submitted by the staff for the HPI/MUP systems that have not already been discussed in terms of the CF and/or LPI/DH systems. The staff determined from a review of the above AMPs there is reasonable assurance that the aging effects applicable to the HPI/MUP system can be adequately managed for the period of extended operation.

Reactor Building Spray

In the LRA, Table 3.3-4, the applicant lists the following AMPs for the RBS system:

- ASME Section XI ISI-IWC (pressure tests)
- boric acid corrosion prevention program
- primary chemistry monitoring program
- secondary chemistry monitoring program
- oil analysis program

The applicant states that the AMPs will manage the effects of aging so that the intended function(s) of the components of the RBS system will be maintained consistent with the CLB under all design loading conditions for the period of extended operation.

The staff reviewed these AMPs for their appropriateness and effectiveness for managing the aging effects of concern as part of its review of the CF system or the LPI/DH system. Because the RBS system has the same materials (stainless and carbon steels), aging effects (loss of material, loss of mechanical closure integrity, cracking and fouling), and environments (borated water, external-ambient and lube oil) as the CF and LPI/DH systems, the previous staff reviews of these AMPs apply to the RBS system, and are not repeated here.

However, the staff did request a clarification on Table 3.3-4 relating to leakage monitoring, and the use of housekeeping to manage the effects of boric acid deposits, as well as the potential loss of material to the external surfaces of the RBS system. In a letter to the NRC dated September 12, 2000, the applicant states that the boric acid corrosion prevention program and

ASME Section XI ISI-IWC activities are used to manage boric acid leakage. It was also explained that most of the system, except for bolting and external valve parts, consists of stainless steel which is not subject to loss of material caused by boric acid leakage. The staff found the applicant's response acceptable.

On the basis of the review described above, the staff found that the AMP identified by the applicant for RBS system can manage the applicable aging effects so that there is reasonable assurance that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

Reactor Building Cooling and Purge

In the LRA, Table 3.3-5, the applicant lists the following AMPs for the RBCP system:

- preventative maintenance program
- Maintenance Rule program

The applicant states that the these two AMPs will manage the effects of aging so that the intended function(s) of the RBCP system will be maintained consistent with the CLB under all design loading conditions for the period of extended operation.

The staff reviewed the AMPs for their appropriateness and potential effectiveness for managing the aging effects of concern to the RBCP system as part of its review of the CF system or the LPI/DH system. Because the RBCP system has an additional material (90/10 CuNi), and an additional operating environment (gas-air) than the CF and LPI/DH systems, the staff performed a review of these AMPs for loss of material, loss of mechanical closure integrity and fouling specific to 90/10 CuNi and a gas-air environment for the RBCP system.

The applicant uses the preventative maintenance program for managing the loss of material for carbon steel components that are exposed to internal gas-air environments, and the potential of the internal surfaces of components being subject to wetting. A description of the preventative maintenance program is given in the LRA, Section 4.15 of Appendix B. The staff considers the preventative maintenance program a common AMP used by more than one system at ANO-1, and evaluated this program in Section 3.3.1 of this SER.

In a letter to the applicant dated June 1, 2000, the staff requested additional information concerning aging management of galvanized steel and duct sealants in the RBCP system. In its response dated September 12, 2000, the applicant states that galvanized steel was conservatively treated as carbon steel, and that duct sealants should be treated in the same manner as gaskets in accordance with a letter from the NRC dated November 26, 1997 (C.I. Grimes to D.J. Firth). The staff found the applicant's response acceptable.

On the basis of the review described above, the staff found that the AMP identified by the applicant for RBCP system can manage the applicable aging effects so that there is reasonable assurance that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

Sodium Hydroxide

The AMPs cited for the sodium hydroxide system in Table 3.3-6 of the LRA are:

- ASME Section XI I IWD (pressure tests)
- NaOH tank level monitoring program
- wall thinning inspection program
- Maintenance Rule

The applicant states that the list of AMPs will manage the effects of aging so that the intended function(s) of the components of the SH system will be maintained consistent with the CLB under all design loading conditions for the period of extended operation. The staff reviewed these AMPs for its appropriateness and effectiveness for managing the aging effect of concern.

The applicant uses the ASME Section XI IWD pressure tests to manage loss of material, cracking, and loss of mechanical closure integrity. The staff concluded that this program of leakage testing is an appropriate program for managing loss of material and mechanical closure integrity caused by exposure of external surfaces of carbon steel components to sodium hydroxide solutions. This program, as described in Section 4.3.3 of Appendix B, focuses on monitoring for leakage during periodic pressure tests and visual examinations to determine the general mechanical and structural condition of components. Leakage from the SH system can result in exposure of carbon steel components to sodium hydroxide solutions. The staff found that the Section XI IWD pressure tests can be effective for managing the applicable aging effects because it can be used to identify leakage so that corrective actions can be taken in a timely manner. The staff considers the ASME Section XI ISI-IWD (pressure tests) to be common to several systems at ANO-1. The staff's evaluation of this program is described in Section 3.3.1 of this SER.

The applicant cited the existing NaOH tank level monitoring program to manage the effects of aging of loss of material and loss of mechanical closure integrity from external surfaces of components exposed to the sodium hydroxide environment of the SH system. The staff reviewed the NaOH tank level monitoring program as described in the LRA, Section 4.21.7 of Appendix B. The following is the staff's review of the NaOH tank level monitoring program:

[*Program Scope*] Because the applicant includes carbon and stainless steel components of piping, valves and the SH tank as being within the scope of the program, the staff found the scope of the program adequate.

[*Preventive/Mitigative Actions*] The applicant did not identify any preventative or mitigative actions for this program nor does the staff see a need for such. However, the staff notes that monitoring of SH tank levels will contribute to prevention/mitigation by providing early indications of leakage so that sources of leakage can be identified and external exposures of components to the SH environment can be minimized.

[*Parameters Monitored*] The applicant states that SH tank level is continuously monitored per ANO-1 TS 3.3.4.B, and that alarms activate when levels move below limits prescribed in ANO-1 TS 3.3.4.B. The staff found the parameters and monitoring frequencies to be acceptable.

[*Detection of Aging Effects*] The NaOH tank level monitoring program provides indications of leakage from the SH system, and can thereby detect the potential exposure of the components to the SH environment. Indications of leakage could then result in activities to establish the source of leakage and the extent of aging effects associated with the leakage. The staff found that the NaOH tank level monitoring program is an effective supplement to the other inspection and maintenance activities which can detect and mitigate aging effects in the SH system.

[*Monitoring and Trending*] The applicant states that no formal monitoring and trending for NaOH tank levels as part of the NaOH tank level monitoring program, nor does the staff see a need for such.

[*Acceptance Criteria*] Corrective actions are initiated in response to the NaOH tank low level alarm. The alarm level is in accordance with ANO-1 TS 3.3.4.B. The staff found the acceptance criteria to be appropriate.

[*Operating Experience*] The applicant states that operating experience with the NaOH Tank level monitoring program has proven successful in the identification and correction of small amounts of leakage. The amount of leakage identified from operating experience to date has been sufficiently small such that no aging resulting in leakage has been identified.

The staff found that the NaOH tank level monitoring program provides reasonable assurance that leakage and the associated effects of material loss will be detected and mitigated on a timely basis. However, as part of its review, the staff requested additional information regarding the detection of aging before the intended function(s) is lost.

In a letter to the applicant dated June 1, 2000, the staff requested additional information regarding the AMPs listed in Table 3.3-6 and their ability to manage loss of material in the stainless steel piping and valves of the SH system. The RAI expressed concern that the application of ASME Section XI ISI -IWD and NaOH tank level monitoring activities were limited to (pressure tests and tank level monitoring) failure detection rather than methods that would detect early signs of aging before leaks or breaks occur. In its response dated September 12, 2000, the applicant states that Table 3.3-6 failed to list other applicable inspections designed to detect cracking or wall thinning. The aging management activities will include a one-time ASME Code Section XI examination. The findings of this examination would be used to initiate and plan corrective actions and/or additional inspections. The staff found this response acceptable.

The wall thinning inspection program and Maintenance Rule activities are the AMPs used to manage the loss of material from internal and external surfaces for the sodium hydroxide tank. The applicant describes the elements of these programs in Sections 3.7 and 4.13 of Appendix B. The wall thinning program uses nondestructive examinations to measure wall thickness at a sample of locations of the wall of the NaOH storage tank. The program will be initiated prior to the end of the current 40-year licensing period with thickness measurements to be made on a periodic basis. The staff considers the wall thinning inspection program and Maintenance Rule to be common to several systems at ANO-1. The staff's evaluation of these programs is described in Section 3.3.1 of this SER.

In a letter to the NRC dated March 14, 2001, the applicant addresses Open Item 2.3.2.6.2-1, and states that the flow control function for the sodium hydroxide in-line flow orifices has been added to the scope of license renewal and subject to an aging management review. The orifices are constructed of stainless steel and are susceptible to cracking and loss of material. The aging management activities used to manage similar applicable aging effects of sodium hydroxide stainless components will be used to manage the aging of the in-line orifices for the flow control intended function. Inspection activities will be completed as part of the new ASME, Section XI, ISI augmented inspections activities evaluated in this SER, Section 3.3.1.4.9.

On the basis of the review described above, the staff found that the AMP identified by the applicant for SH system can manage the applicable aging effects so that there is reasonable assurance that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

Reactor Building Isolation

The AMPs cited for the reactor building isolation system in Table 3.3-7 of the LRA are:

- ASME Section XI ISI-IWC Inspections
- augmented inspections
- wall thinning inspection program
- reactor building leak rate testing program
- Maintenance Rule program
- boric acid corrosion prevention program
- primary chemistry monitoring program
- secondary chemistry monitoring program
- auxiliary systems chemistry monitoring program

The applicant states that these AMPs will manage the effects of aging so that the intended function(s) of the components of the RBI system will be maintained consistent with the CLB under all design loading conditions for the period of extended operation. The staff reviewed these AMPs for their appropriateness and effectiveness for managing the aging effect of concern to the RBI system.

The applicant uses ASME Section XI IWC inspections as a program to manage internal cracking of stainless steels for environments of borated and treated water, loss of material for internal exposures of carbon steels to treated water, and loss of material and mechanical closure integrity for carbon steels exposed to leakage of borated water. The staff determined that the Section XI program of leakage testing is an appropriate program for managing loss of material and mechanical closure integrity. This program, as described in Section 4.3.2 of Appendix B, consists of volumetric (nondestructive), surface, and visual examinations. Such examinations can also detect evidence of internal cracking and external loss of material by corrosion, and can detect these aging effects before loss of pressure boundary integrity and leakage occurs. The ASME Section XI-IWC Program is complimented by additional examinations under programs of augmented Section XI examinations and wall thinning inspections. These inspection programs are common to several systems that are within the scope of license renewal. The staff review of these AMPs is described in Section 3.1 of this SER.

The applicant uses the Maintenance Rule and boric acid corrosion prevention programs to manage the loss of material to external surfaces, and loss of mechanical closure integrity. Sections 4.5 and 4.13 of Appendix B describe the elements of these programs. These two programs are common to several systems that are within the scope of license renewal. The staff's evaluation of this program is described in Section 3.3.1 of this SER.

The reactor building leak rate testing program is used for all components of the RBI system. The relevant part of the plant leak rate testing activities is described in Section 4.16.1 of Appendix B for integrated leak testing. Such testing can detect the presence of advanced levels of degradation sufficient to cause leak rates from the reactor building that exceed allowable limits as defined in the TS. Such testing, in combination with other examinations to detect degradation, provides reasonable assurance that the applicable aging effects of penetrations do not degrade the pressure boundary function of the reactor building. The reactor building leak testing program is common to several systems that are within the scope of license renewal. The staff's evaluation of this program is described in Section 3.3.1 of this SER.

The applicant uses the primary water, secondary water and auxiliary systems chemistry monitoring programs to manage the applicable aging effects of the RBI system. These programs are described in the LRA, Section 4.6 of Appendix B, and can be used to ensure that adverse water chemistries do not contribute to loss of material or cracking. The chemistry monitoring programs are common to several systems that are within the scope of license renewal. The staff's evaluation of these programs is described in Section 3.3.1 of this SER.

On the basis of the review described above, the staff found that the AMP identified by the applicant for RBI system can manage the applicable aging effects so that there is reasonable assurance that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

Hydrogen Control

In the LRA, Table 3.3-8, the applicant lists the following AMPs for the hydrogen control system:

- Maintenance Rule
- preventative maintenance

The applicant states that the AMPs will manage the effects of aging so that the intended function(s) of the components of the HC system will be maintained consistent with the CLB under all design loading conditions for the period of extended operation. The staff reviewed these AMPs for their appropriateness and effectiveness for managing the applicable aging effects.

The Maintenance Rule is used to manage the loss of material from the external surfaces of components. In the LRA, Section 4.13 of Appendix B, the applicant describes the elements of this program. This program is common to several systems that are within the scope of license renewal. The staff's evaluation of this program is described in Section 3.3.1 of this SER.

The applicant uses the preventative maintenance program to manage fouling of stainless steel components (tubes) that are exposed to external gas-air environments. A description of the preventative maintenance program is given in the LRA, Section 4.15 of Appendix B. This program includes inspection, cleaning and lubrication of the hydrogen sampling system cabinet/heat exchanger to manage the effects of fouling. The staff found the preventative maintenance program acceptable for managing the applicable aging effects. This program is common to several systems that are within the scope of license renewal. The staff review of this program (as described in Appendix B to the LRA) is described in Section 3.3.1 of this SER.

On the basis of the review described above, the staff found that the AMP identified by the applicant for HC system can manage the applicable aging effects so that there is reasonable assurance that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

After its initial review, the staff requested that the applicant include in its FSAR Supplement summary description of the Augmented Inspection program a one-time inspection to detect cracking or wall thinning of piping and fittings in the SH system. This was FSAR Item 3.3.3.3 of Open Item 3.3-1.

In its revised summary description of Section 16.2.3.7 of the FSAR Supplement, the applicant includes a one-time inspection to detect cracking and wall thinning of piping and fittings in the SH system in the summary description of the Augmented Inspection program. The staff finds the revised summary description as submitted by the applicant in a letter to the NRC dated March 14, 2001, acceptable.

3.3.3.3 Conclusions

On the basis of the review described above, the staff concludes the applicant has demonstrated that aging effects associated with the engineered safeguards systems can be adequately managed so that there is reasonable assurance that these systems will perform their intended functions in accordance with the CLB for the period of extended operation.

3.3.3.4 References for Section 3.3.3

1. 10 CFR Part 54, “Requirements for Renewal of Operating Licenses for Nuclear Power Plants.”
2. DG-1047, “Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants,” Working Draft, April 21, 2000.
3. “Arkansas Nuclear One–Unit 1, License Renewal Application,” January 31, 2000.
4. 10 CFR 50.55a, “Codes and Standards.”
5. ASME Boiler and Pressure Vessel Code, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” American Society of Mechanical Engineers.

3.3.4 Auxiliary Systems

In the LRA, Section 3.4, “Auxiliary Systems” the applicant describes the AMR for the auxiliary systems. Appendices A, B, and C to the LRA also contain supplementary information relating to the AMR of the auxiliary systems. The staff reviewed Section 3.4 and the applicable portions of these appendices to determine whether the applicant has provided sufficient information to demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation in accordance with 10 CFR 54.21(a)(3) for the auxiliary system SCs determined to be within the scope of license renewal and subject to an AMR.

The ANO-1 auxiliary systems include the following 13 systems:

- spent fuel
- fire protection
- Emergency diesel generator
- auxiliary building sump and reactor building drains
- alternate AC diesel generator
- Halon
- fuel oil
- instrument air
- chilled water
- service water
- penetration room ventilation
- auxiliary building heating and ventilation
- control room ventilation

In the LRA, Section 2.1, “Scoping and Screening Methodology,” the applicant describes the method used to identify the SCs that are within the scope of license renewal and subject to an AMR. The applicant identifies and lists the auxiliary system SCs in Section 2.3.3 of the LRA. The staff’s evaluation of the scoping methodology and the auxiliary system SCs included within the scope of license renewal and subject to an AMR is documented in Sections 2.1 and 2.3.3 of this SER, respectively. In the LRA, Appendix A, “Safety Analysis Report Supplement,” the applicant provides a summary description of the programs and activities used to manage the effects of aging, as required in 10 CFR 54.21(d). The applicant provides a more detailed description of these AMPs for the staff to use in its evaluation in Appendix B of the LRA. In the LRA, Appendix C, the applicant describes the processes used to identify many of the applicable aging effects for the SCs that are subject to an AMR. In the LRA, Appendix D, the applicant states that no changes to the ANO-1 TS have been identified.

3.3.4.1 Summary of Technical Information in the Application

Spent Fuel

In the LRA, Section 2.3.3.1, “Spent Fuel,” the applicant identifies and lists the following SCs of the spent fuel system that are within the scope of license renewal and subject to an AMR:

- spent fuel pool stainless steel liner

- spent fuel pool gates
- spent fuel racks
- mechanical reactor building penetration used for filling and draining the fuel transfer canal
- fuel transfer tube, which is a reactor building penetration, and a blind flange installed on the tube
- Boraflex neutron absorber material

The applicant states that the spent fuel pool safety functions are to maintain an adequate water level in the pool for cooling and shielding, and to maintain a subcritical margin.

In the LRA, Section 3.4 and Table 3.4-1, the applicant identifies the following intended functions that need to be included in the spent fuel pool system AMR:

- maintain system pressure boundary integrity
- provide structural support

The applicant also identifies the SC commodity groups, materials, environments, applicable aging effects, and AMPs for the spent fuel pool system in Table 3.4-1. The applicant includes the following applicable aging effects in its AMR:

- cracking
- loss of material
- loss of mechanical closure integrity

The AMPs and activities used by the applicant to manage the applicable aging effects include the following:

- ASME Section XI ISI - IWD inspection activities
- primary chemistry monitoring program
- boric acid corrosion prevention program
- reactor building leak rate testing
- spent fuel pool level monitoring
- spent fuel pool monitoring

The applicant also identifies an auxiliary system TLAA associated with Boraflex. The applicant analyzes this TLAA in Section 4.7 of the LRA, and the staff's evaluation is presented in Section 4.7 of this SER.

Fire Protection

In the LRA, Section 2.3.3.2, "Fire Protection," the applicant identifies and lists the fire protection SCs that are within the scope of license renewal and subject to an AMR. The applicant groups most of the fire protection components into piping, pumps, valves, and sprinkler head commodities whose system function is to distribute water to various safety-related plant SSCs in the event of a fire.

The applicant specifically lists the following fire protection components as being within the scope of license renewal:

- electric motor-driven fire pump

- diesel-driven fire pump, including the engine gearbox cooler, the jacket water heat exchanger, and the lube oil cooler

- fire water distribution system including the portion of the outside loop, hose stations, standpipes, sectional control valves, and isolation valves

- sprinkler systems protecting safety-related areas, including piping, control valves, and sprinkler heads

- sprinkler systems required to meet 10 CFR 50.48 requirements, including piping, control valves, and sprinkler heads

In response to Open Item 2.3.3.2.2-1, the applicant also added the jockey pump and fire hydrants to the scope of components subject to an AMR consistent with 10 CFR 50.48, and managed the aging of these components consistent with the balance of the fire protection systems piping, as discussed throughout the remainder of this section of the SER.

The applicant evaluates fire hose stations, fire barriers, fire doors, and penetration seals in Section 3.6 of the LRA, "Structures and Structural Components," and the NRC staff evaluates these components in Section 3.3.6 of this SER.

In the LRA, Section 3.4 and Table 3.4-2, the applicant identifies the following intended functions that need to be included in the fire protection AMR:

- maintain system pressure boundary integrity
- provide for heat transfer

The applicant also identifies the SCs commodity groups, materials, environments, applicable aging effects, and AMPs for the fire protection system in Table 3.4-2 of the LRA. The applicant includes the following applicable aging effects in its AMR:

- cracking
- loss of material
- loss of mechanical closure integrity
- fouling

The AMPs and activities used by the applicant to manage the applicable aging effects include the following:

- service water chemical control program

- fire suppression water supply system surveillance
- fire suppression sprinkler system surveillance
- fire water piping thickness evaluation
- control room Halon fire system inspection
- Maintenance Rule activities
- oil analysis program
- preventive maintenance
- ASME Section XI ISI - IWD inspection activities
- reactor building leak rate testing

Emergency Diesel Generator

In the LRA, Section 2.3.3.3, “Emergency Diesel Generator,” the applicant identifies and lists the emergency diesel generator (EDG) SCs that are within the scope of license renewal and subject to an AMR. In this list, the applicant includes components from the EDG starting air system (safety related portions from the receivers to the EDG), the EDG lubrication system, the EDG combustion air intake and exhaust system, and the EDG cooling water system. The applicant performs its scoping, screening, and AMR of the EDG fuel oil and service water systems in the LRA, Section 3.4.

The applicant states that the EDG safety functions are to supply the engineered safeguards bus loads following a design-basis accident (DBA), and to be available following a fire consistent with the requirements of 10 CFR 50.48.

In the LRA, Section 3.4 and Table 3.4-3, the applicant identifies the following intended functions that need to be included in the EDG AMR:

- maintain system pressure boundary integrity
- heat transfer

The applicant also identifies the SCs commodity groups, materials, environments, applicable aging effects, and AMPs for the EDG in Table 3.4-3. The applicant includes the following applicable aging effects in its AMR:

- cracking
- loss of material
- loss of mechanical closure integrity
- fouling

The AMPs and activities used by the applicant to manage the applicable aging effects include the following:

- EDG testing and inspections
- auxiliary systems chemistry monitoring program
- Maintenance Rule
- oil analysis program

Auxiliary Building Sump and Reactor Building Drains

In the LRA, Section 2.3.3.4, “Auxiliary Building Sump and Reactor Building Drains,” the applicant identifies and lists the auxiliary building sump and reactor building drain SCs that are within the scope of license renewal and subject to an AMR. In this list, the applicant includes the following SCs:

- floor and equipment drains
- mechanical components associated with penetrations that are required for reactor building isolation
- reactor building sump inlet and anti-vortex screen
- individual screens on floor drains that drain into the reactor building sump
- valves and piping that isolate the decay heat pump rooms
- reactor coolant pump motor oil leakage collection tanks and piping

The applicant identifies the following auxiliary building sump and reactor building drain system safety functions:

- collect radioactive material from the reactor building penetrations following a LOCA
- prevent debris from entering the reactor building sump and interfering with recirculation post LOCA
- prevents vortexing that could occur under accident conditions when the reactor building is flooded and recirculation of reactor building water is underway
- prevent release of radioactive liquid from the decay heat room to other portions of the auxiliary building following a LOCA
- collect oil leakage from the reactor coolant pump motor to prevent fire

In the LRA, Section 3.4 and Table 3.4-4, the applicant identifies the following intended functions that need to be included in the auxiliary building sump and reactor building drain system AMR:

- maintain system pressure boundary integrity
- eliminate vortexing

The applicant also identifies the SCs commodity groups, materials, environments, applicable aging effects, and AMPs for the auxiliary building sump and reactor building drain system in Table 3.4-4. The applicant includes the following applicable aging effects in its AMR:

- cracking
- loss of material

- loss of mechanical closure integrity

The AMPs and activities used by the applicant to manage the applicable aging effects include the following:

- reactor building sump closeout inspections
- primary chemistry monitoring program
- secondary chemistry monitoring program
- ASME Section XI ISI augmented inspection
- local leak rate testing
- reactor coolant pump oil collection system inspections
- boric acid corrosion prevention program

Alternate AC Generator

In the LRA, Section 2.3.3.5, “Alternate AC Diesel Generator,” the applicant identifies and lists the alternate AC (AAC) diesel generator SCs that are within the scope of license renewal and subject to an AMR. In this list, the applicant includes the following AAC diesel generator pressure boundary components:

- AAC diesel generator starting air subsystem (from the receivers to the AAC generator)

- AAC generator combustion air intake and exhaust subsystem

- AAC diesel generator cooling water subsystem

- AAC diesel generator lubrication subsystem

- two engine room exhaust fans

- exhaust fan providing cooling for the switchgear room and the corresponding inlet air damper

The applicant performs its scoping, screening, and AMR of AAC diesel generator building and AAC diesel generator fuel oil systems along with its corresponding evaluations for ANO-1 structures and the fuel oil system in the LRA, Sections 3.6, and 3.4.

The applicant states that the AAC diesel generator safety functions are to provide backup power in the event of a station blackout.

In the LRA, Section 3.4 and Table 3.4-5, the applicant identifies the following intended functions that need to be included in the AAC diesel generator system AMR:

- maintain system pressure boundary integrity
- provide for heat transfer

The applicant also identifies the SCs commodity groups, materials, environments, applicable aging effects, and AMPs for the AAC diesel generator system in Table 3.4-5.

The applicant includes the following applicable aging effects in its AMR:

- cracking
- loss of material
- loss of mechanical closure integrity
- fouling

The AMPs and activities used by the applicant to manage the applicable aging effects include the following:

- AAC diesel generator testing and inspection
- auxiliary systems chemistry monitoring program
- Maintenance Rule program
- oil analysis program

Halon

In the LRA, Section 2.3.3.6, "Halon," the applicant identifies and lists the Halon SCs that are within the scope of license renewal and subject to an AMR. In this list, the applicant includes the following Halon components:

- Halon cylinders
- actuation valves
- pilot piping
- manual actuator cylinders and valves
- discharge piping
- outlet nozzles

The applicant performs its scoping, screening, and AMR of the electrical systems in the LRA, Sections 2.5 and 3.7, and the staff discusses its evaluation of this information in Sections 2.5 and 3.3.7 of this SER. The applicant performs its scoping, screening, and AMR of Halon structures and structural components (bottle racks, system supports, ceiling tiles, marinite boards, concrete walls, concrete, and false floors) in Sections 2.4.3 and 3.6 of the LRA, and the staff discusses its evaluation in Sections 2.4.3 and 3.3.6 of this SER.

The applicant states that the Halon system function is to provide fire suppression for the ceiling and false ceiling of the control room in accordance with 10 CFR 50.48.

In the LRA, Section 3.4 and Table 3.4-6, the applicant states that maintaining system pressure boundary integrity is the intended function of the Halon system that needs to be considered during its AMR.

The applicant also identifies the SCs commodity groups, materials, environments, applicable aging effects, and AMPs for the Halon system in Table 3.4-6. The applicant includes the following applicable aging effects in its AMR:

- cracking
- loss of material

The AMPs and activities used by the applicant to manage the applicable aging effects include the following:

- control room Halon fire system inspection
- Maintenance Rule activities

Fuel Oil

In the LRA, Section 2.3.3.7, “Fuel Oil,” the applicant identifies and lists the fuel oil SCs that are within the scope of license renewal and subject to an AMR. In this list, the applicant includes the following fuel oil system components:

- bulk fuel oil storage tank

- EDG fuel tanks

- EDG day tank

- safety-related equipment and piping that supports the transfer of fuel to the EDG

- AAC diesel generator day tank

- equipment and piping that supports the transfer of fuel to the AAC diesel generator

- diesel driven fire pump day tank

- equipment and piping that supports the transfer of fuel to the diesel fire pump

- AAC diesel fuel oil return cooler (the heat exchanger)

The applicant states that the safety function of the fuel oil system is to store and supply fuel oil to diesel-driven components. The safety function of the emergency diesel fuel tanks and the EDG day tank is to store and supply the EDG with fuel oil.

In the LRA, Section 3.4 and Table 3.4-7, the applicant identifies the following intended functions that need to be included in the fuel oil system AMR:

- maintain system pressure boundary integrity
- provide for heat transfer

The applicant also identifies the SCs commodity groups, materials, environments, applicable aging effects, and AMPs for the fuel oil system in Table 3.4-7. The applicant includes the following applicable aging effects in its AMR:

- loss of material
- loss of mechanical closure integrity
- fouling

The AMPs and activities used by the applicant to manage the applicable aging effects include the following:

- AC diesel generator testing and inspection
- EDG testing and inspections
- Maintenance Rule activities
- diesel fuel monitoring program
- buried pipe inspection

Instrument Air

In the LRA, Section 2.3.3.8, "Instrument Air," the applicant identifies and lists the instrument air SCs that are within the scope of license renewal and subject to an AMR. In this list, the applicant includes piping, tubing, valves, tanks, and regulators necessary to provide reactor building integrity and to provide gas for operation of the pneumatic actuators of certain safety-related valves and dampers. The applicant listed the following instrument air components as being within the scope of license renewal:

- portions of the system that are part of the reactor building instrument air penetration

- pneumatic accumulator and the components between the accumulator and actuator of the intermediate cooling water supply valve for the reactor coolant pump (RCP) motor air and lube oil coolers

- pneumatic accumulator and the components between the accumulator and the actuators of the intermediate cooling water supply and return valves for the letdown coolers and RCP seal coolers

- high pressure carbon dioxide bottles and components between the carbon dioxide bottles and the actuators of the outside air dampers for the emergency fan filter units of the control room ventilation system

The applicant states that the instrument air safety functions are to provide reactor building integrity, and to provide gas for operation of the pneumatic actuators of certain safety-related valves and dampers.

In the LRA, Section 3.4 and Table 3.4-8, the applicant identifies maintaining system pressure boundary integrity as the intended function that needs to be included in the instrument air system AMR.

The applicant also identifies the SCs commodity groups, materials, environments, applicable aging effects, and AMPs for the instrument air system in Table 3.4-3 of the LRA. The applicant includes the following applicable aging effects in its AMR:

- cracking
- loss of material

The AMPs and activities used by the applicant to manage the applicable aging effects include the following:

- instrument air quality
- Maintenance Rule

Chilled Water

In the LRA, Section 2.3.3.9, “Chilled Water,” the applicant identifies and lists the chilled water system SCs that are within the scope of license renewal and subject to an AMR. In this list, the applicant includes the following chilled water system SCS:

- auxiliary building electrical room emergency chillers
- internal surfaces of the six cooling coils supplied by the chillers
- associated valves and piping
- main chilled water system reactor building penetrations piping and valves

The applicant states that the chilled water system safety function is to supply chilled water for emergency cooling to coolers that service the safety-related electrical equipment located in the auxiliary building electrical equipment rooms.

In the LRA, Section 3.4 and Table 3.4-9, the applicant identifies the following intended functions that need to be included in the chilled water system AMR:

- maintain system pressure boundary integrity
- provide for heat transfer

The applicant also identifies the SCs commodity groups, materials, environments, applicable aging effects, and AMPs for the chilled water system in Table 3.4-9. The applicant includes the following applicable aging effects in its AMR:

- cracking
- loss of material
- loss of mechanical closure integrity
- fouling

The AMPs and activities used by the applicant to manage the applicable aging effects include the following:

- wall thinning inspection
- ASME Section XI ISI - IWC
- auxiliary systems chemistry monitoring program
- Maintenance Rule program
- reactor building leak rate testing
- heat exchanger monitoring program
- oil analysis program

Service Water

In the LRA, Section 2.3.3.10, “Service Water,” the applicant identifies and lists the service water system SCs that are within the scope of license renewal and subject to an AMR. In this section, the applicant states that all passive, long-lived safety-related components in the service water system are within the scope of license renewal and subject to an AMR, and include the following SCs:

- piping to and from the emergency cooling pond (ECP)

- sluice gates

- service water side of each cooler supplied by the service water system

- piping and valves of the four reactor building mechanical penetrations

The applicant performs its scoping, and screening reviews of the reactor building mechanical penetration assemblies, the intake structure, and the ECP in Sections 2.4.1, 2.4.4, and 2.4.5 of the LRA. The AMR of these structures/structural components is discussed in Section 3.6 of the LRA. With the exception of the service water side of each cooler supplied by the service water system, the applicant performs its AMR of coolers and heat exchangers in conjunction with the system being cooled.

The applicant states that the safety function of the service water system is transfer heat from safety-related components to an ultimate heat since, in this case Lake Dardanelle or the ECP. This system also provides an emergency supply of water to the emergency feedwater pumps and the spent fuel pool. The service water system is also required to meet the requirements of 10 CFR 50.48.

In the LRA, Section 3.4 and Table 3.4-10, the applicant identifies the following intended functions that need to be included in the AAC diesel generator system AMR:

- maintain system pressure boundary integrity
- provide for heat transfer

The applicant also identifies the SCs commodity groups, materials, environments, applicable aging effects, and AMPs for the service water system in Table 3.4-10 of the LRA. The applicant includes the following applicable aging effects in its AMR:

- cracking
- loss of material
- fouling

The AMPs and activities used by the applicant to manage the applicable aging effects include the following:

- heat exchanger monitoring program
- ASME Section XI ISI - IWC & IWD

- service water integrity program
- Maintenance Rule
- buried pipe inspection program

Penetration Room Ventilation

In the LRA, Section 2.3.3.11, “Penetration Room Ventilation,” the applicant identifies and lists the penetration room ventilation system SCs that are within the scope of license renewal and subject to an AMR. In this list, the applicant includes the following penetration room ventilation system SCs:

- exhaust fans
- pre-filters
- high-efficiency particulate air filters
- absorber filters
- ductwork and dampers in the exhaust flow path
- dampers that isolate the normal ventilation system

The applicant states that the penetration room ventilation system safety function is to collect and process the radioactivity released to the penetration areas due to post-LOCA reactor building leakage to assure that 10 CFR 100 values are not exceeded.

In the LRA, Section 3.4 and Table 3.4-11, the applicant identifies maintaining system pressure boundary integrity as the intended function that needs to be included in the penetration room ventilation system AMR:

The applicant also identifies the SCs commodity groups, materials, environments, applicable aging effects, and AMPs for the penetration room ventilation system in Table 3.4-11. The applicant identifies loss of material as an applicable aging effects that needs to be considered in its AMR.

The AMPs and activities used by the applicant to manage the applicable aging effects include the following:

- penetration room ventilation system testing
- auxiliary systems chemistry monitoring program

Auxiliary Building Heating and Ventilation

In the LRA, Section 2.3.3.12, “Auxiliary Building Heating and Ventilation,” the applicant identifies and lists the auxiliary building heating and ventilation system SCs that are within the scope of license renewal and subject to an AMR. In this list, the applicant includes ductwork, damper bodies[1], louvers, fans, and the components that maintain the system flow path for the following three subsystems that are safety-related:

---

[1] The fire dampers that form part of the pressure boundary for these subsystems are addressed in Section 2.3.3.12 of the LRA, whereas the other fire dampers are addressed in Section 2.4.6.2 of the LRA.

- switchgear room unit coolers
- decay heat removal room unit coolers
- makeup pump room coolers

The applicant states that the auxiliary building heating and ventilation system safety function is to provide a suitable environment for those areas of the auxiliary building, which contain equipment requiring post-accident cooling.

In the LRA, Section 3.4 and Table 3.4-12, the applicant identifies the following intended functions that need to be included in the auxiliary building heating and ventilation system AMR:

- maintain system pressure boundary integrity
- provide for heat transfer

The applicant also identifies the components and component commodity groups, materials, environments, applicable aging effects, and AMPs for the auxiliary building heating and ventilation system in Table 3.4-12. The applicant includes the following applicable aging effects in its AMR:

- loss of material
- fouling

The AMPs and activities used by the applicant to manage the applicable aging effects include the following:

- EDG testing and inspection
- service water integrity program
- Maintenance Rule program
- preventive maintenance

Control Room Ventilation

In the LRA, Section 2.3.3.13, "Control Room Ventilation," the applicant identifies and lists the control room ventilation system (CRVS) SCs that are within the scope of license renewal and subject to an AMR. In this list, the applicant includes the following SCs:

- damper bodies, cooler housings, blower housings and other components that maintain the system flow path for the emergency ventilation equipment

- normal control room ventilation isolation dampers

- control room emergency unit coolers

- emergency compressor/condensing units

- electrical equipment room 2150 emergency cooling units

- emergency fan filter units

Fire dampers that do not form part of the pressure boundary for the safety-related portions of the CRVS, and the emergency compressor and condensing units heat exchangers that are exposed to service water, are evaluated in the fire protection and service water system evaluations of the LRA.

The applicant states that the control room ventilation system safety function is to isolate the control room under accident conditions.

In the LRA, Section 3.4 and Table 3.4-13, the applicant identifies the following intended functions that need to be included in the control room ventilation system AMR:

- maintain system pressure boundary integrity
- provide for heat transfer

The applicant also identifies the components and component commodity groups, materials, environments, applicable aging effects, and AMPs for the control room ventilation system in Table 3.4-13. The applicant includes the following applicable aging effects in its AMR:

- loss of material
- fouling

The AMPs and activities used by the applicant to manage the applicable aging effects include the following:

- control room ventilation system testing
- preventive maintenance program
- Maintenance Rule program
- oil analysis program

3.3.4.1.1 Effects of Aging

In the LRA, Section 3.4, the applicant performs its AMR for the auxiliary systems. The scoping and screening of the auxiliary systems SCs that are subject to an AMR is provided in Section 2.3.3 of the LRA. The applicant identifies the applicable aging effects for these SCs in Section 3.4.2 and Tables 3.4-1 through 3.4-13 of the LRA. In Appendix C of the LRA, the applicant describes the basic process used for determining these aging effects based on materials and environmental conditions. The applicant also reviews operating history to identify any additional aging effects that need to be included in the AMR. In Section 3.3.4.2.1 of this SER, the NRC staff evaluated the applicable aging effects of the auxiliary systems identified by the applicant.

Spent Fuel

In the LRA, Section 3.4.2 and Table 3.4-1, the applicant identifies the following applicable aging effects for the spent fuel components and commodity groups that are subject to an AMR: cracking, loss of material, and loss of mechanical closure integrity.

The applicant determined the following:

- cracking of the stainless steel liner plate exposed to borated water on the inside surface
- cracking of the stainless steel liner plate exposed to concrete on the outside surface
- cracking of the stainless steel gates, racks, piping, valves, fuel transfer tube, and blind flange exposed to borated water
- loss of material and loss of mechanical closure integrity of carbon steel bolting and external valve parts exposed to borated water

Fire Protection

In the LRA, Section 3.4.2 and Table 3.4-2, the applicant identifies the following applicable aging effects for the fire protection components and commodity groups that are subject to an AMR: loss of mechanical closure integrity, fouling, loss of material, and cracking.

The applicant determined the following:

- loss of mechanical closure integrity, loss of material, and cracking of diesel fire pump carbon steel and aluminum SCs exposed to gas-air mixtures
- loss of mechanical closure integrity of lube oil system carbon steel and cast iron SCs exposed to lubricating oil
- loss of mechanical closure integrity, loss of material and fouling of the cooling water jacket and heat exchangers containing carbon steel, cast iron, brass, and 90/10 Cu-Ni and copper tubing components exposed to lubricating oil and treated water environment

Emergency Diesel Generator

In the LRA, Section 3.4.2 and Table 3.4-3, the applicant identifies the following applicable aging effects for the EDG components and commodity groups that are subject to an AMR: loss of mechanical closure integrity, fouling, loss of material, and cracking.

The applicant determined the following:

- loss of material of the starting air system carbon steel components, unpainted carbon steel internal surfaces, portions of the intake exposed to rain, the intake air aftercoolers, piping and muffler internal surfaces, exhaust components exposed to weather, the lube oil coolers, cooling water carbon steel components, and the cooling water heat exchangers
- loss of (bolted) mechanical closure integrity of EDG skid mounted and connected components exposed to high vibrations
- fouling of EDG cooling water heat exchangers, the lube oil coolers and the intake air aftercoolers exposed to gas-air, treated water, and lubricating oil environments

- cracking of EDG stainless steel components exposed to stainless steel and lubricating oil

Auxiliary Building Sump and Reactor Building Drains

In the LRA, Section 3.4.2 and Table 3.4-4, the applicant identifies the following applicable aging effects for the auxiliary building sump and reactor building drain system components and commodity groups that are subject to an AMR: loss of mechanical closure integrity and loss of material. In a letter to the NRC dated September 12, 2000, the applicant states that brass, bronze, and admiralty valves listed in Table 3.4-4 of the LRA are floor drain check valves in the penetration rooms. These valves are mounted in stainless steel drain pipes. The external surfaces, as well as the internal surfaces, can be exposed to the sump water environment. Therefore, the aging effects are the same for internal and external surfaces.

The applicant determined the following:

- loss of material and loss of mechanical closure integrity of the reactor coolant pump oil leakage collection system components whose external surfaces (including bolting) may be exposed to borated water

- loss of mechanical closure integrity of the reactor coolant pump oil leakage collection system components whose internal surfaces may be exposed to lubricating oil, borated water, and contaminants

AAC Generator

In the LRA, Section 3.4.2 and Table 3.4-5, the applicant identifies the following applicable aging effects for the AAC generator system components and commodity groups that are subject to an AMR: loss of mechanical closure integrity, fouling, loss of material, and cracking.

The applicant determined the following:

- loss of material of the AAC generator system carbon steel, cast iron, copper, brass, bronze, and admiralty components exposed to gas-air and external-ambient environments

- loss of mechanical closure integrity of the AAC generator system carbon steel, cast iron, brass, bronze, admiralty, aluminum, and stainless steel components exposed to gas-air and external-ambient environments

- cracking of the AAC generator system stainless steel components exposed to lubricating oil, gas-air and external-ambient environments
- fouling of the AAC generator system copper, brass, bronze, and admiralty components exposed to treated water and gas-air environments

Halon

In the LRA, Section 3.4.2 and Table 3.4-6, the applicant identifies the following applicable aging effects for the Halon components and commodity groups that are subject to an AMR: loss of material and cracking.

The applicant determined the following:

- loss of material for Halon carbon steel piping, tanks, and discharge tubes, and the stainless steel flexible connectors exposed to external-ambient environment

- cracking of the Halon carbon steel discharge tubes, and the stainless steel flexible connectors exposed the external-ambient environment

- loss of material for the internal surfaces of the Halon aluminum discharge nozzles, the carbon steel discharge tubes, and the stainless steel flexible connectors exposed to Halon and nitrogen gas environments

- no aging effects were identified for the brass valves in the Halon, nitrogen gas, and external-ambient environment; for the internal surfaces of the carbon steel piping and tanks in Halon and gas environment; and the aluminum discharge nozzles in the external-ambient environment

Fuel Oil

In the LRA, Section 3.4.2 and Table 3.4-7, the applicant identifies the following applicable aging effects for the fuel oil system components and commodity groups that are subject to an AMR: loss of mechanical closure integrity, fouling, and loss of material.

The applicant determined the following:

- loss of material and loss of mechanical closure integrity of fuel oil system carbon steel components exposed to fuel oil

- loss of material of fuel oil system carbon steel components exposed to external-ambient environment, which may include humidity, condensation, and airborne contaminants such as sulfur dioxide, chlorine gas, sulfur gas, and ozone

- loss of material for fuel oil buried carbon steel piping

- loss of mechanical closure integrity of fuel oil system stainless steel, brass, bronze, copper, admiralty, and cast iron components (piping, valves, filters, pumps, tubing, thermowells, strainers, and tanks) exposed to fuel oil

- fouling of fuel oil stainless steel heat exchanger tubing exposed to fuel oil on the inside surface and air on the outside surface

Instrument Air

In the LRA, Section 3.4.2 and Table 3.4-8, the applicant identifies the following applicable aging effects for the instrument air system components and commodity groups that are subject to an AMR: loss of mechanical closure integrity, fouling, loss of material, and cracking.

The applicant determined the following:

- loss of material of instrument air system carbon steel valves, piping, flanges, and tanks, exposed to external-ambient environment

- cracking of instrument air system internal surfaces of stainless steel tubing and valves exposed to air

- loss of material of instrument air system internal surfaces of the carbon steel piping, tanks, flanges, and valves; the brass, bronze, and admiralty valves; and copper tubing exposed to air

- no aging effects were identified for the instrument air system stainless steel tubing and valves; the brass, bronze, admiralty valves, and the copper tubing exposed to external-ambient environment; or the internal and external surfaces of the aluminum regulators

Chilled Water

In the LRA, Section 3.4.2 and Table 3.4-9, the applicant identifies the following applicable aging effects for the chilled water system components and commodity groups that are subject to an AMR: loss of mechanical closure integrity, fouling, loss of material, and cracking.

The applicant determined the following:

- loss of material for chilled water system carbon steel, cooper, brass, and bronze components exposed to treated water, external-ambient, and gas-Freon environments

- fouling of chilled water system copper, brass, and bronze components exposed gas-Freon and treated water

- cracking of chilled water stainless steel components exposed to lubricating oil and treated water environments

- loss of mechanical closure integrity of chilled water system carbon steel components in an external-ambient environment

Service Water

In the LRA, Section 3.4.2 and Table 3.4-10, the applicant identifies the following applicable aging effects for the service water system components and commodity groups that are subject to an AMR: fouling, loss of material, and cracking.

The applicant determined the following:

- loss of material of the service water system carbon steel, stainless steel, brass, bronze, cast iron, copper, admiralty, and 90/10 Cu-Ni components and commodity groups exposed to raw water, external-buried, and external-ambient environments

- cracking of the service water system stainless steel, brass, and bronze components and commodity groups exposed to raw water

- fouling of the service water system copper, admiralty, and 90/10 Cu-Ni components exposed to raw water

Penetration Room Ventilation

In the LRA, Section 3.4.2 and Table 3.4-11, the applicant identifies loss of material as the applicable aging effect for the penetration room ventilation components and commodity groups that are subject to an AMR.

The applicant determined the following:

- loss of material of penetration room ventilation system carbon steel ductwork, dampers, valves, blowers, filters, expansion joints and the carbon steel components in the exhaust stack exposed to the external-ambient environment

- no aging effects were identified for penetration room ventilation system stainless steel flow elements, copper and brass tubing, and the carbon steel components exposed to gas-air

Auxiliary Building Heating and Ventilation

In the LRA, Section 3.4.2 and Table 3.4-12, the applicant identifies the following applicable aging effects for the auxiliary building heating and ventilation system components and commodity groups that are subject to an AMR: fouling and loss of material.

The applicant determined the following:

- loss of material of auxiliary building heating and ventilation system carbon steel ductwork, louvers, fans, dampers, and heat exchanger bodies exposed to external-ambient environment

- loss of material of auxiliary building heating and ventilation system of carbon steel heat exchanger bodies whose internal surfaces are wetted by condensation

- fouling of auxiliary building heating and ventilation system copper tubes whose outside surfaces are exposed to gas-air environment

Control Room Ventilation

In the LRA, Section 3.4.2 and Table 3.4-13, the applicant identifies the following applicable aging effects for the CRVS components and commodity groups that are subject to an AMR: fouling and loss of material.

The applicant determined the following:

- loss of material of the CRVS carbon steel ductwork, damper and heat exchanger bodies, fans, filters, and compressor body exposed to external-ambient environment with the exception of the carbon steel heat exchanger (evaporator) bodies

- loss of material of the CRVS carbon steel compressor body and heat exchanger (condenser) body exposed to lubricating oil

- no aging effect was identified for the CRVS carbon steel ductwork, damper and heat exchanger bodies, fans and filters exposed to air; tubing and valves made of copper, brass and admiralty and exposed to air, carbon dioxide or external-ambient environment; or heat exchanger tubing made of copper or 90/10 copper-nickel.

3.3.4.1.2 Aging Management Programs

In the LRA, Table 3.4-1 through 3.4-13, the applicant identifies the AMPs used to manage the effects of aging for the different components, materials, environments and intended functions. The AMPs for each of the auxiliary systems are also listed in Section 3.3.4.2 of the SER. The applicant provides a detailed description of each program in Appendix B of the LRA. The applicant also provides a summary description of each AMP in Appendix A, “Safety Analysis Report Supplement,” of the LRA in accordance with 10 CFR 54.21(d).

The staff’s evaluation of the AMPs that apply to more than one system is provided in Section 3.3.1 of this SER. AMPs specific to the auxiliary systems are evaluated in this SER, Section 3.3.4.3.2.

Spent Fuel

In the LRA, Table 3.4-1, the applicant identifies the following AMPs that are used to manage the aging of the spent fuel pool components. These programs and activities are described in Appendices A and B.

- primary chemistry monitoring
- spent fuel pool level monitoring
- ASME Section XI ISI-IWD (pressure tests)
- boric acid corrosion prevention
- reactor building leak rate testing
- spent fuel pool monitoring

Fire Protection

The applicant identifies several AMPs that are used to manage aging of the fire protection system in Appendix B of the LRA. These are summarized in the LRA, Table 3.4-2, to include the following existing programs and activities being performed in accordance with the CLB:

- fire suppression water supply system surveillance
- fire water piping thickness evaluation
- reactor building leak rate testing
- service water chemical control
- Maintenance Rule activities
- preventive maintenance
- oil analysis

Emergency Diesel Generator

The applicant identifies several AMPs for the EDG system in Appendix B of the LRA. These AMPs are summarized in the LRA, Table 3.4-3, "Emergency Diesel Generator System," to include the following existing programs and activities being performed in accordance with the CLB:

- EDG testing and inspections
- auxiliary systems water chemistry monitoring
- Maintenance Rule activities
- oil analysis

Auxiliary Building Sump and Reactor Building Drains

The applicant identifies several AMPs for the auxiliary building sump and reactor building drains system in Appendix B of the LRA. These are summarized in the LRA, Table 3.4-4, to include the following existing programs and activities being performed in accordance with the CLB:

- boric acid corrosion prevention
- RCP oil leakage collection system inspection
- ASME Section XI ISI - augmented inspections
- reactor building sump closeout inspection
- primary chemistry monitoring
- secondary chemistry monitoring
- reactor building leak rate testing
- local leak rate testing

AAC Generator

The applicant identifies the following AMPs for the AAC diesel generator system in Appendix B of the LRA.

- AAC diesel generator testing and inspections program
- auxiliary systems chemistry monitoring

- Maintenance Rule activities
- oil analysis

Halon

In the LRA, Table 3.4-6, the applicant identifies the following two AMPs for the Halon system. These programs and activities are described in Appendix B of the LRA.

- control room Halon fire system inspection
- Maintenance Rule activities

Fuel Oil

In the LRA, Table 3.4-7, the applicant identifies the following five AMPs for the fuel oil system. These programs and activities are described in Appendices A and B of the LRA.

- EDG testing and inspection
- fuel monitoring
- Maintenance Rule activities
- buried pipe inspection
- AAC diesel generator testing and inspection

Instrument Air

The applicant identifies the following two AMPs for the instrument air system in Appendix B of the LRA:

- instrument air quality program
- Maintenance Rule program

Chilled Water

The applicant identifies several AMPs for the chilled water system in Appendix B of the LRA. These are summarized in Table 3.4-9 of the LRA to include the following programs and activities being performed in accordance with the CLB:

- reactor building leak rate testing
- ASME Section XI ISI - IWC (pressure tests)
- auxiliary systems chemistry monitoring
- wall thinning inspection
- Maintenance Rule activities
- oil analysis
- heat exchanger monitoring

Service Water

The applicant identifies several AMPs for the service water system in Appendix B of the LRA. These are summarized in Table 3.4-10 of the LRA to include the following existing programs and activities being performed in accordance with the CLB:

- service water integrity
- ASME Section XI ISI - IWC & IWD (pressure tests)
- buried pipe inspection
- Maintenance Rule activities
- heat exchanger monitoring

Penetration Room Ventilation

The applicant identifies two AMPs for the penetration room ventilation system in Appendix B of the LRA. These are summarized in Table 3.4-11 of the LRA.

- penetration room ventilation system testing
- Maintenance Rule activities

Auxiliary Building Heating and Ventilation

In the LRA, Table 3.4-12, the applicant identifies the following four AMPs for the auxiliary building heating and ventilation system components. These programs and activities are described in Appendices A and B of the LRA.

- EDG testing and inspections
- preventive maintenance
- Maintenance Rule activities
- service water integrity

Control Room Ventilation

In the LRA, Table 3.4-13, the applicant identifies the following four AMPs for the control room ventilation system. These programs and activities are described in Appendix B.

- Maintenance Rule activities
- preventive maintenance
- control room ventilation testing
- oil analysis

The applicant concludes that these programs and activities can manage the effects of aging in such a way that the intended functions of the safety-related components of the auxiliary systems would be maintained consistent with the CLB, under all design loading conditions during the period of extended operation.

3.3.4.2 Staff Evaluation

The NRC staff reviewed the information provided by the applicant in the LRA, Sections 2.3.3 and 3.4, and Appendices A, B, and C. The staff also reviewed the applicable system drawings provided with the LRA, and the applicable portions of the ANO-1 UFSAR that apply to the 13 auxiliary systems that are within the scope of this review. After the initial review of this information, the NRC staff requested additional information concerning the auxiliary systems in a letter to the applicant dated June 9, 2000. The applicant responded to the NRC staff's RAIs in a letter to the NRC dated September 12, 2000. The staff requested additional clarification on a number of concerns during a telecommunication that took place on October 11, 2000. This telecommunication is documented in a letter to the applicant dated October 20, 2000, and the applicant's response to the requests for clarification is documented in a letter to the NRC dated November 2, 2000. The concerns identified by the staff, and the staff's overall evaluation of the auxiliary systems' AMR provided by the applicant are discussed in the paragraphs below.

In accordance with 10 CFR 54.21(a)(3), the staff reviewed this information to determine if there is reasonable assurance that the effects of aging of the auxiliary system will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

3.3.4.2.1 Effects of Aging

During the review of auxiliary systems, the staff identified a number of aging effects that were determined to be applicable to more than one auxiliary system. The staff evaluation of these common aging effects raised some general concerns applicable to many of the auxiliary systems. These concerns are reviewed in the following paragraphs.

Aging from Fatigue

Many of the auxiliary systems that are within the scope of license renewal are designed to American National Standards Institute (ANSI) Standards B31.1 and B31.7 Code requirements. Although these codes do not require an explicit fatigue analysis, they do specify allowable stress levels based on the number of anticipated thermal cycles. In the LRA, Section 4.3, the applicant identifies metal fatigue as a TLAA, and states that fatigue evaluations were required in the design of the ANO-1 Class 1 components in accordance with the requirements of the applicable design codes.

In a letter to the applicant dated June 9, 2000, the staff requested additional information relating to cyclic loading and fatigue of Class 2 and 3 components. In its response to the NRC dated September 12, 2000, the applicant indicates that thermal fatigue is not a concern for most non-Class 1 systems at ANO-1 because system temperatures remain below the thermal fatigue threshold. In addition, the applicant states that evaluations were performed that determined a 7000 cycle thermal fatigue design criterion for the non-Class 1 systems that operate above the thermal fatigue threshold. The applicant performed an evaluation that determined, on the bases of the current number of fatigue cycles experienced during its current operation history, that ANO-1 will not exceed the 7000 thermal cycles allowed during the period of extended operation.

In a telephone communication on October 13, 2000, the staff requested that the applicant submit additional information to support its conclusion regarding thermal fatigue. In addition, the staff requested that the applicant provide additional clarification regarding operating temperature, the thermal fatigue threshold, and the evaluations that were performed that led to the conclusion that the 7000-cycle thermal fatigue design criterion would not be exceeded. A correction was also requested on the code requirements for the diesel fire pump exhaust, and the use of the term "ASME qualified piping." In a letter to the NRC dated November 2, 2000, the applicant states that reference to the ASME qualified piping was incorrect, and should have been to ANSI B31.1 and B31.7 qualified piping. In addition, the thermal fatigue threshold was defined as 220EF for carbon steel, and 270EF for austenitic stainless steel. The applicant also states that the maximum number of thermal cycles the non-Class 1 system piping is expected to be exposed was conservatively estimated to be 3500 cycles over the proposed 60 years of operation. This is well below the 7000-cycle thermal fatigue design criterion. On the basis of this information, the staff found that the applicant has adequately considered thermal fatigue of the auxiliary systems Class 2 and 3 components.

Aging from Vibration

On the basis of the staff's experience, degradation of piping systems (e.g., cracking of welds) may be caused by vibration (mechanical or hydrodynamic) loading. In Appendix C, Section 9.0, of the LRA, the applicant describes the loss of mechanical closure integrity in high vibration applications such as diesel generators. However, upon reviewing this information it was not clear to the NRC staff that the applicant adequately considered cracking of auxiliary system piping welds, especially socket welds, and cracking of auxiliary building heating and ventilation system and CRVS ventilation ducting in high-vibration environments. In addition, it is not clear that loosening of bolts attached to the base of equipment, such as the fans in the auxiliary building heating and ventilation system (a source of high vibrations) had been considered. The staff raised this concern based upon review of other operating plants where cracking of ductwork has occurred as a result of vibration-induced fatigue and loosening of fasteners exposed to dynamic loads.

In a letter to the applicant dated June 9, 2000, the staff requested that the applicant clarify whether this aging effect had been considered in its AMR for the auxiliary systems. In its response to the NRC dated September 12, 2000, the applicant states that cracking as a result of vibration can generally be attributed to deficiencies in design, and typically occur in a relatively short period of operational time when compared to the overall operational life of the plant. On the basis of its assessment, the applicant concluded that cracking as a result of dynamic (mechanical or hydrodynamic) loads caused by vibration was a potential aging effect that was found not to be applicable to the auxiliary systems. The applicant also states that loss of mechanical closure integrity due to high vibration is an applicable aging effect for skid mounted and connected components of the EDG, the fire protection system, the AAC generator system, and the fuel oil system that are subjected to diesel engine vibration. The applicant's AMR for loss of mechanical closure integrity is evaluated by the staff in Section 3.3.4.3.2 of the SER. The applicant further stated that loss of mechanical closure integrity due to high vibration was not found to be an applicable aging effect for components in the nine other auxiliary systems.

During a telephone communication on October 13, 2000, the staff noted that several ventilation systems discussed in the auxiliary system AMR contain flexible expansion joints, which will age from relative motion between vibrating equipment, warm moist air, and exposure to temperature changes, oxygen and radiation. The aging effects of concern for these expansion joints are hardening and loss of strength. If flexible expansion joints age in these manners, cracking and loosening of fasteners become applicable aging effects.

In its response to the NRC's RAI, the applicant states that preventive maintenance activities are used to manage the aging of these flexible expansion joints. The staff's evaluation of this AMP is contained in Section 3.3.4.3.2 of the SER. On the basis of this information, the staff found that the applicant has adequately considered the effects of vibration for the auxiliary systems.

Loss of Mechanical Closure Integrity

Loss of mechanical enclosure integrity is an applicable aging effect for the following systems:

- spent fuel
- fire protection
- EDG
- auxiliary building sump and reactor building drains
- AAC diesel generator
- fuel oil
- chilled water

In a letter to the NRC dated September 12, 2000, the applicant states that such AMPs as the boric acid corrosion prevention program, ASME Section XI ISI-IWD (pressure testing), and the reactor building leak rate testing program are used to manage the loss of mechanical closure integrity in the spent fuel system, and the loss of mechanical closure integrity for carbon steel commodity groupings is managed by the boric acid corrosion prevention program, as well. In addition, the applicant states that preventive maintenance activities include bolt torque verification used to detect a loss of mechanical closure integrity in such systems as the fire protection system.

In a letter to the NRC dated September 12, 2000, the applicant states that the EDG, the fire protection system, the AAC generator system, and the fuel oil system are subjected to diesel engine vibration. During a telephone conversation on October 26, 2000, the staff noted that the AAC diesel generator test and inspection program does not include verification of bolt torque as required to ensure bolted closure integrity for systems subjected to high vibration. In a letter the NRC dated November 2, 2000, the applicant states that the high vibration components in the fuel oil system subject to potential loss of mechanical closure integrity due to loss of preload are the EDG, the AAC diesel generator, and the fire protection diesel. In addition to the descriptions of the AMPs credited in the LRA (Emergency Diesel Generator Testing and Inspections, the Alternate AC Diesel Generator Testing and Inspections, and the fire protection diesel inspections), the programs include periodic tear-down and inspection of many components and subsequent re-assembly and torquing of bolts. The applicant states that this addresses loss of preload for those components. Vibration monitoring is performed during performance testing of the diesels that would identify significant loss of preload that could lead to loss of function. Other inspections for leaks, loose fasteners, etc., conducted in accordance with manufacturer’s recommendations, would help to ensure that loss of mechanical closure integrity is addressed prior to loss of function. On the basis of this information, the staff found that the applicant has adequately considered the loss of mechanical closure integrity for the auxiliary systems.

Seismic II/I Components and Piping

In accordance with 10 CFR 54.4(a)(2), the applicant is required to include all non-safety-related systems, structures, and components whose failure could prevent satisfactory accomplishment of any safety-related functions identified in 10 CFR 54.4(a)(1). In the LRA, Section 2.1.2, the applicant identifies a few cases where non-safety-related components can impact safety-related functions. In a letter to the applicant dated June 9, 2000, the NRC requested that the applicant clarify whether the SCs of the auxiliary systems discussed in Section 3.4 of the LRA include any Seismic II/I spatially-related components and piping segments. “Seismic II/I” is a non-seismic Category I system, structure, or component whose failure could prevent the satisfactory accomplishment of a seismic Category I structure or component function. The applicant was asked to identify where Seismic II/I components and piping segments are specifically included within the scope of license renewal for auxiliary systems, and to identify the specific AMPs that apply to these components for each applicable aging effect. The applicant responded to this RAI in a letter to the NRC dated September 12, 2000. In a follow-up telephone communication on October 26, 2000, the applicant was asked to clarify the following statement from its response: “non-seismic Category 1 components are assumed to fail during a seismic event and seismic Category 1 structures are designed to prevent these non-seismic Category 1 components from causing loss of safety function of safety-related components and structures.” In a letter to the NRC dated November 2, 2000, the applicant states that this statement should be deleted. The applicant further states that the auxiliary system “has no seismic Category II/I spatially-related components or piping segments that are within the scope of license renewal as discussed in Section 3.4 of the ANO-1 LRA.” On the basis of this information, the staff found that the applicant has adequately considered the effects of Seismic II/I components associated with the auxiliary systems.

External Environment - Buried Piping

Loss of material is considered to be an applicable aging effect for the external surfaces of buried piping. In the LRA, Section 3.4, the applicant identifies only the fuel oil and service water systems as having inspection programs for buried piping. From this information, it is unclear as to whether any additional components in any of the auxiliary systems are exposed to an underground environment. The piping and instrument drawing for the fire protection system indicates that piping is partially routed in soil or encased in concrete, but an underground environment is not identified in Table 3.4-2 of the LRA, “Fire Protection.” In a letter to the applicant dated June 9, 2000, the staff requested that the applicant identify any additional buried components in any of the auxiliary systems, and any applicable aging effects caused by an underground environment. In a letter to the NRC dated September 12, 2000, the applicant states that in addition to the fuel oil and service water systems, the fire protection system contains buried piping that is within the scope of license renewal. The underground environment for the fire protection system piping was inadvertently omitted from Table 3.4-2. In additionally, the applicant verified that no additional auxiliary system components that are within the scope license renewal exist in an underground environment.

For the fire protection system, the applicant states that the buried piping is cast iron and carbon steel and serves as a pressure boundary for fire suppression water. This pipe is coated with a primer, a first coat of hot coal tar, a fibrous-glass mat, a second coat of hot coal tar, a wrap of coal tar saturated asbestos felt, and whitewash or Kraft paper. The outer-surface of this pipe is not expected to experience corrosion as long as the coating and wrapping is intact. The aging effects become an issue if the coatings develop voids or otherwise fail, therefore, aging of the coating needs to be managed. The applicant has developed the fire suppression water supply system surveillance program to manage the loss of material of the fire protection system buried pipe. On the basis of this information, the staff found that the applicant has adequately considered loss of material for buried piping.

Spent Fuel

For the spent fuel pool components subject to an AMR, the applicant identifies loss of material, cracking, and loss of mechanical closure integrity as applicable aging effects. Table 3.4-4 of the LRA summarizes the components, materials, environments and applicable aging effects considered in the AMR. In addition, Section 3.3.4.2.1 of this SER contains a discussion of the applicable aging effects as presented in the LRA. The staff reviewed the information described above, and found that the aging effects identified by the applicant for the spent fuel pool system are consistent with published literature and industry experience.

Fire Protection

For the fire protection components subject to an AMR, the applicant identifies loss of material, cracking, loss of mechanical closure integrity and fouling as the applicable aging effects. In the LRA, Table 3.4-2, the applicant summarizes the components, materials, environments and applicable aging effects considered in the AMR. In addition, Section 3.3.4.2.1 of this SER contains a discussion of the applicable aging effects as presented in the LRA. However, during its review, the staff noted that Table 3.4-2 does not identify fouling (resulting in loss of flow) due to biological species present in the raw water source as an applicable aging effect for piping. In a letter to the NRC dated November 2, 2000, the applicant states that bio-fouling of fire protection lines is managed by the service water integrity program that includes the injection of biocides, internal visual examinations, and periodic flushing. The staff reviewed the information described above, and found that the aging effects identified by the applicant for the fire protection system are consistent with published literature and industry experience.

Emergency Diesel Generator

For the EDG components subject to an AMR, the applicant identifies loss of material, loss of mechanical closure integrity, cracking, and fouling as the applicable aging effects. Table 3.4-3 of the LRA summarizes the components, materials, environments and applicable aging effects considered in the AMR. In addition, Section 3.3.4.2.1 of this SER contains a discussion of the applicable aging effects as presented in the LRA. The staff reviewed the information described above, and found that the aging effects identified by the applicant for the EDG components are consistent with published literature and industry experience.

Auxiliary Building Sump and Reactor Building Drains

The applicant identifies loss of material, cracking, and loss of mechanical closure integrity as applicable aging effects for components in the auxiliary building sump and reactor building drain system that are exposed to treated water, borated water, raw water, oil and external-ambient conditions. Table 3.4-4 of the LRA summarizes the components, materials, environments and applicable aging effects considered in the AMR. In addition, Section 3.3.4.2.1 of this SER contains a discussion of the applicable aging effects as presented in the LRA. The staff reviewed the information described above, and found that the aging effects identified by the applicant for the auxiliary building sump and reactor building drain system components are consistent with published literature and industry experience.

AAC Generator

For the AAC diesel generator system, the applicant identifies loss of material, cracking, loss of mechanical closure integrity and fouling as the applicable aging effects. Table 3.4-5 of the LRA summarizes the components, materials, environments and applicable aging effects considered in the AMR. In addition, Section 3.3.4.2.1 of this SER contains a discussion of the applicable aging effects as presented in the LRA. The staff reviewed the information described above, and found that the aging effects identified by the applicant for the AAC diesel generator system components are consistent with published literature and industry experience.

Halon

For the Halon system, the applicant identifies loss of material, and cracking as the applicable aging effects. Table 3.4-6 of the LRA summarizes the components, materials, environments and applicable aging effects considered in the AMR, and Appendix C, Sections 7.0 and Section 8.0, discuss aging effects applicable to the Halon system. In addition, Section 3.3.4.2.1 of this SER contains a discussion of the applicable aging effects as presented in the LRA.

During its review of the information in the LRA, the staff noted that Section 3.4.2 of the LRA identifies loss of material and cracking as applicable aging effects for the discharge tube assemblies, and the pilot header flexible tubing and fittings. Contrary to this information, in Appendix C, Section 7.3.50, the applicant states that Halon is non corrosive for the materials that are used in the Halon system. In a letter to the applicant dated June 9, 2000, the staff asked the applicant to explain this discrepancy and to provide clarification of all potential aging effects for the Halon system. In its response to the NRC dated September 12, 2000, the applicant states that Table 3.4-6 is incorrect, the external surfaces of the Halon system have no applicable aging effects because the system is located in an air conditioned environment. In addition, the internal surfaces of the aluminum discharge nozzles are not susceptible to a loss of material (from corrosion) or cracking in the Halon or nitrogen gas environment.

In addition, the staff asked the applicant to explain why no aging effects were identified for the internal environments of the carbon steel piping and tanks. In its response, the applicant states that these components are not subject to aging because the Halon and nitrogen environments are non-corrosive. The staff reviewed the information described above, and found that the aging effects identified by the applicant for the Halon system components are consistent with published literature and industry experience.

Fuel Oil

The fuel oil system stores fuel oil and supplies it to the diesel-driven components including the EDG and AAC diesel generator. The components (piping, valves, filters, pumps, tubing, thermowells, strainers, and tanks) in the system are made of carbon steel, stainless steel, brass, bronze, copper, admiralty, and cast iron. These components may experience loss of material if significant amount of oxygenated water and contaminants such as chloride ions are present in fuel oil. The magnitude and type of corrosion damage will depend on component materials and the operating condition in the system. In addition, the applicant states that stainless steel heat exchanger tubes exposed to fuel oil on the inside surface and gas-air environment on the outside may experience fouling, which can affect the heat transfer capability of the heat exchangers. In Appendix C of the LRA, Section 6.3.1, the applicant identifies cracking as an applicable aging effect for stainless steel components exposed to fuel oil in the presence of oxygenated water, as well. The applicant also states that cracking of bolts may lead to loss of mechanical closure integrity.

The staff reviewed the aging effects identified by the applicant as applicable to the fuel oil systems, and in a letter to the applicant dated June 9, 2000, asked the applicant to discuss the potential for the loss of material to the exterior surface of cast-iron components that are in contact with moist air and the protective coatings applied to these surfaces are not in tact. In its response to the NRC dated September 12, 2000, the applicant states that loss of material to the exterior surface of cast iron components was included in its AMR and is managed by Maintenance Rule activities. This aging effect was not identified on Table 3.4-7 because of an administrative error. The staff found the applicant's response acceptable.

The applicant also states that buried carbon steel components can experience loss of material to the exterior surface if it comes in contact with soil or groundwater. The staff agrees that the presence of oxygen, moisture, biological organisms, and contaminants in soil and groundwater may cause general corrosion, microbiological influenced corrosion, and galvanic corrosion of carbon steel components leading to loss of material.

On the basis of the description of the fuel oil system internal and external environments and materials, the staff found that the applicant has considered applicable aging effects that are consistent with published literature and industry operating experience.

Instrument Air

For the instrument air system components subject to an AMR, the applicant identifies loss of material and cracking as applicable aging effects. Table 3.4-8 of the LRA summarizes the components, materials, environments and applicable aging effects considered in the AMR. In addition, Section 3.3.4.2.1 of this SER contains a discussion of the applicable aging effects as presented in the LRA. The staff reviewed the information described above, and found that the aging effects identified by the applicant for the instrument air system components are consistent with published literature and industry operating experience.

Chilled Water

For the chilled water system components subject to an AMR, the applicant identifies loss of material, loss of mechanical closure integrity, fouling, and cracking as applicable aging effects. Table 3.4-9 of the LRA summarizes the components, materials, environments and applicable aging effects considered in the AMR. In addition, Section 3.3.4.2.1 of this SER contains a discussion of the applicable aging effects as presented in the LRA. The staff reviewed the information described above, and found that the aging effects identified by the applicant for the chilled water system components are consistent with published literature and industry experience.

Service Water

For the service water system components subject to an AMR, the applicant identifies loss of material, fouling, and cracking as applicable aging effects. Table 3.4-10 of the LRA summarizes the components, materials, environments and applicable aging effects considered in the AMR. In addition, Section 3.3.4.2.1 of this SER contains a discussion of the applicable aging effects as presented in the LRA. The staff reviewed the information described above, and found that the aging effects identified by the applicant for the service water system components are consistent with published literature and industry experience.

Penetration Room Ventilation

For the penetration room ventilation system components subject to an AMR, the applicant identifies loss of material as the applicable aging effect. Table 3.4-11 of the LRA summarizes the components, materials, environments and applicable aging effects considered in the AMR. In addition, Section 3.3.4.2.1 of this SER contains a discussion of the applicable aging effects as presented in the LRA. During its review, the staff noted that no internal aging effects were identified for carbon steel components exposed to the gas-air environment. In a letter to the applicant dated June 9, 2000, the NRC staff asked the applicant to provide an AMR for these components or a technical justification for excluding loss of material as an applicable aging effect for carbon steel components in the gas-air environment. In its response, the applicant states that carbon steel components exposed to gas-air do not have their internal surfaces wetted by condensation, therefore, loss of material is not an aging effect for these components.

In the LRA, Table 3.4-11, the applicant does not identify an applicable aging effect for the tubing made of copper and brass exposed to external environment. However, in Section 8.0 of Appendix C, the applicant states that loss of material due to general corrosion is an applicable aging effect for these components in ambient air if the external surface is in contact with moist air. The staff requested that the applicant provide a technical justification for excluding loss of material as an applicable aging effect for these components. In its response the applicant states that these components are not located in a high moisture ambient environment, therefore, loss of material is not an applicable aging effect.

Ductwork generally includes isolators (such as flexible collars between ducts and fans, seals in dampers and doors, etc.) made of elastomers, which can harden or experience loss of strength because of relative motion between vibrating equipment, warm moist air, exposure to temperature changes, oxygen, and radiation. As a result of this aging, vibration and subsequent dynamic loads applied to the ductwork and fasteners cannot be eliminated. The staff requested that the applicant provide an AMR for these components or provide a technical justification for excluding them from an AMR. In its response, the applicant states that carbon steel expansion joints are provided in the ventilation piping (ductwork) near the inlet and outlet of the ventilation exhaust fans. Unlike the other auxiliary ventilation systems, the penetration room ventilation system does not contain expansion joints made of elastomers. Therefore, the expansion joints in the penetration room ventilation system are subject to the same aging effects and AMPs as the carbon steel ductwork, and no additional aging effects need to be considered.

The staff reviewed the information described above, and found that the aging effects identified by the applicant for the penetration room ventilation system components are consistent with published literature and industry experience.

Auxiliary Building Heating and Ventilation

For the auxiliary building heating and ventilation system components subject to an AMR, the applicant identifies loss of material and fouling as applicable aging effects. Table 3.4-12 of the LRA summarizes the components, materials, environments and applicable aging effects considered in the AMR. In addition, Section 3.3.4.2.1 of this SER contains a discussion of the applicable aging effects as presented in the LRA. During its review of this information, the NRC staff noted that the applicant did not identify cracking of ductwork due to vibration-induced fatigue, and loosening of fasteners due to dynamic loading, which are common aging effects of heating, ventilation and air conditioning (HVAC) systems, especially in the vicinity of attached device types exposed to dynamic loads such as fans. As previously discussed in this section of the SER, the ductwork generally includes isolators (such as flexible collars between ducts and fans, seals in dampers and doors, etc) made of elastomers, which can harden or experience a loss of strength as a result of relative motion between vibrating equipment, warm moist air, exposure to temperature changes, oxygen, and radiation. Because of this potential aging, vibration and subsequent dynamic loads applied to the ductwork and fasteners cannot be eliminated. The staff requested that the applicant provide an AMR for these components or provide a technical justification for excluding these aging effects from an AMR. In its response, the applicant states that these flexible expansion joints are within the scope of license renewal. Preventive maintenance activities are used to manage aging effects on these flexible expansion joints. Specifically, flexible expansion joints are routinely examined by inspections performed in accordance with preventive maintenance procedures.

In the LRA, Table 3.4-12, the applicant identifies gas-air as an environment for the external surface of the copper and 90/10 copper-nickel heat exchanger tubing with fouling as an aging effect. However, in Appendix C, Section 7.0, of the LRA, the applicant does not identify fouling as an applicable aging effect requiring an AMR in a gas environment. In Appendix C, Sections 4.0 and 10.0, of LRA, the applicant identifies fouling as an applicable aging effect requiring aging management in raw water (service water) and treated water (chilled water) environments. The ANO-1 UFSAR, Section 9.7.2.1, identifies the potential for fouling to take place on the internal surface of the subject tubing with a chilled or raw water environment. In a letter to the NRC dated November 12, 2000, the applicant confirmed that fouling is an applicable aging effect caused by a raw water environment of the service water systems for applicable auxiliary system components

For makeup pump room coolers, the applicant identifies pressure boundary integrity as an intended function for 90/10 copper-nickel tubing, but did not identify heat transfer as an intended function in its AMR. In its response to the NRC, the applicant states that the heat transfer function of the pump room coolers is not required to meet any of the scoping criteria in 10 CFR 54.4(a), therefore, the heat transfer function for these components is not within the scope of license renewal.

The staff reviewed the information described above, and found that the aging effects identified by the applicant for the penetration room ventilation system components are consistent with published literature and industry experience.

Control Room Ventilation

For the CRVS components subject to an AMR, the applicant identifies loss of material and fouling as applicable aging effects. Table 3.4-13 of the LRA summarizes the components, materials, environments and applicable aging effects considered in the AMR. In addition, Section 3.3.4.2.1 of this SER contains a discussion of the applicable aging effects as presented in the LRA. During its review of this information, the NRC staff noted that the applicant did not identify loss of material as an applicable aging effect for the carbon steel evaporator bodies. In its response to this concern the applicant states that these components are located in a “non-harsh indoor environment” (e.g., not exposed to excessive moisture or humidity), and were not subject to loss of material. In addition, the applicant did not identify loss of material as an applicable aging effect for carbon steel ductwork, dampers, heat exchangers, fans, and filters even though these components are exposed to air and carbon dioxide environment. In a letter to the applicant dated June 9, 2000, the staff asked the applicant to provide an AMR for these components or a technical justification for excluding loss of material as an applicable aging effect for these carbon steel components in the air and carbon dioxide environments. In its response, the applicant states that the internal surfaces of the CRVS coolers are exposed to gas-air and are wetted by condensation. However, the remaining carbon steel components do not have their internal surfaces wetted by condensation, therefore, loss of material is not an aging effect for these components. In the same letter to the applicant, the staff also asked the applicant to provide a technical justification for excluding loss of material as an applicable aging effect for tubing and valves made of copper, brass and admiralty and exposed to an external-ambient environment. In its response, the applicant states that these components are not located in a high moisture ambient environment, therefore, loss of material is not an applicable aging effect. The staff concurs with this response.

According to the staff’s experience, cracking of ductwork due to vibration-induced fatigue and loosening of fasteners due to dynamic loading are very common types of aging effects identified in heating, ventilation and air conditioning (HVAC) systems, especially in the vicinity of attached device types exposed to dynamic loads such as fans. The ductwork generally includes isolators. If the isolators degrade, vibration and subsequent dynamic loads applied to the ductwork and fasteners cannot be eliminated. For a discussion of this issue and the applicant’s responses to the appropriate RAIs, refer to the discussion described in the auxiliary building heating and ventilation system.

The staff reviewed the information described above, and found that the aging effects identified by the applicant for the CRVS components are consistent with published literature and industry experience.

3.3.4.2.2 Aging Management Programs

The staff evaluation of the applicant's AMPs focused on the program elements rather than details of specific plant procedures. The staff's approach to evaluating each program and activity used to manage the applicable aging effects is described in Section 3.3.1 of this SER.

Generic Aging Management Programs

Reactor Building Leak Rate Testing

The reactor building leak rate testing program is evaluated separately in Section 3.3.1 (Common AMPs) of this SER. In general terms, the staff found that the specified leak rate testing, in conjunction with other credited programs, can provide reasonable assurance that the aging effects will be managed such that containment isolation components located between the inboard and outboard containment isolation valves, in the spent fuel system, fire protection system, auxiliary building sump and reactor building drain system, and chilled water system will continue to perform their intended functions consistent with the CLB for the period of extended operation.

Maintenance Rule

The Maintenance Rule program is evaluated separately in Section 3.3.1 (Common AMPs) of this SER. In general terms, the Maintenance Rule AMP is used to manage loss of material and loss of mechanical closure integrity in the following auxiliary systems:

- fire protection system
- EDG
- AAC diesel generator
- fuel oil system
- instrument air system
- chilled water system
- service water system
- penetration room ventilation system
- auxiliary building heating and ventilation
- control room ventilation system

The applicant states that the Maintenance Rule program system walkdowns apply to external surfaces only. These walkdowns are performed at least once per refueling cycle, or about every 18 months. The applicant does not identify any trending activities for the Maintenance Rule program, and the staff does not see a need for any. The applicant states that the Maintenance Rule program is implemented consistent with the requirements of 10 CFR 50.65, and that the acceptance criteria are "no unacceptable visual indications of loss of material or deteriorated coating is present." The applicant describes its Maintenance Rule program as a relatively new program; baseline walkdowns were initiated in 1997. The program focuses on detection and repair prior to loss of component function, and the program will be adjusted based on information gained from ongoing site-specific and industry experience. The Maintenance Rule program is implemented by qualified engineers that are trained in accordance with ANO's Engineering Support Personnel (ESP) Training Program (an INPO-accredited program). The Maintenance Rule program is an NRC-approved program that is part of ANO-1's CLB.

The staff's evaluation of the Maintenance Rule program is documented in Section 3.3.1 of this SER. However, the staff agrees that the Maintenance Rule program, in conjunction with other AMPs, provides reasonable assurance that the loss of material and loss of mechanical closure integrity of the external surface of accessible auxiliary system SCs will be managed so that the intended function of these components will be adequately maintained consistent with the CLB for the period of extended operation.

Preventive Maintenance

The preventive maintenance program, in conjunction with other AMP, is used to manage loss of material, cracking, fouling, and loss of mechanical closure integrity in the following auxiliary systems:

- fire protection system
- auxiliary building heating and ventilation
- control room ventilation system

In the LRA, Section 4.15, "Preventive Maintenance," of Appendix B, the applicant describes the attributes of the preventive maintenance program. In addition, Section 3.4 of the LRA discusses the preventive maintenance program as it applies to the auxiliary systems. In this SER, Section 3.3.4.3.1, "Effects of Aging," provides a discussion on the hardening and loss of strength of flexible expansion joints caused by relative motion between vibrating equipment, warm moist air, exposure to temperature changes, oxygen and radiation. In a letter to the NRC dated September 12, 2000, the applicant states that preventive maintenance activities are used to manage the effects of aging on these flexible expansion joints. In its response dated November 2, 2000, the applicant states that the preventive maintenance activities are used to manage sealant aging effects. Sealants are routinely examined by inspections performed in accordance with preventive maintenance procedures, usually on a semiannual frequency. The applicant states that since the aging effects for sealant materials take a long time to become serious enough to impact the function of the system, use of visual inspections is sufficient to manage the effects of aging of sealant materials. In an electronic communication dated November 7, 2000, the applicant clarified that this inspection procedure also applies to flexible expansion joints. Based upon the information provided by the applicant, the staff found the applicant's response regarding the activities used to manage the effects of aging of flexible expansion joints and sealants acceptable.

[*Program Scope*] The preventive maintenance program involves visual inspection of the intake air, exhaust air, lube oil and heat exchanger components within the fire protection system; fuel oil system; external surface of the carbon steel components including ductwork, louvers, fans, dampers, and heat exchanger bodies, exposed to external-ambient environment in the auxiliary building heating and ventilation, and control room ventilation systems. The applicant uses the preventive maintenance program to manage fouling of the copper tubing in the switchgear room and decay heat room coolers (heat exchangers). The program also manages the loss of mechanical closure integrity for the cooling water systems. The staff found the scope of the program to be adequate.

[*Preventive/Mitigative Actions and Parameters Monitored*] The applicant uses the preventive maintenance program to manage the effects of aging associated with the fire protection system, fuel oil system, auxiliary building heating and ventilation, and control room ventilation system. The components inspected for these systems under the preventive maintenance program, along with the parameters that are observed, are as follows:

- Fire Protection System: Visual inspections are required for the following diesel fire pump engine (K5) components: air cleaner and hoses, oil filter and oil filter hoses, turbo charger, water pump and coolant hoses. Parameters observed are loose clamps, cracks, punctures and tears. Corrective actions are completed in accordance with the site corrective action program as necessary. Bolt torque is checked as required to ensure bolted closure integrity. This maintenance will detect loss of material or cracking of these components and help to prevent a loss of bolted closure integrity.

- Fuel Oil System: Preventive maintenance tasks drain and clean the safety-related fuel oil tanks every 10 years to ensure there is no sludge buildup. These tasks will be updated to have the coating inspected to ensure that it is intact (no flaking or any observable damage to coating) and the uncoated areas of the tank internal surfaces inspected for corrosion. These activities will detect loss of material. Section 4.15 of Appendix B of the LRA indicates that the applicant is committed to incorporating these new activities into existing preventive maintenance procedures prior to the end of the initial 40-year license term for ANO-1.

- Auxiliary Building Heating and Ventilation: The preventive maintenance activities listed below currently ensure the cooling coils are clean, and detect whether fouling is occurring. The activities will be modified to specifically document an inspection for loss of material. Section 4.15 of Appendix B indicates that these new activities will be incorporated into existing preventive maintenance procedures prior to the end of the initial 40-year license term for ANO-1.

- Repetitive Work Tasks: The repetitive work tasks include the following activities:

  - cleaning and inspection of VEF-24A/B/C/D exhaust ventilation penthouse assemblies and the portions of the intakes that could be wetted by rain
  - cleaning and inspection of VUC-1A/B/C/D cooling coil and housing assemblies
  - cleaning and inspection of VUC-2B/2D cooling coil and housing assemblies
  - cleaning and inspection of VUC-7A/B/C cooling coil and housing assemblies
  - cleaning and inspection of VUC-14A/B/C/D cooling coil and housing assemblies

- Control Room Ventilation System: The preventative maintenance tasks, 2VUC-27A and 2VUC-27B, are currently used to ensure that the cooling coils are clean, and to detect whether fouling is occurring. The tasks will be modified to specifically document an inspection for loss of material. Section 4.15 of Appendix B of the LRA indicates that the applicant is committed to incorporating these new activities into existing preventive maintenance procedures prior to the end of the initial 40-year license term for ANO-1.

The applicant also states that the AMR did not identify any unique aging effects for inaccessible surfaces of components that are subject to visual inspections. Thus, inspection of the surfaces in the accessible areas is a representative sample of both accessible and inaccessible surfaces. When an unacceptable condition or situation is identified for an accessible surface of a component, a determination will be made as to whether the same condition or situation is applicable to other accessible or inaccessible surfaces of the component or other components, and what additional actions need to be taken.

With the exception of the preventive maintenance activities that require component cleaning, there are no preventive or mitigative actions taken, and the staff did not identify the need for such.

[*Parameters Monitored*] The parameters inspected during these activities are typically direct indications of corrosion (e.g., rust, pits, and wall loss), coating defects (e.g., holidays and scratches) and fouling. The staff found the parameters and the associated techniques are acceptable because they are considered standard for this type of application and have been proven to be effective.

[*Detection of Aging Effects and Monitoring and Trending Activities*] In a letter to the NRC dated September 12, 2000, the applicant discusses how the preventive maintenance program detects aging effects for auxiliary system components before there is a loss of function in the following manner:

- Fire Protection System - For the diesel fire pump engine (K5), inspections are used to identify loss of material, cracking, or loss of mechanical closure integrity of air cleaner and hoses, oil filter and oil filter hoses, turbo charger, water pump, or coolant hoses. Inspections of these components are performed at least once every year. These inspections are visual. If aging effects are identified during these inspections, a condition report is prepared and appropriate corrective actions are taken in accordance with the site corrective action program. On the basis of plant operating history, this preventive maintenance activity in concert with the Fire Suppression Water Supply System Surveillance is sufficient to detect aging effects for the diesel fire pump components before there is a loss of function.

- Fuel Oil System - A visual inspection of the internal surface of the safety-related fuel oil tanks is used to determine if loss of material is occurring. These inspections are performed every 10 years. When aging is identified, condition reports are prepared, defects are evaluated, and corrective actions are taken, as necessary, in accordance with the site corrective action program. On the basis of plant operating history, this preventive maintenance activity in concert with the Diesel Fuel Monitoring Program is sufficient to detect aging effects for the safety-related fuel oil tanks before there is a loss of function.

- Auxiliary Building Heating and Ventilation - Inspection of ventilation components indicates if loss of material or fouling is occurring. Visual inspections of these components are performed at least once every two years. When aging is identified, condition reports are prepared, defects are evaluated, and corrective actions are taken, as necessary, which may include monitoring and trending. On the basis of plant operating history, this preventive maintenance activity is sufficient to detect aging effects for these ventilation components before there is a loss of function.

- Control Room Ventilation System - Inspection of 2VUC27A and 2VUC27B indicates if loss of material or fouling is occurring. Visual inspections of these components are performed at least once every two years. When aging is identified, condition reports are prepared, defects are evaluated, and corrective actions are taken, as necessary, which may include monitoring and trending. On the basis of plant operating history, this preventive maintenance activity in concert with Control Room Ventilation Testing is sufficient to detect aging effects for these ventilation components before there is a loss of function.

On the basis of plant operating history, the staff found that the preventive maintenance program scope, inspection technique, and inspection frequency, when applied in conjunction with other AMPs, is capable of detecting aging effects before there is a loss of component-intended function.

[*Monitoring and Trending*] With the exceptions noted in the discussion above, there are no trending actions taken as part of the preventive maintenance program, and the staff did not see a further need for such.

[*Acceptance Criteria*] In a letter to the NRC dated September 12, 2000, the applicant states that the preventive maintenance inspections for auxiliary system components are visual inspections. The acceptance criteria used for these inspections are “no unacceptable visual indications of cracking, loss of material, loss of mechanical closure integrity, or fouling” depending on the component being inspected. If unacceptable visual indications are identified, a condition report is prepared and appropriate corrective actions are taken, which may include additional monitoring or trending. These preventive maintenance activities will be monitored for effectiveness via ongoing maintenance assessment activities and enhancements will be made if warranted.

The staff found the acceptance criteria for the visual inspections performed as part of the preventive maintenance program to be reasonable based upon standard industry practice.

[*Operating Experience*] The ANO-1 history of successful operation demonstrates that visual inspections have been effective in managing the effects of aging on SCs. Specifically, visual inspections have been shown to identify coating degradation. Thus, based on this experience, the visual inspections performed during component walk-downs will be effective in the future for verifying coating integrity, especially since the inspections incorporate a proven monitoring technique, acceptance criteria, corrective actions, and administrative controls from existing programs and procedures.

The staff found the operating experience supports the attributes of the preventive maintenance program activities. On the basis of plant operating history, the staff found that the preventive maintenance program scope, inspection technique, and inspection frequency, when applied in conjunction with other AMPs, provides reasonable assurance that the preventive maintenance program will manage the applicable aging effects consistent with the CLB for the period of extended operation.

Buried Pipe Inspection

The applicant credits a newly developed buried piping inspection program to manage the loss of material due to external surface corrosion of buried carbon steel piping and valves in the fuel oil, and service water systems. The aging effect on the surfaces of the pipe results from a loss of the protective coating. The staff's evaluation of the buried pipe inspection program is provided in Section 3.3.1 of this SER.

Oil Analysis

The applicant credits the oil analysis program for the components in the auxiliary systems that are in contact with lubricating oil. The staff's evaluation of the oil analysis program is provided in Section 3.3.1 of this SER.

Spent Fuel System

In the LRA, Table 3.4-1, the applicant identifies five AMPs and activities to address the applicable aging effects of the spent fuel system components. These programs and activities are described in Appendices A and B, and are evaluated in the following paragraphs:

Primary Chemistry Monitoring

The applicant credits the primary chemistry monitoring program to manage cracking of the spent fuel system stainless steel liner plate, gates and racks, piping, valves, fuel transfer tube, and blind flanges caused by exposure to borated water. The staff's evaluation of the primary water chemistry program is provided in Section 3.3.1 of this SER.

Spent Fuel Pool Level Monitoring

The applicant credits an existing spent fuel pool level monitoring program to manage cracking in the stainless steel liner plate exposed to chlorides in the external concrete.

[*Program Scope*] The scope of this program includes the stainless steel liner. The staff found this scope acceptable.

[*Preventive/Mitigative Actions*] There is no preventative or mitigative actions associated with this program nor does the staff see a need for such activities.

[*Parameters Monitored*] The applicant states that in conjunction with a new program, (spent fuel pool monitoring described in Section 3.7 of Appendix B to the LRA), the spent fuel pool level monitoring program provides reasonable assurance that through-wall cracking of the liner plate will be detected in a timely manner. The new program includes monitoring of trench drains to detect liner leakage. The licensee records the spent fuel pool level once per shift. A decrease in the spent fuel pool level without any other explanation could possibly indicate leakage because of the presence of a through-wall crack in the spent fuel liner.

[*Detection of Aging Effects*] The spent fuel pool level monitoring program can detect cracking in the spent fuel liner plate only after it becomes through-wall. Because spent fuel pool level is monitored each shift, through-wall cracking will be detected in a timely manner before leakage becomes excessive.

[*Monitoring and Trending*] The licensee records the spent fuel pool level once per shift. In addition, the level is continuously monitored by the low-level alarm feature. No other monitoring or trending activities are associated with this program.

[*Acceptance Criteria*] The applicant states that any unacceptable drop in the pool level would require implementation of corrective actions. The applicant also states that the acceptance criterion for the spent fuel pool level is supplied in site procedures.

[*Operating Experience*] The applicant states that successful operating experience with leakage monitoring of other components indicates that the spent fuel pool monitoring activity will be effective for detecting a through-wall crack in the spent fuel pool liner before the leakage becomes excessive. The applicant states that cracking has been found in the ANO-1 refueling pool liner plate, and that welding of the new permanent seal plate caused the previously existing cracks to propagate.

ASME Section XI ISI-IWD (pressure tests)

The applicant credits the ASME Section XI ISI-IWD inspection program to manage loss of material in carbon steel bolting and external valve parts, and loss of mechanical closure integrity for carbon steel bolting, exposed to borated water. The staff's evaluation of the ASME Section XI ISI-IWD program is provided in Section 3.3.1 of this SER.

Boric Acid Corrosion Prevention

The applicant credits the boric acid corrosion prevention program to manage loss of material and loss of mechanical closure integrity of carbon steel bolting and external valve parts exposed to borated water. The staff's evaluation of the Boric Acid Corrosion Prevention program is provided in Section 3.3.1 of this SER.

Reactor Building Leak Rate Testing

The applicant credits the reactor building leak rate testing program to manage loss of material and loss of mechanical closure integrity of carbon steel bolting and external valve parts exposed to borated water. The staff's evaluation of the Reactor Building Leak Rate Testing program is provided in Section 3.3.1 of this SER.

Conclusion

On the basis of the AMP review described above, the staff finds that there is reasonable assurance that the effects of aging applicable to the spent fuel pool system will be adequately managed so that the intended functions of the spent fuel pool system will be maintained consistent with the CLB for the period of extended operation.

Fire Protection System

In the LRA, Table 3.4-2, the applicant identifies six AMPs and activities to address the applicable aging effects of the fire protection system components. These programs and activities are described in Appendices A and B, and are evaluated in the following paragraphs:

Fire Suppression Water Supply System Surveillance Program

The applicant credits the fire suppression water supply system surveillance program to manage the loss of material, cracking, a loss of mechanical closure integrity of components and fouling of heat exchangers. The applicant describes this program in Appendix B of the LRA, Section 4.8.3, "Fire Suppression Water Supply System Surveillance."

[*Program Scope*] The scope of this program includes the major components of the fire protection system. The staff found this scope acceptable.

[*Preventive/Mitigative Actions*] There are no preventive or mitigative actions, and the staff does not see a need for any such activities.

[Parameters Monitored] The licensee performs periodic flow and functional tests and flushes the system. The applicant states that these tests and flushing of the system ensure the pumps are capable of starting and supplying the required flow rate, and ensure that heat exchangers for the fire pumps operate as required. Flushing and flow-testing of the system helps to ensure that flow is not blocked from the buildup of corrosion products or fouling. In addition, the applicant states that the components observed during testing of the diesel fire pump allow for identification and correction of aging effects. In addition, the ability to readily detect system leakage also addresses aging effects for the fire protection system piping.

[*Detection of Aging Effects*] The applicable aging effects are detected using performance testing and visual examinations for leakage. The frequency of these tests and visual examinations are based on the type of components and are implemented by plant procedures. The staff found that the methods for detecting the effects of aging are adequate because leakage, corrosion products, or other residue on the external surfaces of components can be detected using performance monitoring and visual inspection, and can be used to identify a loss of material, cracking, and a loss of mechanical closure integrity.

[*Monitoring and Trending*] In a letter to the NRC dated November 2, 2000, the applicant states that corrosion coupons are located in the service water system. The service water system and fire protection systems both take suction from the same water source. This is the only monitoring and trending aspect to this program, and the staff found this acceptable.

[*Acceptance Criteria*] The acceptability of surveillance results is determined in accordance with site procedures for each of the activities described in the *"parameters monitored"* above. Implementation of this program using the criteria maintained in site procedures is determined to be adequate on the basis of a demonstration supported by operating history.

[*Operating Experience*] Although not formally credited as part of the fire suppression water supply system surveillance aging management activities, the applicant states that they have the ability to readily detect system leakage during routine operation to address aging of the fire protection system piping. Between surveillance tests, the piping remains pressurized and if leakage from the system piping exceeds the capacity of the jockey pump, then decreasing system pressure will start the electric motor-driven main fire pump. The automatic starting of the fire pump will be indicated in the control room and actions will be taken to determine and correct the cause. In addition, plant walk-downs routinely completed by operations and system engineering personnel would likely detect leakage long before approaching the capacity of the jockey fire pump. The applicant states that anytime a repair or replacement activity is required to address an aging effect that could impact the intended function of the fire protection system, the need for changes in surveillance and inspection activities is assessed. The results of such assessments done under the corrective action program are factored into any needed changes to these programs, including the fire suppression water supply system surveillance. This formal process of continuous feedback ensures that the aging management activities are providing the needed level of protection from potential system loss of intended function. The staff concurs that these tests demonstrate that the components are able to perform their intended functions.

Fire Water Piping Thickness Evaluation Program

The applicant credits the fire water piping thickness evaluation program to manage the loss of material for carbon steel and cast iron piping. The applicant describes this program in Appendix B of the LRA, Section 4.8.5, “Fire Water Piping Thickness Evaluation.”

[*Program Scope*] The applicant states that the scope of this program includes the fire water system piping fabricated of cast iron and carbon steel, which the staff found acceptable.

[*Preventative/Mitigative Actions*] There are no preventive or mitigative actions, and the staff does not see a need for any such activities.

[*Parameters Monitored*] Nondestructive (ultrasonic) thickness measurements are used to detect the deepest pits and average general corrosion through the piping wall. Ultrasonic examination techniques, when applied properly, are capable of detecting minute losses of material in the subject piping. The staff found this examination method acceptable for detecting the loss of material in cast iron and carbon steel.

[*Detection of Aging Effects*] An internal loss of material due to pitting and general corrosion is a primary concern in fire protection system piping exposed to raw water. In a letter to the NRC dated November 2, 2000, the applicant states that the preferred method of examination was to perform full wall thickness measurements via automated equipment. This type of ultrasonic scanning method produces a 100 percent wall thickness profile of the selected area. The applicant further stated that a grid technique is sometimes employed when the automated equipment cannot be used, and that point measurements are not used except to pinpoint the exact area of a suspected defect. The staff found the full wall thickness scans, supplemented by grid measurements, to be an effective ultrasonic method to detect any significant wall losses that might compromise the pressure boundary of this system.

[*Monitoring and Trending*] The frequency and locations for inspection are determined by the ANO-1 fire protection system engineer. The system engineer will take into account wall thickness measurements from, and elapsed time since previous inspections, inspection results from nearby piping and consequences of failure when determining these locations and frequencies. In a letter to the NRC dated November 2, 2000, the applicant states that examinations result not meeting the acceptance standards for wall thickness minimums are evaluated by design engineering for corrective action, and determination of continued service. Typically, an evaluation for continued service will require a recommendation for continued thickness monitoring, and include consideration of further degradation prior to the next inspection. The applicant states that trending is not performed since each evaluation considers expected degradation until repair or the next inspection is performed. The staff found this approach acceptable.

[*Acceptance Criteria*] The applicant states that the acceptability of ultrasonic examination results for each inspection location is determined in accordance with site procedures. These acceptance criteria will vary depending upon the type of component, but are intended to ensure the applicable pipe code minimum wall thickness is maintained. If minimum wall thickness is not violated, the staff concludes that the piping will perform its pressure boundary function.

[*Operating Experience*] Ultrasonic thickness examinations have indicated that pitting corrosion of fire protection system piping is an ongoing degradation mechanism at ANO-1. The applicant further states that pipe repairs resulting from excessive pipe wall-thinning have occurred. In a letter to the NRC dated November 2, 2000, the applicant further describes instances of localized corrosion that caused piping leaks. All were the result of pitting, and no general wall thinning was discovered that might impact the loss of system function. All of these leaks were considered when determining inspection locations and frequencies. Since the program started in 1998, ultrasonic inspections have identified 12 other locations with aging that required engineering evaluation, and were determined to require continued monitoring. On the basis of the applicant's response, the staff found that ultrasonic inspection results are being appropriately considered for the ongoing development of this program.

Reactor Building Leak Rate Testing

The applicant credits the reactor building leak rate testing program to manage the loss of material in fire protection system components exposed to raw water. The staff's evaluation of the reactor building leak rate testing program is provided in Section 3.3.1 of this SER.

Maintenance Rule

The applicant credits the Maintenance Rule program to manage the loss of material in fire protection system components exposed to external-ambient environment. The staff's evaluation of the Maintenance Rule program is provided in Section 3.3.1 of this SER. See the discussion of the Maintenance Rule program in the section above.

Preventive Maintenance

The applicant credits the preventive maintenance program to manage the loss of mechanical closure integrity, loss of material, and cracking in fire protection system components exposed to gas-air environments. Refer to the preventive maintenance program discussion in the section above.

Service Water Chemical Control

The applicant takes credit for the service water chemical control program to manage the loss of material for carbon steel, cast iron, stainless steel, bronze and brass piping and valves, and cracking in stainless steel, brass and bronze components exposed to a raw water environment. The applicant describes this program in Appendix B of the LRA, Section 4.6.5, “Service Water Chemical Control.”

[*Program Scope*] The applicant states that the scope of this program includes the fire protection system components exposed to raw water, which the staff found acceptable.

[*Preventative/Mitigative Actions*] The applicant uses chemical corrosion inhibitors and biocides to reduce the corrosion rates of materials exposed to raw water. The staff concludes the applicant’s raw water chemical control program, when properly administered, contributes to managing the aging effects of loss of material and cracking. However, this program, as a supplement to the larger service water integrity program, which is discussed below with the service water system, reduces the potential for internal corrosion, and does not alone, ensure the intended function of fire protection components.

[*Parameters Monitored*] The applicant monitors concentrations of corrosion inhibitors to determine these chemicals are maintained within effective ranges. General corrosion rates are also monitored by using corrosion coupons, and when the service water system is dewatered, i.e., for maintenance activities, internal visual inspections are performed. The applicant also states that the service water chemical treatment using biocides can contribute to corrosion. However, improved use of corrosion inhibitors minimizes this corrosion. Monitoring is performed to evaluate the effectiveness of the treatment and to make adjustments as needed to effectively manage fouling and corrosion. The injection of corrosion inhibitors has significantly reduced piping corrosion rates in the system. The combination of biocides and corrosion inhibitors in the service water system have been effective in managing biological fouling and in reducing corrosion. The staff found the monitoring of these parameters acceptable in managing the aging effects of corrosion in the applicable components.

[*Detection of Aging Effects*] This is a water chemistry program, and as such, the staff concludes that detection of aging effects is not directly applicable or necessary, because the program is used in conjunction with other programs that should detect the applicable aging effects.

[*Monitoring and Trending*] In a letter to the NRC dated November 2, 2000, the applicant states that corrosion coupons are located in the service water system. The service water system and fire protection systems both take suction from the same water source. This is the only monitoring or trending aspect to this program, and the staff found this acceptable

[*Acceptance Criteria*] The applicant uses acceptance criteria for concentration levels of biocide and corrosion inhibitors that are based on EPRI guidelines and experience at ANO-1. Factors that influence these variable criteria are the sampling schedule and locations, and existing water chemistry and biological conditions in Lake Dardanelle and the ECP, taking into account actual plant operating conditions at that time. The staff found that by using the standards set forth in the EPRI guidance, and an understanding of existing biological conditions and chemistry of the available sources of intake water, the applicant will be able to define appropriate acceptance criteria for reducing the corrosive effects of the raw water in the fire protection system.

[*Operating Experience*] Three instances of fire protection system pipe leaks were due to localized pitting. The nondestructive examination of the affected areas indicated that the degradation was not generalized pipe wall thinning and that the intended function of the system was not lost. Corrective actions for these leaks included development and implementation of the Fire Water Piping Thickness Evaluation Program. The last of the three leaks was found after implementation of the program. It also involved localized pitting that was not generalized wall thinning and did not result in loss of system function. All three of these leaks have been taken into consideration for subsequent inspections of the system piping. Since 1998, when the Fire Piping Thickness Evaluation Program began, approximately 12 ultrasound inspection results showed indication of piping degradation that required engineering evaluation. All of these areas were determined to be acceptable for continued operation, but each of these locations was specified for continued monitoring. In summary, the fire protection system pipe wall thickness monitoring has identified degradation in local areas of the system. On the basis of these findings, piping has been determined to be acceptable for continued operation, but continued monitoring is being required. Loss of function due to these aging effects has not occurred and the monitoring program provides reasonable assurance that aging effects will not lead to a loss of function. The staff determined that operating history and improvements in aging management activities of fire protection system piping provided reasonable assurance that the intended function can be maintained for the period of extended operation.

Oil Analysis

The applicant credits the oil analysis program to manage the loss of mechanical closure integrity, and loss of material in fire protection system components in the lube oil portions of the diesel fire pump subsystem. The staff's evaluation of the Oil Analysis program is discussed in Section 3.3.1 of this SER.

Conclusion

On the basis of the AMP reviews provided above, the staff finds that there is reasonable assurance that the effects of aging applicable to the fire protection system will be adequately managed so that the intended functions of the fire protection system will be maintained consistent with the CLB for the period of extended operation.

Emergency Diesel Generator

In the LRA, Table 3.4-2, the applicant identifies four AMPs and activities to address the applicable aging effects of EDG components. These programs and activities are described in Appendices A and B, and are evaluated in the following paragraphs:

Emergency Diesel Generator Testing and Inspections Program

The applicant takes credit for the EDG testing and inspection program to manage the loss of material, loss of mechanical closure integrity, cracking, and fouling of heat exchangers. The applicant describes this program in Appendix B of the LRA, Section 4.21.5, “Emergency Diesel Generator Testing and Inspections.”

[*Program Scope*] The scope of this program includes the components of the EDG system and associated support components. The staff found this scope acceptable.

[*Preventive/Mitigative Actions*] The applicant does not discuss preventive or mitigative actions for the EDG testing and inspections program and the staff does not see a need for any.

[*Parameters Monitored*] Pressure drop tests are performed on the aftercoolers to detect fouling of associated heat exchangers. Checks for exhaust leaks, cooling water leaks, or lube oil leaks while the engine is running are performed to detect loss of bolted closure integrity, cracking, or loss of material if progressed to the point of leakage. Other inspections include the disassembly and examinations of the interior of the lube oil cooler, the exhaust assembly, and the air start components to allow for detection of loss of material, loss of integrity, and fouling. The staff found that the EDG test and inspection activities (in combination with other AMPs) can be used to demonstrate that these components are able to perform their intended functions and that the parameters monitored or inspected are adequate.

[*Detection of Aging Effects*] The applicable aging effects are detected using performance testing and visual examinations for leakage. The frequency of these tests and visual examinations are managed by plant procedures. The staff found that the methods for detecting the effects of aging are adequate (in combination with the other three programs) because they can detect a loss of material, cracking, and a loss of mechanical closure integrity.

[*Monitoring and Trending*] There are no trending activities associated with this program, and the staff did not identify a need for trending.

[*Acceptance Criteria*] The applicant states that the acceptability of testing and inspections are documented in operation and inspection procedures for the EDG system. On the basis of the nature of the testing performed, the staff found this acceptable.

[*Operating Experience*] The applicant states that operational testing of the emergency diesel generators has proven effective in identifying loss of mechanical closure integrity for the air, lube oil, and fuel oil systems components.

Auxiliary Systems Water Chemistry Monitoring Program

The applicant credits the auxiliary system water chemistry program to manage the loss of material, cracking and fouling in EDG system components exposed to a treated water environment. The staff’s evaluation of the water chemistry program is discussed in Section 3.3.1 of this SER.

Maintenance Rule Program

The applicant takes credit for the Maintenance Rule program to manage loss of material and loss of mechanical closure integrity. See previous discussion relating to the Maintenance Rule program in the generic aging management section of this report.

Oil Analysis Program

The applicant credits the oil analysis program to manage loss of material and cracking in components of the lube oil subsystem of the EDG system. The staff's evaluation of the oil analysis program is discussed in Section 3.3.1 of this SER.

Conclusion

On the basis of the AMP reviews provided above, the staff finds that there is reasonable assurance that the effects of aging applicable to the EDG system will be adequately managed so that the intended functions of the EDG system will be maintained consistent with the CLB for the period of extended operation.

Auxiliary Building Sump and Reactor Building Drains

In the LRA, Table 3.4-2, the applicant identifies nine AMPs and activities to address the applicable aging effects of the auxiliary building sump and reactor building drains system components. These programs and activities are described in Appendices A and B, and are evaluated in the following paragraphs:

Reactor Building Leak Rate Testing and Local Leak Rate Testing

The applicant credits the reactor building leak rate testing and local leak rate testing programs to manage the loss of material in the auxiliary building sump and reactor building drains system components exposed to borated, treated, and raw water. The staff's evaluation of the reactor building leak rate testing program is provided in Section 3.3.1 of this SER.

Primary Chemistry Monitoring

The applicant credits the primary chemistry monitoring program to manage the loss of material in the auxiliary building sump and reactor building drains system components exposed to treated and borated water. The staff's evaluation of the primary chemistry monitoring program is provided in Section 3.3.1 of this SER.

Secondary Chemistry Monitoring

The applicant credits the secondary chemistry monitoring program to manage the loss of material in auxiliary building sump and reactor building drains system components exposed to treated and borated water. The staff's evaluation of the secondary chemistry monitoring program is provided in Section 3.3.1 of this SER.

Boric Acid Corrosion Prevention

The applicant credits the boric acid corrosion prevention program to manage the loss of material and loss of mechanical closure integrity in auxiliary building sump and reactor building drain system components whose external surfaces can be exposed to borated water. The staff's evaluation of the boric acid corrosion prevention program is provided in Section 3.3.1 of this SER.

RCP Oil Leakage Collection System Inspection

The applicant takes credit for the reactor coolant pump oil collection system inspections to manage the loss of material and loss of mechanical closure integrity for carbon steel piping, tanks, bolts, and valves exposed to borated water and oil environments. The applicant describes this program in Appendix B of the LRA, Section 4.8.7, "Reactor Coolant Pump Oil Collection System Inspection."

[*Program Scope*] The applicant states that the reactor coolant pump oil collection system inspection program ensures the integrity of the reactor coolant pump oil collection system. In a letter to the NRC dated June 9, 2000, the applicant confirmed that the shrouds, drip pans, accessible piping, collection tanks, and spray protection in the reactor coolant pump oil collection system are included within the scope of this inspection program. The staff found the scope of the program to be adequate.

[*Preventive/Mitigative Actions*] There are no preventive or mitigative actions taken as part of this program, and the staff did not identify the need for such actions.

[*Parameters Monitored*] The applicant plans to perform a visual inspection of the shrouds, drip pans, dammed areas, accessible piping, collection tanks, and spray protection portions of the reactor coolant pump oil collection system. The applicant states that guidance for this inspection is contained in a site procedure. This visual inspection verifies that there is no accumulation of oil outside the collection system which indicates that the oil is draining and the system is sound. The staff found that these parameters are adequate to indicate an anomaly with the function of this system.

[*Detection of Aging Effects*] The applicant identifies a combination of an adequate scope of inspections, the use of qualified inspection techniques, and adequate inspection timing, therefore the staff concludes that aging effects can be detected before there is a loss of intended function.

[*Monitoring and Trending*] There are no trending activities associated with the reactor coolant pump oil collection system program and the staff did not identify a need for trending.

[*Acceptance Criteria*] The applicant states that the acceptance criteria for the reactor coolant pump oil collection system inspections are based on guidance provided in site procedures, and that there should be no accumulation of oil outside the collection system. In a letter to the NRC dated September 12, 2000, the applicant states that the reactor coolant pump oil collection system inspection program provides for visual inspection of the reactor coolant pump oil collection system. In accordance with the CLB, the auxiliary building sump and reactor building drains system is a non-seismic system. Therefore, a visual inspection to verify that the oil is draining and the system is sound, is adequate for managing aging effects during the period of extended operation. The staff found that the acceptance criteria for this visual inspection are reasonable based upon standard industry practice.

[*Operating Experience*] In Appendix B of the LRA, Section 4.8.7, the applicant discusses operating experience. The applicant states that visual inspections have been effective in identifying deficiencies such as abnormal accumulations of oil, or oil not being collected properly. The applicant also states that the operating history review performed during the AMR did not identify any corrosion issues with the system.

ASME Section XI ISI-Augmented Inspections

The applicant credits the ASME Section XI ISI - augmented inspection program to manage the loss of material, cracking and a loss of mechanical closure integrity in auxiliary building sump and reactor building drains system stainless steel, brass, bronze, and admiralty valves and piping exposed to treated, borated, and raw water. The staff's evaluation of the ASME Section XI ISI - augmented inspection program is provided in Section 3.3.1 of this SER.

Reactor Building Sump Closeout Inspection

The applicant takes credit for the reactor building sump closeout inspection to manage the loss of material and cracking of the screen in the auxiliary building sump and reactor building drains system exposed to lubricating oil, and borated, treated and raw water environments. The applicant describes this program in Appendix B of the LRA, Section 4.17, "Reactor Building Sump Closeout Inspection."

[*Program Scope*] In the LRA, Table 3.4-4, the applicant states that the intended function of the screens is to keep debris out of the sump, and that the purpose of the reactor building sump closeout inspection program is to detect damage of the sump components, and remove any foreign objects that could impede suction from the sump. The applicant also states that the aging effects addressed by the reactor building sump closeout inspection are loss of material for the carbon steel components and cracking for stainless steel components due to the presence of borated water.

In the LRA, Table 3.4-4, the applicant indicates that the scope of the reactor building sump closeout inspection program is limited to the screen in the auxiliary building sump and reactor building drains system. In Appendix B of the LRA, Section 4.17, the applicant states that the scope of the reactor building sump closeout inspection program is the reactor building sump, the area surrounding the sump, screening materials, and equipment and structural components inside the sump. In a letter to the NRC dated June 9, 2000, the applicant confirms that the scope of the program described in Section 4.17 of Appendix B is correct. The grouping of equipment and components inside the sump includes both carbon and stainless steel components. The staff found that the scope of the program is adequate for the inspected components.

[*Preventive/Mitigative Actions*] There are no preventive or mitigative actions taken as part of this program, and the staff did not identify the need for such actions.

[*Parameters Monitored*] In Appendix B of the LRA, Section 4.17, Appendix B of the LRA, the applicant states that the inspection activities performed on the exterior and interior surfaces of the sump include a visual inspection for evidence of structural distress, corrosion, excessive rust, significant physical degradation, obvious loose or missing bolts, excessive hatch gap, or tears in sump screens. During limited scope outages, or when controls for foreign material exclusion area are in place, sump screens are not unbolted, and only exterior surfaces of the sump are inspected. The staff found that the applicant has adequately defined the parameters of an inspection needed to detect loss of material and cracking of reactor building sump components.

[*Detection of Aging Effects*] The applicant identifies a combination of an adequate scope of inspections, the use of qualified inspection techniques, and adequate inspection timing, therefore the staff concludes that aging effects can be detected before there is a loss of intended function. With respect to the inspection timing, the applicant states that this inspection is performed at the end of each refueling outage, or at least once every 24 months. The staff found that the methods and timing are adequate for detecting the applicable aging effects prior to loss of intended function for the components that are within the scope of the reactor building sump closeout inspection program.

[*Monitoring and Trending*] In a letter to the NRC dated June 9, 2000, the applicant states that there are no trending activities under the reactor building sump closeout inspection. The staff did not identify a need for trending.

[*Acceptance Criteria*] The applicant states that the acceptance criteria for the reactor building sump closeout inspection program are based on guidance provided in site procedures. In a letter to the NRC dated June 9, 2000, the applicant states that the visual inspection of the screens is adequate to manage loss of material and cracking. The inspection verifies the screens show no evidence of physical damage that could allow foreign objects to enter the sump. The inspection also looks for structural distress, corrosion, signs of rust, physical degradation, or tears in the screen. This program is similar to the Containment Emergency Sump Inspection Procedures credited for Calvert Cliffs containment sumps, and approved by the NRC in NUREG-1705, Safety Evaluation Report Related to the License Renewal of Calvert Cliffs Nuclear Power Plant, Units 1 and 2.

The staff concurs that the reactor building sump inspection program is acceptable for identifying loss of material and cracking of the reactor building sump SCs before loss of intended function.

[*Operating Experience*] In Appendix B of the LRA, Section 4.17, the applicant discusses operating experience. The applicant states that during the 1998 inspection, wetted portions of the sump were inspected for indications of aging effects. Very little evidence of corrosion was noted on structural members, and light boric acid deposits were removed, which had formed due to a leaking valve above the sump. No service-induced, or environmentally-induced deficiencies were found with respect to carbon steel or stainless steel valve parts. The carbon steel portions of the divider plate exhibited corrosion. Flued heads showed no signs of service- or environmentally-induced pitting, cracking, or corrosion. In a letter to the NRC dated June 9, 2000, the applicant discusses a review of reactor building sump closeout inspection reports. The applicant states that no component repairs or replacements were required. Therefore, the applicant concluded that no program enhancements had been required based on operating experience.

Conclusion

On the basis of the AMP reviews provided above, the staff finds that there is reasonable assurance that the effects of aging applicable to the auxiliary building sump and reactor building drains system will be adequately managed so that the intended functions of the auxiliary building sump and reactor building drains system will be maintained consistent with the CLB for the period of extended operation.

AAC Diesel Generator

In the LRA, Table 3.4-2, the applicant identifies four AMPs to address the applicable aging effects of the AAC diesel generator components. These programs and activities are described in Appendices A and B, and are evaluated in the following paragraphs:

Alternate AC Diesel Generator Testing and Inspections Program

The applicant takes credit for the AAC diesel generator testing and inspection program to manage the loss of material, loss of mechanical closure integrity, cracking and fouling for heat exchangers exposed to gas-air, treated water, and external-ambient environments. The applicant describes this program in Appendix B of the LRA, Section 4.2, "AAC Diesel Generator Testing and Inspections."

[*Program Scope*] The applicant includes the AAC diesel generator system and associated support components in the scope of the AAC diesel generator testing and inspection program. The staff found the scope of the program acceptable.

[*Preventive/Mitigative Actions*] The applicant states that inspections and preventive maintenance performed in accordance with the manufacturer's recommendations would detect loss of material, leakage, and fouling. The staff agrees.

[*Parameters Monitored*] The applicant performs periodic operability testing and surveillance activities on the AAC diesel generator system. The applicant states that such tests and surveillance provide one method for managing aging effects since pressure boundary integrity and heat transfer functions are verified. The applicant also states that other inspections are periodically performed in accordance with the manufacturer's recommendations, and that these inspections are a second method for managing aging effects because the inspections could detect loss of material, leakage, and fouling. The staff found that these tests and inspections (in combination with the other three programs) demonstrate that the AAC generator system is able to perform the intended functions, and that the parameters monitored or inspected are adequate.

[*Detection of Aging Effects*] The applicant states that the effects of aging are detected using operability testing and surveillance. The staff found that the AAC diesel generator testing and inspections program can detect a loss of material, cracking, and a loss of mechanical closure integrity (by appearance of leakage, corrosion products, or other residue on the external surfaces of components) and fouling of the components (by increased temperature). The staff also finds that this program must be used in combination with the other programs for detecting aging effects for the AAC diesel generator system.

[*Monitoring and Trending*] The AAC diesel generator is tested quarterly and once every 18 months in accordance with ANO commitments to station blackout requirements. The staff found this acceptable,

[*Acceptance Criteria*] The applicant states that the acceptance criteria of testing and inspections are documented in site procedures and vendor manuals for the AAC generator system for each of the activities described in the "Parameters Monitored," above. The staff found that these acceptance criteria adequately address (in combination with the other three programs) the aging effects of loss of material, cracking, loss of mechanical closure integrity, and fouling on the basis of a demonstration supported by operating history.

[*Operating Experience*] The applicant states that operational testing of the AAC generator system has identified loss of mechanical closure integrity and leaks in various components. The applicant also stated that these conditions identified through system testing and inspections are typical of those expected due to aging effects.

Auxiliary Systems Water Chemistry Monitoring Program

The applicant credits the auxiliary system water chemistry monitoring program to manage the loss of material, fouling, cracking, and loss of mechanic closure integrity in the AAC diesel generator system components exposed to treated water. The staff's evaluation of the auxiliary system water chemistry monitoring program is provided in Section 3.3.1 of this SER.

Maintenance Rule Program

The applicant credits the Maintenance Rule program to manage the loss of material in AAC diesel generator system components exposed to exterior-ambient, and gas-air environments. The staff's evaluation of the Maintenance Rule program is provided in Section 3.3.1 of this SER. See previous discussion in generic AMPs section of this report.

Oil Analysis Program

The applicant credits the oil analysis program to manage the loss of material and fouling in AAC diesel generator system components exposed to lubricating oil. The staff's evaluation of the oil analysis program is provided in Section 3.3.1 of this SER.

Conclusions

On the basis of the AMP reviews provided above, the staff finds that there is reasonable assurance that the effects of aging applicable to the AAC diesel generator system will be adequately managed so that the intended functions of the AAC diesel generator components will be maintained consistent with the CLB for the period of extended operation.

Halon

In the LRA, Table 3.4-6, the applicant identifies two AMPs to address the applicable aging effects of the Halon system components. In a letter to the NRC dated September 12, 2000, the applicant provides a correction to Table 3.4-6 to identify that the control room Halon fire system inspection should be credited with managing the loss of material and cracking of the carbon steel discharge tube assemblies, the stainless steel flexible tubing, and fittings. These programs are described in Appendices A and B, and are evaluated in the following paragraphs:

Control Room Halon Fire System Inspection

[*Program Scope*] The applicant includes all the components of the Halon system in the scope of this program. The staff found the scope of the program acceptable.

[*Preventive/Mitigative Actions*] There are no preventive or mitigative actions, and the staff does not see a need for these actions.

[*Parameters Monitored*] The applicant performs inspections to ensure system operability. As part of these inspections the pressurized portions of the system are leak tested by weighing the Halon cylinders to determine whether Halon losses are occurring. Inspections verify that nitrogen pressure is adequate and that frequently operated components have not experienced cracking or loss of material. The staff found that the parameters monitored are adequate for detection of loss of material and cracking.

[*Detection of Aging Effects*] Procedural verification is used to ensure correct installation of components, and to detect loss of material of cracking. The staff concludes that control room Halon system inspections are adequate to ensure that loss of material and cracking will be detected before loss of the Halon system function.

[*Monitoring and Trending*] The inspection program was developed along the guidelines of NFPA Standard 12A, and inspections are performed at least once every 6 months. Because the control room Halon system components are not subjected to a corrosive environment, the frequency and level of detail of inspections should provide reasonable assurance that degradation will be detected and corrected in a timely manner.

[*Acceptance Criteria*] The applicant states that the acceptance criteria for minimum Halon cylinder weights, minimum Halon and nitrogen gas pressures, and maximum nitrogen gas pressure are contained in the inspection procedure. Additionally, procedures require inspection of components that are disassembled. The staff concludes that the acceptance criteria are appropriate.

[*Operating Experience*] Previous inspections have identified loss of nitrogen or Halon pressure, and loss of Halon gas, primarily due to removal of gauges for calibration, installation of test gauges, and seal leakage in the cylinder control heads. In order to minimize gas losses, the applicant has permanently installed test gauges. The applicant’s review of industry and site operating experience indicates no other types of degradation to Halon system components. Therefore, the staff concludes that the control room Halon fire system inspection is effective in managing the loss of material and cracking.

Maintenance Rule Program

The applicant credits the Maintenance Rule program to manage the loss of material and cracking in Halon system components exposed to exterior-ambient environments. The staff's evaluation of the Maintenance Rule program is provided in Section 3.3.1 of this SER. See previous discussion in generic AMPs section of this report.

Conclusion

On the basis of the AMP reviews provided above, the staff finds that there is reasonable assurance that the effects of aging applicable to the Halon system will be adequately managed so that the intended functions of the Halon system will be maintained consistent with the CLB for the period of extended operation.

Fuel Oil

In the LRA, Table 3.4-7, the applicant identifies five AMPs and activities to address the applicable aging effects of the fuel oil system components. These programs and activities are described in Appendices A and B, and are evaluated in the following paragraphs:

EDG Testing and Inspection Program

The applicant credits the EDG testing and inspection program to manage the loss of mechanical closure integrity in carbon steel, stainless steel, brass, bronze, copper, admiralty and cast iron components, as well as loss of material in carbon steel piping, valves, filters, pumps, tubing, and tank components in fuel oil system exposed to fuel oil.

[*Program Scope*] The applicant includes all the fuel oil system components except tanks and heat exchanger tubes in the EDG testing and inspection program. The staff found the scope of the program acceptable.

[*Preventive/Mitigative Actions*] The applicant checks bolt torque on a large number of the fuel oil system components during each inspection associated with the EDG testing and inspection program. The staff found this acceptable.

[*Parameters Monitored*] The applicant inspects bolted closures that are exposed to high engine vibrations for fuel oil leaks while the engine is running. The staff found this acceptable.

[*Detection of Aging Effects*] Inspections for fuel oil leaks would detect loss of material in the carbon steel components that had progressed to the point of allowing leakage. Inspection of bolt torques identifies and corrects loss of mechanical closure integrity. In a letter dated November 2, 2000, the applicant states that other inspection activities are conducted as part of the EDG testing and inspection program. For example, several components (e.g., air start components, exhaust manifold, lube oil cooler, etc.) are removed or disassembled and inspected at frequencies specified by the manufacturer (usually on an 18-month schedule). In addition, the EDG is performance tested, which includes vibration, temperature, and pressure monitoring, and inspected for signs of leakage or degradation. These inspections are performed in accordance with manufacturers' recommendations.

[*Monitoring and Trending*] The applicant performs a visual inspection for fuel oil leaks once every 18 months per ANO-1 TS 4.6.1. The staff concludes that the frequency of periodic testing [and checking for leakage] is adequate to ensure that a loss of material in the carbon steel components will be detected prior to an unacceptable loss of pressure boundary integrity.

[*Acceptance Criteria*] The acceptance criteria for testing and inspection are documented in EDG operation and inspection procedures. The acceptance criteria are based on NRC Regulatory Guide 1.108, "Periodic Testing of Emergency Diesel Generator Units Used As Onsite Power Systems at Nuclear Power Plants," and ANO-1 TS 4.6.1. The staff found the acceptance criteria acceptable.

[*Operating Experience*] In a letter to the NRC dated September 12, 2000, the applicant submits additional operating experience information that allowed the staff to determine that the EDG testing and inspection program has been proven effective in identifying loss of material in the carbon steel components. The staff agrees with the applicant that the operational testing of the EDGs has proven effective in identifying loss of mechanical closure integrity of fuel oil system components.

Diesel Fuel Monitoring

The applicant credits the diesel fuel monitoring program to manage the loss of material, fouling and loss of mechanical closure integrity in the fuel oil system components exposed to fuel oil. The staff's evaluation of the diesel fuel monitoring program is provided in Section 3.3.1 of this SER.

Maintenance Rule

The applicant credits the Maintenance Rule program to manage the loss of material in carbon steel piping, valves, filters, pumps, tubing and tanks in fuel oil system components exposed to an external-ambient environment. The staff's evaluation of the Maintenance Rule program is provided in Section 3.3.1 of this SER.

Buried Piping Inspection

The applicant credits the buried piping inspection program to manage the loss of material in fuel oil system carbon steel components exposed to external-buried environments. The staff's evaluation of the buried piping inspection program is provided in Section 3.3.1 of this SER.

Alternate AC Diesel Generator Testing and Inspections

The applicant takes credit for the AAC diesel generator testing and inspection program to manage the effects of aging of loss of material, loss of mechanical closure integrity, cracking, and fouling of heat exchangers as described in Section 4.2, "Alternate AC Diesel Generator Testing and Inspections," of Appendix B of the LRA. Refer to the discussion in Section 3.3.4.3.2, "Managing Loss of Mechanical Closure Integrity," regarding the effects of diesel engine vibration upon this system and the need to include verification of bolt torque to ensure bolted closure integrity.

[*Program Scope*] The applicant includes the fuel oil heat exchanger tubes in the scope of the program. The staff found the scope of the program acceptable.

[*Preventive/Mitigative Actions*] This is a performance monitoring program of heat transfer capacity of the heat exchanger that can allow the detection of fouling prior to a loss of intended function.

[*Parameters Monitored/Inspected*] The applicant states that the periodic testing of the AAC diesel monitors both pressure boundary integrity and heat transfer functions of the heat exchangers. The staff agrees.

[*Detection of Aging Effects*] The applicant states that even a thin layer fouling can adversely impact the heat transfer capacity. The staff agrees that monitoring heat transfer capacity can allow the detection of fouling prior to a loss of intended function.

[*Monitoring/Trending*] The AAC diesel generator is tested quarterly and once every 18 months in accordance with ANO commitments to station blackout requirements as presented in Regulatory Guide 1.155, "Station Blackout." The staff found the testing frequency adequate for detecting fouling before the loss of heat transfer function.

[*Acceptance Criteria*] The applicant states that the acceptance criteria are based on guidance provided in site procedures. The acceptance criteria are based on manufacturer recommendations, and include vibration, temperature, and pressure monitoring.

[*Operating Experience*]   Operating experience associated with operation, testing, and inspection of the AAC diesel generator has not identified aging effects that could cause loss of intended function.

Conclusions

On the basis of the AMP reviews provided above, the staff finds that there is reasonable assurance that the effects of aging applicable to the fuel oil system will be adequately managed so that the intended functions of the fuel oil system will be maintained consistent with the CLB for the period of extended operation.

Instrument Air

In the LRA, Table 3.4-8, the applicant identifies two AMPs to address the applicable aging effects of the instrument air system components. These programs and activities are described in the LRA, Appendices A and B, and are evaluated in the following paragraphs:

Instrument Air Quality Program

The applicant takes credit for the instrument air quality program to manage the loss of material, and cracking in instrument air system components exposed to gas-air environment. The applicant describes this program in Appendix B of the LRA, Section 4.11, "Instrument Air Quality."

[*Program Scope*] The applicant identifies only portions of the instrument air system that are within the scope of license renewal. The applicant includes those portions of the instrument air system that penetrate the reactor building, and the pneumatic accumulators to the valve actuators for three cooling water system valves and two ventilation system dampers. The staff found the scope of the program acceptable because it is consistent with the applicant's AMR.

[*Preventive/Mitigative Actions*] The applicant tests instrument air samples to ensure that the moisture in the air supply is within limits, and that the air is free of contaminants and foreign materials to provide reasonable assurance that corrosion will not occur on the internal surfaces of the instrument air system components. The staff concludes that these actions are acceptable for preventing corrosion of the internal surfaces of the instrument air system.

[*Parameters Monitored*] The instrument air system is monitored for particle size, corrosive contaminants, oil content, and moisture content to prevent degradation of the internal surfaces of the instrument air system. The staff concludes that the monitoring of these parameters is adequate to ensure instrument air quality.

[*Detection of Aging Effects*] Maintaining instrument air quality provides reasonable assurance that the identified internal aging effects will not occur. The staff found this acceptable.

[*Monitoring and Trending*] Instrument air samples are taken quarterly at the outlet of each air dryer, and at the low points of each major piping branch. Trending is performed to monitor the performance of the instrument air system. The staff concludes that the frequency of sampling and the trending of air quality test results is adequate to prevent degradation of the internal surfaces of the instrument air system.

[*Acceptance Criteria*] The applicant states that the acceptability of air quality tests is determined in accordance with "ISA Quality Standard for Instrument Air," ISA S7.3-1975. The staff found these limits acceptable.

[*Operating Experience*] The applicant states that instrument air quality has been maintained and that no age-related failures of instrument air system components subject to an AMR have occurred. Therefore, the staff concludes that the instrument air quality program is effective in managing the aging effects of the internal surfaces of the instrument air system.

Maintenance Rule

The applicant credits the Maintenance Rule program to manage the loss of material in instrument air system components exposed to an external-ambient environment. The staff's evaluation of the Maintenance Rule program is provided in Section 3.3.1 of this SER.

Conclusions

On the basis of the AMP reviews provided above, the staff finds that there is reasonable assurance that the effects of aging applicable to the instrument air system will be adequately managed so that the intended functions of the instrument air system will be maintained consistent with the CLB for the period of extended operation.

Chilled Water

In the LRA, Table 3.4-9, the applicant identifies seven AMPs and activities to manage the applicable aging effects of the chilled water system SCs. These programs and activities are described in the LRA, Appendices A and B, and are evaluated in the following paragraphs.

Reactor Building Leak Rate Testing

The applicant credits the reactor building leak rate testing program to manage the loss of material in chilled water system components exposed to treated water. The staff's evaluation of the reactor building leak rate testing program is provided in Section 3.3.1 of this SER.

ASME Section XI ISI - IWC (pressure tests)

The applicant credits the ASME Section XI ISI - IWC program to manage the loss of material in chilled water system components exposed to treated water. The staff's evaluation of the ASME Section XI ISI - IWC program is provided in Section 3.3.1 of this SER.

Auxiliary Systems Chemistry Monitoring

The applicant credits the auxiliary system chemistry monitoring program to manage the loss of material, cracking and fouling in chilled water system components exposed to treated water environments. The staff's evaluation of auxiliary system chemistry monitoring program is provided in Section 3.3.1 of this SER.

Wall Thinning Inspection

The applicant credits the wall thinning program to manage the loss of material in chilled water system components exposed to treated water. The staff's evaluation of the wall thinning program is provided in Section 3.3.1 of this SER.

Maintenance Rule

The applicant credits the Maintenance Rule program to manage the loss of material in chilled water system components exposed to an external-ambient environment. The staff's evaluation of the Maintenance Rule program is provided in Section 3.3.1 of this SER.

Oil Analysis

The applicant credits the oil analysis program to manage the loss of material in chilled water system components exposed to lubricating oil. The staff's evaluation of the oil analysis program is provided in Section 3.3.1 of this SER.

Heat Exchanger Monitoring

The applicant credits the heat exchanger program to manage the loss of material in chilled water system components exposed to treated water environment. The staff's evaluation of the heat exchanger program is provided in Section 3.3.1 of this SER.

Conclusions

On the basis of the AMP reviews provided above, the staff finds that there is reasonable assurance that the effects of aging applicable to the chilled water system will be adequately managed so that the intended functions of the chilled water system will be maintained consistent with the CLB for the period of extended operation.

Service Water

In the LRA, Table 3.4-10, the applicant identifies five AMPs and activities to address the applicable aging effects of the service water system components. These programs and activities are described in Appendices A and B, and are evaluated in the following paragraphs:

Service Water Integrity

The applicant takes credit for the service water integrity program to manage the loss of material, cracking and fouling of the service water system components exposed to a raw water environment. The applicant describes this program in Appendix B of the LRA, Section 4.19, "Service Water Integrity."

[*Program Scope*] The "Service Water Integrity" program is a multi-parameter program that encompasses chemistry controls, system and component testing, nondestructive examinations, and certain preventive maintenance activities. In a letter to the NRC dated June 9, 2000, the applicant states that performance testing, nondestructive examinations and chemistry control are performed for carbon steel piping, strainers, and valves; stainless steel piping, flow elements, thermowells, and valves; the carbon steel and stainless steel service water pump casings; and the service water side of heat exchangers. Visual inspections are performed on the sluice gates and brass and bronze valves. In a letter to the NRC dated November 2, 2000, the applicant states that corrosion coupons are located in the service water system and that the service water system and fire protection systems both take suction from the same water source. The staff found the scope of the program acceptable as it includes all components subject to an AMR.

[*Preventative/Mitigative Actions*] The applicant identifies several actions intended to reduce the aging effects of loss of material and fouling. These include chemical additions of biocides and corrosion inhibitors, periodic flushing and cleaning of stagnant portions of the system, and periodic cleaning of heat exchanger components to remove fouling. The staff concludes that these actions provide a reasonable basis to expect that the applicable aging effects will be managed during the period of extended operation. However, these actions alone do not ensure the intended function of all components in the service water system. Therefore, they are applied in conjunction with other "Service Water Integrity" program activities. The staff found this acceptable.

[*Parameters]* are listed in Section 4.19 of the LRA. These include flow, pump performance and heat transfer testing, wall thickness measurements and visual inspections, and gate and valve through-leakage tests. In a letter to the NRC dated June 9, 2000, the applicant states, that automated ultrasonic wall thickness examinations are performed at selected locations of carbon steel piping in this system. If applied properly, wall thickness measurements will ensure that a significant loss of material due to corrosion is detected prior to a breach of the pressure boundary. In a letter to the NRC dated June 9, 2000, the applicant lists multiple heat exchangers for which periodic tube-side visual inspections are performed to detect fouling and loss of material. The staff found that the ultrasonic thickness examination, visual inspection, and performance test parameters are adequate to detect aging effects prior to a loss of intended function of applicable service water components.

[*Detection of Aging Effects*] Performance testing and nondestructive examinations are used to manage loss of material, cracking, and fouling in the service water system. These activities were developed in response to NRC Generic Letter 89-13. Because the response to GL 89-13 has been previously reviewed by the staff and these activities fall under the CLB, they are acceptable.

[*Monitoring and Trending*] For ultrasonic thickness mapping, the initial samples were chosen on the basis of potential aging effects, flow conditions, and piping stresses. The applicant uses these wall thickness measurements in conjunction with flow testing data to determine the order of piping replacement since the inception of this program. The applicant states that many of the piping segments had been replaced as a result of these measurements, and that since new piping is in place, fewer sampling locations are now required. Also, recent improvements in the service water chemistry control program have reduced the corrosion potential for the system. The performance testing results are monitored to determine if flow or heat transfer rates are gradually declining. The staff found that the ultrasonic examinations and performance tests are acceptable means for managing loss of material in the service water system.

[*Acceptance Criteria*] The applicant states that acceptance criteria for the system flow tests are “measured flows above minimum flow requirements.” These minimums are adjusted to account for projected degradation between flow tests. Other verification of flows, e.g., reactor building coolers, sluice gates, etc., are based on safety analyses assumed flows and system hydraulic conditions. In the November 2, 2000, letter, the applicant states that the reference to “safety analysis assumed flows” in the response to this RAI refers to the minimum required flow values used in the safety analyses under the ANO-1 CLB. The minimum required flows are adjusted to account for measurement uncertainty and for potential degradation between tests. These adjusted values are the acceptance criteria for the service water system flow tests. The staff found this acceptable.

[*Operating Experience*] In a letter to the NRC dated June 9, 2000, the applicant describes the operating history of service water system components subject to an AMR. The existing service water integrity program has provided significant information to indicate where the aging degradation of loss of material and fouling are occurring. Several portions of piping have been replaced prior to compromising their intended function as a result of this program. Also, the chemical additions have been shown to reduce fouling and system corrosion rates. The staff found the attributes of this program, in conjunction with other service water AMPs, to be effective in managing applicable aging effects for the period of extended operation.

ASME Section XI ISI - IWC & IWD (pressure tests)

The applicant credits the ASME Section XI ISI - IWC inspection program to manage the loss of material and cracking in service water system components exposed to raw water. The staff’s evaluation of the reactor building leak rate testing program is provided in Section 3.3.1 of this SER.

Buried Pipe Inspection

The applicant credits the buried pipe inspection program to manage the loss of material in service water system components exposed to an external-buried environment. The staff's evaluation of the buried pipe inspection program is provided in Section 3.3.1 of this SER.

Maintenance Rule

The applicant credits the Maintenance Rule program to manage the loss of material in service water system components exposed to an external-ambient environment. The staff's evaluation of the Maintenance Rule program is provided in Section 3.3.1 of this SER.

Heat Exchanger Monitoring

The applicant credits the heat exchanger monitoring program to manage the loss of material and cracking in service water system components exposed to raw water. The staff's evaluation of the heat exchanger monitoring program is provided in Section 3.3.1 of this SER.

Conclusion

On the basis of the AMP reviews provided above, the staff finds that there is reasonable assurance that the effects of aging applicable to the service water system will be adequately managed so that the intended functions of the service water system will be maintained consistent with the CLB for the period of extended operation.

Penetration Room Ventilation

In the LRA, Table 3.4-11, the applicant states that the penetration room ventilation system is subjected to environments of gas-air and external-ambient. The applicant identifies Maintenance Rule as the only AMP to address material degradation in the penetration room ventilation system.

Maintenance Rule

The applicant credits the Maintenance Rule program to manage the loss of material in penetration room system components exposed to an external-ambient environment. To manage the external aging effect of loss of material, the applicant cites Section 4.13, "Maintenance Rule," of Appendix B of the LRA. The applicant states in Table 3.4-11 of the LRA that the Maintenance Rule program will manage this aging effect. The staff's evaluation of the Maintenance Rule program is provided in Section 3.3.1 of this SER.

Conclusion

On the basis of the AMP reviews provided above, the staff finds that there is reasonable assurance that the effects of aging applicable to the penetration room ventilation system will be adequately managed so that the intended functions of the penetration room ventilation system will be maintained consistent with the CLB for the period of extended operation. The staff concludes that the applicant has demonstrated that aging effects associated with this system will be adequately managed by the identified AMPs.

Auxiliary Building Heating and Ventilation

In the LRA, Table 3.4-12, the applicant identifies four AMPs and activities to address the applicable aging effects of the auxiliary building heating and ventilation system components. These programs and activities are described in Appendices A and B of the LRA, and are evaluated in the following paragraphs:

EDG Testing and Inspection

The applicant credits the EDG testing and inspections program to manage loss of material in carbon steel exterior ductwork, louvers, and fans, exposed to external-ambient environment.

[*Program Scope*] Because the applicant has included in this program the carbon steel components that are part of the EDG rooms ventilation system within the auxiliary building heating and ventilation system, the staff found the scope of the program acceptable.

[*Preventive/Mitigative Actions*] There are no preventive or mitigative actions associated with the EDG testing and inspections program, and the staff did not identify a need for any.

[*Parameters Monitored*] The applicant inspects for damage caused by general corrosion and pitting on the external surface of the carbon steel components exposed to warm, moist air. This would detect loss of material in the carbon steel components.

[*Detection of Aging Effects*] The applicant performs an inspection of the EDG in accordance with manufacturer recommendations per ANO-1 TS 4.6.1. It is unclear whether any damage detected visually may have only just begun, or progressed to the point of compromising the integrity of the pressure boundary. However, the applicant has cited other AMPs, including preventive maintenance and Maintenance Rule activities, to manage the loss of material on the external surface of the carbon steel components. Therefore, the staff concludes that the EDG testing and inspection program in conjunction with the preventive maintenance program will be effective in detecting damage before the structural integrity of the pressure boundary is compromised.

[*Monitoring and Trending*] The applicant performs a visual inspection of the EDG once every 18 months per ANO-1 TS 4.6.1. The staff is not certain whether the frequency of these periodic inspections is adequate for detecting corrosion damage on the external surface before it may have progressed to the point of compromising the structural integrity of the pressure boundary. However, the staff concludes that the EDG testing and inspection program in conjunction with the preventive maintenance and Maintenance Rule activities can identify the corrosion damage on the external surface of the carbon steel components in a timely manner and ensure that the pressure boundary integrity of carbon steel components will be maintained.

[*Acceptance Criteria*] The acceptance criteria for testing and inspection are documented in EDG operation and inspection procedures. Because this program is based on NRC Regulatory Guide 1.108, "Periodic Testing of Emergency Diesel Generator Units Used As Onsite Power Systems at Nuclear Power Plants," and ANO-1 TS 4.6.1, the staff found this criteria to be acceptable.

[*Operating Experience*] In response to a staff's RAI, the applicant states that the EDG testing and inspection program has proven effective in managing loss of material in components in the EDG system, the auxiliary building heating and ventilation system and the fuel oil system. To confirm this conclusion for the exhaust subsystems, an inspection was performed. Visual inspections as well as ultrasonic test measurements were used. The areas chosen for UT measurements were the areas most likely to trap or store water, the discharge elbow and underside of the silencer. The visual inspections as well as UT measurements indicated little or no thinning or pitting exists from the operation of the units for approximately 20 years. This inspection has therefore shown there is no significant loss of material and the inspections of the engine and exhaust performed as a part of the plant TS maintenance is adequate to manage these limited effects on the internal surfaces of the exhaust subsystem.

Preventive Maintenance

The applicant credits the preventive maintenance program to manage the loss of material, and fouling in auxiliary building heating and ventilation system components exposed to gas-air environments. This program is evaluated in Section 3.3.1 of this SER.

Maintenance Rule

The applicant credits the Maintenance Rule program to manage the loss of material in auxiliary building heating and ventilation system components exposed to an external-ambient environment. The staff's evaluation of the Maintenance Rule program is provided in Section 3.3.1 of this SER.

Service Water Integrity

The applicant credits the service water integrity program to manage the fouling in auxiliary building heating and ventilation system components exposed to gas-air environments. See the discussion of service water integrity program in the section above.

Conclusion

On the basis of the AMP reviews provided above, the staff finds that there is reasonable assurance that the effects of aging applicable to the auxiliary building heating and ventilation system will be adequately managed so that the intended functions of the auxiliary building heating and ventilation system will be maintained consistent with the CLB for the period of extended operation.

Control Room Ventilation

In the LRA, Table 3.4-13, the applicant identifies four AMPs and activities to address the applicable aging effects of the CRVS components. These programs and activities are described in Appendices A and B in the LRA, and are evaluated in the following paragraphs:

Maintenance Rule

The applicant credits the Maintenance Rule program to manage the loss of material in CRVS components exposed to an external-ambient environment. The staff's evaluation of the Maintenance Rule program is provided in Section 3.3.1 of this SER.

Preventive Maintenance

The applicant credits the preventive maintenance program to manage the loss of material from the internal surface of the carbon steel evaporator bodies exposed to gas-air environment. The material loss takes place on surfaces wetted by condensation. The applicant has included in this program the carbon steel evaporator bodies within the CRVS whose internal surfaces are exposed to air.

The applicant credits the preventive maintenance program to manage fouling on the external surface of the copper tubing in the evaporators (heat exchangers), which is exposed to air. The applicant performs periodic inspection and cleaning of the external surface of the copper tubing to detect and remove the deposits of foreign matter (fouling) that adversely affects the heat transfer capacity of tubing. See previous discussion in generic AMPs in Section 3.3.1 of this report.

Control Room Ventilation Testing

The applicant takes credit for the control room ventilation testing program to manage the loss of material and fouling in components exposed to gas-air and carbon dioxide environments. The applicant describes this program in Appendix B of the LRA, Section 4.21.3, "Control Room Ventilation Testing."

[*Program Scope*] Because the applicant includes in this program the evaporator tubing of the control room ventilation system, the staff found the scope of the program adequate.

[*Preventative/Mitigative Actions*] There are no preventive or mitigative actions associated with the control room ventilation testing program, nor did the staff identify a need for such actions.

[*Parameters Monitored*] The applicant performs periodic testing for heat transfer through the tubing. Because the heat transfer capacity is directly related to the presence of fouling, the staff concludes that the parameters being monitored were adequate for detection of significant fouling prior to a loss of intended function.

[*Detection of Aging Effects*] The applicant performs heat transfer testing of the evaporators tubing. Because a thin film of fouling adversely affects heat transfer function of the tubing, such testing would be adequate to detect a presence of a small amount of fouling. Therefore, the staff concludes that this testing is adequate to ensure that fouling on evaporators tubing will be detected before the impairment of heat transfer capacity becomes unacceptable.

[*Monitoring and Trending*] The applicant states that each train of the control room emergency air conditioning system is demonstrated operable at least once per 31 days. This testing provides evidence that excessive fouling is not present. The applicant states that this testing frequency is per ANO-1 TS 4.10. Therefore, the staff concludes that the program can detect fouling on the tubing in a timely manner and ensure that the heat transfer capacity of the evaporators will be maintained at an acceptable level.

[*Acceptance Criteria*] The applicant states that the acceptance criteria for testing are provided in ANO-1 TS 4.10. Therefore, the staff concludes that the acceptance criteria are appropriate.

[*Operating Experience*] The applicant states that a review of ANO-1 condition report summaries did not identify fouling of the evaporator tubing. Therefore, the staff concludes that the control room ventilation testing program is effective in managing fouling on the external surface of the evaporator tubing.

Oil Analysis

The applicant credits the oil analysis program to manage the loss of material in CRVS components exposed to lubricating oil. The staff's evaluation of the oil analysis program is provided in Section 3.3.1 of this SER.

Conclusion

On the basis of the AMP reviews provided above, the staff finds that there is reasonable assurance that the applicable aging effects of the CRVS will be adequately managed so that the intended functions of the CRVS will be maintained consistent with the CLB for the period of extended operation.

3.3.4.3 Conclusions

The staff has reviewed the information presented by the applicant in the LRA and the additional information provided by the applicant in response to the staff RAIs. The staff concludes that the applicant has considered applicable aging effects that are consistent with published literature and industry experience for components subject to license renewal. The various AMPs used by the applicant to manage the effects of aging for the various auxiliary system components are effective as "stand-alone" programs, or in conjunction with other AMPs or activities. On the basis of the reviews described above, the staff found that there is reasonable assurance that the effects of aging applicable to the auxiliary systems will be adequately managed so that the intended functions of the auxiliary systems will be maintained consistent with the CLB for the period of extended operation.

3.3.4.4 References for Section 3.3.4

1. 10 CFR Part 54, “Requirements for Renewal of Operating Licenses for Nuclear Power Plants.”
2. DG-1047, “Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants,” Working Draft, April 21, 2000.
3. “Arkansas Nuclear One–Unit 1, License Renewal Application,” January 31, 2001.
4. ANSI USAS B31.1.0, “USA Standard Code for Pressure Piping,” 1968.
5. ANSI USAS B31.7, “USA Standard Code for Pressure Piping, Nuclear Power Piping,” 1968.
6. 10 CFR 50.55a, “Codes and Standards.”
7. NRC BL 79-17, “Pipe Cracks in Stagnant Borated Water Systems at PWR Plants.”
8. NRC IN 79-19, “Pipe Cracks in Stagnant Borated Water Systems at Power Plants.”
9. NRC IN 79-23, “Emergency Diesel Generator Lube Oil Coolers.”
10. NRC IN 84-18, “Stress Corrosion Cracking in Pressurized Water Reactor Systems.”
11. NRC IN 85-24, “Failures of Protective Coatings in Pipes and Heat Exchangers.”
12. NRC GL 88-14, “Instrument Air Supply System Problems Affecting Safety-Related Equipment.”
13. NRC GL 89-13, “Service Water System Problems Affecting Safety-Related Equipment.”

3.3.5 Steam and Power Conversion Systems

In the ANO-1 LRA, Section 2.3.4, “Steam and Power Conversion Systems,” the applicant describes its process and results for the scoping and screening of the steam and power conversion systems (SPCS) SSCs that are within the scope of license renewal and subject to an AMR. The applicant described its AMR of the SPCSs for license renewal in Section 3.5, “Steam and Power Conversion Systems” and in Appendices B and C of its LRA.

After the initial review of this information, the NRC staff requested additional information concerning these systems in a letter to the applicant dated April 17, 2000. In a letter to the NRC dated September 12, 2000, the applicant provides its response to the staff’s RAIs. The applicant adequately addressed all the staff’s RAIs.

Initially, the staff was concerned with the adequacy of the applicant’s ability to identify loss of material on the interior surface of piping within the SPCS. In the LRA, the applicant states that control of secondary water chemistry will manage the loss of material in treated water environments, while the NRC staff did not agree that control of water chemistry can preclude loss of material due to crevice and pitting corrosion at locations with stagnant flow conditions. Therefore, the NRC staff suggested that a one-time inspection of selected components at susceptible locations may be needed in addition to the water chemistry program to ensure that potential degradation is not occurring and the components’ intended function(s) will be maintained during the period of extended operation.

On October 5, 2000, the staff also initiated a conference call with the applicant to clarify this concern. This conference call is documented in a letter to the applicant dated October 11, 2000. The applicant responded to this request for additional clarification in letters to the NRC dated November 2, 2000, and December 20, 2000. In its responses, the applicant states that maintenance related inspections have been routinely performed and have not identified loss of material due to crevice and pitting corrosion in areas of stagnant flow. The applicant states that maintenance inspections involving internal visual inspections performed by the systems engineer or members of the applicant’s chemistry department have been performed in the areas of the EFW pump, condensate storage tank, the backup condensate pump, and other locations numerous times over the past 25 years. These inspections have not identified the loss of material caused by crevice and pitting corrosion. These inspections are expected to continue during the period of extended operation, and meet or exceed the staff’s minimum requirement of a one-time inspection. Therefore, the staff has no additional concerns associated with the loss of material in stagnant areas of the SPCS.

The NRC staff reviewed the SPCS information included in the LRA to determine whether the applicant has demonstrated that the effects of aging on the SPCS will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation in accordance with 10 CFR 54.21(a)(3).

3.3.5.1 Technical Information in the Application

In the LRA, Section 3.5, the applicant describes the four SPCSs: main steam system (MSS), MFW, EFW, and condensate storage and transfer system (CSTS).

The MSS transports steam from the steam generators to the main turbine, the main feedwater pump turbine, the emergency feedwater turbine during emergency operation, and to a variety of other components during normal operation. The components of the MSS are constructed from carbon steel and stainless steel. All the external surfaces of main steam components are exposed to the ambient environments of the auxiliary, turbine, or reactor buildings. All internal surfaces of main steam components are exposed to a treated water environment that could be heated, superheated, dry, saturated, or partially condensed steam. The MSS also includes the steam supply to the EFS turbine and the nitrogen supply to the steam generators.

The MFW supplies feedwater to the steam generators to support normal plant operation. The components in the MFW are constructed from carbon steel. The internal surfaces of these components are exposed to treated water, and the external surfaces are exposed to the ambient environments of the auxiliary, turbine, and reactor buildings.

The EFS supplies feedwater to the steam generators if the main feedwater supply is lost. The components in the EFS are constructed from aluminum, cast iron, carbon steel, low alloy steel, stainless steel, brass, bronze, and copper. The internal surfaces of these components are primarily exposed to treated water, and the external surfaces are exposed to the ambient environments of the turbine and reactor buildings. The EFS lubricating oil subsystem is exposed to lubricating oil internally and the external-ambient environment. The heat exchanger of the EFS is exposed to treated water and lubricating oil internally and the external-ambient environment.

The CSTS provides water from the condensate storage tanks to the secondary plant systems. It also supplies water to the emergency feedwater pumps during emergency operation. The components in the CSTS are constructed from carbon steel and stainless steel. The internal surfaces of these components are exposed to treated water, and the external surfaces are exposed to the external-ambient environment.

3.3.5.2 Staff Evaluation

In accordance with 10 CFR 54.21(a)(3), the NRC staff reviewed the information included in Sections 2.3.4 and 3.5 of the LRA. The staff evaluation of the applicant's methodology and results for identifying the SSCs included within the scope of license renewal, and the SCs that are subject to an AMR is provided separately in Sections 2.2 and 2.3 of this SER.

3.3.5.2.1 Effects of Aging

The components of the SPCSs are constructed from carbon steel, stainless steel, low alloy steel, brass, bronze, copper, aluminum, and cast iron. They are exposed to the external-ambient (air) environments of the auxiliary, turbine, and reactor buildings, which by itself will not cause any significant aging effects. Internally, the components in the SPCSs are exposed to a treated water and/or steam environment. The following material aging effects were identified in the systems carrying treated water and steam: loss of material from general corrosion and pitting of carbon steel, low-alloy steel, cast iron, brass, bronze, copper, and aluminum; loss of material from erosion/corrosion of carbon steel components; galvanic corrosion of coupled dissimilar materials having different electrochemical potentials; selective leaching of cast iron and loss of material from pitting; cracking of carbon steel and stainless steel components; loss of mechanical closure integrity in components made from carbon steel, stainless steel, and low-alloy steel; and fouling for carbon steel components. Tables 3.5-1 through 3.5-4 of the LRA list the components, material, environment, and applicable aging effects.

3.3.5.2.1.1 Process for Identification of Aging Effects

In the LRA, Exhibit A, Appendix C, “Process for Identifying Aging Effects Requiring Aging Management for Non-Class 1 Mechanical Components,” the applicant described its process for identifying aging effects for the SPCS components for license renewal. To facilitate the identification of the applicable aging effects, the applicant identifies four environments and two “special topics” that apply to the SPCS components that require AMR. The four operating environments and two special topics are described below.

- Treated water is a high-purity demineralized water and is the internal operating environment for all SPCS components within the scope of license renewal using demineralized water.

- The lubricating oil and fuel oil environment is applicable to components holding or using oil lubricants or fuel oil. The heat exchanger of the lubricating oil subsystem of the EFS is exposed to lubricating oil.

- The air/gas internal environment is applicable to the materials exposed to internal environments consisting of gases such as air (at atmospheric pressure and compressed), nitrogen, carbon dioxide, Freon, and Halon. The nitrogen supply to the steam generators of the MSS is exposed to nitrogen internally.

- Component external surfaces may experience aging due to exposure to the external environment. Applicable external environments include the ambient atmosphere (including airborne contaminants and moisture) and leakage. All of the SPCSs are exposed to the ambient air environment externally.

- Bolting applications that are within the scope of license renewal are divided into pressure boundary bolting and structural or component support bolting. Pressure boundary bolting applications include bolted flange connections for vessels (i.e., manways and inspection ports), flanged joints in piping, body-to-bonnet joints in valves, and pressure-retaining bolting associated with pumps and miscellaneous process components. A bolted closure includes the entire bolted joint, seating surfaces (e.g., flange set surfaces), gasket, and pressure-retaining bolting.

- The heat exchanger of the EFS is exposed to treated water and lubricating oil internally and the ambient air environment externally.

On the basis of the environments to which the SPCS components are primarily exposed, the applicant identifies the following aging effects that require management during the period of extended operation.

- Loss of material may be attributed to one or more of the following: general corrosion, pitting corrosion, galvanic corrosion, crevice corrosion, erosion (including erosion caused by abrasive wear, erosive wear, cavitation wear, and droplet impingement wear), erosion/corrosion, microbiological influenced corrosion, or selective leaching.

- Service-induced cracking (initiation and growth) of base metal or weld metal may be attributed to one or more of the following: stress corrosion, intergranular attack, or vibration.

- loss of mechanical closure integrity is an aging effect resulting in failure of a mechanical closure to provide a required pressure boundary. Loss of mechanical closure integrity may be attributed to one or more of the following conditions affecting bolting material. Loss of preload, cracking of bolting material, loss of bolting material, and reduction of fracture toughness of bolting material (only applicable to RCS components).

- Fouling is caused by macro-organisms, contaminants, precipitation, or silting. Fouling is not a material degradation phenomenon, but is an aging effect that could cause loss of the heat transfer intended function, or a reduction in flow rate, for components in the SPCSs.

3.3.5.2.1.1.1 Applicable Aging Effects for a Treated Water Environment

Treated water is a high-purity demineralized water, and is the base water for all “clean” systems.
Depending on the system, treated water may require additional processing. Treated water could be deaerated, include corrosion inhibitors, biocides, or some combination of these treatments. In the determination of aging effects, steam is considered treated water.

The MFW chemistry is closely monitored to minimize the potential for degradation of the once-through steam generators. The pH of the feedwater is maintained by the addition of amines to reduce the potential for flow-assisted corrosion by reducing iron transport. Deaeration and the addition of hydrazine control dissolved oxygen. Impurities such as chlorides and sulfates are controlled to reduce the stress corrosion cracking of the OTSG tubes. The demineralized or makeup water chemistry requirements are stringent because makeup water is used for reactor coolant, secondary, and other auxiliary systems in which high-quality water is required. The EFS and the CSTS have strict limits on contaminants, but the safety-related condensate storage tank is vented to atmosphere so these systems contain water saturated with oxygen. The auxiliary systems use treated water with corrosion inhibitors.

The applicant identifies the following materials of the SPCSs that are exposed to an internal treated water environment: stainless steels, carbon steel, cast iron, copper, brass, and bronze.

Loss of material due to general corrosion is an applicable aging effect for carbon steel, cast iron, copper, brass and bronze materials in a treated water environment. Stainless steel resists general corrosion in treated water environments.

Loss of material due to pitting corrosion is an applicable aging effect for the different materials found in the SPCS in a treated water environment under certain conditions. For a treated water environment, two sets of conditions can lead to pitting corrosion. First, the presence of halogens in excess of 150 ppb, oxygen in excess of 100 ppb and stagnant or low-flow conditions. Second, the presence of sulfates in excess of 150 ppb, oxygen in excess of 100 ppb and stagnant or low-flow conditions.

Loss of material due to galvanic corrosion can occur in oxygenated water only when materials with different electrochemical potentials are in close proximity.

Loss of material due to erosion-corrosion is an applicable aging effect for carbon steel in treated water under high energy flow conditions. Fluid conditions in the main steam and main feedwater systems can lead to erosion-corrosion.

Cracking due to stress corrosion of stainless steel materials in treated water is an applicable aging effect under certain conditions. For stress corrosion cracking to occur in stainless steel, the concentration of halogens or sulfates must exceed 150 ppb. In addition, stress corrosion cracking of stainless steel has been observed in high-purity water (i.e., sulfates and halogens less than 150 ppb) at temperatures greater that 200°F with dissolved oxygen levels greater than 100 ppb.

In order to verify that all the applicable aging effects for components exposed to a treated water environment have been identified, the applicant reviewed industry and ANO-1 plant specific experience. The applicant confirms that they were unable to identify any additional aging effects beyond those identified and described in this section.

The NRC staff performed its own review of up-to-date industry and ANO-1 plant-specific experience, and did not identify any omissions.

3.3.5.2.1.1.2 Applicable Aging Effects for a Lubricating Oil and Fuel Oil Environment

The applicant conducted separate evaluations for lubricating oil and fuel oil. The SPCS components are exposed only to a lubricating oil environment, as discussed above. The SPCS lubricating oils are low- to medium-viscosity hydrocarbons used for bearing, gear, and engine lubricating. The materials used in SPCS mechanical components that are within the scope of license renewal, and exposed to lubricating oil and fuel oils include stainless steels, carbon steel, cast iron, aluminum, and copper.

Loss of material due to general corrosion is an applicable aging effect for carbon and low-alloy steel in a lubricating oil environment at locations containing water. The stainless steel, brass, admiralty, cast iron, glass, aluminum, bronze, and copper in a lubricating oil environment are inherently resistant to general corrosion.

Loss of material due to pitting corrosion is an applicable aging effect for brass, bronze, carbon steel, low-alloy steel, copper, and stainless steel materials in a lubricating oil environment at locations containing oxygenated water with contaminants such as halide ions, particularly chloride ions. In addition, loss of material due to galvanic corrosion in a lubricating oil environment can occur only when materials with different electrochemical potentials are in contact in the presence of water. Loss of material due to crevice corrosion can also occur in brass, bronze, carbon steel, copper, and stainless steel materials in a lubricating oil environment at locations containing oxygenated water. Oxygen is required for the initiation of crevice corrosion. Lube oil that is not contaminated with water does not contain oxygen in sufficient quantities for crevice corrosion to occur.

Cracking due to stress corrosion of the stainless steel material in a lubricating oil environment is an applicable aging effect at locations containing oxygenated water, as well.

Water contamination of lubricating oil can occur, and is required for the introduction of oxygen. Although, only high quality (water and contaminant free) lubricating oil is received, and periodic sampling is performed to ensure the quality is maintained, the potential contamination of lubrication oil makes loss of material due to general corrosion, pitting, galvanic corrosion and crevice corrosion an applicable aging effect for stainless steel, brass, admiralty, cast iron, glass, aluminum, bronze, and copper materials exposed to lubricating oil in the SPCSs.

Loss of material due to microbiological influenced corrosion is an applicable aging effect for brass, carbon steel, copper, and stainless steel materials exposed to lubricating oil. The applicant treats the lubricating oil with biocides to limit the presence of microbiological organisms and, therefore, microbiological influenced corrosion has not been a concern for those portions of the SPCSs that are within the scope of license renewal, and the associated materials exposed to lubricating oil. However, the potential for the presence of microbiological organisms to be found in lubricating oil makes microbiological influenced corrosion an applicable aging effect for brass, carbon steel, copper, and stainless steel materials exposed to lubricating oil in the SPCSs.

In order to validate the applicable aging effects for components exposed to a lubricating oil, the applicant reviewed industry and ANO-1 plant specific experience. The review revealed no additional aging effects beyond those identified and described in this section.

The NRC staff performed its own review of up-to-date industry and ANO-1 plant-specific experience, and did not identify any omissions.

3.3.5.2.1.1.3 Applicable Aging Effects for Gas Environments

The gas environments that are within the scope of license renewal for the SPCSs include atmospheric air (filtered and unfiltered) and compressed nitrogen.

Air is composed mostly of nitrogen and oxygen with smaller fractions of various other constituents. The internal surfaces of some components are, at times, exposed to air. Where air is the internal medium (e.g., compressed air and instrument air systems), it is supplied in either its natural state or in a “dry” condition. The instrument air system supplies air free of water or contaminants.

Nitrogen is an inert gas used in many nuclear plant applications to place components in a dry lay-up condition or to provide a cover gas to prevent exposure to oxygen. The commercial grade nitrogen used at ANO-1 is a high-quality product with little, if any, contaminants.

The materials exposed to air or nitrogen includes stainless steels, carbon steel, brass, bronze, and copper.

In general, air provides an environment for aging only in the presence of moisture or other contaminants. At ordinary temperatures, oxygen and moisture are the basic factors for the general corrosion of iron. Both oxygen and moisture must be present because oxygen alone or water free of dissolved oxygen does not corrode iron to any significant extent. Carbon and low-alloy steels, as well as cast iron are susceptible to general corrosion. Stainless steels, nickel-based alloys, aluminum, copper alloys and galvanized steel are inherently resistant to general corrosion. General corrosion is an electrolytic reaction and, regardless of the particular gas environment, is dependent on the presence of oxygen and moisture. Corrosion in a non-aqueous environment only occurs by direct chemical reaction and only at high temperatures well above those encountered in the applications under review. Nitrogen and Halon environments have negligible amounts of free oxygen. Therefore, corrosion of carbon steel and cast iron components in these environments is not a concern. The air environments within plant systems and components can vary from clean, dry air to moist, contaminated air whose purity is dictated by the source of the air. Portions of compressed air system contain air that has been processed through dryers, which provide dry, oil free air to the downstream portions of the system. Therefore, moisture is not a concern for these portions of systems and general corrosion is not expected.

The severity of galvanic corrosion depends largely on the type and amount of moisture present. Galvanic corrosion does not occur when the metals are completely dry since there is no electrolyte to carry the current between the two electrode areas. Any gas and moisture environments that contain dissimilar materials with significant potential differences can result in galvanic corrosion. Air systems that contain dissimilar materials with potential differences, and the potential for moisture in crevices and other low points of systems can be susceptible to galvanic corrosion. Aluminum to brass connections, as well as steel-to-copper connections, are susceptible to galvanic corrosion.

Crevice corrosion is not a concern with any gas environment other than air, where the oxygen content may be low enough to preclude crevice corrosion concerns. Crevice corrosion is a concern where moisture may occur in the presence of contaminants such as halides or sulfates.

All materials of interest are susceptible to pitting corrosion under certain conditions. Most pitting is associated with the presence of halide ions, chlorides, bromides, or hypochlorites.

Microbiological organisms that could induce corrosion are generally not found in a gaseous environment. Microbiological influenced corrosion, therefore, is only a potential problem where contamination from untreated water or soil may have introduced bacteria. Air and gas systems are only affected where stagnant conditions and the pooling of an untreated aqueous solution provide an environment suitable for propagation of the mechanism.

Carbon steel, cast iron and stainless steel in a nitrogen environment have no aging effects since the nitrogen has negligible amounts of free oxygen.

In order to validate the applicable aging effects for components exposed to gas environments, the applicant reviewed industry and ANO-1 plant specific experience. The review revealed no additional aging effects beyond those identified and described in this section. The NRC staff also performed its own review of up-to-date industry and ANO-1 plant-specific experience, and did not identify any omissions.

3.3.5.2.1.1.4 Applicable Aging Effects for External Surface Environments

The purpose of this section is to identify aging effects applicable for external surfaces of the SPCS SCs. External environments include the ambient atmosphere and leakage. The external-ambient atmospheric environment consists of atmospheric conditions, which may include humidity, condensation, and airborne contaminants such as sulfur dioxide, chlorine gas, sulfur gas, and ozone. The leakage environment is created when fluid escapes from a system boundary (usually from bolted closures), and comes in contact with the external surfaces of components. The fluids typical of concern with respect to SPCS and leakage environments are borated, treated, and raw water.

The materials of the SPCSs exposed to external environments include stainless steels, carbon and low-alloy (chrome-moly) steel, cast iron, brass, bronze, and copper.

Loss of material due to general corrosion is an applicable aging effect for carbon steel and low-alloy steel, admiralty, brass, cast iron, and copper materials in ambient air environments if these materials come in contact with moist air. Paint or protective coatings applied to external surfaces will prevent this aging effect.

Loss of material due to galvanic corrosion can occur when dissimilar materials with different electrochemical potentials come in contact with water or moisture, which is needed to establish the galvanic couple. Systems continually operating under conditions that can cause surface condensation in an ambient air environment can have water present on their external surfaces and, therefore, are susceptible to galvanic corrosion.

Stress corrosion cracking in low-strength carbon steel exposed to atmospheric conditions is not a concern in materials with a yield strength of less than 100 ksi. The yield strength of carbon steels typically used in non-Class 1 systems is on the order of 30 to 45 ksi. Industry data does not indicate a problem of stress corrosion cracking in low-strength carbon steels. For these reasons, stress corrosion cracking is not an applicable aging effect for carbon and low-alloy steel materials. For stainless steel components exposed to atmospheric conditions, stress corrosion cracking is only applicable when exposed to high levels of contaminants (e.g., saltwater environment) and, then, only if the material is in a sensitized condition.

Loss of material due to boric acid wastage is an applicable aging effect for carbon steel and low-alloy chrome-moly materials exposed to leakage of borated water. Leaking fluid from a borated water system may expose the external surfaces of components made from these materials to a concentrated boric acid solution that can cause loss of material.

Loss of material due to general corrosion is an applicable aging effect for admiralty brass, brass, carbon steel, cast iron, copper, and low-alloy steel exposed to treated water leakage due to the presence of oxygen. Stainless steel materials are resistant to general corrosion. Paint or protective coatings applied to external surfaces of components susceptible to general corrosion can prevent this aging effect from occurring.

Loss of material due to pitting corrosion is an applicable aging effect for admiralty brass, brass, carbon steel, cast iron, copper, low-alloy steel, and stainless steel materials exposed to treated water leakage. Paint or protective coatings applied to external surfaces can prevent this aging effect from occurring.

Loss of material due to general corrosion is an applicable aging effect for admiralty brass, brass, bronze, carbon steel, cast iron, copper, and 90-10 copper-nickel component materials exposed to raw water leakage. Stainless steel materials are resistant to general corrosion due to raw water leakage.

Loss of material due to pitting corrosion is an applicable aging effect for admiralty, brass, bronze carbon steel, cast iron, copper, 90-10 copper-nickel, and stainless steel materials exposed to raw water leakage.

Loss of material due to galvanic corrosion can occur in materials exposed to raw water leakage.

Loss of material due to microbiological influenced corrosion is an applicable aging effect for admiralty, brass, bronze, carbon steel, cast iron, copper, 90-10 copper-nickel and stainless steel materials exposed to raw water leakage.

Paint or protective coatings applied to external surfaces can prevent exposure to raw water to prevent the potential for loss of material.

Galvanic corrosion in an underground environment can occur when materials with different electrochemical potentials are in close proximity of each other, and are in the presence of water. In the underground environment, galvanic corrosion can occur between the material and the surrounding soil and groundwater.

In order to validate the applicable aging effects to the external surfaces of components in the SPCSs, the applicant reviewed industry and ANO-1 plant specific experience. The review revealed no additional aging effects beyond those identified and described in this section.

The NRC staff performed its own review of up-to-date industry and ANO-1 plant-specific experience, and did not identify any omissions.

3.3.5.2.1.1.5 Applicable Aging Effects for Bolted Closures

In the LRA, the applicant discusses aging effects applicable to bolted closures of non-Class 1 mechanical components whose intended function is to maintain pressure boundary. The bolting closures addressed in these discussions included the following connections: bolted flange connections for vessels (i.e., manways and inspection ports), flanged joints in piping, body-to-bonnet joints in valves, and pressure-retaining bolting on pumps and miscellaneous process components.

A bolted closure includes the seating surfaces (e.g., flange set surfaces), gasket, and pressure retaining bolting. Pressure boundary bolting, typically referred to as threaded fasteners, include nuts, bolts, studs, and cap-screws. Gaskets do not require an AMR because gaskets are only a part of the bolted connection and exist to minimize leakage rather than to directly support the pressure boundary.

Bolted closures that are within the scope of license renewal are found in all non-Class 1 mechanical systems. The non-Class 1 bolting and threaded connections are subject to external-ambient environments, and are potentially susceptible to being exposed to process fluids as a result of leakage.

The materials used in bolting and threaded connections that are within the scope of license renewal are primarily carbon, low-alloy steels, and stainless steel.

The governing aging effect to consider for bolted closures is loss of mechanical closure integrity. Loss of mechanical closure integrity may be attributed to one or more of the following conditions:

- The loss of pre-load for bolted closure may be attributed to embedment, cyclic load embedment, gasket creep, thermal effects (e.g., yield stress effect, modulus of elasticity effect, and stress relaxation), and self-loosening.

- Loss of material due to boric acid wastage is the most common aging effect observed for ferritic fasteners. Stainless steel fasteners are immune to loss of material due to general corrosion. Most bolting is normally in a dry environment and is coated with a lubricant and general corrosion is not expected. General corrosion of ferritic fasteners has only been observed due to leaking joints.

- Wear could lead to the loss of bolting material in connections subject to frequent operational use. Proper maintenance and tool handling practices and infrequent opening and closing preclude significant wear on most bolted connections.

- Cracking of bolting materials may be caused by stress corrosion cracking or fatigue. Stress corrosion cracking may occur in bolting materials subjected to water or steam (e.g., from leakage) that contains various contaminants. When leakage is combined with contaminants such as sulfides or chlorides, an aggressive environment that can promote stress corrosion cracking may result.

Threaded connections are assumed to be subject to loss of mechanical closure integrity in high-vibration applications, such as on diesel generators. This mechanism is independent of material and environment.

In order to validate the applicable aging effects for bolted connections, the applicant reviewed industry and ANO-1 plant specific experience. The review revealed no additional aging effects beyond those identified and described in this section. The NRC staff performed its own review of up-to-date industry and ANO-1 plant-specific experience, and did not identify any omissions.

3.3.5.2.1.1.6 Applicable Aging Effects in Heat Exchangers

The applicant indicates that although most heat exchanger aging effects are material and environment driven, heat exchangers are also evaluated for the component-specific mechanisms of erosion and wear. In addition, heat exchangers are typically exposed to several different environment and material combinations (i.e., tube material and fluid are different from shell material and fluid).

The internal environments of the heat exchangers (both the internal surfaces of the tubes and the internal surfaces of the shells) are treated water and lubricating oil. External heat exchanger surfaces are exposed to the external-ambient atmosphere.

In general, heat exchanger components are subject to the same aging mechanisms as other non-Class 1 mechanical components of similar materials exposed to similar environments as described in Appendix C of the LRA. The focus of this section is on the aging mechanisms that affect the heat exchangers in addition to the standard material and environment mechanisms.

Loss of material due to vibration and rubbing of the heat exchanger internal components is an applicable aging effect for heat exchangers. The applicability of this mechanism is dependent on the heat exchanger configuration, the specific component geometry and contents of the entrained fluids. Erosion and erosion-corrosion can also result in a loss of material on internal heat exchanger surfaces in locations with high flow velocities (typically, where the velocity is high in a localized area).

Fouling is any change in the heat transfer surfaces such that it impairs the heat transfer intended function. The impairment is primarily from a decrease in the heat transfer coefficient due to entrained materials precipitating out of the fluid, and adhering to the heat exchanger surfaces. This precipitation and adherence process, and the resulting fouling is influenced by local flow patterns and temperature conditions. Many corrosion mechanisms also generate corrosion byproducts that adhere to the surface of the corroding metal. Since even a thin layer of the materials that adhere to the heat transfer surfaces can impact the heat transfer capability of heat exchangers, fouling is considered an applicable aging effect for non-Class 1 heat exchangers subject to an AMR.

In order to validate the applicable aging effects in heat exchangers, the applicant reviewed industry and ANO-1 plant specific experience. The review revealed no additional aging effects beyond those identified and described in this section.

The NRC staff performed its own review of up-to-date industry and ANO-1 plant specific experience, and did not identify any omissions.

The applicant supplies references specifically pertaining to ANO-1, as well as industry-wide experience to support its identification of applicable aging effects for the SPCSs. On the basis of the description of internal and external environments, and material of construction for these systems, the NRC staff did not identify any omission from the applicable aging effects identified by the applicant for the SPCSs. The applicable aging effects identified by the applicant are consistent with the published literature, industry guidance, and industry operating experience.

3.3.5.2.2 Aging Management Programs

The applicant identifies a number of AMPs for controlling the effects of aging in the SPCSs. The programs were developed from industry wide data, industry developed methodologies, NRC documents, and the applicant's own experience. The applicant concludes that these programs will adequately manage the effects of aging of the SPCSs so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation. In the LRA, the applicant identifies the following AMPs for the SPCSs:

- ASME Section XI ISI - IWC and IWD inspections
- reactor building leakage detection program
- flow-accelerated corrosion prevention program
- wall thinning inspection program
- secondary chemistry monitoring program
- Maintenance Rule program
- augmented ASME Section XI ISI special inspections
- EFW pump testing program
- oil analysis program
- heat exchanger monitoring program
- condensate storage tank level monitoring program

The applicant describes these programs in the LRA, Appendix B.

The NRC staff evaluated the applicant's AMPs in order to determine if they contain the ten essential elements needed to provide adequate aging management of the components in the SPCSs so that the components will perform their intended functions in accordance with the CLB during the period of extended operation.

The ASME Section XI ISI programs, the reactor building leakage detection program, flow-accelerated corrosion prevention program, wall thinning inspection program, oil analysis program, heat exchanger monitoring program, and the condensate storage tank level monitoring program are evaluated in Section 3.3.1.4 of this SER. The secondary chemistry control and the Maintenance Rule programs are evaluated by the NRC staff in Sections 3.3.1.1 and 3.3.1.3 of the SER, respectively.

In the LRA, the applicant states that the activities for license renewal will be conducted in accordance with programs meeting the requirements of 10 CFR Part 50, Appendix B, and cover all SCs that are subject to an AMR. The staff evaluated the applicant's Quality Assurance Program in Section 3.3.1.2 of this SER, and its ability to address the corrective actions, confirmatory actions, and administrative control elements of an adequate AMP. Presented below are the results of the NRC staff's evaluation of the applicant's programs for the remaining seven essential elements of an AMP used for monitoring and controlling aging effects on the SPCSs.

Listed below is the NRC staff evaluation of the bolting and torquing activities, condensate storage tank level monitoring program, augmented ASME Section XI ISI special inspections, and the Emergency Feedwater Pump Testing programs.

Bolting and Torquing Activities

The NRC staff reviewed the information in Appendix B, Section 4.4, "Bolting and Torquing Activities."

[Program Scope] The applicant includes the flanged joints in piping, body-to-body joints in valves, and pressure-retaining bolting associated with valves, which are the typical component groups that require torquing activities, in the scope of this program. The NRC staff found no omission from the scope selected by the applicant.

[Preventative/Mitigative Actions] The applicant did not identify any preventive or mitigative actions associated with the inspection program. The NRC staff did not identify a need for preventive and mitigative actions.

[Parameters Monitored] The applicant selects the male and female thread inspection activities and inspection for adequate thread engagement, loose fasteners, and appropriate torque values as the parameters to monitor for detecting the aging effects associated with cracking, loss of material, and loss of mechanical closure integrity. The NRC staff did not identify the need for any additional parameters to monitor.

[Detection of Aging Effects] The applicant concludes that the detection of aging effects before there is a loss of intended function can be reasonably expected from the inspection program because of the adequate inspection scope, technique, and frequency. The applicant noted that satisfactory operating experience to date supports this conclusion. The staff agreed with this conclusion.

[Monitoring and Trending] The applicant determines that the monitoring and trending activities associated with this inspection program are sufficient to predict the extent of degradation so timely corrective or mitigative actions are possible. This is accomplished by performing these activities when maintenance activities involving threaded fasteners are involved. Further, the procedures for these activities are based on generic industry guidance and have been effective in the past. The NRC staff did not identify the need for any additional detection activities.

[Acceptance Criteria] The acceptance criteria for this activity are specifically documented and controlled in site procedures that have been effective at maintaining bolt torquing in the past. The NRC staff finds this criteria acceptable because of its past effectiveness and any indication of degradation of bolting requires the applicant to implement corrective actions.

[Operating Experience] The applicant's operating experience demonstrates that the inspection program will be effective for managing aging effects because it incorporates proven monitoring techniques, acceptance criteria, corrective actions, and administrative controls from existing programs and procedures.

The NRC staff identified sufficient information in the LRA to show that the bolting and torquing activities can effectively prevent degradation of bolting, or identify and correct degradation of bolting of the flanged joints in piping, body-to-body joints in valves, and pressure-retaining bolting associated with pumps and valves in the components of the SPCSs.

Emergency Feedwater Pump Testing

The NRC staff reviewed the information in Appendix B, Section 4.21.6, "Emergency Feedwater Pump Testing."

[Program Scope] The applicant includes the turbine-driven, and the electric motor-driven emergency feedwater pumps, which are the typical components that require these kinds of testing activities, that are within the scope of this program. The NRC staff found no omission from the scope selected by the applicant.

[Preventative/Mitigative Actions] The applicant does not identify any preventive or mitigative actions associated with this performance monitoring activity. The NRC staff did not identify a need for preventive and mitigative actions.

[Parameters Monitored] The applicant monitors the pump discharge pressure and flow temperature of the turbine lube oil and the pump bearing. The NRC staff did not identify the need for any additional parameters to monitor.

[Detection of Aging Effects] The applicable aging effects include fouling, loss of material, and loss of mechanical closure integrity. Fouling causes abnormally high temperature in lube oil or pump bearing which would be detected by the performance monitoring provided by the emergency feed pump testing identifying fouling prior to loss of function.

[Detection of Aging Effects] The detection of fouling, loss of material, and loss of mechanical closure integrity can be identified before there is a loss of intended function can be reasonably expected to occur. This program requires startup, operation and flow verification of applicable pumps which would detect loss of heat transfer capability which would be identified by abnormally high temperature in the lube oil or pump bearing. The staff agreed with this conclusion.

[Monitoring and Trending] The applicant determined that the monitoring and trending activities associated with this performance testing program are sufficient to predict the extent of degradation so timely corrective or mitigative actions can be implemented. This is accomplished by early notification of high lube oil and pump bearing temperatures. The NRC staff did not identify the need for any additional detection activities.

[Acceptance Criteria] The acceptance criteria for this activity are documented and controlled by the applicant's TS, Section 4.8. The NRC staff finds this criteria acceptable because any indication of reduced performance would require the applicant to implement corrective actions.

[Operating Experience] The applicant's operating experience demonstrates that this performance monitoring program will be effective for managing aging effects because it incorporates proven monitoring techniques, acceptance criteria, corrective actions, and administrative controls from existing programs and procedures.

The NRC staff identified sufficient information in the LRA to show that the emergency feedwater pump testing program can effectively prevent fouling of the SPCSs heat exchangers.

3.3.5.3 Conclusions

On the basis of the above review, the NRC staff concludes that the applicant has demonstrated that effects of aging associated with the SPCSs will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

3.3.5.4 References for Section 3.3.5

1. 10 CFR Part 54, “Requirements for Renewal of Operating Licenses for Nuclear Power Plants.”
2. DG-1047, “Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants,” Working Draft, April 21, 2000.
3. Arkansas Nuclear One - Unit 1, “License Renewal Application” January 31, 2000.
4. NRC BL 79-13, "Cracking in Feedwater System Piping."
5. NRC BL 87-01, "Thinning of Pipe Walls in Nuclear Power Plants."
6. NRC IN 80-29, "Broken Studs on Terry Turbine Steam Inlet Flanges."
7. NRC IN 81-04, "Cracking in Main Steam Lines."
8. NRC IN 84-87, "Piping Thermal Deflection Induced by Stratified Flow.”
9. NRC IN 86-106, "Feedwater Line Break."
10. NRC IN 87-36, "Significant Unexpected Erosion of Feedwater Lines."
11. NRC IN 88-17, "Summary of Responses to NRC Bulletin 87-01,Thinning of Pipe Walls in Nuclear Power Plants."
12. NRC IN 91-18, "High-Energy Piping Failures Caused by Wall Thinning.”
13. NRC IN 91-19, "Steam Generators Feedwater Distribution Piping Damage."
14. NRC IN 91-28, "Cracking in Feedwater System Piping."
15. NRC IN 91-38, "Thermal Stratification in Feedwater System Piping."
16. NRC IN 92-07, "Rapid Flow-Induced Erosion/Corrosion of Feedwater Piping."
17. NRC GL 79-20, "Information Requested on PWR Feedwater Lines" NRC Generic Letter 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning."

THIS PAGE IS INTENTIONALLY LEFT BLANK

3.3.6 Structures and Structural Components

This section of the SER documents the staff's evaluation of the ANO-1 AMR for structures and structural components presented in Section 3.6, Sections 3.6.1 through 3.6.7 and Tables 3.6-1 through 3.6-8 of the ANO-1 LRA. In Section 3.6 of the LRA, the AMPs which are credited for managing aging of structures and structural components are identified, but are not described. The AMP descriptions are presented in Appendix B of the LRA.

In the ANO-1 LRA, Section 2.4, "Structures and Structural Components Scoping and Screening Results," the applicant describes its process and results for the scoping and screening of the structures and structural components that are within the scope of license renewal and subject to an AMR. The applicant describes its AMR of the structures and structural components for license renewal in Section 3.6, "Structures and Structural Components," and in Appendix B of its LRA. In the LRA, Tables 3.6-2 through 3.6-8, the applicant presents the seven structures/structural groupings defined in Section 2.4, and provides detailed listings of the components and commodities included in each of the seven structures/structural groupings. For each table entry, the intended function(s), material, environment, aging effect, and applicable AMPs are identified. These tables provide the link between the license renewal scope for structures and structural components and the AMR results.

After the initial review of this information, the NRC staff requested additional information in a letter to the applicant dated June 1, 2000. The applicant responded to these RAIs in a letter to the NRC dated September 7, 2000. The staff requested additional clarification on a number of concerns during a telephone conference call that took place on October 16, 2000. This telephone communication is documented in a letter to the applicant dated October 20, 2000, and the applicant's response to the requests for clarification is documented in a letter to the NRC dated November 2, 2000. The concerns identified by the staff, and the staff's overall evaluation of the structural AMR provided by the applicant are discussed in the paragraphs below.

In the LRA, Section 3.6, the applicant also identifies two TLAAs applicable to structures and structural components that include the following analyses:

- concrete reactor building tendon prestress
- reactor building liner plate and penetrations fatigue

These TLAAs are discussed in Sections 4.5 and 4.6 of the LRA, and the staff's evaluations are documented in Sections 4.5 and 4.6 of this SER.

In the LRA, Section 3.4, "Auxiliary Systems," the applicant states that the spent fuel pool stainless steel liner is within the scope of license renewal and subject to an AMR. The applicant credits the following AMPs with managing applicable aging effects of the spent fuel pool liner:

- spent fuel pool monitoring (App. B, 3.6)
- spent fuel pool level monitoring (App. B, 4.21.8)

NUREG-1557, "Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal," is the guiding document used by the staff for the AMR of structures and structural components. This document includes the stainless steel liner for the spent fuel pool as a structural component. For consistency with NUREG-1557, the AMR of the spent fuel pool stainless steel liner is evaluated in Section 3.3.6.2 of the SER.

3.3.6.1 Structural Steel

In the LRA, Section 3.6.1, “Steel,” the applicant describes the AMR of the ANO-1 structural steel components that are subject to an AMR. In this section, the applicant describes the materials, environments, and applicable aging effects, and identifies the AMP(s) that will be used to manage the different aging effects, which are described in Appendix B of the LRA. In addition, structures and structural components are described in the ANO-1 UFSAR, Section 5. The NRC staff reviewed this information, as well as the applicant’s responses to the staff’s RAIs and requests for clarification, to determine whether the applicant has adequately demonstrated that the requirements of 10 CFR 54.21(a)(3) have been met for structural steel components.

3.3.6.1.1 Technical Information in the Application

In the LRA, Section 3.6.1, “Steel,” the applicant presents the results of the AMR for the ANO-1 structural steel components that are made of carbon steel, stainless steel, and galvanized steel and include such components as the reactor building liner plate, threaded fasteners, anchorage/embedment/attachment, emergency personnel hatches, equipment hatch, and electrical penetrations. The applicant further states that there are no components made of low-alloy steel subject to an AMR. The environments applicable to stainless steel are interior ambient conditions or borated water. The new fuel racks made of aluminum are in an air environment and consequently no aging effects were identified. The reactor building prestress system and threaded fasteners are addressed in Sections 3.6.3 and 3.6.4 of the LRA. In addition, the applicant addresses the spent fuel pool steel gates and the spent fuel steel racks in Section 3.4 of the LRA.

The applicant states that structural steel is exposed to the following environments: protected, non-protected (exterior weather conditions), raw water, borated water, high temperatures, humidity, and radiation. These environments make the structural steel components susceptible to loss of material, cracking, and changes in material properties. The applicant states that “[o]ther potential aging effects and aging mechanisms do not apply to ANO-1 steel components and commodities because of the absence of susceptible material and environmental conditions.”

In the LRA, Section 3.6.1.1, the applicant states that ANO-1 exposed steel is normally coated to manage the loss of material as a result of corrosion. For example, consistent with the ANO-1 UFSAR, Section 5.2.1.3.5, and UFSAR Table 5-2, the reactor building containment liner plate, liner plate attachments, penetrations, and hatches are coated to prevent loss of material as a result of corrosion. Steel encased in concrete is not subject to loss of material because the adjacent concrete provides an alkaline environment, which is an effective corrosion inhibitor. Carbon steel components and commodities that are not coated can experience loss of material as a result of general corrosion in protected and non-protected environments. Loss of material as a result of general corrosion is an applicable aging effect for galvanized steel exposed to weather, but not in a protected environment. For the intake structure, loss of material as a result of corrosion is an applicable aging effect for carbon steel that is submerged, or wetted by raw water, but not for galvanized steel in raw water. Stainless steel in a borated water environment is potentially susceptible to loss of material as a result of pitting corrosion, when significant chloride levels exist in stagnant and low flow areas. However, borated water, in general, does not contain chlorides in sufficient concentration for pitting to occur in stainless steel. Loss of material as a result of boric acid corrosion is an aging effect requiring management for the lower control rod drive service structure skirt, reactor vessel support skirt, flange, anchor bolts, and shear pins.

In the LRA, Section 3.6.1.2, the applicant identifies cracking of steel as a result of fatigue as a TLAA for the reactor building containment liner plate. The results of the TLAA are presented in Section 4.6 of the LRA. For other steel components and commodities, cracking as a result of fatigue was considered based on the original design specification, American Institute of Steel Construction (AISC), for fatigue loading of 2 x 10^6 cycles. However, cracking of welds associated with the control rod drive service structure and reactor vessel support skirt as a result of fatigue is a potential aging effect for ANO-1 that needs an AMR. For reactor building (containment) penetrations, cracking of welds involving dissimilar metals as a result of fatigue and thermal stresses during plant operation is not likely because the welds are located in non-aggressive environments. In addition, reactor building internals include some stainless steel components that are exposed to a borated water environment, at locations not necessary for structural integrity and, therefore, do not require an AMR.

In the LRA, Section 3.6.1.3, the applicant determines that a change in material properties as a result of radiation embrittlement or intermetallic embrittlement is not a potential aging effect for ANO-1 because of limited exposure to radiation as a result of shielding and distance.

In the LRA, Section 3.6.1.4, the applicant summarizes the results of the AMR by identifying the structural components and commodities and the applicable aging effects that require aging management at ANO-1:

C loss of material for the reactor building (containment) liner plate, the control rod drive service structure and reactor vessel support skirt, hatches, penetrations, and steel components and commodities

- cracking of the control rod drive service structure and reactor vessel support

The applicant also performed a review of industry information, NRC generic communications, and ANO-1 operating experience, and did not identify any additional aging effects beyond those discussed in this section.

In the LRA, Section 3.6.1.4, “Conclusion of Aging Effects for Steel,” the applicant identifies a number of AMPs for managing the loss of material and cracking in structural steel components. These AMPs are evaluated by the staff in Section 3.3.6.2.2.2 of this SER and include the following programs and activities:

- Maintenance Rule program
- reactor building leak rate testing program
- ASME Section XI, Inservice Inspection Program (IWE and IWF)

- inspection and preventive maintenance of the ANO-1 polar crane
- service water chemical control program
- battery quarterly surveillance
- boric acid corrosion prevention program
- fire protection program (fire barrier inspections and fire hose station inspections)

3.3.6.1.2 Staff Evaluation

The NRC staff reviewed the information in the LRA, Section 3.3.6.2 and Appendix B, and the ANO-1 UFSAR, Section 5, “Structures.” Upon completing its initial review, the staff requested additional information in a letter to the applicant dated June 1, 2000. The applicant provides the additional information in a letter to the NRC dated September 7, 2000. In a conference call on October 13, 2000, the staff requested additional information, and clarification of the information provided by the applicant in its LRA and RAI responses. The conference call is documented in a letter to the applicant dated October 20, 2000, and the applicant’s response is documented in a letter to the NRC dated November 2, 2000. The NRC staff reviewed this information to determine whether the applicant has adequately demonstrated that the requirements of 10 CFR 54.4, 10 CFR 54.21(a)(1), and 54.21(a)(2) have been met for the reactor building structure and structural components.

3.3.6.1.2.1 Effects of Aging

In the LRA, Section 3.6.1, the applicant identifies carbon steel, stainless steel and galvanized steel as the materials that makeup the structural steel commodity group. The applicant states that structural steel is exposed to protected, non-protected, raw water, boron, high temperatures, humidity, and radiation environments. The staff evaluated this information, and did not identify any omissions in the materials and environments considered by the applicant in its AMR.

The applicant recognizes the following three aging effects for the steel structures and structural components at ANO-1:

- loss of material as a result of general corrosion, galvanic corrosion, crevice corrosion, pitting corrosion, erosion/erosion-corrosion, MIC, and boric acid corrosion
- cracking as a result of fatigue, stress corrosion, and intergranular attack
- change in material properties as a result of radiation embrittlement and intermetallic embrittlement

In Section 3.6.1.4 of ANO-1 LRA, the applicant refers to ANO-1 operating experience as a basis for concluding that no additional aging effect (beyond those discussed above) was identified for steel structures and structural components. However, the applicant did not discuss the operating experience used to support this conclusion, therefore, the staff requested that the applicant provide a description of ANO-1 plant specific operating experience for steel structures and structural components. In a letter to the NRC dated September 7, 2000, and during a conference call on October 13, 2000, the applicant describes the results of plant specific inspections conducted under the existing IWF and boric acid corrosion programs. In addition, the applicant provides a summary of the IWF and boric acid corrosion program results that supported its conclusion.

In the LRA, Tables 3.6-1 through 3.6-8, the applicant summarizes the results of its AMR by including the different structures and structural components within the scope of license renewal. These tables summarize the intended functions, materials, environments, and aging effect(s) that apply to each component/commodity group, and the AMPs that will be used to manage the applicable aging effects. The staff reviewed these tables and identified a number of areas where additional information was needed.

Fretting and lockup as a result of mechanical wear of hinges, locks and closure mechanisms for the containment hatches were excluded from the AMR in the LRA. In a letter to the applicant dated June 1, 2000, the staff requested that the applicant identify where in the LRA is the AMR for these components, or provide a technical justification for not considering these aging effects in the AMR. In a letter to the NRC dated September 7, 2000, and during a conference call on October 13, 2000, the applicant states that the components in question are active and, therefore, are not subject to an AMR. The applicant was advised that the current staff position considers these components to be within the scope of license renewal and subject to an AMR for mechanical wear because they support the pressure boundary intended function by keeping the hatch properly aligned without moving, or without a change of configuration or properties. In a letter to the NRC dated December 12, 2000, the applicant states that (after reviewing the "Generic Aging Lessons Learned Report") it will include hinges, locks and closure mechanisms within the scope of components subject to an AMR, and to manage the potential loss of material using ASME Section XI, IWE inspections, as is currently required by 10 CFR 50.55(a). The staff found this response acceptable.

ASME Section XI, Subsection IWF identifies the loss of component support as an applicable aging effect for ASME Class 1, 2, 3 and MC piping and component supports. The applicant did not include this potential aging effect in the ANO-1 AMR. The staff requested that the applicant provide a technical basis for excluding this aging effect from its AMR. In its response to the NRC dated September 7, 2000, the applicant explained that component support is an intended function, and the aging effect that affect this function is loss of material. As discussed in Section 3.6.1 of the LRA, the loss of material of steel piping components is included in the AMR, and is managed by the IWF AMP. In response to a staff's request for additional clarification during an October 13, 2000, conference call, the applicant confirms that its IWF AMP will be implemented consistently with the requirements of 10 CFR 50.55(a) throughout the period of extended operation, satisfying the staff's concern with the aging management of ASME Class 1, 2, 3 and MC piping and component supports.

In the LRA, Section 2.4.1, the applicant states that the welds of integral attachments to the reactor building liner were reviewed with the reactor building internals AMR. As indicated in Table 3.6-3, items in this component/commodity group are managed using the Maintenance Rule and IWF AMPs. It is the staff's position that welds of integral attachments to the liner plate are liner plate components that need to be subject to an AMR, and that are adequately managed by ASME Section XI, Subsection IWE. The staff requested additional clarification relating to this concern in a letter to the applicant dated June 1, 2000. In a letter to the NRC dated September 7, 2000, the applicant confirms that the integral attachments to the liner plate will be managed by the ASME, Section XI IWE activities consistent with the requirements of 10 CFR 50.55(a) throughout the period of extended operation, as well as the Maintenance Rule AMPs, satisfying the staff's concern.

In the LRA, Table 3.6-8, the applicant includes thermashield as a subcategory of the steel components commodity. Thermashield is also described in the LRA, Section 3.6.1, as a subcategory of steel, but no further reference to this material is made anywhere else in the LRA. The staff requested additional information on thermashield in its June 1, 2000, letter to the applicant. In a letter to the NRC dated September 7, 2000, the applicant describes the material, its intended function, and its use as a structural component. With this additional information, the staff was able to complete its review of thermashield without any additional concerns.

In the LRA, Section 3.6.1, the applicant states that the spent fuel pool steel liner is addressed in Sections 2.3.3 and 3.4 of the LRA. As noted above, the staff considers the spent fuel pool liner to be a structural steel component, and its AMR is evaluated in the following paragraphs. In Section 3.4.2 of the LRA, the applicant states that cracking is a potential aging effect for the spent fuel pool liner. In Section 3.4.3, the applicant lists the spent fuel pool monitoring program (a new program described in App. B, Section 3.6) and the spent fuel pool level monitoring program (described in App. B, Section 4.21.8) as the AMPs for the spent fuel pool liner plate. The spent fuel pool monitoring program will be used to monitor the spent fuel pool monitoring trench drains. The spent fuel pool level monitoring program is used to monitor the water level in the spent fuel pool. The staff has previously concluded that stress corrosion cracking and crevice corrosion of the spent fuel pool stainless steel liners are adequately managed by periodic monitoring of the leak chase system drain lines. The program defined by the applicant correctly identifies the aging effects of concern, but the description of these two programs in the LRA, alone, was not sufficient for the staff to complete its evaluation. Therefore, the staff requested additional information about the leakage collection system for the spent fuel pool, and the two credited AMPs. In a letter to the NRC dated September 7, 2000, the applicant describes the spent fuel pool leakage collection system, and explained that the monitoring of the spent fuel pool water level and trench system drain will adequately provide for fuel pool leakage detection. The staff reviewed this additional information, and found it to be acceptable.

3.3.6.1.2.2 Aging Management Programs

The LRA, Appendix B, contains the applicant's description of the AMPs used to manage the effects of aging of SCs such that the intended functions will be maintained consistent with the applicant's CLB for the period of extended operation. The programs that are credited with managing applicable aging effects of steel structures and structural components at ANO-1 include the following:

- Maintenance Rule program (Section 4.13)
- reactor building leak rate testing program (Section 4.16)
- ASME Section XI, Inservice Inspection Programs IWE and IWF (Sections 4.3.5 and 4.3.4)
- inspection and preventative maintenance of the ANO-1 polar crane (Section 4.10)
- service water chemical control program (Section 4.6.5)
- battery quarterly surveillance (Section 4.21.2)
- boric acid corrosion prevention (Section 4.5)
- fire protection program (Sections 4.8.1 and 4.8.2)

The staff reviewed the information provided in the LRA for these AMPs, and determined that the Maintenance Rule program (Appendix B, Section 4.13), reactor building leak rate testing (Appendix B, Section 4.16), inspection and preventative maintenance of the ANO-1 polar crane (Appendix B, Section 4.10), service water chemical control (Appendix B, Section 4.6.5), battery quarterly surveillance (Appendix B, Section 4.21.2), and fire protection programs (Appendix B, Sections 4.8.1 and 4.8.2) are acceptable to manage the aging specified in each program's purpose and scope statements. A summary description for each of the programs found acceptable are provided in the following paragraphs.

Maintenance Rule Program

The Maintenance Rule AMP involves structural walkdowns and visual inspections of structures, consistent with the requirements of 10 CFR 50.65, to detect indications of cracking, loss of material and/or change in material properties. The LRA, Appendix B, Section 4.13 provides a description of the Maintenance Rule AMP, which is credited for managing aging for a wide variety of structural elements within the license renewal scope, including many steel elements. As stated in this description, the walkdowns are performed periodically at a frequency that varies depending on the structure or component being inspected, and the acceptance criterion is no visual indication of cracking, loss of material, or change of material properties. The Maintenance Rule AMP is evaluated in Section 3.3.1 of this SER, and the staff found that this program satisfies the ten attributes for an acceptable AMP, as defined in the SRP for license renewal, Section 3.0, and can be credited for license renewal. Refer to Section 3.3.1 of this SER for a detailed review of this AMP.

Reactor Building Leak Rate Testing Program

The reactor building leak rate testing program measures the reactor building overall integrated leakage-rate and the local leakage-rate across individual penetration components in accordance with the requirements of 10 CFR 50 Appendix J. The integrated leakage test is performed every 10 years while the local leakage tests are performed as needed to ensure penetration component integrity. The leak rate results provide indication of containment component degradation, and containment pressure boundary integrity. The reactor building leak rate testing AMP is evaluated in Section 3.3.1 of this SER, and the staff found that this program satisfies the ten attributes for an acceptable AMP, as defined in the SRP for license renewal, Section 3.0, and can be credited for license renewal. Refer to Section 3.3.1 of this SER for a detailed review of this AMP.

ASME Section XI, Inservice Inspection Program IWE

The ASME Section XI, Inservice Inspection Program IWE involves inspections to identify and correct degradation of Class MC pressure-retaining components and their integral attachments. In Appendix B, Section 4.3.4 of the LRA, the applicant states that the scope of the ANO-1 program includes inspections of the reactor building liner plate. The applicant further states that the ASME B&PV Code, Section XI, 1992 Edition, 1993 Addendum for pressure testing was used to develop the program. The ASME Section XI, Inservice Inspection Program IWE AMP is evaluated in Section 3.3.1 of this SER, and the staff found that this program satisfies the ten attributes for an acceptable AMP, as defined in the SRP for license renewal, Section 3.0, and can be credited for license renewal. Refer to Section 3.3.1 of this SER for a detailed review of this AMP.

Inspection and Preventive Maintenance Program for the ANO-1 Polar Crane

The inspection and preventive maintenance program for the ANO-1 polar crane provides for inspections, tests and maintenance of the polar crane in compliance with ANSI B30.2 to satisfy commitments to NUREG-0612. The polar crane steel components are inspected annually. The inspections ensure the structural integrity of the polar crane.

[Scope of Program] Structural Steel associated with the ANO-1 polar crane is maintained to NUREG-0612 "Control of Heavy Loads at Nuclear Power Plants."

[Preventive Action] No preventive actions were identified, and the staff did not find a need for such action.

[Parameters Monitored/Inspected] Deformation, cracking, loss of material due to corrosion, and the crane bridge system bolt tightness are the parameters monitored by the inspection and preventive maintenance program for the ANO-1 polar crane.

[Detection of Aging Effects] Polar crane is inspected, tested, and maintained in compliance with ANSI B30.2. Visual inspection of polar crane steel components is performed and bolt tightness is tested by hand. Inspection is conducted every 18 months, in conjunction with other periodic crane inspection activities.

[Monitoring and Trending] No trending activities are identified, and the staff did not find a need for trending.

[Acceptance Criteria] No visual indications of deformation, cracking, loose or corroded members, and no loose or missing bolts.

[Operating Experience] The applicant states that inspections have not identified loss of material due to corrosion. However, visual inspections have proven effective in identifying indications of the applicable aging effects for the polar crane SCs. Continuation of program should provide reasonable assurance that aging of the polar crane will be adequately managed consistent with the CLB for the period of extended operation.

The staff found the inspection and preventive maintenance program for the ANO-1 polar crane program acceptable for managing the applicable effects of aging so that the intended function(s) can be maintained consistent with the CLB for the period of extended operation.

Service Water Chemical Control Program

The service water chemical control program maintains service water chemistry consistent with the guidance provided by NRC Generic Letter 89-13. Sampling is performed daily, biweekly, weekly, or as required, based on plant conditions, and acceptance criteria are consistent with industry guidance. The proper control of water chemistry reduces the likelihood of loss of material. The service water chemical control AMP is evaluated in Section 3.3.1 of this SER, and the staff found that this program satisfies the ten attributes for an acceptable AMP, as defined in the SRP for license renewal, Section 3.0, and can be credited for license renewal. Refer to Section 3.3.1 of this SER for a detailed review of this AMP.

Battery Quarterly Surveillance Program

The battery quarterly surveillance program includes inspection of the battery racks and associated components for physical damage, deterioration and material loss in accordance with the recommendations in ANSI/IEEE Standard 450-1980. The battery racks are visually inspected on a quarterly basis in association with other battery maintenance activities. The inspections ensure the structural integrity of the battery racks.

[Scope of Program] Inspection of seismically-qualified battery racks is necessary to ensure structural integrity. ANSI/IEEE Standard 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations" is referenced.

[Preventive Action] No preventive actions were identified, and the staff did not see a need for such action.

[Parameters Monitored/Inspected] The parameters monitored include physical damage or abnormal deterioration of the battery racks and associated threaded fasteners including loss of material.

[Detection of Aging Effects] Visual inspection is performed on a quarterly basis. The staff found this inspection interval acceptable.

[Monitoring and Trending] No trending activities are identified and the staff did not see a need for such activities.

[Acceptance Criteria] No visual indications of degradation. When damage is noted, deficiencies are evaluated to determine if an intended function is compromised.

[Operating Experience] ANO-1 condition report summaries have not identified damage to or deterioration of battery racks.

The staff concludes that continuation of this program can adequately manage aging of battery racks consistent with the CLB for the period of extended operation.

Boric Acid Corrosion Prevention Program

The boric acid corrosion prevention (BACP) program is credited for monitoring the boric acid corrosion of carbon steel surfaces exposed to leakage from borated water. The BACP AMP is evaluated in Section 3.3.1 of this SER, and the staff found that this program satisfies the ten attributes for an acceptable AMP, as defined in the SRP for license renewal, Section 3.0, and can be credited for license renewal. Refer to Section 3.3.1 of this SER for the staff's review of this AMP.

Fire Protection Program

The elements of the fire protection program involve visual inspection of fire barriers and fire hose reels to ensure that they can perform their intended functions in accordance with 10 CFR 50.48, Appendix R to 10 CFR Part 50, and Operating License DPR-51.2.c.8.

[Scope of Program] The scope of the fire protection program includes the fire walls and fire floors, as indicated on the ANO-1 fire protection drawings, that are required to meet 10 CFR 50.48. Fire doors and hatches, fire damper mountings, fire wraps, and penetration fire stops associated with the fire walls and fire floors are also within the scope of this program.

[Preventive Action] No preventive actions were identified, and the staff did not see a need for such action.

[Parameters Monitored/Inspected] Discussed in conjunction with "Detection of Aging Effects," in the following paragraph.

[Detection of Aging Effects] Visual inspections are used to detect (1) cracking of masonry walls; (2) loss of material for fire doors and hatches, including threaded fasteners; (3) loss of material, cracking, delaminating, change in material properties for fire wraps and associated banding;
(4) loss of material, cracking, delaminating, separation, and change in material properties for penetration fire stops. Fire doors and hardware and fire barriers except penetration seals are inspected at least once per 18 months. At least 10 percent of sealed penetrations are inspected every 18 months. The inspection scope is expanded if degradation is found until a 10 percent sample of the degraded type of seal is found with no visual degradation. Each penetration seal is inspected at least once every 15 years.

[Monitoring and Trending] No trending activities are identified, and the staff did not see a need for such activities.

[Acceptance Criteria] Acceptance criteria are documented in the fire barrier inspection procedures. The visually observed condition must be the same as the as-designed condition.

[Operating Experience] This program has detected voids and gaps in fire barrier penetration seals; torn boot seals; loose damming material due to building vibration or temperature-related contraction; and cracks in block walls near penetrations. Damaged penetration seals and fire barriers have been repaired. Detection and correction of degradation by this program are evidence of its effectiveness, and provide reasonable assurance that aging of fire barriers will be adequately managed consistent with the CLB for the period of extended operation.

The staff found the fire protection program acceptable for managing the applicable effects of aging so that the intended function(s) can be maintained consistent with the CLB for the period of extended operation.

In a letter to the NRC dated September 7, 2000, the applicant clarifies that the multiple AMPs used in the tables throughout Section 3 of the LRA are meant to be redundant in many cases to provide additional assurance that the applicable aging effects will be adequately managed for the period of extended operation.

Upon completing its initial review of the ASME Section XI, Inservice Inspection Program IWE, the staff requested that the applicant provide additional clarification to verify that the IWE activities will be implemented, in accordance with 10 CFR 50.55a (1992 Edition, 1992 Addenda), and that the additional requirements for the inspection of inaccessible areas defined in NUREG-1611 should be credited as an AMP for the steel components of the containment.

In response to a staff's request for additional clarification during an October 13, 2000 conference call, the applicant confirmed that its IWE AMP will be implemented consistently with the requirements of 10 CFR 50.55(a) throughout the period of extended operation, satisfying the staff's concern with the aging management of Class MC pressure-retaining components and its integral attachments. With respect to the staff requirements for the inspection of accessible and inaccessible areas defined in NUREG-1611, the applicant informed the staff that both the accessible and inaccessible concrete were designed and constructed to have low permeability and high resistance to aggressive chemical solutions. Concrete structures were designed and constructed in accordance with ACI 318-63, and the relevant ACI and ASTM standard to ensure a high cement content, low water-cement ratio, and proper curing. In addition, groundwater at the time of construction was verified not to be aggressive. In additional, testing of ground water in 1996 verified that the ground water continues to be non-aggressive over the operating history of the plant. On the basis of these past plant operating data and experience, the staff found this information acceptable to address IWE activities and inaccessible areas.

The ASME Section XI, Inservice Inspection Program IWF involves inspections to identify and correct degradation of ASME Class 1, 2, and 3, or MC component supports. As stated in Appendix B, Section 4.3.5, of the ANO-1 LRA, the scope of the program credited for license renewal includes component supports for ASME Class 1, 2, 3 or MC components. It is further stated that the ASME Section XI, 1992 edition, 1993 Addenda for pressure testing was used to develop this program. Upon completing its initial review, the staff requested that the applicant provide additional clarification to verify that the IWF activities will be implemented throughout the period of extended operation, in accordance with 10 CFR 50.55a. In response to a staff's request for additional clarification during an October 13, 2000 conference call, the applicant confirms that its IWF AMP will be implemented consistently with the requirements of 10 CFR 50.55a throughout the period of extended operation, satisfying the staff's concern with the aging management of ASME Class 1, 2, 3, and MC component supports.

In a letter to the NRC dated September 7, 2000, the applicant provides additional information with regard to applicable code editions. The applicant indicates that the correct code edition for Section XI, IWE, IWF and IWL inspections credited for license renewal is the 1992 Edition, including the 1992 Addenda. This is consistent with the code editions endorsed by the latest revision to 10 CFR 50.55a. The applicant also states that the programs meet the scope and requirements of 10 CFR 50.55a as well as the NRC approved alternatives (relief requests) for the CLB. During a conference call on October 13, 2000, the staff advised the applicant that current relief requests are not applicable to license renewal because they are time-limited and will expire before entering the period of extended operation. The applicant states that they did not intend to imply that any relief request will restrict their compliance to code requirements during the period of extended operation. In a letter to the NRC dated November 2, 2000, the applicant states that they will continue to comply with all applicable requirements of 10 CFR 50.55a during the period of extended operation, thus resolving this concern.

In the LRA, Section 4.5 of Appendix B, the applicant states that the boric acid corrosion prevention (BACP) program is credited for monitoring the boric acid corrosion of carbon steel surfaces exposed to leakage from borated water. This program includes visual inspections to identify pressure boundary leaks. In addition, the BACP program considers the possibility for boric acid leakage from other systems to result in aging of carbon steel components, or the accumulation of boric acid in insulation, but did not clearly describe visual inspections of adjacent structures, components and supports for evidence of leakage and corrosion. The staff requested additional information to clarify the scope of the BACP program. In a letter to the NRC dated September 7, 2000, the applicant confirms that adjacent components and structures are included in the scope of the BACP program, thus resolving the staff's concern.

3.3.6.1.3 Conclusions

On the basis of the review described above, the staff finds that there is reasonable assurance that the applicant has demonstrated that the effects of aging of steel components at ANO-1 will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

3.3.6.2 Concrete

3.3.6.2.1 Technical Information in the Application

In the LRA, Section 3.6.2 "Concrete," the applicant presents the results of the AMR for the concrete components with the exception of the prestressed concrete in the reactor building. The applicant also states that the ANO-1 concrete components and commodities are designed in accordance with ACI 318-63 and constructed in accordance with ACI 301 using material designs conforming to ACI and ASTM standards, which provide a good-quality, dense, low-permeability concrete.

The applicant identifies the following environments for the four major structures included within the scope of license renewal and subject to an AMR:

- Reactor Building
  - internal - high temperature, humidity, radiation
  - external, above grade - external atmospheric conditions
  - external, below grade - backfill and groundwater
- Auxiliary Building
  - internal - elevated temperature, humidity, radiation
  - external, above grade - hot and cold temperatures, elevated humidity, rainfall, snowfall, wind
  - external, below grade - chemicals in groundwater, contact with backfill materials

C Intake Structure

- S internal - elevated temperatures and humidity
- S external, above grade - rainfall, snowfall, wind
- S external, below grade - chemicals in raw water (i.e., groundwater, Lake Dardanelle, emergency cooling pond), contact with backfill materials

C Q-Condensate Storage Tank (Q-CST) Foundation

- drilled piers for foundation - chemicals in groundwater, contact with backfill materials

These environments also apply to the remaining ANO-1 concrete structures.

The applicant identifies the following applicable aging effects for concrete structures and structural components:

- loss of material
- cracking
- change in material properties, as a result of elevated temperature

The applicant also states that other potential aging effects are not applicable to ANO-1 concrete structures because of the absence of susceptible material and environmental conditions.

Loss of material as a result of abrasion is an applicable aging effect for the intake structure exterior concrete wall at the lake level. Loss of material (as a result of cavitation and elevated temperature) is not an applicable aging effect because concrete structures are not exposed to water velocity sufficient to cause aging, and are not exposed to temperatures that exceed code-imposed temperature limits for concrete. Cracking caused by restraint against free contraction is an applicable aging effect for masonry block walls. Cracking and changes in material properties as a result of elevated temperature is not an applicable aging effect because concrete structures, structural components, and commodities are not exposed to temperatures that exceed code-imposed temperature limits. In the LRA, Section3.6.2.4, “Conclusion of Aging Effects for Concrete,” the applicant identifies the AMPs for managing the loss of material and cracking of concrete. These AMPs are evaluated by the staff in Section 3.3.6.2.2.2 of this SER and include the following programs and activities:

- Maintenance Rule activities
- fire barrier inspection

3.3.6.2.2 Staff Evaluation

3.3.6.2.2.1 Effects of Aging

In the LRA, Section 3.6.2, the applicant considers all the concrete structures that are designed and constructed in accordance with ACI and ASTM standards, and that are within the scope of license renewal. In general, the applicant describes the environments experienced by ANO-1 structures similar to internal, external, and below grade environments that are typical for nuclear power plant. However, the applicant does identify some less-common environments as well, such as chemicals in groundwater and chemicals in raw water. The applicant also identifies loss of material, cracking, and change in material properties as the applicable aging effects for concrete (including non-shrink grout, epoxy grout, embedments and reinforcement).

The applicant states that the loss of material as a result of abrasion and cavitation is limited to concrete that is continuously exposed to flowing water. The applicant has identified that the intake structure exterior wall at the normal lake level is the only concrete structure to which this aging effect is applicable. The applicant confirms that the highest average water velocity in the intake structure does not exceed the threshold for cavitation damage and, therefore, cavitation is not applicable. The staff did not find any concerns in the applicant's assessment of loss of material as a result of abrasion and cavitation.

The applicant concludes that loss of material, cracking, and change in material properties as a result of elevated temperature are not applicable aging effects that need to be managed at ANO-1. In accordance with NUREG-1557, degradation from exposure to elevated temperatures is not a concern for concrete structures maintained at operating temperatures less than 150EF and local area temperatures are less than 200EF. The applicant states that ANO-1 concrete structures, structural components, and commodities are not exposed to temperatures above these thresholds. The staff did not identify any concerns with the applicant's assessment of loss of material, cracking, and change in material properties as a result of elevated temperature.

The applicant identifies cracking as a result of restraint against free contraction for masonry block walls as an aging effect that needs to be managed. The staff did not identify any concerns with the applicant's inclusion of this aging effect in its AMR of masonry block walls.

The applicant states that they reviewed its plant operating experience to determine that no additional aging effects, beyond those discussed above, have been identified, but has not presented any specific information substantiating its position. Consequently, in a letter to the applicant dated June 1, 2000, the staff requested that the applicant discuss the results of any inspection activities that support this conclusion. In a letter to the NRC dated September 7, 2000, the applicant indicates that a Maintenance Rule baseline inspection was recently performed. Cracking in concrete was observed, however, the cracks did not exceed established acceptance criteria. Leakage of water into the tendon access gallery was identified, minor cracking and leaching were observed on reactor building concrete surfaces, and some areas of concrete had exposed rebar that were slightly rusted. The applicant assessed these occurrences of degradation as having no effect on the applicable intended function(s) and, therefore requiring no corrective action. In a telephone conversation with the applicant on October 13, 2000, the staff requested additional description of the rusting condition of the rebar, and a technical justification for the statement that the intended function of concrete is not challenged by this degradation. In a letter to the NRC dated November 2, 2000, the applicant identifies the exposed, rusting rebar as a small area of exposed rebar in the northwest corner of the intake structure's exterior wall, which was caused by mechanical damage, not aging. This mechanical damage was determined not to have an effect on the load-bearing capacity of the structure, and the rust was characterized as "slight" and, therefore, not having an effect on the intended function(s). The staff had no additional concerns relating to the results reported from the Maintenance Rule baseline inspection.

The staff was also concerned that the description of potential concrete aging effects and the corresponding aging mechanisms in the LRA were incomplete. A number of potential aging effects and aging mechanisms identified in NUREG-1557 were not specifically discussed in the LRA. These include the following items:

- scaling, cracking, and spalling as a result of freeze/thaw
- increase of porosity and permeability, loss of strength as a result of leaching of calcium hydroxide
- increase of porosity and permeability, cracking, and spalling as a result of aggressive chemical attack
- expansion and cracking as a result of reaction with aggregates
- cracking, spalling, loss of bond, and loss of material as a result of corrosion of embedded steel
- cracks, distortion, and increase in component stress level as a result of settlement
- reduction in foundation strength as a result of erosion of porous concrete sub-foundation

In a letter to the NRC dated November 2, 2000, the applicant provides the following information that, consistent with 10 CFR Part 54, identifies the aging effects specifically evaluated, and presented in its LRA:

Loss of Material

- freeze-thaw
- abrasion and cavitation
- elevated temperature
- aggressive chemicals
- corrosion of embedded steel reinforcing

Cracking

- freeze-thaw
- reaction with aggregates
- shrinkage
- elevated temperature
- radiation
- fatigue
- settlement

Change in Material Properties

- leaching of $Ca(OH)2$
- aggressive chemicals
- elevated temperature

- radiation
- creep

In the same letter, the applicant addresses settlement and reduction in foundation strength due to porous concrete. The applicant states that the structures that are within the scope of license renewal are located on either bedrock or compact fill, and are not susceptible to settlement. The applicant also states that its AMPs for containment concrete includes ASME Section XI IWL containment concrete surface inspections, as well as Maintenance Rule structural inspection activities. In addition, concrete structures were designed and constructed in accordance with ACI 318-63, “Building Code Requirements for Reinforced Concrete,” and its relevant ACI standards and ASTM standards resulting in accessible and inaccessible concrete with high cement content, low water-cement ratio, and proper curing. Therefore, the concrete of concern has low permeability. The staff reviewed this information and found no omissions in the aging effects considered in the applicant’s AMR of concrete structures, structural components, and commodities.

The applicant does not address aging effects for inaccessible areas in the LRA. As stated above, the applicant identifies chemicals in groundwater and chemicals in raw water as environments applicable to ANO-1 concrete. The applicant does not indicate whether such environments, or other aging mechanisms, have an adverse effect on concrete structures in inaccessible areas. Therefore, additional information and clarification were requested. In letters to the NRC dated September 7, and November 2, 2000, the applicant states that the concrete structures at ANO-1 were designed and constructed in accordance with ACI 318-63, “Building Code Requirements for Reinforced Concrete,” and its relevant ACI standards and ASTM standards resulting in accessible and inaccessible concrete with high cement content, low water-cement ratio, and proper curing. Thus, the concrete has low permeability and a high resistance to aggressive chemical solutions. In addition, the plant ground water was not aggressive during plant construction, and remains non aggressive as supported by the most recent ground water test data from May 6, 1996. The staff reviewed the additional information and found no concerns regarding the aging of inaccessible areas.

3.3.6.2.2.2 Aging Management Programs

The LRA, Appendix B, contains the applicant’s description of the AMPs used to manage the effects of aging such that the intended function(s) of the SCs that are within the scope of license renewal and subject to an AMR (including the concrete structures and structural components), will be maintained consistent with the applicant’s CLB for the period of extended operation. In the LRA, Section 3.6.2.4, the applicant identifies the following AMPs that are used to manage the effects of aging for concrete structures:

- Maintenance Rule program
- fire protection program (fire barrier inspections)

The staff reviewed the information provided in the LRA for the AMPs used by the applicant to manage the aging of concrete structures, including the Maintenance Rule program (Appendix B, Section 4.13), and fire protection programs (Appendix B, Sections 4.8.1 & 4.8.2). Refer to Section 3.3.1 of this SER for a detailed review of the Maintenance Rule AMP, and Section 3.3.6.1.2.2 of this SER for the review of the fire protection program.

3.3.6.2.3 Conclusions

On the basis of the review described above, the staff finds that there is reasonable assurance that the applicant has demonstrated that the effects of aging of concrete structures at ANO-1 will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

3.3.6.3 Prestressed Concrete

3.3.6.3.1 Technical Information in the Application

In the LRA, Section 3.6.3, "Prestressed Concrete," the applicant presents the results of the AMR for the ANO-1 reactor building (containment) tendon wires, tendon anchorage, dome and cylinder walls. The applicant states that the prestressed concrete components are designed in accordance with ACI 318-63 and mixed in accordance with ACI 301 using ingredients conforming to ACI and ASTM standards, which provide a good-quality, dense, low-permeability concrete. The ANO-1 UFSAR, Section 5.2, is referenced for the specific codes and editions used in the reactor building design and fabrication.

Except for tendon end caps, post-tensioning system components are not exposed to external atmospheric conditions. The bottom anchorage for the vertical tendons is exposed to the tendon gallery environment. Tendon wires are encased in bulk fill grease, and the tendon anchors are enclosed in sealed end caps.

The applicant identifies the following four aging effects for the ANO-1 prestressed concrete in an external-ambient environment:

- loss of material
- cracking
- change in material properties
- loss of prestress

The applicant also states that other potential aging effects do not apply to ANO-1 prestressed concrete components as a result of the absence of susceptible material and environmental conditions.

In the LRA, Section 3.6.3.1, the applicant discusses the locations and circumstances where loss of material is a potentially applicable aging effect. Although stressed components of the post-tensioning system are normally well protected from corrosion, the carbon steel tendon wires and tendon anchorage are potentially susceptible to loss of material as a result of general corrosion. The tendons and anchorage are enclosed in ducts and end caps, which are filled with bulk-fill grease, to prevent corrosion. The bottom vertical tendon anchorage is exposed to moisture in the tendon gallery, but the end caps are coated for corrosion protection, the anchorage is encased in grease, and they are not directly in contact with water. The applicant states that corrosion at this location has not been identified at ANO-1. However, potential grease leakage could occur, most likely at the tendon anchorage, exposing the tendons to environmental conditions.

In the LRA, Section 3.6.3.2, the applicant discusses two additional aging effects, and the locations and circumstances for aging to occur in prestressed concrete structures. Minor cracking of concrete has been observed on a few exposed concrete surfaces of the ANO-1 reactor building, and cracking of the reactor building dome and cylinder wall is an applicable aging effect. In addition, change in material properties is an applicable aging effect for the reactor building dome and cylinder wall. The applicant also identifies loss of prestress as a result of material strain as a TLAA, which is evaluated in LRA Section 4.5. The applicant also states that as a result of a review of industry information and NRC generic communications, no additional aging effects requiring management beyond those discussed in this section have been identified.

In the LRA, Section 3.6.3.5, “Conclusion of Aging Effects for Prestressed Concrete,” the applicant identifies the following programs and activities that will be used to manage the effects of aging applicable to ANO-1 prestressed concrete:

- ASME Section XI Inservice Inspection Program - IWL Inspections
- Maintenance Rule program

3.3.6.3.2 Staff Evaluation

3.3.6.3.2.1 Effects of Aging

In the LRA, Section 3.6.3, the applicant identifies the prestressed concrete structures and structural components for the ANO-1 reactor building that are within the scope of license renewal and subject to an AMR. These structures and structural components consist of tendon wires, tendon anchorage, dome, and cylinder walls, which is prestressed. The staff’s evaluation of the SCs that are within the scope of license renewal and subject to an AMR is provided in Section 3.3.6.4.2.2 of this SER.

The applicant identifies loss of material, cracking, change in material properties, and loss of prestress (as a result of material strain) as the applicable aging effects for prestressed concrete structures and structural components (including non-shrink grout, epoxy grout, embedments and reinforcement). The applicant states that other potential aging effects do not apply to ANO-1 prestressed concrete components because of the absence of susceptible material and environmental conditions. Each of these four potential aging effects are evaluated separately in the LRA, Sections 3.6.3.1 for loss of material, 3.6.3.2 for cracking, 3.6.3.3 for change in material properties, and 3.6.3.4 for loss of prestress.

The staff reviewed this information and found no omissions in the aging effects considered in the applicant’s AMR of prestressed concrete structures and structural components. However, in Section 2.4.1.2 of the LRA, the applicant identifies the tendon access gallery as a separate structure from the reactor building, which does not perform an intended function, and consequently is outside the scope of license renewal. Although the staff found this assessment consistent with 10 CFR 54.4, adverse environmental conditions in the tendon access gallery can have an aging effect on the lower tendon anchor components and surrounding concrete as documented in NUREG-1522, "Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures," June 1995. Therefore, the staff requested that the applicant provide additional information to address this concern. In a letter to the NRC dated September 7, 2000, the applicant describes the tendon access gallery as having minor water in-leakage not requiring corrective action. Since the tendon gallery is open to the auxiliary building, and the ventilation fans of the tendon access gallery operate in conjunction with the auxiliary building ventilation system, the environment in the gallery is essentially the same as the auxiliary building. The applicant states that they have not observed abnormal levels of humidity or contaminants in the tendon access gallery. Corrosion was identified in anchorage components during the 10-year and 15-year inservice tendon inspections, but this loss of material did not adversely affect the intended function(s) of these components. The staff reviewed this information, and found no additional concerns relating to the prestressed concrete aging effects identified by the applicant.

3.3.6.3.2.2 Aging Management Programs

The LRA, Appendix B, contains the applicant's description of the AMPs used to manage the effects of aging such that the intended function(s) of the SCs that are within the scope of license renewal and subject to an AMR (including the prestress concrete structures and structural components) will be maintained consistent with the applicant's CLB for the period of extended operation. In the LRA, Section 3.6.3.5, the applicant identifies the following AMPs that are used to manage the effects of aging for the prestressed concrete structures:

- ASME Section XI Inservice Inspection Program - IWL inspections
- Maintenance Rule program

The staff reviewed the information provided in the LRA for the AMPs used by the applicant to manage the aging of prestressed concrete structures. The staff determined that the Maintenance Rule program (Appendix B, Section 4.13), and ASME Section XI Program ISI - IWL inspection programs (Appendix B, Section 4.3.6) are acceptable to manage the aging specified in each programs purpose and scope statements. Refer to Section 3.3.1 of this SER for a detailed review of these AMPs.

The ASME Section XI Program, IWL inspections are credited with managing the applicable aging effects of the ANO-1 prestressed concrete. The NRC staff reviewed the description of the IWL Inspection Program as presented in Section 4.3.6 of Appendix B. The applicant states that the purpose of the IWL Inspection program is to provide instructions and documentation requirements for assessing the quality and structural performance of the reactor building's post-tensioning systems and concrete surfaces. In addition, the IWL inspections are performed on the reactor building's post-tensioning systems and concrete components that are subject to an AMR as identified in Sections 2.4 and 3.6 of the LRA. Upon completing its initial review, the staff requested that the applicant provide additional clarification to verify that the IWL activities will be implemented in its entirety, in accordance with ASME Section XI, Subsection IWL, and the additions and modifications specified in 10 CFR 50.55a. In response to a staff's request for additional clarification during an October 13, 2000, conference call, the applicant confirmed that its IWL AMP will be implemented consistently with the requirements of 10 CFR 50.55(a) throughout the period of extended operation, satisfying the staff's concern with the aging management of the reactor building post-tensioning systems and concrete components.

In addition, the description of the IWL Inspection Program in Section 4.3.6 of Appendix B does not adequately address how aging effects of prestress concrete components in inaccessible areas are managed. In accordance with 10 CFR 50.55a licensees are required to evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas. For license renewal, applicants' must also evaluate, on a case-by-case basis, the acceptability of inaccessible areas when conditions in accessible areas may not indicate the presence of or result in degradation to such inaccessible areas, in order to ensure that the intended functions of the SCs will be maintained consistent with the CLB during the period of extended operation. In its September 7, and November 2, 2000, responses to this concern, the applicant states that the concrete structures at ANO-1 were designed and constructed in accordance with ACI 318-63, "Building Code Requirements for Reinforced Concrete," and its relevant ACI standards and ASTM standards resulting in accessible and inaccessible concrete with high cement content, low water-cement ratio, and proper curing. Thus, the concrete has low permeability and a high resistance to aggressive chemical solutions. In addition, the plant ground water was not aggressive during plant construction, and remains non-aggressive as supported by the most recent ground water test data from May 6, 1996. The staff reviewed this information and found no additional concerns relating to the aging of inaccessible areas.

3.3.6.3.3 Conclusions

On the basis of the review described above, the staff finds that there is reasonable assurance that the applicant has demonstrated that the effects of aging of prestressed concrete components at ANO-1 will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

3.3.6.4 Threaded Fasteners

3.3.6.4.1 Technical Information in the Application

In the LRA, Section 3.6.4, "Threaded Fasteners," the applicant states that threaded fasteners within the scope of license renewal, are made of the same materials, and are subjected to the same environmental conditions as the ANO-1 steel components, and commodities, which are described in LRA, Section 3.6.1. High-strength bolts in Category 1 applications are either ASTM-A325 carbon steel or ASTM-A490 alloy steel, and ASTM-A307 carbon steel, with or without zinc coating.

The applicant also states that, except for embedded bolts, fasteners associated with the spent fuel pool gates and liner, the spent fuel racks, and sluice gates are addressed in Sections 2.3.3 and 3.4 of the LRA. Embedded bolts are addressed in Section 3.6.2 of the LRA.

The applicant identifies the following applicable aging effects for threaded fasteners:

- loss of material
- cracking
- change in material properties

The applicant states that other potential aging effects do not apply to ANO-1 threaded fasteners due to the absence of susceptible materials and environmental conditions.

In the LRA, Section 3.6.4.1, the applicant identifies loss of material as an applicable aging effect. The applicant states that boric acid wastage is generally found in closures of reactor coolant systems and in structural steel connections, if closures leak. It is also considered an aging effect for non-boron treated threaded fasteners in the vicinity of the spent fuel pool.

The applicant states that loss of material from general corrosion is typically attributed to leaking joints. It is an aging effect for carbon steel and low-alloy steel threaded fasteners protected from, or exposed to weather and raw water, for galvanized steel threaded fasteners exposed to weather, and for stainless steel threaded fasteners exposed to raw water. The applicant states that loss of material is also an aging effect for stainless steel threaded fasteners in raw water.

In the LRA, Section 3.6.4.2 the applicant discusses cracking. The applicant states that cracking is an aging effect requiring management for stainless steel threaded fasteners in raw water, but not in borated water. For ASTM-A325 and ASTM-A490 bolts, cracking is not an applicable aging effect since the yield strength for these materials is below the threshold for these forms of attack.

Change of material properties is discussed in Section 3.6.4.3 of the LRA. The applicant states that threaded fasteners at ANO-1 are not subject to radiation embrittlement since they will not experience exposure above the threshold necessary to cause embrittlement. Change in material properties is also not an applicable aging effect for galvanized steel fasteners since they are not used on high-temperature piping systems.

In the LRA, Section 3.6.4.4, the applicant states that a review of industry literature, and ANO-1 operating experience did not identify any additional applicable aging effects. The applicant also states that the structural-related walkdowns performed in response to NRC Bulletins 79-02 and 79-14, and a more recent Maintenance Rule walkdown did not reveal any other forms of degradation. In addition, ANO-1 was designed with consideration for adequate preload and installation requirements. Therefore, self-loosening by vibration need not be considered an applicable aging effect. Threaded fasteners are identified and evaluated as a component commodity group in Tables 3.6-2 through 3.6-8 of the LRA.

The applicant concludes that the AMPs credited for managing aging of steel components and commodities (Section 3.6.1) can effectively manage the aging of threaded fasteners.

3.3.6.4.2 Staff Evaluation

3.3.6.4.2.1 Effects of Aging

In the LRA, Section 3.6.4, the applicant identifies the threaded fasteners that are within the scope of license renewal and subject to an AMR. The applicant states that the material and environments for these fasteners are the same as the materials and environments for steel components and commodities discussed in Section 3.6.1 of the LRA and, therefore, are effectively managed by the same AMPs. The NRC staff's review of the aging effects associated with steel components and commodities are discussed and evaluated in Section 3.3.6.2.2.1 of this SER. The NRC staff reviewed this information for threaded fasteners and found no omissions in the aging effects considered in the applicant's AMR of steel SCs and commodities.

The applicant identifies the following applicable aging effects for threaded fasteners:

- loss of material
- cracking
- change in material properties

The applicant states that loss of material is an applicable aging effect for threaded fasteners in the reactor coolant system, for structural supports and connections, and for bolted connects that are exposed to boron, raw water, and unsheltered environments (exterior ambient weather conditions). The staff reviewed this information and found no omissions in the applicant's assessment of loss of material for threaded fasteners.

Cracking was identified as an aging effect for stainless steel threaded fasteners in a raw water environment. The applicant concludes that other bolts at ANO-1 were not subject to cracking because their specified yield strengths are below the recognized threshold stress level ($S_y \geq$ 150 ksi) for these aging mechanisms. The NRC staff found the applicant's assessment of cracking for threaded fasteners to be acceptable.

The applicant assesses the possibility of a change in material properties as a result of radiation embrittlement, and intermetallic embrittlement of threaded fasteners. The applicant concludes that threaded fasteners are not exposed to a radiation threshold level that can result in this change in material properties. The applicant also rules out intermetallic embrittlement as an applicable aging effect for threaded fasteners because galvanized steel fasteners are not exposed to high temperatures (≥ 400°F) needed to cause this change in material. The NRC staff found the applicant's assessment relating to change in material properties for threaded fasteners to be acceptable.

The applicant assesses the potential for the loss of preload for bolted connections as a result of vibration. The applicant determines that the application of adequate preload, together with proper bolted connect design, provide sufficient justification for determining loss of preload as a non-applicable aging effect. The staff agreed that proper design and installation practices should minimize the likelihood of loss of preload of bolted connections as a result of vibration. However, the NRC staff was concerned with loosening of expansion and undercut anchors in concrete that can become loose as a result of local degradation of surrounding concrete, as a result of vibratory loads. Therefore, the NRC staff requested additional information by letter to the applicant dated June 1, 2000, and additional clarification in a letter to the applicant dated October 20, 2000. In the applicant's final response to the NRC dated November 2, 2000, the applicant states that loss of material and cracking of concrete around anchors are applicable aging effects that are managed by the Maintenance Rule program. The NRC staff had no additional concerns relating to the concrete anchors and vibratory loads.

3.3.6.4.2.2 Aging Management Programs

The LRA, Appendix B, contains the applicant's description of the AMPs used to manage the effects of aging for threaded fasteners so that the intended function(s) will be maintained consistent with the applicant's CLB for the period of extended operation. In the LRA, Section 3.6.4.4, the applicant states that the AMPs used to manage the aging of threaded fasteners are the same AMPs used to manage the effects of aging of steel components and commodities.

The programs that are credited with managing age-related degradation of threaded fasteners at ANO-1 include the following:

- Maintenance Rule program (Section 4.13)
- reactor building leak rate testing program (Section 4.16)
- ASME Section XI Inservice Inspection Programs IWE and IWF (Sections 4.3.5 and 4.3.4)
- inspection and preventative maintenance of the ANO-1 polar crane (Section 4.10)
- service water chemical control program (Section 4.6.5)
- battery quarterly surveillance (Section 4.21.2)
- boric acid corrosion prevention (Section 4.5)
- fire protection program (Sections 4.8.1 and 4.8.2)

The staff reviewed the information provided in the LRA for the AMPs used by the applicant to manage the aging of threaded fasteners, and determined that the AMPs identified above are acceptable to manage the aging specified in each program's purpose and scope statements. Refer to Section 3.3.1 or Section 3.3.6.1.2.2 of this SER for the review of these AMPs.

The NRC staff's assessment of the AMPs used by the applicant to manage the aging of threaded fasteners is provided in Sections 3.3.1 and 3.3.6.2..2 of this SER. However, the staff's review of Tables 3.6-1 through 3.6-8 of the LRA identified the following two omissions from the tables: the loss of material caused by boric acid wastage for threaded fasteners for structural connections in the vicinity of the spent fuel pool; and stress corrosion cracking and intergranular attack of stainless steel threaded fasteners in raw water. In response to an NRC staff request for additional clarification dated November 2, 2000, the applicant verifies that the boric acid corrosion prevention program is credited for managing the loss of material of threaded fasteners in the vicinity of the spent fuel pool. In the same response, the applicant verifies that the Maintenance Rule program, IWF Inspections, and the service water chemical control program are used to manage cracking of stainless steel bolts in raw water environments. The NRC staff had no additional concerns relating to the aging management of threaded fasteners.

3.3.6.4.3 Conclusions

On the basis of the review described above, the staff finds that there is reasonable assurance that the applicant has demonstrated that the effects of aging associated with threaded fasteners will be adequately managed so that the intended function will be maintained consistent with the CLB for the period of extended operation.

3.3.6.5 Fire Barriers

3.3.6.5.1 Technical Information in the Application

In the LRA, Section 3.6.5, “Fire Barriers,” the applicant states that fire barrier commodities including fire wraps and fire stops (penetration sealant), which are within the scope of license renewal and subject to an AMR. These SCs are potentially susceptible to the following applicable aging effects:

- loss of material
- cracking
- delaminating and separation
- change in material properties

The applicant also states that other aging effects do not apply to fire barriers due to the absence of susceptible material and environmental conditions.

In the LRA, Section 3.6.5.1, the applicant identifies the loss of material as an applicable aging effect for fire wraps and fire stops. In Section 3.6.5.2 of the LRA, the applicant identifies cracking that results in delaminating as an applicable aging effect for both fire wraps and fire stops, and cracking that results in separation as an applicable aging effect for fire stops. Cracking is also an applicable aging effect for the sealants used in floor and wall piping penetrations that are in contact with piping.

In LRA Section 3.6.5.3, the applicant also identifies the change in material properties as an applicable aging effect for fire barrier commodities within the reactor building and some rooms of the auxiliary building. On the basis of a review of industry experience, NRC generic communications, and ANO-1 operating experience, the applicant concludes that no additional aging effects are applicable to the fire barrier commodities. The NRC staff reviewed this information and found no omissions in the applicable aging effects identified by the applicant for fire barrier commodities.

3.3.6.5.2 Staff Evaluation

3.3.6.5.2.1 Effects of Aging

In the LRA, Section 3.6.5, the applicant identifies the fire wraps and fire stops (penetration sealant) that are within the scope of license renewal and subject to an AMR.

Fire wraps are either sprayed on, troweled on, or wrapped on members. Fire stops are the materials used to close gaps in penetrations. This commodity is comprised of a variety of materials exposed to an internal-ambient environments. The staff found the applicant’s assessment of the environments for fire barriers to be acceptable.

The applicant identifies the following applicable aging effects for fire barriers:

- loss of material
- cracking, delaminating , and separation
- change in material properties

On the basis of a review of industry experience, NRC generic communications, and ANO-1 operating experience, the applicant concludes that no additional aging effects are applicable to the fire barrier commodities. The NRC staff reviewed this information and found no omissions in the applicable aging effects identified by the applicant for fire barrier commodities.

The applicant concludes that, on the bases of industry information, NRC generic communications and ANO-1 operating experience, loss of material, cracking, delaminating, separation and change in material properties are the applicable aging effects requiring an AMP for fire barrier commodities. The staff performed a review of industry operating experience, and did not identify any additional aging effects requiring an AMR.

3.3.6.5.2.2   Aging Management Programs

In the LRA, Appendix B, the applicant describes the AMPs used to manage the effects of aging such that the intended function(s) of the SCs that are within the scope of license renewal and subject to an AMR (including the fire barrier commodities) will be maintained consistent with the applicant's CLB for the period of extended operation. In the LRA, Section 3.6.5.4, the applicant states that the fire protection program-fire barrier inspection is the AMP credited for managing the applicable aging effects for fire barrier commodities. This AMP is described in Appendix B, Section 4.8.1, of the LRA.

The scope of fire protection program-fire barrier inspection activities includes fire wraps and fire stops, as well as other fire barrier components that perform the function of separating redundant safe shutdown systems. The fire protection program is required to meet the provisions of
10 CFR 50.48, Appendix R to 10 CFR Part 50, and Operating Licence DPR-51.2.c.8. The following is the staff's evaluation of the fire barrier inspection program:

[Scope of Program] Fire walls and fire floors, as indicated on the ANO-1 fire protection drawings are required to meet 10 CFR 50.48. Fire doors and hatches, fire damper mountings, fire wraps, and penetration fire stops associated with the fire walls and fire floors are also within the scope of license renewal.

[Preventive Action] The applicant does not identify any preventive actions, and the staff does not see a need for such actions.

[Parameters Monitored/Inspected] See the discussion in the "Detection of Aging Effects" in the following paragraph.

[Detection of Aging Effects] Visual inspection is used to detect cracking of masonry walls, loss of material from fire doors and hatches, including threaded fasteners, loss of material, cracking, delaminating, change in material properties for fire wraps and associated banding, and loss of material, cracking, delaminating, separation, change in material properties for penetration fire stops. Fire doors and hardware and fire barriers with the exception of penetration seals are inspected at least once per 18 months. At a minimum, 10 percent of sealed penetrations are inspected every 18 months; the inspection scope is expanded if degradation is found until a 10 percent sample of the degraded type of seal is found with no visual degradation. Each penetration seal is inspected at least once every 15 years.

[Monitoring and Trending] No trending activities are identified and the staff does not see a need for such trending.

[Acceptance Criteria] Acceptance criteria are documented in the fire barrier inspection procedures. The visually observed condition must be the same as the as-designed condition.

[Operating Experience] The fire protection program has detected voids and gaps in fire barrier penetration seals; torn boot seals; loose damming material due to building vibration or temperature-related contraction; and cracks in block walls near penetrations. Damaged penetration seals and fire barriers have been repaired. Detection and correction of degradation by this program are evidence of its effectiveness, and provide reasonable assurance that aging of fire barriers will be adequately managed consistent with the CLB for the period of extended operation.

3.3.6.5.3 Conclusions

On the basis of the review described above, the staff finds that the applicant has demonstrated that the aging effects associated with fire barrier commodities will be adequately managed so that there is reasonable assurance that intended function(s) can be maintained consistent with the CLB for the period of extended operation.

3.3.6.6 Earthen Embankments

3.3.6.6.1 Technical Information in the Application

In the LRA, Section 3.6.6, “Earthen Embankments,” the applicant states that the emergency cooling pond (ECP), the intake canal, and the discharge canal are within the scope of license renewal and subject to an AMR. The applicant identifies the loss of form as the only applicable aging effect that requires aging management for the ANO-1 earthen embankments.

The applicant states that loss of form as a result of sedimentation in the ECP requires aging management, however, loss of form is not an applicable aging effect for the intake and discharge canals. The intake and discharge canals are designed to prevent loss of form from affecting maximum flow.

The applicant identifies the annual emergency cooling pond sounding as the AMP used to manage the aging effect for the ECP.

3.3.6.6.2 Staff Evaluation

3.3.6.6.2.1 Effects of Aging

In the LRA, Section 2.4.5, the applicant identifies the intake and discharge canals to Lake Dardanelle and the ECP as the earthen embankments that are within the scope of license renewal and subject to an AMR. The applicant states that earthen embankments provide a heat sink during DBA or station blackout conditions.

Upon completing its initial review, the NRC staff requested additional information regarding the intended function of Lake Dardanelle, which was not identified as having an intended function.

If the lake performs an intended function, then associated water control structures such as dams, should be included in the license renewal scope. In a letter to the NRC dated September 7, 2000, the applicant confirms that the ultimate heat sink complex consists of both the emergency cooling pond and Lake Dardanelle. However, only the portions of Lake Dardanelle controlled by the applicant, the intake and discharge canals, are considered within the scope of license renewal. The Dardanelle Dam is also needed to maintain the water inventory of the lake, but the dam is under the jurisdiction of the U.S. Army Corps of Engineers. The U.S. Army Corps of Engineers inspection and maintenance programs have been determined by the staff to be adequate for managing the effects of aging of dams.

The intake and discharge canals are designed to prevent loss of form from affecting maximum flow preventing the need to manage this aging effect. The applicant states that this design feature is controlled through power operations to ensure that sediment build-up does not affect safety systems. The NRC staff reviewed this information, and did not identify any concerns relating to the aging effects of earthen embankments.

3.3.6.6.2.2 Aging Management Programs

In the LRA, Section 3.6.6, the applicant identifies the annual emergency cooling pond sounding as the AMP used to manage the loss of forms of the ECP. The applicant describes this program in Appendix B, Section 4.21.1, of the LRA.

The applicant states that under this program the ECP and surrounding structural components are inspected annually for loss of form as a result of sedimentation. The following is the staff evaluation of this emergency cooling pond sounding program:

[Scope of Program] The annual emergency cooling pond sounding program verifies the availability of a sufficient supply of cooling water to handle DBAs, with a concurrent loss of the Dardanelle Reservoir. The scope includes the pond and surrounding structural elements. The surveillance requirements are contained in ANO-1 TS 4.13.

[Preventive Action] The applicant does not identify any preventive actions, and the staff does not see a need for any such actions.

[Parameters Monitored/Inspected]   Excessive sedimentation, erosion, degradation of rip rap, and silt buildup are the parameters monitored to ensure a loss of form does not result effect the intended function.

[Detection of Aging Effects] Visual inspections and soundings for pond level are conducted annually.

[Monitoring and Trending] No trending activities are identified, and the staff does not see a need for any trending.

[Acceptance Criteria] A sufficient water inventory must be maintained in the pond to meet applicable site procedures, based on TS

[Operating Experience] ANO-1 in-house documentation indicates that pond deficiencies have been identified during ECP inspection activities, including torn sandbags, out-of-place rip rap,

eroded banks, and a broken drain. Corrective measures are taken when deficiencies are identified. The applicant concluded that continued implementation of this program should provide reasonable assurance that the effects of aging will be adequately managed during the period of extended operation.

The NRC staff reviewed this information, and concludes that this program can provide reasonable assurance that aging of ECP will be adequately managed consistent with the CLB for the period of extended operation. Therefore, the staff found the emergency cooling pond sounding program to be acceptable.

3.3.6.6.3 Conclusions

On the basis of the review described above, the staff finds that the applicant has demonstrated that aging effects associated with the intake and discharge canals to Lake Dardanelle and the ECP will be adequately managed so that there is reasonable assurance that the intended function will be maintained consistent with the CLB for the period of extended operation.

3.3.6.7 Elastomers and Teflon

3.3.6.7.1 Technical Information in the Application

In the LRA, Section 3.6.7, “Elastomers and Teflon,” the applicant states that elastomers and Teflon that are within the scope of license renewal and subject to an AMR. Elastomers that are subject to an AMR can be made of rubber, neoprene, or material with similar properties. Teflon (polytetrafluoroethylene material) is used on sliding surfaces. In the LRA, Table 3.6-8, “Bulk Commodities,” the applicant identifies the locations and structures where elastomers (waterstops) and Teflon are used.

The applicant states that the environment that elastomers are exposed to is described as various air conditions, fluids, and radiation. Teflon is subjected to the interior environments of the applicable structures. The applicant identifies the following applicable aging effects for elastomers and Teflon that are within the scope of license renewal and subject to an AMR:

- cracking
- change in material properties

The applicant also states that other potential aging effects do not apply to ANO-1 elastomer and Teflon components and commodities due to the absence of susceptible material and environmental conditions.

In the LRA, Section 3.6.7.1, the applicant discusses the basis for concluding that cracking of elastomers, used as waterstops, is not an aging effect. Cracking as a result of ultraviolet radiation (sunlight) is not an applicable aging effect because the components are concealed in the exterior walls of ANO-1 structures and not exposed to sunlight. Cracking as a result of thermal exposure is not an applicable aging effect because the temperatures experienced by these components will be less than 95EF. Cracking as a result of ionizing radiation is not an aging effect since radiation levels will be less than 10^6 rads.

In the LRA, Section 3.6.7.2, the applicant states that for elastomers in exterior walls (waterstops), a change in material properties as a result of exposure to ultraviolet radiation, thermal exposure, and ionizing radiation is not an aging effect. For Teflon materials in the reactor building and auxiliary building, change in material properties is an applicable aging effect because the radiation dose of $10^{4\text{ rads}}$ may cause breaking of the chemical bonds.

In the LRA, Section 3.6.7.3, the applicant summarizes the results of the AMR for elastomers and Teflon. The only aging effect requiring aging management is the change in material properties for Teflon. The applicant states that no additional aging effects beyond those discussed above were identified for elastomers or Teflon in the applicant review of industry correspondence and ANO-1 operating experience.

The applicant states the following AMPs are used to manage the aging of Teflon:

- Maintenance Rule program
- ASME Section XI Inservice Inspection Program-IWF

3.3.6.7.2 Staff Evaluation

3.3.6.7.2.1 Effects of Aging

In the LRA, Section 3.6.7, “Elastomers and Teflon,” the applicant states that elastomers are used as waterstops and Teflon is used for sliding surfaces. The applicant does not clearly identify non-metallic joints or sealants as being within the scope of its review of elastomers and/or Teflon. Therefore, the staff requested additional information relating to the AMR of expansion joint sealants, structural sealants (other than elastomer waterstops), and caulking, or to provide a technical justification for excluding these components from an AMR.

In a letter to the NRC dated September 7, 2000, the applicant states that expansion joint sealants, structural sealants, and caulking are considered to be parts of the structural components or commodities to which they are connected, and are not evaluated separately as elastomers or Teflon components.

In the LRA, Section 3.6.7.1, the applicant identifies its basis for concluding that cracking is not an applicable aging effect for elastomers used as waterstops because of the absence of environmental conditions (no ultraviolet radiation (sunlight), no temperatures above 95EF, no ionizing radiation greater than 10^6 rads) that are needed for cracking to occur. In addition, the applicant provides additional justification in Section 3.6.7.2 of the LRA for not considering a change in material properties as an applicable aging effect for elastomers used as waterstops for similar reasons.

In the LRA, Section 3.6.7.2, the applicant states that change in the material properties of Teflon materials in the reactor building and auxiliary building is an applicable aging effect. A radiation dose of 10^4 rads can affect the chemical bond, requiring aging management of Teflon in locations where radiation doses can exceed 10^4 rads.

In the LRA, Section 3.6.7.3, the applicant summarizes the results of the AMR for elastomers and Teflon, and states that a change in material properties of Teflon is the applicable aging effect. The applicant also states that no additional aging effects were identified for elastomers or Teflon in its review of industry correspondence, and ANO-1 operating experience.

The applicant does not discuss seals used in personnel hatches and the equipment hatch in the reactor building in Section 3.6.7 of the LRA. In the LRA, Section 2.4.1.1, the applicant states that these seals are not long-lived components, and are not subject to an AMR because they are monitored in accordance with ASME Section XI, IWE inspections consistent with 10 CFR 50.55a, and are replaced on an as needed basis to ensure its intended function(s) is maintained. In addition, the seals are replaced at least once every 12 years in accordance with the site preventive maintenance program. The staff found this response acceptable.

As a result of its initial review, the NRC staff requested additional information regarding potential cracking and change in material properties caused by aggressive chemicals present in the groundwater. Industry experience has shown that aging of water stops can occur as a result of an aggressive chemical environment caused by aggressive chemicals in the groundwater. In its September 7, and November 2, 2000, responses to this concern, the applicant states that the plant ground water was not aggressive during plant construction, and remains non-aggressive as supported by the most recent ground water test data from May 6, 1996. The staff reviewed this information and found no additional concerns relating to cracking and/or changes in material properties of water stops.

The staff also requested additional information regarding any observable seepage or leaching through concrete walls below grade, which could be indicative of aging of waterstops, waterproofing membranes, caulking, and/or sealants. Seepage through these materials has been identified in other nuclear power plant structures. In a letter to the NRC dated September 7, 2000, and November 2, 2000, the applicant states that indications of in-leakage of water through cracks in the tendon access gallery concrete had been observed. However, the tendon access gallery performs no intended function and is outside the scope of license renewal. The applicant also states that except for cracks in concrete components in other below grade locations (none of which exceeded the acceptance criteria), no other indications have been observed to date. The NRC staff reviewed this information and found no additional concerns regarding potential aging of waterstops, waterproofing membranes, caulking, and/or sealants.

3.3.6.7.2.2 Aging Management Programs

In the LRA, Section 3.6.7.3, the applicant identifies the following two AMPs used to manage the change in material properties of Teflon:

- Maintenance Rule program
- ASME Section XI Inservice Inspection Program-IWF

In the LRA, Table 3.6-8 and Section 3.6.7.3, the applicant is not clear as to the scope of the IWF Inspection Program when used as an AMP for Teflon. This issue is addressed in Section 3.3.6.1 of this SER. The applicant confirms that the IWF AMP will be implemented consistently with the requirements of 10 CFR 50.55(a) throughout the period of extended operation

The description of the IWF Inspection program is presented in the LRA, Section 4.3.5 of Appendix B, and evaluated by the staff in Section 3.3.1.4 of this SER. The applicant states that the purpose of the IWF inspections is to identify and correct degradation of ASME Class 1, 2, and 3, and MC component supports in accordance with 10 CFR 50.55a and ANO-1 TS 4.0.5. The description presented for the IWF Inspection program includes the scope, aging effects, inspection method, industry code or standards, frequency, and acceptance criteria or standards. The NRC staff reviewed this information and found no additional concerns regarding IWF inspections used to manage the aging of Teflon.

The applicant also identifies the Maintenance Rule program as an AMP used to manage the change in material properties of Teflon. The applicant describes the Maintenance Rule program in the LRA, Section 4.13 of Appendix B. The NRC staff reviewed this information and found no additional concerns regarding the Maintenance Rule program used to manage aging of Teflon.

3.3.6.7.3 Conclusions

On the basis of the review described above, the staff concludes that the applicant has demonstrated that the aging effects associated with elastomers and Teflon will be adequately managed so that there is reasonable assurance that intended function(s) will be maintained consistent with the CLB for the period of extended operation.

3.3.6.8 References for Section 3.3.6

1. 10 CFR Part 54, “Requirements for Renewal of Operating Licenses for Nuclear Power Plants.”
2. DG-1047, “Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants,” Working Draft, April 21, 2000.
3. “Arkansas Nuclear One - Unit 1, License Renewal Application,” January 31, 2000.
4. NUREG-0612, “Control of Heavy Loads at Nuclear Power Plant.”
5. NRC GL 89-13, “Alternate Waste Management Procedures in Case of Denial of Access to Low-Level Waste Disposal Sites.”
6. ANSI/IEEE Standard 450-1980, “IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Storage Batteries for Generating Stations and Substations.”
7. NUREG-1611, “Aging Management of Nuclear Power Plant Containments for License Renewal,” September 1997
8. ACI 318-63, “Building Code Requirements for Reinforced Concrete,” American Concrete Institute.
9. ACI 301, “Specifications for Structural Concrete for Buildings,” American Concrete Institute.
10. NUREG-1557, "Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal," October 1996
11. NUREG-1522, "Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures," June 1995.
12. ASTM A325, “Standard Specification for Structural Bolts, Steel, Heat-Treated, 120-ksi and 105-ksi Minimum Tensile Strength,” American Society for Testing and Materials.
13. ASTM A490, “Standard Specification for Heat-Treated Steel Structural Bolts, 150-ksi Minimum Tensile Strength,” American Society for Testing and Materials.

14. ASTM A307, "Standard Specification for Carbon Steel Bolts and Steels, 60,000-psi Tensile Strength," American Society for Testing and Materials.
15. NRC Bulletin 79-02, Revision 0, "Pipe Support Base Plate Designs Using Expansion Anchor Bolts," March 8, 1979.

3.3.7 Electrical Instrumentation and Controls AMR

In the LRA, Section 2.5, “Electrical and Instrumentation and Controls System Scoping and Screening Results,” the applicant describes its process and results for the scoping and screening of electrical and instrumentation and controls (EIC) SSCs that are within the scope of license renewal and subject to an AMR. The AMR for these components is provided in Section 3.7, “Electrical and Instrumentation and Controls,” of the ANO-1 LRA. In Table 3.7.1 of the LRA, “Passive Electrical Components,” the applicant identifies the passive electrical components, environments, aging effects, and programs and activities used to manage these effects. Appendix B, Section 3.2, “Electrical Component Inspection,” of the LRA contains a description of the programs and activities used to manage the aging of electrical components. The applicant states that the electrical component inspection program will be used to manage the aging of splices, terminal blocks, connectors, and cables located in environments that may be conducive to aging.

3.3.7.1 Technical Information in the Application

The applicant follows the “plant spaces” approach and methodology as described in the Sandia Report entitled “Aging Management Guideline for Commercial Nuclear Power Plants–Electrical Cable and Terminations” (SAND96-0344). In summary, Sandia evaluated aging mechanisms and consolidated historical maintenance and industry operating information into one source. The report contains the following primary conclusions:

- Cables and terminations that are included in the scope of the AMR are highly reliable and can be expected to perform their intended safety function during the initial license term and for license renewal.

- On the basis of the analysis of failure data, the number of cable and termination failures that have occurred throughout the industry is extremely low in proportion to the general population.

- The environmental conditions and aging mechanisms affecting cables and terminations are generally well understood and characterized.

- Cable aging can be evaluated on an ongoing basis using theoretical techniques, measurement of physical properties, and periodic inspection.

The Sandia report also concluded that detailed component review was only required for those passive electrical components that are located near heat or radiation sources, are subject to continuous or near continuous loading at a significant percentage of the cable ampacity limits, are exposed to wetting (medium voltage only) or adverse chemical environments, or are subject to repeated or damaging mechanical stress. Low-voltage instrument circuits that are sensitive to small variations in impedance were determined to be potentially affected by oxidation of connector or termination contacts. The report also recommended that special consideration be given to the possibility of damage during installation.

Heat stress can cause accelerated aging for some passive electrical components and is primarily a concern for cable insulation. For ANO-1, temperature thresholds of 105EF outside the reactor building and 120EF inside the reactor building have been chosen for the purpose of license renewal evaluation. These temperature thresholds are consistent with the ANO-1 EQ Program ambient temperatures assumed in EQ analyses. The applicant states that as long as the temperature remains below these thresholds, further evaluation of heat stress and thermal aging of passive electrical components is not required.

The applicant states that cables that are subject to an AMR, but not within the scope of the ANO-1 EQ Program are similar to cables that are covered by the EQ Program, and have been tested and qualified to at least 2.0×10^8 rads. The applicant also states that for the license renewal evaluation, a total integrated radiation dose of 1.0×10^8 rads will be the established threshold for requiring aging management of cables and cable terminations that are not within the scope of the ANO-1 EQ Program. In addition, cables that are continuously operated at a significant percentage of their current carrying capacity operate closer to their temperature limits as a result of ohmic heating. These cables have been identified, and are evaluated in Section 3.7.6, “Cables,” of the LRA.

The applicant also states that the majority of cables at ANO-1 are located in dry locations and are not exposed to an adverse chemical environment. However, the cables that are potentially exposed to wet conditions or chemical environments have been identified, and are evaluated in Section 3.7.6 of the LRA.

The applicant states that passive electrical components exposed to repeated mechanical stress are those cables and connectors that must be periodically moved or disconnected for plant outages or surveillance testing and that these components have been identified and are evaluated in Sections 3.7.4, “Connectors,” 3.7.5, “Terminal Blocks,” and 3.7.6, “Cables.”   In addition, low-voltage instrument cables and connectors that are sensitive to small variations in impedance have been identified, and evaluated in Sections 3.7.4 through 3.7.6 of the ANO-1 LRA.

The applicant performs an AMR for passive electrical components that are located in the portions of the reactor building where the ambient temperature is above the 120EF temperature threshold for aging and/or the integrated radiation dose is above the 1.0×10^8 rad dose threshold for aging. The applicant also includes in its review the passive electrical components that are located in the few areas of the auxiliary building and the turbine building that have experienced temperatures above the 105EF temperature threshold.

In addition, the applicant preforms an AMR for cables that are exposed to a wetted medium or potentially hazardous chemicals, power cables that are continuously loaded above 50 percent of their ampacity rating, cables and connectors that are subject to frequent manipulation (i.e., being disconnected/reconnected more than once per refueling cycle), and low-voltage instrument cables and connectors that operate at low currents or are otherwise sensitive to small variations in impedance.

The main structures that house passive electrical equipment that are within the scope of license renewal are the reactor building, the auxiliary building, the turbine building, and the intake structure. There are also a limited number of passive electrical components in specialized structures such as the fuel oil storage vault and duct banks and manholes. Passive electrical components in the portion of the reactor building, where the ambient temperature is above the 120°F license renewal threshold value, or the high radiation threshold value is exceeded, are evaluated. The auxiliary building and intake structure controlled environments are mostly benign spaces for the passive electrical equipment in the scope of license renewal. For the few areas in the auxiliary building that have equipment temperature above the 105°F threshold value, the passive electrical components are evaluated. Passive electrical components in the turbine building that are above the 105°F threshold have been evaluated. Therefore, using the plant spaces approach, the Sandia selection of significant stressors, and NEI 95-10 passive component categorization, the following subsets of passive electrical components that are within the scope of license renewal and subject to an AMR require component specific evaluation:

- passive electrical equipment in elevated temperature locations
- passive electrical equipment in areas of the reactor building that are above the threshold radiation level
- wetted medium voltage power cables and cables exposed to potentially hazardous chemicals
- power cables loaded above a significant fraction of their ampacity rating (significant fraction is defined as 50 percent of the ampacity rating)
- cables and connectors subject to frequent manipulation (frequent is defined as being disconnected/reconnected more than once per refueling cycle)
- low voltage instrument cables and connectors that operate at low currents or are otherwise sensitive to small variation in impedance

Connectors

The applicant states that the potential aging mechanisms that were considered for ANO-1 connectors include metal corrosion, electrical stresses, water or humidity effects, mechanical stresses (including wear), and thermal or radiation aging of the organic components. The applicant also states the following:

- Plant connectors are exposed primarily to dry conditions and, therefore, significant corrosion is not expected during normal plant operation.
- The connectors do not provide any substantial mechanical support and, therefore, mechanical stresses are not an aging mechanism of concern.
- The electrical stresses for most connectors are not an aging mechanism concern because in general they are designed to carry larger current than they actually carry.
- The organic portions of connectors (such as insulating materials, O-rings, filler materials, cases, etc.) are susceptible to aging from both thermal and radiation exposure.
- Frequent manipulation can cause wear on the surfaces in contact and loosen the sealing material or the cases.

The applicant also states that it has identified a small number of splices as being subject to the aging effects that are discussed in the Sandia Report for splices that are exposed to moisture and elevated temperature. In order to manage the effects of aging on these splices and demonstrate acceptable performance during the extended license term, an electrical component inspection program will be established to inspect and monitor the condition of the identified splices. They also identified connectors that are subject to frequent manipulation. The connector types identified consist of terminal blocks, multi-pin connector, screw terminals, and battery terminal posts. Terminal blocks are addressed in the terminal block section. For the other connector types, the applicant will rely on good maintenance practices to ensure that frequent manipulation does not unacceptably degrade connectors during the extended license term. Inspections of connectors are completed during the reconnection of connectors following any maintenance that required the connector to be disconnected. The effects of frequent manipulation (wear, loose fittings, cracking, etc.) are easily detected by visual inspection. Electrical checks of many vital functions are performed after reconnection of connectors to verify continuity. Therefore, no additional measures are necessary to manage aging for these connectors.

Coaxial and triaxial connectors that are terminating impedance-sensitive circuits have also been identified. Aging caused by oxidation or corrosion of the connector pins could interfere with the operation of these circuits. In order to ensure this does not happen, the applicant indicates that the Electrical Component Inspection Program will be established to periodically inspect these connectors. This program will ensure the proper operation of these circuits during the period of operation. This program is described in Section 3.2 of the Appendix B of LRA.

Terminal Blocks

The applicant states that the potential aging mechanisms for ANO-1 terminal blocks include corrosion of exposed metal surfaces, electrical stresses, mechanical stresses, and thermal or radiation aging. The applicant also states that the following aging can occur during routine operations:

- Corrosion of exposed metal surfaces is possible especially in high humidity environments, however, since none of the terminal blocks subject to an AMR are located in a high humidity environment, corrosion is not an aging mechanism of concern.

- Deterioration from electrical stress is not expected in terminal blocks because current carrying capability is typically much larger that the actual current.

- Mechanical stresses are not aging mechanism of concern because terminal blocks are not utilized as structural members.

- Thermal and radiation aging of organic portions of the terminal blocks such as insulating and filler materials can be susceptible to aging for elevated temperature or radiation. Phenolic material is very resistant to aging from both thermal and radiation exposure. For example, terminal block manufacturer, Buchanan, lists 302°F as a maximum continuous service temperature. The EPRI Data Bank qualification for radiation for the General Electric or the Buchanan terminal blocks is over 10^8 rads. The applicant concludes that thermal and radiation aging are not significant aging effects for the phenolic terminal block at ANO-1.

The applicant also states that it has identified numerous terminal blocks subject to various aging effects. One terminal block was found in an area where it could be exposed to elevated temperatures. The aging effect for the remaining terminal blocks includes damage due to frequent manipulation of the wires connected to the blocks.

The applicant states that terminal blocks are highly resistant to aging from elevated temperature. Testing of terminal blocks of the type used at ANO-1 demonstrates a qualified life significantly longer than the period of extended operation. The terminal block identified above is only exposed to elevated temperature, and not exposed to moisture or hazardous chemicals. Therefore, no additional actions are necessary to demonstrate the integrity of this terminal block through the period of license renewal.

The terminal blocks that are subject to frequent manipulation have been identified. Periodic maintenance and surveillance procedures call for the lifting of leads from terminal blocks for testing purposes. These same procedures require that the leads are reconnected and an independent verification performed. Good maintenance practices are relied upon at ANO-1 to ensure that frequent manipulation of the connections does not degrade any terminal block during the license renewal term. Therefore, the applicant concludes that no additional measures are necessary to manage aging of terminal blocks.

Cables

The applicant states that aging effects considered for cables at ANO-1 include corrosion of conductors, electrical stresses, water and humidity effects, thermal and radiation aging, and mechanical stresses. Corrosion of conductors is not expected since the conductors are covered by insulation, and deterioration of low-voltage cables from electrical stresses is not expected because voltage is very low with respect to material capabilities.

The applicant also stated the following:

- Ohmic heating can be very significant for those cables that are routinely or continuously operated with high current relative to their ampacity rating. However, since the majority of emergency equipment is only placed in operation following an accident, the “Q” switchgear cables, load center transformers, load centers and motor control centers are normally loaded to only a small percentage of their rating. In addition, the “Q” supply for the essential 4.16kV switchgear is from the EDGs, which is normally not in service.

- Moisture aging is not a concern for the majority of cables at ANO-1 because they are located in dry environments. However, for buried medium- or high-voltage cables, exposure to a wetted environment can present a significant aging effect resulting in a reduced insulation resistance to ground, and potential electrical failure due to moisture intrusion, water treeing, and contamination.

- Chemical attack of organic materials used in cables may occur due to the exposure to hydraulic fluids, fuel oils, lubricating oils, or other chemicals resulting in softening, flowing, cracking, or discoloration of the jacket, as well as reduced insulation resistance to ground and a potential for electrical failure.

- Thermal aging of organic materials in cables can cause embrittlement, cracking, and discoloration of the insulation. Such effects can result in reduced insulation resistance to ground and a potential electrical failure.

- Radiation stress is only significant for cables that are exposed to greater than the threshold value of $1x10^8$ rads, and cables that are subject to an AMR will not reach the radiation threshold.

- Mechanical stresses in cable systems will not change significantly during the license renewal period, and the installation practices at ANO-1 and operating experience indicate that insulation cut is not a concern. Therefore, mechanical stress is not a concern at ANO-1 except for those cables that are frequently manipulated. Cables that are frequently manipulated are covered under good maintenance practices.
- Low-voltage instrument and control circuits primarily degrade because of environmental influences that result in oxidation, increases in circuit resistance and, the reduction in signal strength. The applicant also states that it has identified cables that are within the scope of license renewal and are exposed to the various aging effects, subject to frequent manipulation, and used in impedance-sensitive circuits.

The applicant identifies cables that are in scope of license renewal, and are exposed to various aging effects. To ensure that the aging effects due to these various conditions do not degrade the ability of these cables to functions, the Electrical Component Inspection Program will be established in order to monitor the condition of the cables for the period of extended operation. The program will periodically inspect the cables, and document the results for trending purposes.
The applicant also identifies cables that are within the scope of license renewal subject to frequent manipulation. The connectors and terminal blocks to which these cables are terminated are discussed in other sections. The discussion on managing aging effects for connectors are also applied to the cables that are frequently manipulated. Inspections of these cables are completed during the reconnection process. The effects of frequent manipulation are easily detected by visual inspections. Therefore, the applicant concludes that no additional measures are necessary to manage aging of these cables.

For cables used in impedance-sensitive circuits, corrosion of the connectors is the applicable aging effect that needs to be considered. The connectors in this category were discussed in a previous section. The cables do not have an aging effect that can impact the operation of the circuit in terms of impedance change. Therefore, the applicant concludes that there is no need to consider these cables further.

In the LRA, Section 3.2 of Appendix B, the applicant identifies an AMP to manage aging effects of splices, connectors, and cables. In this section, the applicant states the following:

- The purpose of the Electrical Component Inspection Program is to inspect splices, connectors, and cables located in areas that may be conducive to accelerated aging.

- The scope of the inspection includes cables that are exposed to elevated temperatures, wet environments, or corrosive chemicals. The scope also includes cables that can experience elevated temperatures associated with the current they carry, connectors used in impedance-sensitive circuits, and cable splices subject to aging-related stressors.

- The aging effect of concern for cables and cable splices is a change in the material properties of insulation, as evidenced by cracking or discoloration of the insulation. The aging effect for impedance-sensitive circuits is the corrosion of connector pins.

- Visual inspection will be used to detect aging effects in cables, splices, and connectors in the scope of this program.

- Samples may be used for this program. If used, an appropriate sample size will be determined before the inspection or test begins.

- No industry codes or standards are applicable.

- Cables, splices, and connectors in the selected sample(s) will be inspected at least once every 10 years.

- No unacceptable visual indications of age related degradation of the cables, splices, and connectors in the scope of the AMR is the acceptance criteria or standard.

- The Electrical Component Inspection Program will be formally implemented, and the first inspection of in-scope cables, splices, and connectors will be completed before the expiration of the initial 40-year licensing term.

- The Electrical Component Inspection Program will be effective in the future for managing aging effects at ANO-1 because it incorporates proven monitoring techniques, corrective actions and administrative controls from existing programs, and procedures. The implementation of this inspection program provides reasonable assurance that the effects of aging will be adequately managed so that components that are within the scope of this program will perform their intended functions consistent with the CLB for the period of extended operation.

The applicant also reviewed industry and plant operating experience to ensure that no unique aging effects exist beyond those discussed in Section 3.7 for cables, connectors, splices, and terminal blocks.

In its response to the NRC dated July 31, 2000, the applicant also states the following:

- According to the DOE/Sandia aging management guideline, dirt, dust, and other types of contamination do not directly produce stress, but may intensify the effects of other stressors acting on cables and related components. The DOE/Sandia aging management guideline also notes that the combination of moisture and contaminants is considered to be the only stressor of any significance involving contaminants. The LRA discusses the cable aging effects attributable to moisture and humidity. The effects of contaminants are mentioned in the ANO-1 LRA, Section 3.7.6.

- The DOE/Sandia aging management guideline does not consider vibration to be a significant and observable aging mechanism. The effects of vibration are considered event-driven and are not age-related. Specifically, vibration induced effects result from poor design or poor installation practices. At ANO-1, the corrective action process will detect and correct any problems associated with vibration. Operating experience does not show any adverse trends associated with this condition. Therefore, vibration is not considered in the LRA.

- During the AMR of cables and connections at ANO-1, no unique aging effects for inaccessible cables or connections were identified in comparison to accessible cables or connectors. The areas in the cable inspection program may contain both accessible and inaccessible cables and connectors. Thus, the cables and connections in the accessible areas can be thought of as a sample, representative of all cables and connections. When an unacceptable condition or situation is identified for an accessible cable or connection, a determination will be made as to whether the same condition or situation is applicable to other accessible or inaccessible cables or connections and what additional actions need to be taken.

- EPRI TR-109619, “Guideline for the Management of Adverse Localized Equipment Environments,” will be used as guidance in implementing the inspection program. Cables and connections will be judged to be acceptable if there are no visual indications of cable and connection jacket surface anomalies (e.g., embrittlement, cracking, or discoloration) that suggest that conductor insulation degradation exists. Terminal blocks and pin-type connectors will also be inspected for corrosion. The personnel performing the inspections will possess the qualifications and training necessary to adequately detect degradation. When the acceptance criteria are not met, further investigation will be performed on electrical cables and connections in question. Corrective actions may include, but are not limited to, testing, shielding, or otherwise changing the environment, and relocating or replacing the affected cables or connections. Specific corrective actions will be implemented consistent with the ANO-1 corrective action process in accordance with 10 CFR Part 50, Appendix B.

- Underground cables that may be exposed to a wetted environment at ANO-1 (e.g., service water pump motor cables) will be included in the scope of the electrical component inspection program. As stated in the “Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station, Units 1, 2, and 3" (NUREG-1723), the NRC staff is evaluating the root cause of the Davis-Besse cable failure event, as well as the results of related tests and experience, to determine whether any further actions are necessary or whether to pursue this matter as a generic safety issue. On the basis of the results of this evaluation, the applicant will take additional action as necessary.

3.3.7.2 Staff Evaluation

The NRC staff evaluated the AMR information presented in the LRA, Appendix A, Section 3.7, and Appendix B, Section 3.2, and in the applicant’s response to staff RAIs, dated July 31, 2000, to determine if there is reasonable assurance that the effects of aging applicable to electrical equipment will be adequately managed, consistent with its CLB for the period of extended operation, in accordance with 10 CFR 54.21(a)(3).

Equipment that is within the scope of license renewal, that contains materials such as polymers, coatings, oil and grease lubricants, and ferrous and non-ferrous metals, is known to be susceptible to degradation under plant service conditions. Tables 3.11-1 through 3.11-3 of the SRP for license renewal identify the environment, aging mechanisms, aging effects, and appropriate monitoring methods for each of the aging effects identified. The applicable environmental conditions include temperature, radiation, humidity, water, electrical, mechanical, vibration, chemical, electrochemical, and contaminants.

Electrical Component Inspection Program

The staff evaluation of the applicant's AMPs focused on the program elements rather than details of specific plant procedures. The staff's approach to evaluating each program and activity used to manage the applicable aging effects is described in Section 3.3.1 of this SER.

[Program Scope] The scope of inspection includes cables exposed to elevated temperatures, wet environments, or corrosive chemicals. The scope also includes cables that can experience elevated temperature due to the current they are carrying and connectors used in impedance-sensitive circuits and cable splices subject to aging-related stressors. The staff found the scope of the program acceptable because it includes cables and connections that are subject to potentially adverse localized environments that can result in applicable aging effects on these insulated cables and connections. However, for buried (inaccessible) medium-voltage cables that are within the scope of license renewal and subject to an AMR, exposure to a wetted environment can present a significant aging effect resulting in a reduced insulation resistance to ground, and potential electrical failure due to moisture intrusion, water treeing, and contamination. For cables exposed to these conditions, the staff does not agree that visual inspection is an adequate AMP.

[Preventive/Mitigative Actions] There are no preventive or mitigative actions taken as part of this program, and the staff did not identify the need for such actions

[Parameter Inspected/Monitored] Visual inspection will be used to detect aging effects in cables, splices, and connectors in the scope of this program. Samples may be used for this program. If used, an appropriate sample size will be determined prior to the inspection or test. Cables, splices, and connectors in the selected sample will be inspected at least every 10 years.

In letter dated July, 31, 2000, the applicant states that based on the AMR of cables and connections at ANO-1, no aging effects for inaccessible cables or connections were found in comparison to accessible cables or connectors. The areas in the cable inspection program may contain both accessible and inaccessible cables and connectors. Thus, the cables and connections in the accessible areas can be thought of as a sample, representative of all cables and connections. When an unacceptable condition or situation is identified for an accessible cable or connection, a determination will be made as to whether the same condition or situation is applicable to other accessible or inaccessible cables or connections and what additional actions need to be taken. Cables may be inaccessible due to the presence of fire wrapping material or due to the conduit or cable tray blocking access to the cable. For these situations, and evaluation will be performed to ensure that the accessible cables in the same area are truly representative of the inaccessible cables. The staff found this approach acceptable because it provides the means for monitoring the applicable aging effects for inaccessible in-scope cables and connections, with one exception. Buried (inaccessible) medium-voltage cables exposed to ground water typically do not have comparable accessible cables exposed to a similar environment that can serve as a sample for the inaccessible cables. For buried cables that are subject to an AMR, and that are exposed to ground water, visual inspection is not sufficient for managing the significant aging effect resulting in a reduced insulation resistance to ground, and potential electrical failure due to moisture intrusion, water treeing, and contamination. This item is discussed further in Section 3.3.7.2.4 of this SER.

[Detection of Aging Effects] As discussed above and with the exception of buried cable exposed to ground water, the staff found the inspection scope and inspection technique for both accessible and inaccessible in-scope cables and connections acceptable on the basis that the AMP is focused on detecting change in material properties of the conductor insulation, which is the applicable aging effect when cables are exposed to an adverse, localized environment.

[Monitoring and Trending] AMP monitoring and trending actions as discussed above are focused on the applicable aging effect of change in material properties of conductor insulation. The staff found that these actions are necessary because they provide predictability of the extent of conductor insulation degradation for timely corrective action actions.

[Acceptance Criteria] EPRI TR-109619, Guideline for the Management of Adverse Localized Equipment Environments, will be used as a guidance in implementing the inspection program. Cables and connections will be judged to be acceptable if there are no visual indications of cable and connection jacket surface anomalies (e.g., embrittlement, cracking, or discoloration) that suggest that conductor insulation degradation exists. Terminal blocks and pin type connectors will also be inspected for corrosion. The personnel performing the inspection will possess the qualifications and training necessary to adequately detect degradation. With the exception of buried cable exposed to ground water, the staff found these acceptance criteria acceptable because it should ensure that the cable and connection intended function are maintained under all CLB design condition during the period of extended operation.

[Operating Experience] The ANO-1 Electrical Component Inspection Program is a new program; thus, the applicant did not provide specific operating experience.

3.3.7.2.1 Review of the ANO-1 AMR Methodology and Plant Spaces Approach

The NRC staff has evaluated the information presented in Sections 3.7.1, 3.7.2 and 3.7.3 of the ANO-1 LRA to determine if there is reasonable assurance that the applicant has identified the applicable aging effects and the bounding conditions for electrical components. The process to determine the applicable aging effect on these components is based on industry literature defining the operating environments and the operating stresses for each of the components subject to an AMR. The NRC staff reviewed each of the environments and resulting mechanisms and effects as they apply to the electrical component commodities discussed below.

3.3.7.2.2 Connectors

3.3.7.2.2.1 Effects of Aging

The potential aging mechanisms considered for connectors include metal corrosion, electrical stresses, water or humidity effects, mechanical stresses (including wear), and thermal or radiation aging of the organic components. During plant operating conditions, connectors are typically exposed to dry conditions, and significant corrosion is not expected. The connectors do not provide any substantial mechanical support and, therefore, mechanical stresses are not significant. Electrical stresses are not a concern because, by design, connectors are sized to current carrying capacity for each application. The organic portions of connectors (such as insulating material, O-rings, filler materials, cases, etc.) are susceptible to aging from exposure to chemicals or contaminants, elevated temperature, radiation exposure, and wear (wear on the surface of contact points, loosening of sealing material, and degradation of the casing itself) from frequent manipulation.

3.3.7.2.2.2 Aging Management Programs

The NRC staff has evaluated the information on connectors presented in Section 3.7.4, Table 3.7-1, and Section 3.2 of Appendix B of the ANO-1 LRA to determine if there is reasonable assurance that the applicant has demonstrated that the aging effects for connectors will be adequately managed, consistent with the applicant's CLB for the period of extended operation.

In its AMR, the applicant considers the following environmental conditions: temperature, radiation, humidity, water, chemical, wear caused by excess manipulation, and contaminants. Programs and activities used by the applicant to manage the aging of connectors include the Electrical Inspection Programs, good maintenance practices, and electrical checks to verify continuity

For the small number of splice connectors that are subject to aging caused by moisture and elevated temperatures, the Electrical Component Inspection Program, as evaluated above, will be used to manage the effects of aging.

The applicant also identifies connectors that are subject to aging attributed to frequent manipulation (terminal blocks, multi-pin connectors, screw terminals and battery terminal posts), and that rely on good maintenance practices to ensure that frequent manipulation does not unacceptably degrade the connectors during the extended license period. Wear, loose fittings, cracking, etc., should be detected by visual inspections. In addition, electrical checks of many vital functions are performed after reconnecting to verify continuity. Terminal blocks were addressed separately in Section 3.7.5 of the ANO-1 LRA and are evaluated in Section 3.3.7.3.3 of the SER.

For connectors that are used to terminate impedance sensitive circuits, (e.g., coaxial and triaxial), the applicant identifies loss of material caused by oxidation or corrosion of connector pins as aging concerns. Periodic inspections of these connectors will be used to ensure proper operation during the period of extended operations.

On the basis of this review, the NRC staff found that, good maintenance practices the electrical component inspection program, and tests including continuity checks provide reasonable assurance that the aging effects for connectors will be adequately managed so that the intended function will be maintained consistent with the CLB for the period of extended operation.

3.3.7.2.3 Terminal Blocks

3.3.7.2.3.1 Effects of Aging

The potential aging mechanisms for terminal blocks at ANO-1 include corrosion of exposed metal surfaces, electrical and mechanical stresses, and thermal or radiation aging of the organic components. In addition, aging mechanisms, aging effects, and monitoring methods used to detect aging effects are required to be evaluated for electrical SSCs in the scope of license renewal. Programs and activities used by the applicant to manage the aging of terminal blocks are good maintenance practices.

At ANO-1, no terminal blocks that are subject to an AMR are located in high humidity environments and corrosion of terminal blocks is not a concern. Terminal blocks are not used as mechanical supports, therefore, mechanical stressors are also not a concern. Electrical stresses are not a concern because, by design, they are sized to carry current that is typically much greater that the actual current for each application. The organic portions of the terminal blocks (insulating material, filler materials, etc.) are not typically susceptible to aging from elevated temperature or radiation levels except in one application, however, frequent manipulation of the wires connecting to the terminal blocks can cause damage to the terminal blocks.

3.3.7.2.3.2 Aging Management Programs

The NRC staff has evaluated the information relating to terminal blocks presented in Section 3.7.5, and Table 3.7-1of the ANO-1 LRA to determine if there is reasonable assurance that the applicant has adequately addressed the environmental conditions, related aging effects, monitoring methods, and acceptance criteria used to detect aging effects for terminal blocks in the scope of license renewal. From the terminal blocks that are subject to an AMR, the applicant identifies one terminal block that is located in a high-temperature area. Terminal blocks are highly resistant to aging from elevated temperature. The testing of terminal blocks used at ANO-1 demonstrates a qualified life significantly longer than the period of extended operation. The terminal block exposed to an elevated temperature is not exposed to moisture or hazardous chemicals. Therefore, no additional actions are necessary to demonstrate the integrity of this terminal block through the period of license renewal.

The remaining terminal blocks were found to be susceptible to damage as a result of frequent manipulation of wires connected to each block. For these terminal blocks, periodic maintenance and surveillance procedures call for the lifting of leads from the terminal blocks for testing purposes. These same procedures require that the leads are reconnected, and that an independent verification is performed. Therefore, the applicant uses good maintenance practices to ensure that frequent manipulation of the connections will not degrade any terminal blocks during the license renewal term.

On the basis of this review, the NRC staff finds that the good maintenance practices provide reasonable assurance that the aging effects for terminal blocks will be adequately managed so that the intended function will be maintained consistent with the applicant’s CLB for the period of extended operation.

3.3.7.2.4 Cables

3.3.7.2.4.1 Effects of Aging

Environmental conditions that should be evaluated with regard to the cables at ANO-1 include temperature, radiation, humidity, water, chemical, mechanical, and electrical stressors, vibration and exposure to chemicals and contaminants. In addition, aging mechanisms, aging effects, and monitoring methods used to detect these aging effects of cables are required to be evaluated. To manage the effects of aging on cables, the applicant relies on the Electrical Inspection Program as evaluated above.

3.3.7.2.4.2 Aging Management Programs

The NRC staff has evaluated the information Section 3.7.6, Table 3.7-1, and Appendix B, Section 3.2, of the LRA, as well as the applicant's responses to RAIs provided in a letter to the NRC dated July 31, 2000, to determine if there is reasonable assurance that the applicant has demonstrated that the aging effects for cables will be adequately managed consistent with the CLB for the period of extended operation.

In summary, cables that are within the scope of license renewal, and that are exposed to environments that can lead to aging will be monitored for the extended period of operation through the electrical component inspection program, Section 3.2 of Appendix B. In addition, cables subject to frequent manipulation have been identified, and are managed along with connectors and terminal blocks through inspections during reconnecting. Finally, cables in impedance sensitive circuits have also been identified. Cables do not have an aging effect that can impact the operation of the circuit in terms of impendent change.

With respect to buried (inaccessible) medium-voltage cables exposed to ground water, the staff raised the concern that these cables typically do not have comparable accessible cables exposed to a similar environment that can serve as a sample for the inaccessible cables. For buried cable that are subject to an AMR, and that are exposed to ground water, visual inspection is not sufficient for managing the a significant aging effect resulting in a reduced insulation resistance to ground, and potential electrical failure due to moisture intrusion, water treeing, and contamination. This was Open Item 3.3.7.2-1.

In response to this concern, the applicant committed to implement either a testing or replacement program for the cables of concern. If a testing program is implemented, inaccessible medium-voltage cables exposed to moisture and voltage will be tested for the presence of aging. The specific type of test that will be performed will be identified and implemented before year entering the period of extended operation. This test will provide an indication of insulation integrity. Along with this test, the applicant will monitor and manage groundwater in manholes containing in-scope medium-voltage cables to reduce the exposure of these cables to moisture.

The applicant is also considering a periodic replacement program based on industry and site-specific operational experience, as an alternate approach to testing and monitoring. If the applicant determines periodic replacement to be a more effective means of managing aging of these cables, the program will be implemented prior to entering the period of extended operation. Preventing failures by replacing components before failure is expected to occur has been determined by the staff to be an acceptable means of managing aging of cables consistent with the discussion in Section III.f(i)(b) of the statement of considerations for the License Renewal Rule, 60 FR 22478. The staff found this resolution to Open Item 3.3.7.2-1 acceptable.

In the LRA, Appendix A, Safety Analysis Report Supplement," the applicant did not provide a summary description for the medium voltage buried cable AMP. Therefore, the staff requested that the applicant provide a summary description that adequately describes the medium voltage buried cable AMP, as it applies to license renewal in accordance with 10 CFR 54.21(d). This was FSAR Item 3.3.7.4 of Open Item 3.3-1.

In its revised summary description of Section 16.1.2 of the FSAR Supplement for inaccessible medium-voltage cables exposed to significant moisture and voltage, the applicant states that it will either test for the presence of aging effects or implement a periodic replacement program for these cables. If periodic replacement of medium-voltage underground cables is determined to be the most effective action for this type of cable, ANO-1 will define the frequency for replacement prior to the expiration of the initial 40-year licensing term. The frequency will be based on site-specific and industry operating experience. The staff finds the revised summary description as submitted by the applicant in a letter to the NRC dated March 14, 2001, acceptable, and considers FSAR Item 3.3.7.4 of Open Item 3.3-1 resolved.

3.3.7.3 Conclusions

On the basis of the staff's evaluation described above, the staff finds that there is reasonable assurance that the effects of aging of connectors, terminal blocks, and cables at ANO-1 will be adequately managed so that the intended function will be maintained consistent with the applicant's CLB for the period of extended operation in accordance with 10 CFR 54.21(a)(3).

3.3.7.4 References for Section 3.3.7

1. 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."
2. DG-1047, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," Working Draft, April 21, 2000.
3. "Arkansas Nuclear One - Unit 1, License Renewal Application," January 31, 2000.
4. First Energy, Davis-Besse Nuclear Generating Station, "Root Cause Analysis Report, #2 CCW Pump Trip, CR-1999-1648," October 2, 1999.
5. "Aging Management Guideline for Commercial Nuclear Power Plants–Electrical Cable and Terminations" (SAND96-0344)

4 TIME-LIMITED AGING ANALYSES

4.1 Identification of Time-Limited Aging Analyses

In the LRA, Section 4.1, the applicant identifies the time-limited aging analyses (TLAAs) applicable to Arkansas Nuclear One, Unit No. 1 (ANO-1). The NRC staff reviewed the information in the license renewal application (LRA) to determine whether the applicant provides adequate information to meet the requirements stated in 10 CFR 54.21(c)(1).

4.1.1 Summary of Technical Information in the Application

In the LRA, Table 4.1-1, the applicant identifies the calculations and analyses that satisfied the six criteria of 10 CFR 54.3 for a TLAA. The applicant identifies the following as TLAAs:

RCS Piping

- metal fatigue
- analytical evaluation of flaws
- leak before break analysis
- thermal stratification

Pressurizer

- metal fatigue
- analytical evaluation of flaws

Reactor Vessel

- metal fatigue
- analytical evaluation of flaws
- intergranular separation
- thermal shock
- flow-induced vibration (FIV) analysis

Reactor Vessel Internals

- metal fatigue
- analytical evaluation of flaws
- FIV analysis
- stress and deflection analyses

Once-Through Steam Generators

- metal fatigue
- analytical evaluation of flaws

Reactor Coolant Pumps

- metal fatigue

- analytical evaluation of flaws

Control Rod Drive Mechanism Pressure Boundary

- metal fatigue
- analytical evaluation of flaws

Concrete Reactor Building Tendon

- loss of prestress

Reactor Building Liner Plate and Penetrations

- fatigue analysis

Spent Fuel Racks

- aging of Boraflex

Electrical Equipment

- environmental qualification

Reactor Coolant Pump Motor Flywheels

- fatigue crack growth

4.1.2 Staff Evaluation

In the LRA, Section 4.1, the applicant describes the requirements for identifying and evaluating TLAAs and plant-specific exemptions based on TLAAs. The applicant reviewed plant-specific documents including the ANO-1 licensing correspondence file, the ANO-1 updated final safety analysis report (UFSAR), Babcock and Wilcox (B&W) topical reports referenced in correspondence and in the UFSAR, and American Society of Mechanical Engineers (ASME) Section XI summary reports. The information provided by the applicant was reviewed by the NRC staff to determine which analyses and calculations met the six criteria defining TLAAs in 10 CFR 54.21(c)(1).

4.1.3 Conclusions

The NRC staff concludes that the applicant has provided a list of acceptable TLAAs as defined in 10 CFR 54.3, and that no 10 CFR 50.12 exemptions have been granted on the basis of a TLAA as defined in 10 CFR 54.3.

4.1.4 References for Section 4.1

1. 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."
2. DG-1047, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," Working Draft, April 21, 2000.
3. "Arkansas Nuclear One - Unit 1, License Renewal Application," January 31, 2000.
4. NEI 95-10, "Industry Guidelines for Implementing the Requirements of 10 CFR Part 54 – The License Renewal Rule," Revision 0, March 1996.

THIS PAGE IS INTENTIONALLY LEFT BLANK

4.2 Reactor Vessel Neutron Embrittlement

The TLAAs for evaluating the effects of neutron irradiation on the ability of the reactor vessel to resist failure during a pressurized thermal shock (PTS) event, and the maintenance of acceptable Charpy upper shelf energy (USE) levels are discussed in Section 4.2 of the LRA.

4.2.1 Technical Information in the Application

The applicant participated in a Babcock and Wilcox Owners Group (B&WOG) effort that produced a series of topical reports to demonstrate that the aging effects for the reactor coolant system are adequately managed for the period of extended operation. One report, BAW-2251A, "Demonstration of the Management of Aging Effects for the Reactor Vessel," addresses the reactor vessel and the applicable TLAAs. Staff review of this topical report and the applicant's responses to the review are given in Section 3.3.2.4 of this safety evaluation report (SER).

The TLAAs evaluated in the ANO-1 LRA include analyses and calculations performed to show compliance with 10 CFR 50.60, and 50.61, and Appendix G to 10 CFR Part 50, concerning PTS and acceptable Charpy USE levels. These are reviewed by the staff in the following paragraphs.

4.2.2 Staff Evaluation

BAW-2251A, "Demonstration of the Management of Aging Effects for the Reactor Vessel," was reviewed and approved by the NRC staff in a letter dated April 26, 1999. In the LRA, Section 4.18, the applicant states that the ANO-1 reactor vessel integrity program is being utilized to ensure that the time dependent parameters used in the TLAA evaluations, as reported in BAW-2251A, are tracked such that the TLAA analyses remain valid for the period of extended operation. The staff reviewed the reactor vessel integrity program in Section 3.3.2.4.2.2 of this SER, and finds it acceptable for the period of extended operation. The two aspects of reactor vessel embrittlement are reactor vessel resistance to failure during PTS events and the maintenance of acceptable Charpy USE levels. Both are connected with neutron irradiation, for which the maximum anticipated effects would be in the reactor vessel beltline region at the end of the period of extended operation. A discussion of the two TLAAs is provided below.

Pressurized Thermal Shock

Rules for protecting against PTS in pressurized water reactors are given in 10 CFR 50.61(b)(1). Licensees are required to perform an assessment of the reactor vessel material's projected values of PTS reference temperature, RT_{PTS}, through the end of their operating license. With the potential approval of its application for an extended period of operation for ANO-1, this period will be through 48 EFPY.

In the LRA, Section 4.2.1, the applicant includes a description of the two options for determining RT_{PTS} for reactor vessel materials. As stated in 10 CFR 50.61(b)(1) the two acceptable methods for determining RT_{PTS} are as follows:

- Position 1 - for material that does not have surveillance data available

C Position 2 - for material that has surveillance data

Availability of surveillance data is not the only measure of whether Position 2 may be used. The data must also meet the credibility criteria given in the PTS rule (10 CFR 50.61).

Using the terminology in 10 CFR 50.61, RT_{PTS} is the sum of the initial (unirradiated) reference temperature, $RT_{NDT(u)}$, the shift in reference temperature caused by neutron irradiation (?RT_{NDT}), and a margin term (M) to account for uncertainties.

$RT_{NDT(u)}$ is determined using the method of Section III of the ASME Boiler and Pressure Vessel Code. That is, $RT_{NDT(u)}$ is the greater of the drop weight nil-ductility transition temperature or the temperature that is 60EF below that at which the material exhibits Charpy test values of 50 ft-lbs and 35 mils lateral expansion. For a material for which test data are unavailable, generic values may be used if there are sufficient test results for that class of material. For Linde 1092, 0091, and 124, the generic value of $RT_{NDT(u)}$ is -56EF. For Linde 80 weld material, with the exception of WF-70, the $RT_{NDT(u)}$ is taken to be the currently NRC-accepted values of -5EF or -7EF. The value of -5EF or -7EF is the statistical mean value of Linde 80 welds tested by B&W as documented in topical reports BAW-2166 or BAW-1803, respectively. The ANO-1 reactor vessel does not contain any Linde 80 WF-70 weld material. For forgings and plate material, measured values are used where appropriate data are available. Where not available, a B&W generic value of +3EF is used for forgings.

For Position 1 materials (surveillance data not available), ?RT_{NDT} is defined as the product of the chemistry factor and the fluence factor. The chemistry factor is a function of the material's copper and nickel content expressed as weight percent. Although not explicitly discussed by the applicant, the "best estimate" copper and nickel contents will normally be the mean of the measured values for a plate or forging. For a weld, the best estimate values will normally be the mean of the measured values from weld deposits made using the same weld wire heat number as the limiting weld. For ANO-1, best estimate values were obtained from BAW-2251A. The value of the chemistry factor is directly obtained from tables in 10 CFR 50.61. The fluence factor is calculated using end-of-license peak fluence at the clad-to-base metal interface for the material's location. Fluence values were obtained by extrapolation to 48 EFPY from the current 32 EFPY values.

For Position 2 materials (surveillance data available), the discussion above for Position 1 applies except for determination of the chemistry factor, which in this instance is a material-specific value calculated as follows:

- multiply each ?RT_{NDT} value by its corresponding fluence factor
- sum these products
- divide this sum by the sum of the squares of the fluence factors.

The applicant does not discuss the ratio procedure in 10 CFR 50.61. If surveillance data are being used and there is clear evidence that the copper and nickel content of the surveillance weld differs from the vessel weld (i.e., differs from the average for the weld wire heat number associated with the vessel weld and the surveillance weld), the measured values of ?RT_{NDT}

must be adjusted for differences in copper and nickel by multiplying them by the ratio of the chemistry factor for the vessel weld to that for the surveillance weld.

The margin term (M) is generally determined as follows:

$M = 2 (s_I^2 + s_?^2)^{0.5}$
where s_I is the standard deviation for $RT_{NDT(u)}$
and $s_?$ is the standard deviation for $?RT_{NDT}$

For determining M, $s_I = 0$ if a measured value is used. If a generic value is used, s_I is the standard deviation of the set of values used to obtain the mean value. For $?RT_{NDT}$, $s_?$ = 28EF for welds and 17EF for base metal (plate and forging), except that $s_?$ need not exceed one-half of the mean value of $?RT_{NDT.}$ Note that when using Position 2, the same method for determining the s values is used except that $s_?$ values may be halved (14EF for welds and 8.5EF for base metal).

In accordance with 10 CFR 50.61(b)(2), the screening criteria for RT_{PTS} is 270EF for plates, forgings, and axial welds, and 300EF for circumferential welds. The values of RT_{PTS} at 48 EFPY for ANO-1 are given in Appendix A, Table A-1, of BAW-2251A. The RT_{PTS} values are shown to be below the screening criteria through 48 EFPY.

In a letter to the NRC dated July 1, 1998, the applicant submitted its response to an RAI regarding Supplement 1 to GL 92-01, Revision 1, "Reactor Vessel Structural Integrity." The information was also contained in the B&WOG topical report BAW-2325. In this response, the applicant states that after review of BAW-2325, the staff noted changes in the transition temperature shift data for certain surveillance capsules and issued several requests to the B&WOG for additional information. Subsequent interactions between the B&WOG and the staff resulted in the publication of Revision 1 to BAW-2325 in February 1999. Since BAW-2251A was completed prior to the BAW-2325, Revision 1, an assessment was performed by the applicant relative to the staff's findings regarding chemistry factors reported in BAW-2251A. The chemistry factors reported in BAW-2251A are equivalent to, or exceed, the chemistry factors reported in BAW-2325, Revision 1, for the limiting beltline welds at ANO-1. In addition, ANO-1 has recalculated the 48 EFPY fluence for the beltline region using the methodology described in BAW-2251A, Appendix D, and BAW-2241AP and has determined that the 48 EFPY fluence estimates reported in BAW-2251A remain conservative. Therefore, the 48 EFPY RT_{PTS} values for the limiting beltline welds reported in BAW-2251A, Table A-1, remain conservative for ANO-1 since both the chemistry and fluence estimates remain conservative.

In order to avoid exceeding the PTS screening criteria at ANO-1 during the period of extended operation, the applicant utilizes low leakage core designs. In addition, the applicant is involved with various industry activities that provide new information or new analysis techniques associated with the reactor vessel beltline region.

The limiting material for ANO-1 at the end of the license renewal period (48 EFPY) is projected to be weld WF-112 (weld wire heat number 406L44). The RT_{PTS} value was calculated using Position 1 in 10 CFR 50.61. The limiting projected RT_{PTS} value for ANO-1 is below the screening criterion at the end of the license renewal period. The limiting weld is the upper to lower shell circumferential weld with material identification WF-112 and weld wire heat number

406L44. It has a projected value of RT_{PTS} at 48 EFPY of 278EF (the screening criterion is 300EF for circumferential welds). Therefore, the staff found that, with respect to PTS events, the ANO-1 reactor vessel has sufficient margin to perform its intended function over the period of extended operation.

Charpy Upper-Shelf Energy

Although not discussed by the applicant, Appendix G to 10 CFR Part 50 requires that reactor vessel beltline materials have Charpy USE levels in the transverse direction for the base metal and along the weld for the weld material according to the ASME Code, of no less than 75 ft. lbs. (102 J) initially, and must maintain Charpy USE levels throughout the life of the vessel of no less than 50 ft. lbs. (68 J). However, Charpy USE levels below these criteria may be acceptable if it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that the lower values of Charpy upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code.

The B&WOG position on USE for 32 EFPY is documented in its responses to GL 92-01, Revision 1, "Reactor Vessel Structural Integrity" as reported in BAW-2166 and BAW-2222. The B&WOG position on USE for 48 EFPY is documented in BAW-2275, which is included in BAW-2251A as Appendix B.

RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," provides two positions for determining Charpy upper-shelf energy (C_vUSE). Position 1 is for material that does not have surveillance data available and Position 2 is for material that does have surveillance data. For Position 1, the percent drop in C_vUSE, for a stated copper content and neutron fluence, is determined by reference to Figure 2 of RG 1.99, Revision 2. This percent drop is then applied to the initial C_vUSE to obtain the adjusted C_vUSE. For Position 2, the percent drop in C_vUSE is determined by plotting the available surveillance data on Figure 2 of RG 1.99, Revision 2 and fitting the data with a line drawn parallel to the existing lines that upper bounds all the plotted points. Again, the percent drop is determined, and used to adjust the initial C_v USE value.

Charpy USE issues are discussed in Section 4.2.2 of the application. The 48 EFPY C_v USE values determined for the ANO-1 reactor beltline materials are given in BAW-2251A, Table 4-4. The T/4 fluence values in this table were calculated in accordance with the ratio of the clad-to-base metal interface fluence to T/4 fluence values (i.e., neutron fluence lead factors at T/4) determined in the last reactor vessel surveillance program report. Table 4-4 shows that the C_vUSE is maintained above 50 ft-lbs for all base materials (plates and forgings), but weld materials nearly always fall below the 50 ft-lb limit at 48 EFPY. Appendix G of 10 CFR Part 50 provides for this situation by allowing lower values of C_vUSE if it is demonstrated that the lower C_vUSE will provide margins of safety against fracture equivalent to those required by Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code. An equivalent margins analysis was performed for 48 EFPY, and the results reported in Appendix A to BAW-2251A for service levels A, B, C, and D. For service levels A and B, the results demonstrate that there is sufficient margin beyond that required by the acceptance criteria of Appendix K to Section XI of the ASME Code (1995 Edition). For service levels C and D, the most limiting transient was evaluated. Again, the results showed that there is a sufficient margin beyond that required by the acceptance criteria of Appendix K to Section XI of the ASME Code.

As mentioned earlier in this evaluation, the applicant submitted a response to an RAI for ANO-1 regarding Supplement 1 to GL-92-01, Revision 1. This response was BAW-2325, Revision 1. The “best estimate” chemistry composition (copper and nickel) was reported in BAW-2325, Revision 1. Best estimate chemistry compositions were also reported in BAW-2251A, and were summarized in Table A-1 of Appendix A to BAW-2251A for the various reactor vessel materials. The copper composition reported in BAW-2251A is equivalent to, or exceeds, the copper content reported in BAW-2325, Revision 1. In addition, the 48 EFPY fluence estimates were recalculated using the methodology described in Appendix B of BAW-2251A. It was shown that the fluence estimates listed in BAW-2251A remain conservative. Therefore the C_vUSE values, given in Table 4-4 of BAW-2251A, remain conservative.

The Appendix K analysis, from Section XI of the ASME Boiler and Pressure Vessel Code involves a quantitative assessment of the impact of low C_vUSE on reactor vessel integrity. In Appendix K analysis, cracks are postulated at the inner reactor vessel wall. Since the neutron fluence decreases with depth into the vessel, the Appendix K analysis method assumes the fracture toughness at the crack tip will be greater than that at the inner wall of the vessel. The applicant’s analysis was carried out using conservative stress assumptions for service levels A, B, C, and D for 48 EFPY. The analysis, given in Appendix B of BAW-2251A, shows that for service levels A and B, there is sufficient margin beyond that required by the acceptance criteria of Appendix K to Section XI of the ASME Code (1995 Edition). For service levels C and D, the most limiting transient was evaluated, and again the analytical results demonstrated that there is a sufficient margin beyond that required by Appendix K to Section XI of the ASME Code. The applicant concludes that evaluations for all four service levels show the adequacy of safety against fracture for the ANO-1 vessel for 48 EFPY.

The staff found the B&WOG evaluation of the Charpy USE acceptable for all ANO-1 materials for the period of extended operation because the 48 EFPY analysis reported in Appendix B of BAW-2251A, and referenced in this application, meets the provisions of 10 CFR 54.21(c)(1)(ii) and applies to ANO-1.

4.2.3 Conclusions

The staff has reviewed the TLAAs concerning irradiation-induced changes in reactor vessel material that affect PTS resistance and Charpy USE levels. On the basis of its review, the staff concludes that the applicant’s PTS and USE analyses satisfy the requirements of 10 CFR 54.21(c)(1)(ii).

4.2.4 References for Section 4.2

1. 10 CFR Part 54, “Requirements for Renewal of Operating Licenses for Nuclear Power Plants.”
2. DG-1047, “Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants,” Working Draft, April 21, 2000.
3. “Arkansas Nuclear One – Unit 1, License Renewal Application,” January 31, 2000.

4. BAW-2251A, “Demonstration of the Management of Aging Effects for the Reactor Vessel,” B&WOG Generic License Renewal Program, June 1996.
5. NRC GL 92-01, Revision 1, Supplement 1, “Reactor Structural Integrity,” May 19, 1995.
6. BAW-2325, “Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity – Generic Letter 92-01, Revision 1, Supplement 1,” B&WOG, May 1998.
7. BAW-2325, “Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity, Revision 1, Supplement 1,” B&W Owners Group, January 1999.
8. BAW-2241AP, “Fluence and Uncertainty Methodologies,” April 1997.
9. BAW-2166, “Response to Generic Letter 92-01,” June 1992.
10. BAW-2222, “Response to Closure Letters to Generic Letter 92-01, Revision 1,” June 1994.
11. ASME Boiler and Pressure Vessel Code, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components.”
12. ASME Boiler and Pressure Vessel Code, Section III, “Rules for Construction of Nuclear Power Plant Components.”
13. 1CAN079801, Letter from D. James (ANO) to the NRC, “Generic Letter 92-01, Supplement 1, Reactor Vessel Structural Integrity, Request for Additional Information,” July 1, 1998.

4.3 Metal Fatigue

A metal component subjected to cyclic loads may fail at a load magnitude less than its ultimate load capacity due to metal fatigue, initiating and propagating cracks in the material.
The fatigue life of a component is a function of its material, its environment, and the number and magnitude of the applied cyclic loads. Fatigue was a design consideration for piping and components in the ANO-1 RCS and, consequently, fatigue is part of the current licensing basis (CLB) for ANO-1. The applicant identifies fatigue and flaw growth evaluations as TLAAs for the piping and components of the RCS. The staff reviewed Section 4.3 of the LRA, which discusses thermal fatigue and flaw growth, as well as other related fatigue programs.

4.3.1 Technical Information in the Application

In the LRA, Section 4.3, the applicant discusses design criteria for thermal fatigue of RCS piping and components. The B&W scope of supply includes major components in the RCS, and the associated interconnecting piping. Vessels were designed in accordance with ASME Section III, 1965 edition, with addenda through the summer 1967. Reactor coolant pumps were designed in accordance with ASME Section III, 1968 edition. RCS piping supplied by B&W was designed to Nuclear Piping Code, USAS B31.7 Class 1. Bechtel-supplied piping includes the Class 1 portions of ancillary systems that are attached to the B&W scope of supply and miscellaneous vents, drains, and instrumentation lines. Bechtel-supplied piping was designed to Nuclear Piping Code USAS B31.7, Class 1, dated February 1968, and as corrected by Errata date of June 1968, or later appropriate ASME Section III Code sections, provided they have been reconciled.

The applicant's TLAA evaluation addresses the following topics:

- thermal fatigue, with separate consideration of environmentally-assisted fatigue, thermal stresses in piping connected to the RCS, and pressurizer line thermal stratification

- the ANO-1 transient cycle logging program

- flaw growth evaluation to demonstrate compliance with the ASME Section XI Inservice Inspection Requirements

4.3.2 Staff Evaluation

As discussed in the previous section, components of the RCS were designed to codes that contained explicit criteria for the fatigue analysis. Consequently, the applicant identifies the fatigue analyses and the flaw growth evaluations of the RCS components as TLAAs. The staff reviewed the applicant's evaluation of RCS components for compliance with the provisions of 10 CFR 54.21(c)(1).

The specific design criterion for RCS components involves calculating the cumulative usage factor (CUF). The fatigue damage caused by each thermal or pressure transient depends on the magnitude of the stresses caused in the component by a transient. The CUF sums the fatigue resulting from each transient. The applicant indicates that it addresses fatigue by ensuring that its effects are adequately managed for the period of extended operation.

For the B&W-supplied components, the design cyclic loadings and thermal conditions are defined by the component design specifications. The component design specification defines the transient cycle assumptions used in the fatigue evaluations for the component. As part of the B&W Generic License Renewal Program, the applicant was involved in a review to determine which Class 1 components were more sensitive to fatigue (environmentally-assisted fatigue was not considered), and which transients caused the greatest impact in terms of fatigue stress on the components. For this set of design transients, the number of transients accrued was compiled and a conservative projection was made to determine if the number of design transients would be exceeded in the period of extended operation. The applicant determines that, in no instance for ANO-1, did the extrapolation exceed the number of allowable design cycles prior to 60 years of operation.

For the Bechtel-supplied piping, the design cycle loading and thermal conditions are defined in a Bechtel Class 1 piping design specification. Existing cumulative usage factors and analyzed thermal transients documented in thermal fatigue calculations for the piping were reviewed by the applicant. On the basis of the number of transient cycles accrued for ANO-1 and the rate these cycles have been accumulated, the number of transient cycles that were originally projected for the current license term of 40 years envelopes the number of projected cycles to the end of a 60 year operating life.

The applicant has a process to log transient history and operating transient cycles. Applicable site procedures contain the responsibilities, logging requirements, reporting requirements and transient type definitions. Guidance is provided for collection of the necessary plant data and for projection of the number of cycles to end-of-life. The ANO-1 operating transient cycle logs are retained for the duration of the licence, per site procedures and ANO-1 TS. The applicant's procedure provides assurance that the number of plant transient cycles assumed in the design of the RCS components will not be exceeded.

Generic Safety Issue (GSI)-166, "Adequacy of the Fatigue Life of Metal Components," raised concerns regarding the conservatism of the fatigue curves used in the design of the RCS components. Although GSI-166 was resolved for the current 40-year design life of operating components, the staff identified GSI-190, "Fatigue Evaluation of Metal Components for 60-year Plant Life," to address the period of extended operation for license renewal. The NRC closed GSI-190 in December 1999, concluding the following:

> *The results of the probabilistic analyses, along with the sensitivity studies performed, the iterations with industry (NEI and EPRI), and the different approaches available to the licensees to mange the effects of aging, lead to the conclusion that no generic regulatory action is required, and that GSI-190 is closed. This conclusion is based primarily on the negligible calculated increases in core damage frequency in going from 40 to 60 year lives. However, the calculations supporting resolution of this issue, which included consideration of environmental effects, and the nature of age-related degradation indicate the potential for an increase in the frequency of pipe breaks as plants continue to operate. Thus, the staff concludes that, consistent with existing requirements in 10 CFR 54.21, licensees should address the effects of coolant environment on component fatigue life as AMPs are formulated in support of license renewal.*

From this NRC guidance on addressing environmental effects on fatigue, the applicant has adopted a procedure in which the specific locations in ANO-1 that are most susceptible to failure from thermal fatigue, and other degradation mechanisms, are determined by analysis. The calculations include consideration of stress level and lower bound material properties. From this information the applicant includes the most susceptible components in an augmented inservice inspection program.

The applicant lists the following critical component locations in B&W plants that are applicable to ANO-1:

- reactor vessel shell and lower head
- reactor vessel inlet and outlet nozzles
- pressurizer surge line
- makeup/high pressure injection nozzles
- reactor vessel core flood nozzle
- decay heat removal system Class 1 piping

The B&WOG conducted an environmentally-assisted fatigue analysis for the reactor vessel and documented it in BAW-2251A. This study derived environmental fatigue factors based on the model described in NUREG/CR-6335, "Fatigue Strain-Life Behavior of Carbon and Low-Alloy Steels, Austenitic Stainless Steels, and Alloy 600 in LRA Environments." These factors were applied to the reactor vessel shell and lower head, the vessel inlet and outlet nozzles, and the core flood nozzles. The study concluded that, after an accounting for environmentally-assisted fatigue, the reactor vessel fatigue usage factors remain acceptable for the period of extended operation. The applicant, therefore, concludes that reactor vessel shell, lower head, inlet and outlet nozzles, and core flood nozzles are no longer issues with respect to environmentally-assisted fatigue during the period of extended operation. In its April 26, 1999, SER for BAW-2251A, the NRC staff concluded that the environmental effects on the fatigue life of reactor vessel components had been adequately addressed for license renewal.

The applicant indicates that the three remaining locations (the pressurizer surge line, makeup/high pressure injection nozzles, and the decay heat removal system Class 1 piping) are included in the risk-informed ISI (RI-ISI) program which has recently been approved by the NRC staff as an alternative to requirements of ASME Section XI Inservice Inspection. The primary objective of the program was to identify "risk important" piping sections for inspection based on the analysis of the probability, and the consequences of piping failure. The applicant concludes that implementation of the RI-ISI program will ensure that inspections at ANO-1 will be performed in locations where degradation mechanisms, including thermal fatigue, are most likely to occur. In a letter to the applicant dated May 5, 2000, the staff requested additional information regarding the potential for fatigue cracking in the three remaining locations considering the data contained in NUREG/CR-5704, "Effects of LWR Coolant Environment on Fatigue Design Curves of Austenitic Stainless Steels."

In its response, the applicant outlines the environmentally-assisted fatigue analyses that were carried out for the three components specified by the staff. Specifically, the applicant states that the work to close out GSI-190 includes a review of the results of the INEEL studies published in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," as well as later experimental studies by ANL to account for the detrimental effects of primary coolant on the fatigue life. Using the environmental fatigue data in NUREG/CR-5704, the applicant's evaluation indicates that the surge line and the HPI/MU nozzles and safe ends have CUFs that may exceed 1.0 during the period of extended operation. For the decay heat removal piping, the CUF calculated from 1986 Code rules is less than 1.0. To address the locations where the CUF may exceed 1.0 when environmental effects are considered, the applicant proposed a program to manage the effects of fatigue. This program will be undertaken prior to the period of extended operation and will include one or more of the following options:

- refinement of the fatigue analysis in an attempt to lower the CUF to less than 1.0

- repair of affected locations

- replacement of affected locations

- management of the effects of fatigue during the period of extended operation using a program that will be reviewed and approved by the staff

The applicant commits to provide the NRC with the inspection details of the aging management program (AMP) requiring staff approval for managing the effects of fatigue prior to the period of extended operation if the last option is selected. As indicated by the applicant, the use of an AMP to manage fatigue will require prior staff review and approval. The staff found the applicant's proposed program an acceptable plant specific approach to address environmentally-assisted fatigue during the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1). However, in accordance with 10 CFR 54.21(d), this information needs to be added to the FSAR Supplement. This was FSAR Item 4.3.4 of Open Item 3.3-1.

The applicant also discusses actions taken in response to the NRC's Bulletin (BL) 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems." In NRC BL 88-08, the staff requested that licensees review their RCS designs to identify any connected, non-isolable sections of pipe that could be subjected to temperature distributions that would result in unacceptable stresses. The applicant reviewed 23 different piping configurations connected to the RCS. As a result of its BL 88-08 reviews, the applicant added temperature monitoring devices to the HPI lines. The decay heat system suction line from the RCS also required monitoring and evaluation due to packing leaks on an isolation valve.

The applicant indicates that, because of the detection of stratified flow in ANO-2 lines, ANO-1 systems were reviewed again, to identify systems with attributes similar to the ANO-2 stratified lines. Four lines were found to require monitoring and evaluation. They are the pressurizer main spray, decay heat drop leg, RCS drains, and the RCS letdown drains. The applicant states that temperature monitoring and evaluation have demonstrated that these ANO-1 lines are qualified for their service conditions.

In response to BL 88-08, the applicant commits to perform enhanced ultrasonic examinations of 17 HPI welds, and visual inspection of two segments of HPI piping as part of its 10-year interval Inservice Inspection Plan. Subsequently, the scope of the ISI for the HPI lines and pressurizer surge line was modified based on an ANO-1 risk analysis performed consistent with the requirements of ASME Code case N-560, "Alterative Examination Requirements for Class 1, Category B-J Piping Welds, Section XI, Division 1." This commitment will be continued by the applicant through the period of extended operation.

In a letter to the applicant dated May 5, 2000, the applicant was asked to describe its modified inspection program for HPI welds and piping. In its response dated September 6, 2000, the applicant states that, initially, the inspections were to be ultrasonic for the welds and visual for the piping segments, in response to BL 88-08. As a result of the implementation of Code Case N-560, a new RI-ISI program was developed based on volumetric examination of the 13 most susceptible welds. Visual examination of the piping segments was eliminated from the program. The staff approved this modified program by a letter dated August 25, 1999.

The applicant discussed actions taken in response to NRC BL 88-11, "Pressurizer Surge Line Thermal Stratification." In BL 88-11, the staff requested that licensees establish and implement a program to confirm the integrity of the surge line, and to inform the NRC of actions taken to resolve the issue.

Originally, the applicant committed to performing enhanced ultrasonic examination of two elbows of the pressurizer surge line as part of the ANO-1 10-year interval ISI plan in response to BL 88-11. Subsequently, the scope of ISI inspections of the surge line was modified based on an ANO-1 risk analysis performed consistent with the requirements of ASME Code Case N-560, "Alternative Examination Requirements for Class 1, Category B-J Piping Welds, Section XI, Division 1." The applicant commits to continuing the examination through the period of extended operation.

In a letter to the applicant dated May 5, 2000, the staff requested clarification on possible modifications to the ISI procedure for the ultrasonic examination of the two elbows in the surge line as a result of the adoption of a new RI-ISI plan. In its response to the NRC dated September 6, 2000, the applicant states that the commitments made for the ISI for the elbows in the surge line in response to BL 88-11, had not changed as a result of the implementation of risk-informed evaluations required by Code Case N-560.

In the LRA, Section 4.3.4.4, the applicant discusses the actions taken in response to cracking of HPI/MU nozzle cracking in B&W plants, described in Information Notice 82-09, "Cracking in Piping of Makeup Coolant Lines at B&W Plants"; Generic Letter 85-20, "Resolution of Generic Issue 69: High-Pressure Injection/Makeup Nozzle Cracking in Babcock and Wilcox Plants," and NRC Information Notice 97-46, "Unisolable Crack in High-Pressure Injection Piping." On the basis of the recommendations by a B&WOG task force, actions taken by the applicant include repair of nozzles with loose or damaged sleeves, maintenance of adequate minimum flow, implementation of augmented inspection programs for the nozzles, and performance of stress analysis with modified thermal sleeves. The augmented ISI program for the HPI/MU nozzles is consistent with the methodology and scope of inspection recommended by the B&WOG Safe-End Task Force. The applicant commits to ultrasonic testing of the knuckle region of the HPI nozzles every fifth refueling cycle, and radiography of the thermal sleeves will continue through the period of extended operation. In a letter to the NRC dated May 5, 2000, the applicant states that there will be radiographic testing of the sleeves and the gap between the safe-ends and the sleeves every fifth refueling cycle to monitor for cracking during the period of extended operation. The staff agrees that the augmented inspection program provides an acceptable method to manage cracking of the HPI/MU nozzles during the period of extended operation.

In the LRA, Section 4.3.6, the applicant discusses flaw growth evaluation. The applicant states that indications detected during ISI that exceed that specified acceptance criterion could be analytically evaluated using crack growth analysis. The crack growth analyses would consider the same design transient cycle assumptions used in the original design. Since the analyses were performed using the full number of design transient cycles, which have been demonstrated to be applicable over 60 years of operation, these flaw growth calculations remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

The applicant uses ASME Code Case N-481 to evaluate RCP weld flaws. The Code Case N-481 flaw tolerance evaluation was reviewed by the applicant to determine if the evaluation is acceptable for the period of extended operation. A separate effort was carried out to evaluate the acceptability of a Code Case N-481 flaw growth analysis for the RCS pump casings for the period of extended operation, taking into consideration the effects of thermal aging on fracture toughness. The fatigue growth calculation performed, included an assumption of 240 heatup and cooldown cycles. Since the applicant has not increased the number of design transients for license renewal, the applicant states that the flaw growth evaluation for pump casings is acceptable for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i). The staff found that the applicant's TLAA evaluation meets the requirements of 10 CFR 54.21(c)(1).

After its initial review, the staff requested that the FSAR Supplement include a summary description of the applicant's evaluation of NUREG/CR-6260 components for environmental fatigue including the options for future evaluations of the surge line and HPI/MU nozzles and safe ends in accordance with 10 CFR 54.21(d). This was FSAR Item 4.3.4 for Open Item 3.3-1.

In its revised summary description of the FSAR Supplement, Section 16.3.2, the applicant provides a description proposed program as described above to address environmental effects of fatigue including the options for future evaluations that meets the requirements of 10 CFR 54.21(d). The staff found the revised summary description submitted by the applicant in a letter to the NRC dated March 14, 2001, acceptable.

4.3.3 Conclusions

On the basis of its projection of the number of expected transients the applicant concludes that the fatigue analysis of RCS components and the flaw growth evaluation of indications found during component inspections will remain valid for the period of extended operation. The applicant also has a process to maintain a record of these transients and that process will continue during the period of extended operation. In addition, the applicant commits to implement a program, prior to the period of extended operation, to manage two locations in the RCS where environmental effects on the fatigue life are significant. On the basis of the applicant's TLAA evaluations, and its commitment to implement a program to manage the environmental effects on fatigue at critical locations prior to the period of extended operation, the staff concludes that the applicant's actions and commitments satisfy the requirements of 10 CFR 54.21(c)(1).

4.3.4 References for Section 4.3

1. 10 CFR Part 54, “Requirements for Renewal of Operating Licenses for Nuclear Power Plants.”
2. DG-1047, “Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants,” Working Draft, April 21, 2000.
3. “Arkansas Nuclear One - Unit 1, License Renewal Application,” January 31, 2000.
4. ASME Boiler and pressure Vessel Code, Section III, “Rules for Construction of Nuclear Power Plant Components.”
5. USAS B31.7, “Nuclear Power Piping.”
6. NUREG/CR-6335, “Fatigue Strain-Life Behavior of Carbon and Low-Alloy Steels, Austenitic Stainless Steels, and Alloy 600 in LRA Environments,” August 1995.
7. NUREG/CR-5704, “Effects of LWR Coolant Environment on Fatigue Design Curves of Austenitic Stainless Steels,” April 1999.
8. NRC BL 88-08, “Thermal Stresses in Piping Connected to Reactor Coolant Systems,” June 22, 1988.
9. NRC BL 88-11, “Pressurizer Surge Line Thermal Stratification,” December 20, 1988.
10. NRC IN 82-09, “Cracking in Piping of Makeup Coolant Lines at B&W Plants,” March 31, 1982.
11. NRC GL 85-20, “Resolution of Generic Issue 69: High Pressure Injection/Makeup Nozzle Cracking in Babcock and Wilcox Plants,” November 11, 1985.
12. NRC IN 97-46, “Unisolable Crack in High-Pressure Injection Piping,” July 9, 1997.

THIS PAGE IS INTENTIONALLY LEFT BLANK

4.4 Environmental Qualification

The ANO-1 10 CFR 50.49 Environmental Qualification (EQ) Program has been identified as a TLAA for the purposes of license renewal. The TLAA of EQ components includes all long-lived, passive and active electrical components and commodities located in a harsh environment that are important to safety, including safety-related and Q-list equipment, non-safety-related equipment whose failure could prevent satisfactory accomplishment of any safety-related function, and the necessary post-accident monitoring equipment.

The NRC staff has reviewed Section 4.4, “Environmental Qualification,” of the LRA to determine whether the applicant submitted adequate information to demonstrate that they meet the requirements in 10 CFR 54.21(c)(1) regarding an evaluation of the EQ TLAA. In addition, the staff reviewed Section 4.4.69, “GSI-168 ‘EQ of Electrical Components’.”

On the basis of this review, the NRC staff requested additional information in letters to the applicant dated April 17, 2000, and April 25, 2000. The applicant responded to these RAIs in a letter to the NRC dated July 6, 2000. In addition, the NRC staff met with the applicant on May 25, 2000, to review related EQ calculations. The results of this meeting are documented in a letter from the NRC to the applicant dated June 13, 2000.

4.4.1 Technical Information in the Application

In the LRA, Section 4.4, the applicant describes the TLAA evaluation methodology and how the results from these evaluations were used to demonstrate that (i) the analyses remain valid for the period of extended operation; (ii) the analyses have been projected to the end of the period of extended operation; or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation. The following is a summary of the methodology used by the applicant to evaluate the EQ TLAAs and the results from this evaluation.

Scope of EQ Equipment

The qualification requirements for electrical equipment originally installed at ANO-1 are based on NRC IE Bulletin 79-01B, “Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors,” which is now referred to as the Division of Operating Reactors (DOR) Guidelines. The applicant’s EQ program complies with the scope of 10 CFR 50.49 requirements, and was “grandfathered” by 10 CFR 50.49 thereby allowing qualification in accordance with the DOR Guidelines. Therefore, the DOR Guidelines document is the CLB for the ANO-1 EQ program.

The Environmental Qualification Program at ANO-1 is a centralized plant support program administered by design engineering in order to maintain compliance with 10 CFR 50.49. The scope of the EQ program includes the following categories of electrical equipment located in a harsh environment:

- safety-related equipment
- non-safety-related equipment whose failure could adversely affect safety-related equipment

- the necessary post-accident monitoring equipment

The identification of EQ equipment is specified by procedural controls, and a component database is utilized to maintain an EQ equipment master list.

EQ Process

The EQ Program includes three main elements:

- establish and control a list of equipment and service conditions

- establish and control equipment documentation

- maintain (or preserve) qualification through preventive maintenance, the procurement process and corrective actions

As part of the first element, the applicant has established, and currently controls, an EQ master list of equipment, and the service condition for the harsh environment plant areas. The applicant has also established, and currently controls, the qualification documents, including vendor test reports, vendor correspondence, calculations, evaluations of equipment tested conditions as compared to plant required conditions, and determinations of configuration and maintenance requirements. Finally, the applicant established the following required processes to maintain the qualification:

- a preventive maintenance process for replacing parts and the equipment at required intervals

- a design control process to ensure that changes to the plant are evaluated to assess the potential impact on the EQ program

- a procurement process to ensure new and replacement equipment is purchased in accordance with applicable EQ requirements

- a corrective action process to identify and correct problems

Replacement of Equipment

As a normal part of the ANO-1 EQ process, when the EQ documentation process establishes that equipment, or parts thereof, have a limited life, the preventive maintenance process ensures that the equipment or parts are replaced before the expiration of the qualified life. The ANO-1 EQ program ensures that replacement equipment is purchased in accordance with applicable EQ requirements.

Reanalysis of the Qualified Life

If excess conservatism exists in the original qualified life determination, then reanalysis could be performed to extend the qualified life. The reanalysis would then become a part of the EQ documentation. Parameter conservatism may exist in the ambient temperature of the equipment, in an unrealistically low activation energy, and in the application of the equipment. The primary method used for reanalysis is to reduce excess conservatism in the equipment service temperature by using temperature values closer to an actual temperature measured in the area around the equipment being analyzed. This reanalysis is performed as follows:

- Analytical Methods - This reanalysis method uses standard EQ techniques, such as the Arrhenius methodology for thermal aging effects. Moisture has not been identified as a significant aging mechanism for ANO-1. The analytical method used for radiation analysis is to identify the 40-year radiation dose for the area where the equipment is installed, multiply that value by the ratio of the evaluation period divided by 40 years (i.e., 60 years/40 years = 1.5), and add the applicable accident radiation dose to obtain the total integrated dose for the equipment.

- Data Collection and Reduction Methods - The primary method used for reanalysis is to reduce excess conservatism in the equipment service temperatures. The applicant describes the following activities used to obtain temperature data for the reanalysis of EQ equipment:

    - A plant modification installed a temporary temperature monitoring system for the ANO-1 reactor building, and data were collected from 1989 to 1996. This system included temperature elements that monitored 21 different area ambient temperatures (at various elevations and azimuths) and 11 different EQ equipment surface temperatures.

    - In May 1989, the applicant conducted EQ walkdowns to determine the EQ equipment surface temperatures in the auxiliary building, and the temperatures in the associated general area.

    - Self-contained temperature data loggers were initially installed in the ANO-1 reactor building in 1993, and were used to gather additional temperature data.

    - In August/September 1997, the applicant conducted an environmental walkdown and documented area temperatures and any hot spots in several different buildings, including the auxiliary building. These measurements were taken at a single point in time with a hand-held digital thermometer or infrared camera.

    - For the reactor building, the applicant measured temperature on, next to, or within close proximity to the EQ equipment. Measuring devices were located in expected hot areas, such as the D-rings of the containment. Measurements were taken with the temporary monitoring system on most normal working days, and data logger measurements were taken continuously. From the data obtained, the applicant determined the overall operating temperature, which is generally several degrees above the average temperature value, and is visually selected so that most data points fall on or below this value.

    - For the auxiliary building, the applicant measured temperatures on equipment surfaces and in the areas of the auxiliary building containing EQ equipment at a single point in time. To provide conservative values of the operating temperatures, the applicant took these measurements while the plant was operating during warm months of the year. An infrared camera was used to specifically identify hot spots.

- Underlying Assumptions - ANO-1 was one of several plants cited in NRC Information Notice 89-30 as having experienced elevated temperatures in the plant. For ANO-1, the reactor building experienced the elevated temperatures. This event was identified and the conditions were evaluated in 1987, including revising EQ analyses. In an effort to reduce the reactor building temperatures, the applicant installed larger chilled water pumps, an additional chiller, and an additional air-handling unit. These major plant modifications reduced the reactor building operating temperatures. Plant modifications or initiatives are controlled by procedures that include determining the related impact on EQ analyses. There have been no major plant modifications or events at ANO-1 that have changed the radiation values used in the EQ analyses.

Refurbishment of EQ Electrical Equipment

Refurbishment is an option at ANO-1. EQ equipment that is in need of refurbishment is refurbished in place or is replaced with new equipment or previously refurbished equipment taken out of storage before exceeding its qualified life. Refurbishment is a process that preserves the qualification status of equipment and is typically accomplished by replacing items such as gaskets, seals, and wires that are the limiting components or sub-components for the qualified life. The EQ documentation identifies limited-life replacement parts for specific equipment, manufacturers, and models. The replacement option discussed for several equipment types would effectively involve refurbishment.

Ongoing Qualification/Retesting

For EQ equipment with a qualified life less than the required design life of the plant, “ongoing qualification” is a method of long-term qualification involving additional testing. Ongoing qualification or retesting, as described in IEEE Standard 323-1974, Section 6.6, “Ongoing Qualification,” is not currently considered by the applicant to be a viable option, and there are no plans to implement such an option. If this option becomes viable in the future, the applicant would perform ongoing qualification or retesting in accordance with accepted industry and regulatory standards.

Procurement of EQ Equipment

The ANO-1 EQ program includes procurement processes to ensure that new and replacement equipment is purchased in accordance with applicable EQ requirements.

Plant Environmental Changes

Controls used to monitor changes in plant environmental conditions involve temperature monitoring in the reactor building. For areas of the auxiliary building, the applicant relies upon normal operator rounds, personnel performing routine maintenance work, and periodic engineering walkdowns and inspections to identify changes in normal operating temperature conditions that might exceed the design value of 105EF.

EQ Generic Safety Issue (GSI)

To resolve GSI-168, “Environmental Qualification of Electrical Components,” the applicant has chosen to submit a technical rationale demonstrating that the effects of aging will be managed in accordance with 10 CFR 54.21(c)(1)(iii) until some future point in time when other more reasonable options become available.

4.4.2 Staff Evaluation

The NRC staff reviewed Section 4.4 of the ANO-1 LRA to determine whether the applicant submitted adequate information to meet the requirements of 10 CFR 54.21(c)(1). In addition, the NRC staff met with the applicant to obtain clarifications, to review specific EQ calculations, and to review the applicant’s response to RAIs.

The NRC staff verified that the applicant is using standard, approved EQ methodologies and acceptance criteria as defined by NRC IE Bulletin 79-01B (DOR Guidelines), including Supplements 1, 2, and 3; NUREG-0588, “Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment,” Revision 1; 10 CFR 50.49, “Environmental Qualification for Electric Equipment Important to Safety for Nuclear Power Plants”; Regulatory Guide 1.89, “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants,” Revision 1; and various EQ-related NRC generic letters, information notices; and SERs. The current ANO-1 actions, such as refurbish and replace, for short-lived EQ equipment are also acceptable for long-lived EQ equipment.

TLAA Demonstration for Option 10 CFR 54.21(c)(1)(i)

In the LRA, Section 4.4.18, “General Atomic Radiation Detectors,” the applicant states that it uses 10 CFR 54.21(c)(1)(i) in its TLAA evaluation to demonstrate that the analyses remain valid for the period of extended operation. The applicant applies this method to the detector assemblies, including their connectors, which are constructed of metal, ceramic, quartz cloth insulation, and Rexolite. With the exception of Rexolite, these materials are inorganic, and are not susceptible to thermal or radiation age degradation. The Rexolite is used in the connectors as a locator during assembly, and has no required function after assembly. On the basis of the NRC staff’s review of the information submitted by the applicant regarding the General Atomic radiation detectors, and its materials of construction, the NRC staff finds the applicant’s demonstration to be consistent with 10 CFR 54.21 (c)(1)(i).

TLAA Demonstration for Option 10 CFR 54.21(c)(1)(ii)

For the following list of electrical equipment identified in Section 4.4 of the LRA, the applicant uses 10 CFR 54.21(c)(1)(ii) in its TLAA evaluation to demonstrate that the analyses have been projected to the end of the period of extended operation:

- 4.4.1 Allis Chalmers Motors
- 4.4.2 Anaconda Instrumentation Cable, FR-EP Insulation

- 4.4.3 Anaconda Control and Power Cable, EP Insulation
- 4.4.4 Anaconda EPR Insulated Instrumentation, Control/Power Cable
- 4.4.8 Buchanan Terminal Blocks, Outside Reactor Building
- 4.4.9 Buchanan Terminal Blocks, Inside Reactor Building
- 4.4.10 Conax Thermocouples
- 4.4.11 Conax Resistance Temperature Detectors
- 4.4.12 Conax Multipin Connector
- 4.4.13 Conax Electrical Penetration Assemblies
- 4.4.14 Conax Electrical Connection Seal Assembly
- 4.4.15 Conax Electrical Feedthrough Adapters
- 4.4.16 Eaton Flame Retardant Ethylene Propylene Diene Monomer Insulated Cable
- 4.4.17 Gems De Laval Level Sensors
- 4.4.19 General Electric Terminal Blocks
- 4.4.21 Limitorque Motor-Operated Valve Actuators; Alternating Current/Inside Reactor Building (most applications)
- 4.4.22 Limitorque Motor-Operated Valve Actuators; Alternating Current/Outside Reactor Building
- 4.4.23 Limitorque Motor-Operated Valve Actuators; Direct Current/Outside Reactor Building
- 4.4.26 NAMCO EA-740 Limit Switches with NAMCO Connectors
- 4.4.27 NAMCO EA-740 Limit Switches
- 4.4.28 NAMCO Quick Connectors
- 4.4.29 Okonite 5 kV Power Cable with EPR Insulation and an Okolon Jacket
- 4.4.30 Okonite 2 kV Power and Control Cable with Okonite or Okoguard Insulation and Okoprene or Okolon Jackets
- 4.4.31 Okonite 600V Power Cable with Okonite Insulation and an Okolon Jacket (most applications)
- 4.4.32 Okonite 600V Power Cable with FMR Insulation (most applications)
- 4.4.33 Okonite T-95 and No. 35 Splicing Tapes (most applications)
- 4.4.34 Raychem 600V Flamtrol XLPE Cable
- 4.4.35 Raychem Cable Splice and Jacket Repair Tape (type NJRT)
- 4.4.36 Raychem Cable Splices (types WCSF-N, NPK, NMCK, ANK, etc.)
- 4.4.37 Reliance Electric, Electric Motors
- 4.4.38 Rockbestos Coaxial Cable (most applications)
- 4.4.39 Rockbestos Firewall III Irradiation Cross-Linked Polyethylene Cable
- 4.4.40 Rockbestos Firezone R Silicone Rubber High-Temperature Cable (some applications)
- 4.4.41 Rockbestos Firewall III Chemically Cross-Linked Polyethylene Cable
- 4.4.44 Rotork Motor Operated Valve Actuators, Model NA1
- 4.4.45 Target Rock Solenoid-Operated Valves (Report 2375)
- 4.4.47 Target Rock Modulating Solenoid Operated Valves (Report 3414)
- 4.4.48 Target Rock Solenoid-Operated Valves (Reports 2375 and 1827)
- 4.4.49 TEC Valve Flow Monitoring System (some subcomponents)
- 4.4.50 TEC Reactor Vessel Level Monitoring System (some subcomponents)
- 4.4.51 Weed Resistance Temperature Detectors
- 4.4.52 Dow-Corning 3145 Silicone Sealant
- 4.4.55 Westinghouse Motors, Models TBFC and SBDP

- 4.4.56 Babcock & Wilcox Core Exit Thermocouples (pin half connector with mineral-insulated cable)
- 4.4.57 Gamma Metrics Neutron Detectors and Cable Assemblies (organic cable)
- 4.4.58 Brand Rex Cross-Linked Polyethylene Coaxial Cable
- 4.4.59 Brand Rex Cross-Linked Polyethylene Power and Control Cable
- 4.4.60 NDT International Acoustic Sensor, Connector and Cable
- 4.4.61 American Insulated Wire 600V Instrumentation Cable
- 4.4.62 American Insulated Wire 600V Power and Control Cable
- 4.4.63 AMP Pre-insulated Butt Splices
- 4.4.64 EGS Quick Disconnect Electrical Connectors (except connector o-rings)
- 4.4.65 EGS Grayboot Electrical Connectors
- 4.4.67 Valcor Model V526-5961-1 Solenoid Operated Valve
- 4.4.68 General Cable Corporation 5 kV Power Cable

In its response to the NRC's RAIs, the applicant supplied the following clarifications regarding two of the above components:

- Target Rock Solenoid-Operated Valves - the EQ documentation shows that the valves are qualified for more than 60 years at 180EF. Most applications are at or below 180EF. Applications at temperatures above 180EF have been evaluated separately, and replacements are identified as a normal part of the ANO-1 EQ process. The applicant concludes that EQ aging analyses of Target Rock solenoid-operated valve have been projected to the end of the period of extended operation for most applications, and the remainder have scheduled replacements before they exceed the qualified life. Therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

- Westinghouse Motors - The EQ documentation shows that these motors are qualified for 78,840 hours of operation at their maximum operating temperature of 120EC (248EF). During plant operation, the motors do not run (i.e., they only run during surveillance). Between the surveillance and a conservative 1-year post-accident operating time (8,760 hours), a conservative total run-time of 18,000 hours is estimated. The applicant concludes that this is significantly less than the 78,840 hours for which these motors are qualified, and would conservatively consider them qualified (accounting for runtime and non runtime) through the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

On the basis of the review for the thermal and radiation summaries for the electrical equipment discussed above, the review of EQ calculations, and the responses to the NRC's RAI, the NRC staff found that the applicant has demonstrated that the analyses have been projected to the end of the period of extended operation consistent with 10 CFR 54.21(c)(1)(ii)

TLAA Demonstration for Option 10 CFR 54.21(c)(1)(iii)

For the following list of electrical equipment identified in Section 4.4 of the LRA, the applicant uses 10 CFR 54.21(c)(1)(iii) in its TLAA evaluation to demonstrate that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation:

- 4.4.5 ASCO Solenoid Valves, Outside Reactor Building (some applications)
- 4.4.6 ASCO Solenoid Valves, Inside Reactor Building
- 4.4.7 Boston Insulated Wire, Instrumentation, Control, and Power Cable
- 4.4.20 ITT/General Controls Electro-Hydraulic Actuators
- 4.4.21 Limitorque Motor-Operated Valve Actuators; Alternating Current/Inside Reactor Building (some applications)
- 4.4.24 NAMCO EA-170 Limit Switches
- 4.4.25 NAMCO EA-180 Limit Switches
- 4.4.26 NAMCO EA-740 Limit Switches with NAMCO Connectors
- 4.4.31 Okonite 600V Power Cable with Okonite Insulation and an Okolon Jacket (some applications)
- 4.4.32 Okonite 600V Power Cable with FMR Insulation (some applications)
- 4.4.33 Okonite T-95 and No. 35 Splicing Tapes (some applications)
- 4.4.38 Rockbestos Coaxial Cable (some applications)
- 4.4.40 Rockbestos Firezone R Silicone Rubber High-Temperature Cable (some applications)
- 4.4.42 Rosemount Model 1153 Series D Pressure Transmitters
- 4.4.43 Rosemount Model 1154 Pressure Transmitters
- 4.4.46 Target Rock Solenoid-Operated Valves (Reports 2375 and 3996)
- 4.4.49 TEC Valve Flow Monitoring System (some subcomponents)
- 4.4.50 TEC Reactor Vessel Level Monitoring System (some subcomponents)
- 4.4.53 Westinghouse Hydrogen Recombiners
- 4.4.54 Westinghouse Motors, Model ABDP
- 4.4.56 Babcock & Wilcox Core Exit Thermocouple (except the pin half connector with the mineral-insulated cable)
- 4.4.57 Gamma Metrics Neutron Detectors and Cable Assemblies (except organic cable)
- 4.4.64 EGS Quick Disconnect Electrical Connectors (connector o-rings)
- 4.4.66 Valcor Model V526-5683 Solenoid Operated Valve

The NAMCO EA-740 Limit Switches with NAMCO connectors are replacement components for equipment removed from service in 1986. These replacements have a qualified life of 47.1 years at 105EF, and their qualified life expires in 2033, which is 1 year before the end of the period of extended operation. The applicant will replace this equipment in accordance with the ANO-1 EQ program before the end of the qualified life, unless an analysis is performed to extend the qualified life.

The Gamma Metrics Neutron Detectors and Cable Assemblies are not original plant equipment; they were installed in 1984 and are qualified in accordance with Regulatory Guide 1.97. The detector assemblies and junction box o-rings have a qualified life of 40 years at 120EF, and their qualified life expires in 2024, which is 10 years before the end of the period of extended operation. The applicant will replace this equipment in accordance with the ANO-1 EQ program before the end of the qualified life, unless an analysis is performed to extend the qualified life. Mineral insulated cable extending from the detector is non-age-sensitive. The organic cable is qualified for 50 years at 180EF, and its qualified life expires in 2034, which makes these cables qualified through the period of extended operation because the application is below 180EF.

The EGS Quick Disconnect Electrical Connectors in two applications at or below 150EF are not original plant equipment; they were installed in 1995. These connectors have a qualified life of

40 years at 150EF, and their qualified life expires in 2035, therefore, they are qualified to the end of the period of extended operation. The connector o-rings are qualified for 10 years at 150EF, and their qualified life expires in 2005, which is 29 years before the end of the period of extended operation. The applicant will replace this equipment in accordance with the ANO-1 EQ program before the end of the qualified life, unless an analysis is performed to extend the qualified life.

The remaining components are original plant equipment with a qualified life of 40 years or less. In a response to the NRC staff's RAI, the applicant addressed the options of replacement, refurbishment or reanalysis for the above components. The applicant has no current plans to reanalyze and extend the qualified life of this equipment and will replace or refurbish the equipment before its qualified life expires, in accordance with the ANO-1 EQ program.

In cases where replacement is the current option, the applicant states that replacement requirements are managed through the preventive maintenance program in which a work package is automatically initiated and implemented to perform and document the replacement before exceeding the qualified life. Additionally, Westinghouse motors, Model ABDP are qualified in accordance with NRC IE Bulletin 79-01B requirements and, therefore, the components would be upgraded if they were to be replaced. If reanalysis is performed for this equipment, it would follow the same process discussed in Section 4.4.2 under "Reanalysis of the Qualified Life." The applicant has not yet decided which option will be used for this equipment.

The applicant did not identify any specific cases where refurbishment is the current option. However, in a response to the NRC's RAI, the applicant states that the replacement option discussed for several equipment types would effectively involve refurbishment. The staff found this acceptable because it is consistent with 10 CFR 50.49 and 54.21 (c)(1)(iii).

4.4.3 Conclusions

On the basis of the review described above, the NRC staff has determined that there is reasonable assurance that the applicant has evaluated the TLAAs for EQ of electrical equipment in accordance with 10 CFR 54.21(c)(1).

4.4.4 References for Section 4.4

1. 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."
2. DG-1047, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," Working Draft, April 21, 2000.
3. "Arkansas Nuclear One - Unit 1, License Renewal Application," January 31, 2000.
4. IEEE Std. 323-1974, "Qualifying Class 1E Equipment for Nuclear Power Generating Stations," 1974.
5. C. I. Grimes letter to D. Walters (NEI), "Guidance on Addressing GSI 168 for License Renewal," Project 690, dated June 2, 1998.
6. 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

7. 10 CFR 50.49, “Environmental Qualification of Electric Equipment to Safety for Nuclear Power Plants.”
8. 10 CFR 50.34(a)(1), “Contents of application; Technical Information.”
9. 10 CFR Part 100, “Reactor Site Criteria.”
10. NRC BL 79-01B, “Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors.”
11. NRC IN 89-30 and IN 89-30, Supplement 1, “High-Temperature Environments at Nuclear Power Plants.”
12. NRC IN93-39, “Radiation Beams from Power Reactor Biological Shields.”

4.5 Concrete Reactor Building Tendon Prestress

The applicant identifies loss of reactor building prestress as a TLAA in the LRA. This section of the report documents the staff's safety evaluation of the TLAA for the reactor building tendon prestress based on information presented in Section 4.5 of the LRA.

4.5.1 Technical Information in the Application

In the LRA, Section 4.5, "Concrete Reactor Building Tendon Prestress," the applicant presents the results of the TLAA for the loss of prestress in the post-tensioning system. The applicant states that the ANO-1 reactor building post-tensioning system is designed in accordance with ACI 318-63 for prestress losses caused by:

- seating anchorage
- elastic shortening of concrete
- creep of concrete
- relaxation of prestressed steel
- frictional loss due to curvature in the tendons and contact with tendon conduit

At the time of initial licensing, the initial stress from tensile loading and the appropriate prestress loss parameters were used by the applicant to calculate the design losses and the final effective prestress at the end of 40 years for the dome, vertical, and hoop tendons. In the LRA, the applicant states that this analysis is described in the ANO-1 UFSAR, Section 5.2.4.2.1, and identifies it as a TLAA requiring review for license renewal.

The applicant describes the requirements in ASME Code Section XI, Subsection IWL for the inservice inspection, repair, and replacement activities of the post-tensioning components of concrete containments, and identifies that tendon force and elongation measurements are required to evaluate the prestress forces in the post-tensioning system.

The applicant states that "ANO-1 is completing a calculation of the final effective tendon prestress based on additional information on concrete creep from existing creep tests and results of the tendon surveillance testing." The applicant indicates that the calculation will confirm projections on the relaxation of the tendons and this will show that the tendons will be acceptable for the period of extended operation.

The applicant also indicates that the ASME Section XI Inservice Inspection Program, IWL Inspections, will be adequate to manage the effects of aging on the intended function for the period of extended operation. The applicant states that the "implementation of this program dispositions this TLAA in accordance with 10 CFR 54.21(c)(1)(iii)."

4.5.2 Staff Evaluation

From the description provided in the LRA, Section 4.5, the applicant is currently performing a calculation that will confirm projections on the relaxation of the tendons and this will show that the tendons will be acceptable for the period of extended operation. This type of analysis would be consistent with a TLAA performed in accordance with 10 CFR 54.21(c)(1)(ii). However, after describing the ASME Section XI Inservice Inspection Program, IWL Inspections, the applicant states that “this program dispositions this TLAA in accordance with 10 CFR 54.21(c)(1)(iii).” Therefore, it is not clear which approach is being taken to address the TLAA for loss of tendon prestress.

A TLAA performed in accordance with 10 CFR 54.21(c)(1)(iii) must demonstrate that the effects of aging on the intended functions will be adequately managed for the period of extended operation. The information contained in LRA Section 4.5 is not sufficient for the staff to conclude that the loss of prestress will be adequately managed for the period of extended operation. If the applicant is addressing this TLAA in accordance with 10 CFR 54.21(c)(1)(iii), a description of the attributes of the AMP is needed, with special emphasis on parameters monitored, monitoring and trending, acceptance criteria, corrective actions, and operating experience.

In the LRA, Section 4.5, the applicant indicates that the analysis for prestress losses and final effective prestress at the end of 40 years is summarized in the ANO-1 UFSAR, Section 5.2.4.2.1. The NRC staff reviewed Section 5.2.4.2.1 as well as Section 5.2.4 of the ANO-1 UFSAR (Amendment 15), and could not locate the applicable information. Therefore, the staff requested that the applicant identify the section in the UFSAR that contains a description of the tendon prestress calculations corresponding to the end of 40 years. In order to understand the applicant’s approach to address the TLAA for reactor building tendon prestress, the staff requested additional technical information in a letter to the applicant dated May 5, 2000.

In its response to the NRC dated September 7, 2000, the applicant states that the TLAA for loss of tendon prestress has been addressed in accordance with 10 CFR 54.21(c)(1)(iii). However, the additional information provided in the September 7, 2000, letter did not adequately address monitoring and trending, acceptance criteria, and corrective action.

For the purpose of monitoring and trending of tendon prestress, the following parameters need to be plotted against time and projected for the period of extended operation: the predicted lower limit (PLL), the minimum required value (MRV), and the trend line representing the measured prestress forces. 10 CFR 50.55a(b)(2)(viii)(B) specifies acceptance criteria for trending of tendon forces, in addition to the criteria contained in ASME Section XI, Subsection IWL. For corrective action, the types of corrective measures that will be considered (e.g., retensioning, tendon replacement, or reanalysis) need to be described, pending receipt and staff review of this additional information. This was Open Items 4.5.2-1.

After a number of discussions with the applicant, in a letter to the NRC dated March 14, 2001, the applicant provided the following additional information:

[*Prestress Forces Monitoring and Trending*] - The tendon surveillance is conducted every five years as required by ASME, Section XI, Subsection IWL. Trending is accomplished as required by 10 CFR 50.55a(b)(2)(ix)(B). The requirements for tendon surveillance and tendon force graphs for ANO-1 are documented and controlled in the site tendon surveillance program procedures. The IWL Inspection Program provides for the random selection of tendons. The surveillance of the selected tendons includes the following activities: inspection of the tendon components, analysis of wire and grease samples, inspection of the concrete around the tendons anchorage, and determining residual tendon force.

During the surveillance, lift-off forces for the tendons are measured and evaluated for adequacy as required by IWL. Graphs for each group of tendons (hoop, dome, and vertical tendons) provide the age related expected normalized tendon force plotted on a log-normal graph. These graphs are developed based on the tendon group and the aging effects on the reactor building concrete properties, the wire properties, and the initial prestress force. The lift-off values obtained during tendon surveillance are plotted on the graphs and trended to determine if the tendon system is performing as expected.

[*Acceptance Criteria*] - The acceptance criteria are included in the site procedures for the reactor building tendon surveillance and concrete inspections. The tendon force graphs are compared with the actual forces found during the surveillance to determine if the residual prestress in the reactor building meets the minimum required prestress.

The minimum required tendon force for each of the tendon groups is 1233 kips for the hoop tendons, 1274 kips for the vertical tendons, and 1252 kips for the dome tendons. Corrective actions will be taken should the projected tendon force for a tendon group fall below the minimum required value before the next scheduled tendon surveillance.

[*Corrective Actions*] - Conditions that do not meet the acceptance criteria in the site procedures are documented in the site condition reporting system. Evaluations are performed and acceptability is determined. Corrective actions that are needed are tracked to completion through the site condition reporting system.

Should trending indicate that prestress in a tendon group may be inadequate to meet the minimum required prestress before the next scheduled tendon surveillance, action will be taken to correct the problem. This may include re-tensioning, replacing tendons, or reanalysis of the reactor building to assure adequate prestress to meet design requirements.

The staff found this additional information acceptable to resolve Open Item 4.5.2-1.

After its initial review, the staff requested that the FSAR Supplement include a summary description of the applicant’s prestress monitoring and trending activities, the acceptance criteria, and corrective actions when acceptance criteria are not met. This was FSAR Item 4.5.5 of Open Item 3.3-1.

In its revised summary description of Sections 16.2.3.6 and 16.3.4 of the FSAR Supplement, the applicant includes a description that adequately summarizes the prestress monitoring and trending activities, the acceptance criteria, and corrective actions as described above for managing prestress tendons of the ANO-1 containment in the FSAR Supplement consistent with 10 CFR 54.21(d). The staff finds the revised summary description as submitted by the applicant in a letter to the NRC dated March 14, 2001, acceptable.

4.5.3 Conclusions

On the basis of the review described above, the NRC staff has determined that there is reasonable assurance that the applicant has evaluated the TLAAs for prestress tendon force for the containment structure in accordance with 10 CFR 54.21(c)(1).

4.5.4 References for Section 4.5

1. 10 CFR Part 54, “Requirements for Renewal of Operating Licenses for Nuclear Power Plants.”
2. DG-1047, “Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants,” Working Draft, April 21, 2000.
3. “Arkansas Nuclear One – Unit 1, License Renewal Application,” January 31, 2000.

4.6 Reactor Building Liner Plate Fatigue Analysis

Fatigue associated with the reactor building liner plate has been identified in the ANO-1 LRA as a TLAA. This section of the report documents the staff's safety evaluation of the fatigue TLAA for the reactor building liner plate, based on the information presented in Section 4.6 of the LRA.

4.6.1 Technical Information in the Application

In the LRA, Section 4.6, "Reactor Building Liner Plate Fatigue Analysis," the applicant presents the results of the fatigue TLAA for the reactor building liner plate and piping penetrations. The interior surface of the reactor building is lined with welded carbon steel plate to provide an essentially leak tight barrier. The applicant states that design criteria are applied to the liner to assure that a specified leak rate is not exceeded under DBA conditions. "Reactor Building Liner Plate and Penetrations – Fatigue," is listed in the LRA, Table 4.1-1, "List of ANO-1 Time Limited Aging Analyses."

In the LRA, Section 4.6, the applicant lists the following fatigue conditions, as described in UFSAR, Section 5.2.1.4.7.3, that were considered in the CLB design of the liner plate:

- 40 thermal cycles corresponding to 40 years of annual outdoor temperature variations
- 500 thermal cycles corresponding to reactor building interior temperature variations during reactor coolant system startup and shutdown
- one thermal cycle corresponding to DBA conditions

The design analysis of the liner plate, which considers these fatigue conditions, is considered to be a TLAA for the purposes of license renewal.

The applicant evaluates each of the above fatigue conditions for continued operation for up to 60 years. For the thermal cycles corresponding to annual outdoor temperature variations, the increase in the number of cycles from 40 to 60 is considered to be insignificant. For the thermal cycles corresponding to reactor building interior temperature variations, based on ANO-1 operating experience, the projected cycles for 60 years of operation was determined to be less than the original 500 cycle design assumption. For the thermal cycles corresponding to DBA conditions, the assumed value is considered to remain valid for 60 years of operation.

The applicant also considers additional load cycles on the liner caused by the integrated leak rate tests. Due to the limited number of these tests, the additional load cycles were stated to be bounded by the 500 cycle startup and shutdown fatigue condition.

The applicant states that the design of the reactor building piping penetrations meets the general requirements of the ASME Boiler & Pressure Vessel Code Section III for thermal cycling. Also, by design, the liner plate penetrations are isolated from thermal load cycles in the piping by concentric sleeves between the pipe and liner plate.

The applicant identifies the feedwater and main steam lines as high-temperature lines penetrating the reactor building wall and the liner plate. The applicant states that the design number of thermal load cycles in these two systems is greater than the design number of heatup and cooldown cycles of the reactor coolant system. The applicant further states that the projected number of cycles for ANO-1 through 60 years of operation has been determined to be less than the original design assumptions.

The applicant concludes that the assumed fatigue conditions used in the reactor building liner plate fatigue analysis are bounding for 60 years of plant operation. Therefore, this TLAA remains valid for the period of extended operation and meets the criteria of 10 CFR 54.21(c)(1)(i).

4.6.2 Staff Evaluation

In the LRA, Section 4.6, the applicant describes four cyclic-loads that could affect the results of the original fatigue evaluation of the containment liner plate for the period of extended operation. The applicant concludes that extrapolation of these loads from 40 to 60 years would not have a significant effect on the fatigue of the containment liner plate and that the existing fatigue analysis remains valid. The staff evaluated the information contained in LRA Section 4.6 and found it to be insufficient to support this conclusion.

The staff noted that there is no discussion of containment pressure cycling due to integrated leak rate testing. Pressure cycling and thermal load cycling may have significantly different effects on the liner plate state of stress. It is not evident from the discussion in Section 4.6 of the LRA as to how this is considered for the period of extended operation. Also, there is no definition of the projected number of these pressure cycles through the period of extended operation. To complete the review of fatigue for the liner plate, additional information on cyclic loading due to both pressure and temperature was requested in a letter to the applicant dated May 5, 2000.

In the LRA, Section 4.6, the applicant states that the number of heatup and cooldown cycles assumed in the design basis (500) envelopes the number of such cycles projected through the extended period of operation. In a letter to the applicant dated May 5, 2000, the staff requested justification for this statement.

In the LRA, Section 4.6, the applicant does not provide any information on the actual pressure and temperature cycles which are included in the calculation of cumulative fatigue usage factors for any of the penetrations through the liner plate. To complete its review, the staff requested a definition of the events, the number of occurrences assumed for design, and the projected number of occurrences through the period of extended operation for each penetration subjected to cycling loading. This information was requested of the applicant in the May 5, 2000, letter.

On the basis of the information provided in the LRA, Section 4.6, the main steam and feedwater line penetrations appear to be subject to the greatest number of thermal load cycles. There is no discussion of the effects of pressure cycling in these lines, which may also induce cyclic stresses in the penetrations. The UFSAR figure which shows the details of these penetrations was reviewed. The evaluation boundary between the liner plate penetration and the piping was not obvious and needed to be defined. This information was requested of the applicant in the May 5, 2000, letter.

In its September 7, 2000, letter to the NRC, the applicant submits supplementary information on cycle loading due to pressure and temperature. The applicant's response essentially restated what is already contained in Section 4.6 of the LRA. During a telephone conference with the applicant on October 13, 2000, the staff requested additional clarification on how past operating experience justifies the conservatism of the design-basis heatup-cooldown cycles (500) for the period of extended operation, and additional justification of their statement that pressure cycling due to integrated leak rate testing is not applicable to cumulative fatigue. In a letter to the NRC dated November 2, 2000, the applicant addresses these questions. The applicant states that within the last ten years, ANO-1 has experienced an average of approximately one heatup and cooldown per year. Thus, assuming three heatup/cooldown cycles each year through the period of extended operation is conservative, and results in a total number of cycles well below the design-basis number. The staff found the applicant's assessment acceptable.

With regard to pressure cycling, the applicant states that fatigue due to integrated leak rate testing pressure cycling loads are implicitly accounted for in the fatigue analysis by the bounding number of thermal cycles specified for fatigue evaluation. The staff notes that although pressure cycling and thermal load cycling would produce different states of stress in the liner plate, the number of cycles associated with leak rate testing is small and would have a minimal contribution to the fatigue usage factor through the period of extended operation. Since the analysis was based on a conservative number of thermal cycles, the staff's concerns were resolved.

In the same response, the applicant states that the applicable TLAAs are limited to the main steam lines and main feedwater lines mechanical penetrations. The loading conditions for these penetrations are the same as those defined in the ANO-1 UFSAR for the liner plate, namely, 500 thermal cycles of RCS startup and shutdown. This number of cycles bounds the projected number of heatup and cooldown cycles for the RCS, and is therefore acceptable. The staff found this response acceptable.

The applicant's response concerning the evaluation boundary for the main steam and feedwater line penetrations states that the evaluation boundary for mechanical penetrations, which includes the main steam and feedwater line penetrations, consists of the penetration assembly, and the weld to the process piping, but does not include the process piping within the penetrations. The staff found this response acceptable.

In a subsequent telephone conference with the applicant on October 19, 2000, concerning details of the penetration fatigue analyses, the applicant described the original fatigue analysis, which considered both through wall thermal gradients in the penetration nozzles and piping expansion loads, induced by heatup and cooldown cycling. This analysis was similar to other penetration fatigue analyses previously found acceptable by the staff. The number of heatup and cooldown cycles was shown to envelop the number of cycles projected through the extended period of operation previously discussed. Therefore, the original fatigue evaluation is considered valid for the period of extended operation. On the basis of this information, the staff considers this concern resolved.

4.6.3 Conclusions

On the basis of the review described above, the staff finds that the applicant has demonstrated that, pursuant of 10 CFR 54.21(c)(1)(i), the existing fatigue TLAA for the containment liner plate and piping penetrations remains valid for the period of extended operation.

4.6.4 References for Section 4.6

1. 10 CFR Part 54, “Requirements for Renewal of Operating Licenses for Nuclear Power Plants.”
1. DG-1047, “Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants,” Working Draft, April 21, 2000.
2. “Arkansas Nuclear One - Unit 1, License Renewal Application,” January 31, 2000.

4.7 Aging of Boraflex in Spent Fuel Pool Racks

Aging of Boraflex in the spent fuel pool racks plate has been identified in the LRA as a TLAA. This section of the report documents the staff's safety evaluation of the TLAA for aging of Boraflex, based on the information presented in Section 4.7 of the LRA.

4.7.1 Technical Information in the Application

In the LRA, Section 4.7, the applicant describes the TLAA for the degradation of Boraflex, which is currently used in the ANO-1 Region I spent fuel storage racks as a neutron absorber. The applicant states that the potential stressors for the Boraflex in the pool include the chemical environment of borated water and gamma radiation, which changes the material characteristics of the base polymer.

The applicant references the following NRC Information Notices (IN) and Generic Letter (GL) that identified the concern of aging of Boraflex neutron-absorbing material:

- IN 87-43, "Gaps in Neutron-Absorbing Material in High-Density Spent Fuel Storage Racks"
- IN 93-70, "Degradation of Boraflex Neutron Absorber Coupons"
- IN 95-38, "Degradation of Boraflex Neutron Absorber in Spent Fuel Storage Racks"
- GL 96-04, "Boraflex Degradation in Spent Fuel Pool Storage Racks"

In the response to Generic Letter 96-04, the applicant commits to continue monitoring and performing analyses of the Boraflex degradation at ANO-1. In the LRA, Section 4.7, the applicant states that it will continue the existing coupon monitoring program as required into the period of extended operation. The applicant also commits to the continued monitoring of spent fuel pool silica levels, and performing silica evaluations. These evaluations are based on the EPRI RACKLIFE system or its equivalent. Projected Boraflex performance will be assessed to confirm that a 5-percent subcriticality margin will be maintained as required.

The applicant states that degradation of Boraflex is treated as a TLAA at ANO-1 because it meets the six criteria of 10 CFR 54.3. In addition, the analysis meets 10 CFR 54.21(c)(1)(ii) and the sampling actions meet 10 CFR 54.21(c)(1)(iii). On the basis of these activities, the applicant concludes that the TLAA is valid for the period of extended operation.

4.7.2 Staff Evaluation

In order to determine whether the TLAA meets the requirements of 10 CFR 54.21(c), the staff reviewed the applicant's response to GL 96-04.

In the response to GL 96-04, the applicant states that long-term and accelerated test location coupon specimens are periodically removed and inspected. It is stated that "the inspections provide an indication of the general condition of the Boraflex, including gross or unusual degradation." Long-term coupons are tested approximately every five years, while accelerated coupons are tested after each refueling. The applicant also stated that it will continue to monitor spent fuel pool silica levels, perform silica evaluations based on the EPRI RACKLIFE system or its equivalent, and assess projected Boraflex performance to confirm that the 5-percent subcriticality margin will be maintained through the next evaluation period. The applicant has committed to continuing these assessments each refueling cycle prior to fuel receipt.

On the basis of its review of the information in Section 4.7 of the LRA, and the applicant’s response to GL 96-04, the NRC staff cannot conclude that the effects of aging will be adequately managed for the period of extended operation. It was unclear as to the frequency of inspection and testing will be during the period of extended operation and whether there will be sufficient long-term and accelerated coupons to continue the existing monitoring program. In addition, it was unclear as to the physical condition of the coupons that were observed during inspection, and whether gap formation, and a decrease in boron density will be monitored. The applicant also did not describe the current trending analyses that have been obtained by use of the RACKLIFE code, and whether these results demonstrate that a 5-percent subcriticality margin of the spent fuel racks will be maintained for the period of extended operation. If not, the applicant needs to describe the corrective actions that will be implemented to ensure that the 5-percent subcriticality margin will be maintained through the period of extended operation.

In order to complete the evaluation of this TLAA, the staff requested additional information in a letter to the applicant dated May 5, 2000. In a letter to the NRC dated September 7, 2000, the applicant states that since the submittal of the ANO-1 LRA, Boraflex monitoring has revealed that the Boraflex is degrading more rapidly than expected. This condition has been documented in accordance with the onsite Appendix B corrective action program, and is currently being evaluated in order to determine the appropriate action. It has been determined that the Boraflex, as incorporated in the initial spent fuel pool rack design, will not last through the current 40-year licensing term, and therefore, should no longer be considered a TLAA with respect to license renewal. The applicant is evaluating several options including a revised criticality analysis, a modification of the existing spent fuel pool racks with a different neutron absorber, or a combination thereof. The applicant plans to complete the evaluation, and identify a corrective action plan for the remainder of a 60-year operating term by the fourth quarter of 2002, and plans to submit a license amendment in accordance with 10 CFR 50.90. The applicant is scheduled to complete the ANO-1 license renewal process by January 2002. Therefore, the final resolution of this concern for the entire term of the operating license at the time of submittal will be subject to NRC review and approval in accordance with 10 CFR 50.90.

The staff disagreed with the applicant’s conclusion. Irrespective of the results of the condition monitoring, the Boraflex design appears to meet the definition of a TLAA in 10 CFR 54.3, as was originally stated in the application. While the applicant may continue to pursue various corrective actions in the future, the applicant will continue to rely on monitoring, evaluation, and design criteria to decide on the extent and timing of corrective actions so that the spent fuel pool design will maintain the structural and criticality design margins in accordance with the CLB. Therefore, the staff concluded that the applicant needed to provide the basis upon which the staff can conclude that there is reasonable assurance that the effects of aging of Boraflex will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation, in accordance with 10 CFR 54.21(c)(1). This was Open Item 4.7.2-1.

In a letter to the NRC dated March 14, 2001, the applicant acknowledges the analysis of Boraflex in the spent fuel storage racks as a time limited aging analysis. The applicant further states that the existing analysis is not valid through the license renewal period and cannot be acceptably projected to the end of the license renewal period (as discussed in its letter to the NRC dated September 6, 2000). The applicant also agrees to continue its boraflex monitoring program to provide reasonable assurance that the effects of aging on the intended function will be adequately managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii). As a result of crediting the boraflex monitoring program in response to this concern, the applicant also provides the following information regarding the boraflex monitoring program requested by the staff in a letter to the applicant dated May 5, 2000:

- The applicant states that the frequency of the inspection and testing will be the same for the extended period of operation as stated in its response to Generic Letter 96-04 (0CAN109605), where Entergy Operations committed to continued monitoring and analysis of Boraflex degradation at ANO-1. The applicant states that it will continue the coupon monitoring program into the period of extended operation. Although all the accelerated coupons have been used and are no longer available, the long-term coupons are tested once every 5 years. These coupons have provided indications of Boraflex degradation. Entergy Operations will continue to monitor spent fuel pool silica levels and perform silica evaluations once per cycle. These evaluations are based on the EPRI RACKLIFE system. Boraflex performance will be projected to confirm the 5 percent subcriticality margin will be maintained as required.

- There are a sufficient number of long-term coupons to continue the existing program through the period of extended operation. The portion of the program for which the accelerated coupons were designed is complete.

- ANO-1 currently has a procedure in place for examining and testing the spent fuel pool Boraflex test coupons. The coupon inspections consist of taking thickness measurements, density determinations, general visual inspection, and hardness testing. Neutron attenuation testing is also performed on the sampled Boraflex coupons. This testing more accurately determines areal Boron density.

- The minimum as-designed Boraflex dimensions and the minimum designed areal Boron 10 densities with an assumed degradation of 10 percent are used in the ANO-1 criticality analysis. The ANO-1 criticality analysis assumes all the shrinkage is on the ends, which is more conservative than gap formation assumptions for the ANO-1 rack geometry. The Boraflex panels are assumed to shrink 4.1 percent in width. Current RACKLIFE analysis indicates that there is less than 10 percent boron degradation.

  These assumptions and analytical calculations have been correlated to industry data obtained through in-situ testing of a similar rack design to ANO-1. The results from the tested racks are conservatively applied to the ANO-1 racks based upon the tested racks having been subjected to higher doses, the spent fuel pool silica levels exceeding the concentrations seen at ANO-1, and the tested racks have a higher peak panel degradation. The tested racks have also been in service longer than the ANO-1 racks.

C The results of the current Boraflex trending analysis demonstrate that the 5 percent subcriticality margin is being maintained; however, it will not be maintained for the period of extended operation. As previously discussed, this condition has been documented in accordance with the onsite Appendix B corrective action program. Corrective actions will be implemented to ensure that the 5 percent subcriticality margin will be maintained through the period of extended operation. Corrective actions may include modification of the spent fuel racks to incorporate a different neutron absorber material. Entergy Operations is committed to resolving this issue as documented in correspondence dated September 6, 2000 (1CAN090002).

The staff found this resolution to Open Item 4.7.2-1 acceptable.

After its initial review, the staff also requested that the FSAR Supplement include a summary description of the applicant's monitoring, evaluation activities, optional corrective actions, and decision criteria for the aging of Boraflex in the spent fuel pool. This was FSAR Item 4.7.3 of Open Item 3.3-1.

In its revised summary description of Section 16.3.6 of the FSAR Supplement, the applicant included a description of the monitoring, evaluation activities, optional corrective actions, and decision criteria for the aging of Boraflex in the spent fuel pool consistent with the information described above. The staff finds the revised summary description as submitted by the applicant in a letter to the NRC dated March 14, 2001, to be acceptable and, therefore, finds FSAR Item 3.3.1.2.3 of Open Item 3.3-1 resolved.

4.7.3 Conclusions

On the basis of the review described above, the staff finds that the applicant has demonstrated that there is reasonable assurance, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended function of Boraflex will be adequately managed for the period of extended operation.

4.7.4 References for Section 4.7

1. 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."
2. DG-1047, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," Working Draft, April 21, 2000.
3. "Arkansas Nuclear One – Unit 1, License Renewal Application," January 31, 2000.
4. NRC IN 87-43, "Gaps in Neutron-Absorbing Material in High-Density Spent Fuel Storage Racks."
5. NRC IN 93-70, "Degradation of Boraflex Neutron Absorber Coupons."
6. NRC IN 95-38, "Degradation of Boraflex Neutron Absorber in Spent Fuel Storage Racks."
7. NRC GL 96-04, "Boraflex Degradation in Spent Fuel Pool Storage Racks."

4.8 Other Time-Limited Aging Analyses

The TLAAs evaluated in this section of the SER include the following:

- reactor vessel underclad cracking
- reactor vessel incore instrumentation nozzle FIVs
- leak-before-break in RCS piping
- reactor coolant pump motor flywheels

4.8.1 Reactor Vessel Underclad Cracking

In the LRA, Section 4.8, the applicant discusses the TLAA for the intergranular separations.

4.8.1.1 Technical Information in the Application

In this TLAA, the applicant addresses the issue of intergranular separations (underclad cracking) in low-alloy steel heat-affected zones under austenitic stainless steel weld cladding in SA 508, Class 2 reactor vessel forgings with coarse grain structures. The applicant references the topical report BAW-10013, "Study of Intergranular Separations in Low-Alloy Steel Heat-Affected Zones Under Austenitic Stainless Steel Cladding," as containing a fracture mechanics calculation that demonstrates that the critical crack size required to initiate fast fracture is several orders of magnitude greater than the assumed maximum flaw size plus predicted flaw growth due to fatigue cycling. The analysis concluded that intergranular separation in B&W vessels would not lead to failure. This conclusion, according to the applicant, was accepted by the Atomic Energy Commission. To cover the period of extended operation, the applicant performs a calculation using current ASME Code requirements. This analysis is given in Appendix C to BAW-2251A.

4.8.1.2 Staff Evaluation

The NRC staff has evaluated the B&WOG approach to resolving underclad cracking issues in Appendix C to the final SER for BAW-2251A. The B&WOG approach includes the following conservatisms:

- a maximum crack depth of 0.165 inch reported by industry as the initial crack depth, instead of the 0.1 inch size reported for reactor pressure vessels

- a total effective crack depth of 0.353 inch (nominal cladding thickness of 0.1875 inch plus 0.165 inch crack depth in the underlying alloy steel)

- assumption that all the cracks are surface cracks

- use of the fatigue crack growth rate for surface flaws in a water reactor environment

- use of a safety factor of 17 percent more than that specified by the ASME Code for Levels A and B loading, and 72 percent more for Levels C and D loading

The maximum crack growth and applied stress intensity factor for the normal and upset conditions were found to occur near the nozzle belt region. The maximum crack growth considering all of the normal and upset transients for 48 EFPY was determined by B&WOG to be 0.180 inch. This gives a final crack depth of 0.533 inches (0.353 inch plus 0.180 inch). The maximum applied stress intensity factor for the normal and upset conditions results in a fracture toughness margin of 3.6, which is greater than the ASME IWB-3612 acceptance criterion of 3.16. The maximum applied stress intensity factor for the emergency and faulted conditions was shown by B&WOG to be in the closure head to head flange region and the fracture toughness margin was found to be 2.24, which is greater than the ASME IWB-3612 acceptance criterion of 1.41.

The NRC staff found that, consistent with the final SER for BAW-2251A, the B&WOG underclad cracking flaw analysis, performed in accordance with 10 CFR 54.21(c), is acceptable for the period of extended operation.

4.8.1.3 Conclusions

On the basis of its review, the staff concludes that the B&WOG's underclad cracking flaw analysis satisfies the requirements of 10 CFR 54.21 (c).

4.8.2 Reactor Vessel Incore Instrumentation Nozzle – FIV Endurance Limit

In the LRA, Section 4.8.2, the applicant presents a description of a TLAA for flow-induced fatigue in the reactor vessel incore instrumentation nozzles.

4.8.2.1 Technical Information in the Application

The applicant discusses the evaluation of FIV of the reactor vessel incore instrumentation nozzles. For the current licensing period, BAW-10051, "Flow Induced Vibration Endurance Limit Assumptions," contains an analysis of the stresses in the reactor vessel incore instrumentation nozzles. The topical report compares the stresses to the fatigue endurance limits. However, the analysis does not cover the period of extended operation.

4.8.2.2 Staff Evaluation

In the initial calculation, given in BAW-10051, the endurance limits were based on 10^{12} cycles over 40 years. The applicant states that the fatigue cycles were extended to 60 years in a new calculation. The component stress values were acceptable, when compared to the newly calculated endurance limits for 60 years.

In a letter to the NRC dated September 6, 2000, the applicant describes the new analysis that covers the period of extended operation. In the analysis, the number of fatigue cycles over the 60-year operating period was conservatively assumed to be 10^{13}, which is an order of magnitude greater than that estimated for the current 40 year licensing period. The fatigue endurance limit was assumed to be reduced by 4 percent for each decade of cycles. This assumption is consistent with that given in BAW-10051, Appendix A. Thus, the applicable fatigue endurance limit from ASME III, Division 1, which extends only to 10^{11} cycles, is reduced by a factor of $(0.96)^2$ at 10^{13} cycles. A correction factor of 0.9 is also applied to the ASME fatigue curve. The fatigue curve is for room temperature, and the correction factor is added to account for the reduction in Young's modulus at the nozzle operating temperature. The applicable endurance limit at 10^{13} cycles becomes 13,700 psi. A similar calculation was carried out for high strength bolting and gave an endurance limit of 9,100 psi. From Table 5.1 of BAW-10051, the alternating stresses for the incore instrumentation nozzles and bolting were shown to be at least 19 percent lower than the calculated endurance limits at 10^{13} cycles. This indicates that fatigue failure is highly unlikely. On the basis of the methodology used and the conservative results from implementing this methodology, the staff found the applicant's TLAA evaluation of the reactor vessel incore instrumentation nozzles acceptable for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.8.2.3 Conclusions

On the basis of its review, the staff concludes that the applicant's FIV analysis satisfies the requirements of 10 CFR 54.21(c)(1)(ii).

4.8.3 Leak-Before-Break

The applicant's leak-before-break (LBB) analysis is given in Section 4.8.3 of the application.

4.8.3.1 Technical Information in the Application

The application describes the LBB approach for the RCS main coolant loop piping.
It is based on the analysis given in topical report BAW-1847, Revision 1, "The B&W Owners Group Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping of B&W Designed NSSS." This report provides the technical basis for evaluating postulated flaw growth in the main RCS piping under normal plus faulted conditions and was approved by the staff for the current licensing period.

The LBB analyses described in the LRA include the following items:

- fatigue flaw growth
- thermal aging effects in cast austenitic stainless steel (CASS) reactor coolant pump (RCP) inlet and discharge nozzles

For fatigue flaw growth, the applicant uses an analysis to show that the number of fatigue cycles that were originally defined for the current period of operation will not be exceeded during the period of extended operation. In the case of thermally-induced embrittlement of CASS RCP inlet and discharge nozzles, the applicant calculates the maximum anticipated crack lengths in the nozzles, and estimated the margin of safety. The two analyses are reviewed in the next section of this SER.

4.8.3.2 Staff Evaluation

Fatigue Flaw Growth

The LBB analysis, described in BAW-1847, Revision 1, was carried out in accordance with guidance given in NUREG 1061, Volume 3, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Potential for Pipe Breaks." Specifically, a surface flaw is postulated at selected locations of the piping system (i.e., highest stress coincident with the lower bound of materials properties for base metal, welds, and safe-ends). The analysis seeks to demonstrate that such a surface flaw will propagate through the wall and cause an identifiable leak, before it can propagate circumferentially around the pipe to such an extent that it could cause a double-ended pipe rupture under faulted conditions.

In the LRA, Section 4.3.5, the applicant describes its program to monitor the number of transients for the current licensing period. The applicant intends to undertake corrective actions if the number of transient cycles exceeds the allowable design limit. Currently, the design limit, as given in Section 4.3, Table 4-3, of BAW-1847, Revision 1, is 240 heatup and cooldown cycles and 22 cycles of safe shutdown earthquake. The applicant states that the flaw growth evaluation in BAW-1847, Revision 1, is applicable to 60 years of operation since it has not revised the transients defined in the RCS design specifications for the period of extended operation.

The staff found the TLAA acceptable since it covers the period of extended operation, in accordance with the requirements of 10 CFR 54(c)(1)(i).

Thermal Aging of CASS Reactor Coolant Pump Suction and Discharge Nozzles

The applicant identifies thermal aging of CASS components as a potential problem with respect to maintenance of sufficient piping material fracture toughness. The applicant references the review in BAW-1847, Revision 1, and NUREG/CR-6177, "Assessment of Thermal Embrittlement of Cast Stainless Steels." The latter report showed that prolonged heating of CASS to reactor coolant temperatures could lead to a loss in fracture toughness.

In a letter to the applicant dated May 5, 2000, the staff requested assurance that the d-ferrite content of the CASS RCP nozzles were within the bounds of applicability of data in NUREG/CR-6177, which gives guidance on LBB analyses for CASS components. In its response to the NRC dated September 6, 2000, the applicant states that the d-ferrite content of the CASS RCP nozzles was 14.2 percent, and that this was within the bounds reported in NUREG/CR-6177. A review of NUREG/CR-6177 confirmed that the normal ferrite content of domestic CASS is #15 percent.

The applicant provides a flaw stability analysis in Section 4.8.3 of the LRA to show the acceptability of the LBB concept for the RCS main coolant piping over the period of extended operation. The analysis was performed on suction and discharge nozzles of the RCP casings since the applicant states that they were susceptible to loss of fracture toughness due to thermal aging. In the analysis, the lower bound CASS fracture toughness properties were used.

In the applicant's analysis, bounding 10 gpm crack sizes (margin of 10 on the plant's leak detection capability) for the RCP suction and discharge nozzles were determined using a method consistent with that reported in BAW-1847, Revision 1. In the revised analysis, the applied loadings were considered, using the absolute sum load combination method. The leakage crack length (twice the leakage flaw size) for the suction nozzle was determined to be 8.62 inches, and for the discharge nozzle it was found to be 8.86 inches. In addition, a crack extension value of 0.6 inches was considered in the flaw stability analysis. The flaw stability analysis was performed for the suction and discharge nozzles for the reactor coolant pump. The discharge nozzle was found to be bounding. The critical crack length was found to be 21.6 inches. Therefore, the margin was determined to be 2.4.

The applicant describes an additional calculation in the LRA for crack propagation in thermally-aged SMAW material connecting the stainless steel transition pieces to the RCP nozzles since the structure of the welds is similar to CASS. Using data from the literature, it was shown that the LBB analyses for aged CASS material bounds that for the SMAW material.

On the basis of the large margin of safety on the calculated critical crack length for CASS RCS components, the staff found the above analysis, conducted in accordance with 10 CFR 54.21(c)(1)(i), to be acceptable.

4.8.3.3 Conclusions

The staff accepts the TLAA regarding LBB, fatigue flaw growth, and thermal aging of the CASS RCP inlet and discharge nozzles as demonstration that the appropriate bases for LBB will be maintained through the extended period of operation for ANO-1. On the basis of its review, the staff finds that the applicant's LBB analysis satisfies the requirements of 10 CFR 54.21(c) (1)(i).

4.8.4 Reactor Coolant Pump Motor Flywheels

The applicant evaluates the TLAA relating to fatigue of the reactor coolant pump (RCP) motor flywheel in Section 4.8.4 of the LRA.

4.8.4.1 Technical Information in the Application

The RCP motors are large, vertical, squirrel cage motors. The motors have flywheels to increase rotational inertia thus prolonging pump coastdown, and ensuring a more gradual loss of main coolant flow to the core in the event pump power is lost. The aging mechanism of concern is fatigue crack growth of pre-existing cracks in the flywheel bore keyway from stresses due to starting the motor. Therefore, this topic is considered a TLAA for license renewal. The applicant addresses the TLAA by projecting the existing analysis to the end of the period of extended operation.

4.8.4.2 Staff Evaluation

The applicant references a crack growth evaluation, which shows that crack sizes remain acceptable for 4,000 startup/shutdown cycles. This number of cycles is reported in the LRA to exceed the number of design cycles by a factor of 8. The applicant identifies that the RCP pump starts normally occur once every 200 to 300 days, on average; this conservative design is considered valid for the period of extended operation. On this basis, the NRC staff concludes that the applicant has provided an acceptable basis for extending the TLAA for the RCP flywheel to cover the period of extended operation and, therefore, meets the requirements of 10 CFR 54.21(c)(1)(ii).

4.8.4.3 Conclusions

The staff concludes that the applicant has provided an acceptable TLAA involving components of the RCP flywheel as defined in 10 CFR 54.3 and meets 10 CFR 54.21(c)(1)(ii).

4.8.5 References for Section 4.8

1. 10 CFR Part 54, “Requirements for Renewal of Operating Licenses for Nuclear Power Plants.”
2. Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants, Draft Regulatory Guide DG-1047.
3. “Arkansas Nuclear One – Unit 1, License Renewal Application,” January 31, 2000.
4. BAW-10051, “Design of Reactor Internals and Incore Instrument Nozzles for Flow-Induced Vibrations,”September, 1972.
5. BAW-10013, “Study of Intergranular Separations in Low-Alloy Steel Heat-Affected Zones Under Austenitic Stainless Steel Weld Cladding,” B&W Nuclear Power Generation, December 1971.
6. “ASME Boiler and Pressure Vessel Code,” American Society of Mechanical Engineers.
7. BAW-2251A, “Demonstration of the Management of Aging Effects for the Reactor Vessel,” The B&W Owners Group Generic License Renewal Program, June 1996.
8. ASME Boiler and Pressure Vessel Code, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” American Society of Mechanical Engineers.
9. BAW-1847, “The B&W Owners Group Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping of B&W Designed NSSS,” Revision 1, B&W Owners Group, September 1985.
10. BAW-2243A, “Demonstration of the Management of Aging Effects for the Reactor Coolant System Piping,” The B&W Owners Group Generic License Renewal Program, June 1996.

5 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

During the 482th meeting of the Advisory Committee on Reactor Safeguards (ACRS) on March 1, 2001, the ACRS reviewed the NRC staff's safety evaluation report (SER) related to the license renewal application (LRA) for Arkansas Nuclear One, Unit 1 (ANO-1). The ACRS Subcommittee on Plant License Renewal initially reviewed the SER prior to its meeting with the NRC staff and the applicant on February 22, 2001, and presented its findings during the March 1, 2001 ACRS meeting. Because of the small number and subject matter of the open items, the subcommittee recommended not issuing an ACRS interim letter on its review of the ANO-1 license renewal SER with open items. The subcommittee also recommended that the final SER be presented directly to the ACRS without a separate, second subcommittee meeting to review the resolution of the six open items. The staff submitted the final SER related to the LRA for ANO-1 with the resolution to the open items on April dd, 2001. The staff briefed the ACRS full-committee on May 10, 2001, regarding the resolution of open items.

During the 487th meeting on May 10, 2001, the ACRS will complete its review of the ANO-1 LRA, and will document its findings in letter that will be describe within upon completion of the ACRS review.

THIS PAGE IS INTENTIONALLY LEFT BLANK

6 CONCLUSIONS

In accordance with Federal regulations under Title 10 of the *Code of Federal Regulations*, Part 51 and Part 54, and the NRC draft "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," dated September 1997, the staff has completed its review of the Arkansas Nuclear One, Unit 1(ANO-1) license renewal application and supporting documentation, and has documented its finding in this safety evaluation report (SER). The standards for issuance of a renewed license are set forth in 10 CFR 54.29.

In the SER issued on January 10, 2001, regarding the review of the ANO-1 license renewal application, the staff identified six open items. Those open items have been resolved, as discussed in this SER. On the basis of its evaluation of the ANO-1 license renewal application and the applicant's response to the open items as discussed within this SER, the staff concludes the following:

1. actions have been identified and have been or will be taken with respect to managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require an aging management review under 10 CFR 54.21(a)(1)

2. actions have been identified and have been or will be taken with respect to time-limited aging analyses that have been identified to require review under 10 CFR 54.21(c)

Accordingly, the staff finds that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis for ANO-1. The staff notes that the results of the staff's environmental review are documented in the final plant-specific supplement to the Generic Environmental Impact Statement.

THIS PAGE IS INTENTIONALLY LEFT BLANK

APPENDIX A
CHRONOLOGY

This appendix contains a chronological listing of routine licensing correspondence between the U.S. Nuclear Regulatory Commission (NRC) staff and Entergy Operations, Inc., and other correspondence regarding the NRC staff's review of the Arkansas Nuclear One, Unit 1 (under Docket Nos. 50-313) application for license renewal.

.January 31, 2000	In a letter (signed by J. Vandergrift) Entergy submitted its License Renewal Application (LRA) for Arkansas Nuclear One, Unit 1 (ANO-1) as well as a copy of the boundary drawings to the NRC.
February 4, 2000	In a letter (signed by C. Grimes) NRC informed Entergy that the NRC received ANO-1 LRA on February 1, 2000, and that Mr. Robert J. Prato was appointed as the project manager for ANO-1 License Renewal Application.
February 14, 2000	In a letter (signed by J. Vandergrift) Entergy informed NRC that as of February 12, 2000, Mr. Craig G. Anderson replaced Mr. Randy Hutchinson as Vice President, Operations, at ANO-1.
February 28, 2000	In a letter (signed by D. Mathews) NRC informed Entergy that the NRC staff has determined that Entergy has submitted sufficient information that is complete and acceptable for docketing.
March 7, 2000	In a letter (signed by R. Prato) NRC informed Entergy of the schedule for the conduct of review of the ANO-1 LRA.
April 11, 2000	In a letter (signed by J. Vandergrift) Entergy submitted corrections to the LRA Environmental Report (ER) and also provided information on severe accident mitigation alternatives (SAMA).
April 12, 2000	In a letter (signed by R. Prato) NRC issued a public meeting notice to the stakeholders and the public and informed that a meeting to be held on May 17, 2000, with Entergy to discuss the status of review of license renewal application for ANO-1.
April 12, 2000	In a letter (signed by T. Kenyon) NRC requested Entergy for additional information (RAI) regarding severe accident mitigation alternatives for ANO-1.
April 17, 2000	In a letter (signed by R. Prato) NRC requested Entergy to provide additional information (RAI) on Sections 2.5, 3.5, and portions of 4.4 of the ANO-1 LRA.
April 25, 2000	In a letter (signed by R. Prato) NRC requested Entergy to provide additional information (RAI) on Sections 3.3.2.4, 3.3.2.5, 3.3.2.6, 3.7, 4.1, 4.2, 4.3, 4.4, 4.8.1, 4.8.2, and 4.8.3 of the ANO-1 LRA.

May 2, 2000	In a letter (signed by R. Prato) NRC requested Entergy to provide additional information (RAI) on Sections 2.3.2.4, 2.3.2.5, 2.3.2.6, 2.3.2.7, 2.3.2.8, 2.3.3.11, 2.3.3.12, 2.3.3.13, 2.4, and 3.7 of the ANO-1 LRA.
May 5, 2000	In a letter (signed by R. Prato) NRC requested Entergy to provide additional information (RAI) on Sections 2.2, 2.3.3.1, 2.3.3.2, 2.3.3.3, 2.3.3.4, 2.3.3.5, 2.3.3.6, 2.3.3.7, 2.3.3.8, 2.3.3.9, 2.3.3.10, 2.3.4, 3.3.2.2, 3.3.2.3, 3.3.2.7, 3.3.2.8, 4.5, 4.6, and 4.7 of the ANO-1 LRA.
June 1, 2000	In a letter (signed by R. Prato) NRC requested Entergy to provide additional information (RAI) on Sections 2.3.1, 3.3, and 3.6 of the ANO-1 LRA.
June 5, 2000	In a letter (signed by T. Kenyon) NRC requested Entergy to provide additional information (RAI) regarding its January 2000 Environmental Report for ANO-1.
June 6, 2000	In a letter (signed by J. Vandergrift) Entergy provided its response to the NRC RAIs on Section 4.4 of the ANO-1 LRA requested on April 17, 2000, and April 25, 2000.
June 9, 2000	In a letter (signed by R. Prato) NRC requested Entergy to provide additional information (RAI) on Sections 3.3.1.1, 3.3.1.2, and 3.3.4 of the ANO-1 LRA.
June 23, 2000	In a letter (signed by R. Prato) NRC requested Entergy to provide additional information (RAI) on Sections 2.3.1, 3.3, and 3.6 of the ANO-1 LRA.
July 6, 2000	Response to NRC staff requests for additional information regarding equipment qualification.
July 31, 2000	In a letter (signed by J. Vandergrift) Entergy provided its response to the NRC RAIs on Sections 2.5 and 3.7 of the ANO-1 LRA requested on April 17, 2000, April 25, 2000, and May 2, 2000.
August 24, 2000	In a letter (signed by J. Vandergrift) Entergy provided its response to the NRC RAIs on Sections 2.3.1.5, 2.3.1.6, 3.2.4, 3.2.5, and 4.2 of the ANO-1 LRA requested on April 25, 2000, and June 1, 2000.
August 30, 2000	In a letter (signed by J. Vandergrift) Entergy provided its response to the NRC RAIs on Section 2.0 of the ANO-1 LRA requested on May 2, 2000, May 5, 2000, June 1, 2000.

September 6, 2000	In a letter (signed by J. Vandergrift) Entergy provided its response to the NRC RAIs on Sections 2.3.1.3, 2.3.1.4, 2.3.1.7, 3.2.2, 3.2.3, 3.2.6, 3.2.7, 3.2.8, 4.3, 4.7, and 4.8 of the ANO-1 LRA requested on April 25, 2000, May 5, 2000, and June 1, 2000.
September 7, 2000	In a letter (signed by J. Vandergrift) Entergy provided its response to the NRC RAIs on Sections 3.1.3, 3.6, 4.5, and 4.6 of the ANO-1 LRA requested on May 5, 2000, June 1, 2000, and June 23, 2000.
September 12, 2000	In a letter (signed by J. Vandergrift) Entergy provided its responses to the NRC RAIs on Section 3.0 of the ANO-1 LRA requested on April 17, 2000, May 5, 2000, June 1, 2000, and June 9, 2000.
October 3, 2000	In a letter (signed by J. Vandergrift) Entergy provided additional clarifications and/or corrections to its responses to the NRC RAI #s 3.3.2.2.2.2-1(d), 4.2.3-3, 2.3.1-4, 2.4-5, 3.3.3.1-7, 3.3.3.1-2(b), 3.3.3.1-6(a), and 3.3.1.4.4-2. These RAIs were requested on August 24, 2000, August 30, 2000, September 6, 2000, and September 12, 2000.
October 11, 2000	In a letter (signed by R. Prato) NRC provided the summary of conference calls between the NRC staff and members of ANO-1in order to obtain clarifying information for the Entergy's responses to the staff's RAIs. These conference calls were conducted on September 13, 2000, September 18, 2000, September 20, 2000, and October 3, 2000.
October 20, 2000	In a letter (signed by R. Prato) NRC provided the summary of conference calls between the NRC staff and members of ANO-1in order to obtain clarifying information for the Entergy's responses to the staff's RAIs. These conference calls were conducted on October 11, 2000, October 12, 2000, and October 13, 2000.
November 2, 2000	In a letter (signed by J. Vandergrift) Entergy provided additional clarifications and/or corrections to its responses to the NRC RAIs from Sections 2.3.3.2, 2.3.3.6, 3.3.2.2.2.2, 3.3.2.2.2, 3.3.2.3.2.2, 3.3.2.6.2.2, 3.3.4.3.1, 3.3.4.3.2, 3.3.4.3.2.5, 3.3.4.3.1, 3.3.4.3.1.8, 3.3.4.3.2, 3.3.4.3.2.2, 3.3.4.3.2.3, 3.3.4.3.2.9, 3.3.4.3.2.10, 2.3.3.9, 3.3.5, 3.3.6, 4.5, and 4.6.
December 4, 2000	Telecommunication for clarification of information relating to the ANO-1 LRA and Site summary visit.
December 20, 2000	Clarification to request for additional information relating to the ANO-1 LRA.

January 10, 2001	The NRC staff issued the license renewal safety evaluation report with open items for ANO-1
February 21, 2001	The NRC Staff issue notice of forthcoming public meeting with Entergy Operations, Inc., on license renewal fire protection scoping for Arkansas Nuclear One, Unit 1
February 22, 2001	ACRS Subcommittee for license renewal meet with the NRC staff and Entergy Operation, Inc., to discuss the ANO-1 license renewal applicant and the staff’s safety evalution with open items
March 1, 2001	The ACRS subcommittee for license renewal met with the ACRS full committee to summarize its findings related to the ANO-1 license renewal application as was documented in the January 10, 2001 SER with open items.
March 14, 2001	In a letter (signed by J. Vandergrift) Entergy provided its responses to the license renewal Safety Evaluation Report Open Items
April 2, 2001	In a letter to the applicant, the staff sent a revised schedule reducing the duration of the ANO-1 license renewal application review from 25-months to 18-months
April 9, 2001	The NRC staff issued the final license renewal safety evaluation report for ANO-1
April dd, 2001	The final EIS regarding the license renewal of ANO-1 was issued
May dd, 2001	Regional Administrator, Region IV, submits his recommendation regarding the license renewal of ANO-1
May 10, 2001	The staff met with the ACRS full committee to summarize its findings related to the ANO-1 license renewal application as was documented in the April, dd, 2001, SER
May dd, 2001	The ACRS documents its findings regarding the ANO-1 LRA and submits its recommendation to the Commission

APPENDIX B
REFERENCE DOCUMENTS

This appendix contains a listing of references used in the preparation of the Safety Evaluation Report prepared during the review of the license renewal application for Arkansas Nuclear One Unit 1 under Docket Numbers 50-313

American Concrete Institute (ACI)

ACI 301, "Specifications for Structural Concrete for Buildings."

ACI 318-63, "Building Code Requirements for Reinforced Concrete."

American Society of Mechanical Engineers (ASME)

ASME Boiler and Pressure Vessel Code.

ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Power Plant Components through Summer 1979.

ASME Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components.

ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, 1995 Edition through 1996 Addenda.

American Society for Testing Materials

ASTM A307, "Standard Specification for Carbon Steel Bolts and Steels, 60,000 psi Tensile Strength."

ASTM A325, "Standard Specification for Structural Bolts, Steel, Heat-Treated, 120 ksi and 105 ksi Minimum Tensile Strength."

ASTM A490, "Standard Specification for Heat-Treated Steel Structural Bolts, 150ksi Minimum Tensile Strength."

ASTM D975-1981, "Standard Specification for Diesel Fuel Oils."

Babcock and Wilcox

BAW-1347, "The B&W Owners Group Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping of B&W Designed NSSS," Revision 1, B&WOG, September 1985.

BAW-2166, "Response to Generic Letter 92-01," June 1992.

BAW-2222, "Response to Closure Letters to Generic Letter 92-01, Revision 1," June 1994.

BAW-2243A, “Demonstration of the Management of Aging Effects for the Reactor Coolant System Piping,” B&WOG Generic License Renewal Program, June 1996.

BAW-2244A, “Demonstration of the Management of Aging Effects for the Pressurizer,” B&WOG Generic License Renewal Program, December 1997.

BAW-2248A, “Demonstration of the Management of Aging Effects for the Reactor Vessel Internals,” B&WOG Generic License Renewal Program, December 1999.

BAW-2251A, “Demonstration of the Management of Aging Effects for the Reactor Vessel,” B&WOG Generic License Renewal Program, June 1996.

BAW-2325, “Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity - Generic Letter 92-01, Revision 1, Supplement 1,” B&WOG, May 1998.

BAW-2325, “Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity,” Revision 1, B&WOG, January 1999.

BAW-10013, “Study of Intergranular Separations in Low-Alloy Steel Heat-Affected Zones Under Austenitic Stainless Steel Weld Cladding,” B&W Nuclear Power Generation, December 1971.

BAW-10051, “Design of Reactor Internals and Incore Instrument Nozzles for Flow Induce Vibrations,” September 1972.

Entergy Operations, Inc. (Entergy)

Correspondence

Letter from C. Randy Hutchinson (Entergy) to NRC “Response to NRC Request Under 10 CFR 50.54(f) Regarding Adequacy and Availability of Design Bases Information,” February 7, 1997.

Letter from Jimmy D. Vandergrift (Entergy) to Document Control Desk (NRC) “Arkansas Nuclear One - Unit 1 Additional Information in Support of Risk-Informed Inservice Inspection Pilot Application.”

1CAN079801, Letter from D. James (ANO) to Document Control Desk (NRC) “Generic Letter 92-01, Supplement 1, Reactor Vessel Structural Integrity, Request for Additional Information,” dated July 1, 1998.

Arkansas Nuclear One Power Plant Procedures

Procedure GES-26, “ULD Writers Guide,” Revision 1.

Procedure NES-16, “Accident Analysis ULD and AIM Basis Document Format and Content,” Revision 1.

Procedure 1000.150, “Licensing Document Maintenance,” Revision2.

Procedure 1409.66, “Component Level Q-List Project Design Review,” Revision 0.

Procedure 5010.004, “Design Document Changes,” Revision 3.

Procedure 5010.007, “Control of Upper Level Documents,” Revision 3.

Reports

ULD-0-TOP-22, ANO Unit 1 and 2, “ANO Component Classification Topical,” Revision 0.

93-R-1009-01, “ANO-1 License Renewal Project Methodology and Management Plan,” Revision 0.

93-R-1010-01, “ANO-1 License Renewal Integrated Plant Assessment System and Structures Screening,” Revision 0.

Submittals

Arkansas Nuclear One - Unit 1, License Renewal Application dated January 31,2000.

Electric Power Research Institute (EPRI)

TR-105714, “PWR Primary Water Chemistry Guidelines.”

TR-102135-R4, “PWR Secondary Water Chemistry Guidelines.”

TR-107396, “Closed Cooling Water Chemistry Guidelines.”

TR-106229, “Service Water System Chemistry Addition Guidelines.”

First Energy

CR-199901648, Davis-Besse Nuclear Generating Station, “Root Cause Analysis Report, #2 CCW Pump Trip,” October 2,1999.

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

ANS/IEEE Std. 450-1980, “IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Storage Batteries for Generating Stations and Substations.”

IEEE Std. 323-1974, “Qualifying Class 1E Equipment for Nuclear Power Generating Stations,” 1974.

Nuclear Energy Institute

NEI 95-10, “Industry Guideline for Implementing the Requirements of 10 CFR Part 54—The License Renewal Rule,” Revision 0, March 1996.

NEI 95-10, “Industry Guideline for Implementing the Requirements of 10 CFR Part 54—The License Renewal Rule,” Revision 1, January 2000.

U.S. Nuclear Regulatory Commission (NRC)

Bulletins (BL)

NRC BL-79-01B, “Guidelines for Evaluation Environmental Qualification of Class IE Electrical Equipment in Operating Reactors.”

NRC BL-79-02, Revision 0, “Pipe Support Base Plate Designs Using Expansion Anchor Bolts,” March 8, 1979.

NRC BL-79-13, “Cracking in Feedwater System Piping.”

NRC BL-79-17, “Pipe Cracks in Stagnant Borated water Systems at PWR Plants,”

NRC BL-87-01, “Thinning of Pipe Walls in Nuclear Power Plants.”

NRC BL-88-08, “Thermal Stresses in Piping Connected to Reactor Coolant Systems,” June 22, 1988.

NRC BL-88-11, “Pressurizer Surge Line Thermal Stratification,” December 20, 1988.

Code of Federal Regulations

10 CFR Part 50.34, “Contents of application; technical information,” Section (a)(1).

10 CFR Part 50.48, “Fire Protection”

10 CFR Part 50.49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants.”

10 CFR Part 50.55a, “Codes and Standards.”

10 CFR Part 50.60, “Acceptance Criteria for Fracture Prevention Measures for Light water Nuclear Power Reactors for Normal Operation.”

10 CFR Part 50.61, “Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events.”

10 CFR Part 50.62, “Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants.”

10 CFR Part 50.63, “Loss of All Alternating Current Power.”

10 CFR Part 50.Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.”

10 CFR Part 50.Appendix G, “Fracture Toughness Requirements.”

10 CFR Part 50.Appendix H, “Reactor Vessel Material Surveillance Program Requirements.”

10 CFR Part 54, “Requirements for Renewal of Operating Licenses for Nuclear Power Plants.”

10 CFR Part 100, “Reactor Site Criteria.”

Generic Letters (GL)

NRC GL 79-20, “Information Requested on PVR Feedwater Lines

NRC GL 85-20, “Resolution of Generic Issue 69: High Pressure Injection/Makeup Nozzle Cracking in Babcock and Wilcox Plants,” November 11,1985.

NRC GL 88-14, “Instrument Air Supply System Problems Affecting Safety-Related Equipment.”

NRC GL 89-13, “Alternate Waste Management Procedures in Case of Denial of Access to Low-Level Waste Disposal Sites.”

NRC GL 89-13, “Service Water System Problems Affecting Safety-Related Equipment.”

NRC GL 92-01, Revision 1, Supplement 1, “Reactor Vessel Structural Integrity,” May 18, 1995.

NRC GL 96-04, “Boraflex Degradation in Spent Fuel Pool Storage Racks.”

Information Notices (IN)

NRC IN 79-19, “Pipe Cracks in Stagnant Borated Water Systems at Power Plants.”

NRC IN 79-23, “Emergency Diesel Generator Lube Oil Coolers.”

NRC IN 80-29, “Broken Studs on Terry Turbine Steam Inlet Flanges.”

NRC IN 81-04, “Cracking in main Steam Lines.”

NRC IN 82-09, “Cracking in Piping of Makeup Coolant Lines at B&W Plants,” March 31, 1982.

NRC IN 84-18, “Stress Corrosion Cracking in Pressurized Water Reactor Systems.”

NRC IN 84-87, “Piping Thermal Deflection Induced by Stratified Flow.”

NRC IN 85-24, “Failures of Protective Coatings in Pipes and Heat Exchangers.”

NRC IN 86-106, “Feedwater Line Break.”

NRC IN 87-36, “Significant Unexpected Erosion of Feedwater Lines.”

NRC IN 87-43, “Gaps in Neutron Absorbing Material in High Density Spent Fuel Storage Racks.”

NRC IN 88-17, “Summary of Responses to NRC Bulletin 87-01, Thinning of Pipe Walls in Nuclear Power Plants.”

NRC IN 89-30 and IN 89-30, Supplement 1, “High Temperature Environments at Nuclear Power Plants.”

NRC IN 91-18, “High-Energy Piping Failures Caused by Wall Thinning.”

NRC IN 91-19, “Steam Generators Feedwater Distribution Piping Damage.”

NRC IN 91-28, “Cracking in Feedwater System Piping.”

NRC IN 91-38, “Thermal Stratification in Feedwater System Piping.”

NRC IN 92-07, “Rapid Flow-Induced Erosion/Corrosion of Feedwater Piping.”

NRC IN 93-39, “Radiation Beams from Power Reactor Biological Shields.”

NRC IN 93-70, “Degradation of Boraflex Neutron Absorber Coupons.”

NRC IN 95-38, “Degradation of Boraflex Neutron Absorber in Spent Fuel Storage Racks.”

NRC IN 97-46, “Unisolable Crack in High-Pressure Injection Piping,” July 9,1997.

Correspondence

Letter from C.I. Grimes (NRC) to D. Walters (NEI), “Guidance on Addressing GSI 168 for license Renewal,” Project 690, dated June 2, 1998.

Letter from T. Martin (NRC) to T. Tipton (NEI), dated October 1, 1996.

Reports

NUREG-0612, “Control of Heavy Loads at Nuclear Power Plant.”

NUREG-1522, "Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures," June 1995.

NUREG-1526, “Lessons Learned from Early Implementation of Maintenance Rule at Nine Nuclear Power Plants.”

NUREG-1557, “Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal,” October 1996

NUREG-1611, “Aging Management of Nuclear Power Plant Containments for License Renewal,” September 1997

NUREG/CR-5704, “Effects of LWR Coolant Environment on Fatigue Design Curves of Austenitic Stainless Steels,” April 1999.

NUREG/CR-6335, “Fatigue Strain-Life Behavior of Carbon and Low-Alloy Steels, Austenitic Stainless Steels, and Alloy 600 in LRA Environments,” August 1995.

NUREG/CR-6384, “Literature Review of Environmental Qualification of Safety-Related Electric Cables,” Vol. 1, April 1996, Brookhaven National Laboratory, Prepared for U. S. Nuclear Regulatory Commission.

NRC Regulator Guide 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials,” May 1988.

Working Draft NRC Generic Aging Lessons Learned (GALL) Report, August 2000.

Standard Review Plan (SRP)

DG-1047, “Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants.”

USA Standards Institute (USAS)

ANSI USAS B31.1.0, “USA Standard Code for Pressure Piping,” 1968.

ANSI USAS B31.7, “USA Standard Code for Pressure Piping, Nuclear Power Piping,” 1968.

USAS B31.7, “Nuclear Power Piping.”

THIS PAGE IS INTENTIONALLY LEFT BLANK

APPENDIX C
PRINCIPAL CONTRIBUTORS

NAME	RESPONSIBILITY
H. Ashar	Structural Engineering
G. Bagchi	Structural Engineering
M. Banic	Materials Engineering
W. Bateman	Technical Support
S. Coffin	Materials Engineering
J. Davis	Materials Engineering
D. Diec	Plant Systems
T. Eaton	Plant Systems (Fire Protection)
J. Fair	Mechanical Engineering
Z. Fu	Materials Engineering
G. Galletti	Quality Assurance
G. Georgiev	Structural Engineering
C. Gratton	Plant Systems
B. Grenier	Technical Support
C. Grimes	Technical Support
F. Grubelich	Mechanical Engineering
J. Guo	Plant Systems
M. Hartzman	Mechanical
C. Holden	Electrical Engineering
S. Hou	Structural Engineering
D. Jeng	Structural Engineering
M. Khanna	Materials Engineering
Y. Kim	Mechanical Engineering
C. Lauron	Structural Engineering
A.D. Lee	Mechanical Engineering
A.J. Lee	Mechanical Engineering
C. Li	Plant Systems
Y. Li	Mechanical Engineering
J. Ma	Structural Engineering
K. Manoly	Electrical Engineering
P. Milano	Safety and Environment
M. Mitchell	Materials Engineering
J. Moore	Legal Councel
C. Munson	Structural Engineering
D. Nguyen	Electrical Engineering
A. Pal	Electrical Engineering
P. Patnaik	Structural Engineering
K. Parczewski	Chemical Engineering
J. Peralta	Quality Assurance
J. Rajan	Mechanical Engineering
J. Raval	Plant Systems
K. Rico	Technical Support
J. Strosnider	Technical Support
E. Sullivan	Materials Engineering
O. Tabatabai-Yazdi	Technical Support

R. Wessman	Technical Support
K. Wichman	Technical Support

CONTRACTORS

Argonne National Laboratory	Aging Management Review
Brookhaven National Laboratory	Aging Management Review
Idaho National Engineering and Environmental Laboratory	Aging Management Review
Pacific Northwest National Laboratory	Aging Management Review