Safety Evaluation Report

Related to the License Renewal of the Fort Calhoun Station, Unit 1

Docket No. 50-285

Omaha Public Power District

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

April 2003

ABSTRACT

This safety evaluation report (SER) documents the technical review of the Fort Calhoun Station, Unit No. 1 (FCS), license renewal application (LRA) by the U.S. Nuclear Regulatory Commission staff (staff). By letters dated January 9 and April 5, 2002, Omaha Public Power District (OPPD or the applicant) submitted the LRA for FCS in accordance with Title 10 of the *Code of Federal Regulations* Part 54 (10 CFR Part 54 or the Rule). OPPD is requesting renewal of the operating license for Unit 1 (license numbers DPR-40) for a period of 20 years beyond the current expiration of midnight, August 9, 2013.

The FCS site is located in Washington County, Nebraska, on the west bank of the Missouri River, approximately 19 miles north of Omaha, Nebraska. The construction permit was issued by NRC on June 7, 1968, and the operating license was issued August 9, 1973. The unit consists of a Combustion Engineering (CE) pressurized-water reactor (PWR) nuclear steam supply system designed to generate 1500 MW-thermal, or approximately 475 MW-electric.

This SER presents the status of the staff's review of information submitted to the NRC through April 4, 2003, the cutoff date for consideration in the SER. The staff has identified open items that must be resolved before the staff can make a determination on the application. These items are summarized in Section 1.6 of this report. In order to close these items, the staff requires the additional information identified in the open items. The staff will present its final conclusion on the review on the FCS application in its update to this SER.

ABBREVIATIONS

AB-FO	auxiliary boiler fuel oil
AC	alternating current
ACRS	Advisory Committee on Reactor Safeguards
ACSR	aluminum conductor, steel reinforced
AERM	aging effect requiring management
AFW	auxiliary feedwater
AMP	aging management program
AMR	aging management review
ANSI	American National Standards Institute
AOV	air-operated valve
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	anticipated transient without scram
AWWA	American Water Works Association
B&W	Babcock & Wilcox
B&WOG	Babcock & Wilcox Owners Group
BAC	boric acid corrosion
BL	Bulletin
BTP	Branch Technical Position
BWR	boiling-water reactor
CA	compressed air
CAP	corrective action program
CASS	cast austenitic stainless steel
CCNPP	Calvert Cliffs Nuclear Power Plant
CCW	component cooling water
CE	Combustion Engineering, Control element
CEA	control element assembly
CEDM	control element drive mechanism
CEOG	Combustion Engineering Owners Group
CFR	Code of Federal Regulations
CI	confirmatory item
CIAS	containment isolation actuation signal
CIV	containment isolation valve
CLB	current licensing basis
CMAA	Crane Manufacturers Association of America
CQE	critical quality element
CR	condition report
CRD	control rod drive
CS	containment spray
CSB	core support barrel
CUF	cumulative usage factor
CVCS	chemical and volume control system
DBA	Design-Basis Accident
DBD	design basis document
DBE	Design-Basis Event

ABBREVIATIONS (con't)

DC	direct current
DG	diesel generator
DGFO	emergency diesel generator fuel oil
DGLO	emergency diesel generator lube oil
DSS	diverse scram system
EA	engineering analysis
ECCS	emergency core cooling system
ECT	eddy current testing
EDG	emergency diesel generator
EFPY	effective full power years
EFWST	emergency feedwater storage tank
EOCI	Electric Overhead Crane Insititute
EPRI	Electric Power Research Institute
EEQ	electrical equipment qualification
EQ	environmental qualification
ESF	engineered safety features
ESFAS	engineered safety features actuation system
FAC	flow-accelerated corrosion
FACTS	Fort Calhoun Automatic Cable Tracking System
FCS	Fort Calhoun Station, Unit 1
FHA	fire hazards analysis
FMP	fatigue monitoring program
FP	fire protection
FP-FO	fire protection fuel oil
FPP	fire protection program
FSAR	Final Safety Analysis Report
FW	feedwater
GALL	Generic Aging Lessons Learned
GE	General Electric
GEIS	generic environmental impact statement
GL	Generic Letter
GWD	gaseous waste disposal
HELB	high energy line break
HEPA	high-efficiency particulate air
HPCI	high-pressure coolant injection
HPSI	high-pressure safety injection
HVAC	heating, ventilation, and air conditioning
I&C	instrumentation and control
IA	instrument air
IASCC	irradiation-assisted stress corrosion cracking
IN	Information Notice
ICI	in-core instrumentation
IGA	intergranular attack
IGSCC	intergranular stress corrosion cracking
LBB	leak-before-break
LOCA	loss-of-coolant-accident

ABBREVIATIONS (con't)

LPSI	low-pressure safety injection
LTOP	low-temperature overpressure
LRA	license renewal application
LRDB	license renewal database
ISG	interim staff guidance
ISI	inservice inspection
LER	licensee event report
LWD	liquid waste disposal
MCRE	main control room envelope
MFW	main feedwater
MIC	microbiologically-influenced corrosion
Mo	molybdenum
Mn	managanese
MS	main steam
MSIV	main steam isolation valve
MW	megawatt
n/cm^2	neutrons per square centimeter
NDE	non-destructive examination
NEI	Nuclear Energy Institute
NEPA	National Environmental Policy Act
NFPA	National Fire Protection Association
NG	nitrogen gas
Ni	Nickel
NPS	nominal pipe size
NSSS	nuclear steam supply system
OD	outside diameter
ODCM	off-site dose calculation manual
ODSCC	outer diameter stress corrosion cracking
OI	open item
OPPD	Omaha Public Power District
P&ID	piping and instrumentation diagram
PBD	program basis document
PM	preventive maintenance
POI	potential open item
PORV	power-operated relief valve
PRA	probability and risk assessment, probabilistic risk assessment
PS	primary sampling
PS/PMP	periodic surveillance and preventive maintenance program
P/T	pressure/temperature
PTS	pressurized thermal shock
PVC	polyvinyl chloride
PWR	pressurized-water reactor
PWSCC	primary water stress corrosion cracking
QA	quality assurance
RAI	request for additional information

ABBREVIATIONS (con't)

RAMS	Resource Acquisition Management System
RC	reactor coolant
RCIC	reactor core isolation cooling
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RG	Regulatory Guide
RMS	radiation monitoring system
RPS	reactor protection system
RS	reactor systems
RT_{NDT}	nil ductility reference temperature
RT_{PTS}	PTS reference temperature
RTD	resistance temperature detector
RV	reactor vessel, relief valve
RVI	reactor vessel internals
RVII	reactor vessel internals inspection
RVIP	reactor vessel integrity program
RW	raw water
SBO	station blackout
SC	structure and component
SCC	stress corrosion cracking
SDC	shutdown cooling
SER	safety evaluation report
SFP	spent fuel pool
SFPC	spent fuel pool cooling
SG	steam generator
SGIS	steam generator isolation signal
SGP	steam generator program
SI	safety injection
SI&CS	safety injection and containment spray
SIAS	safety injection actuation signal
SIRWT	safety injection refueling water tank
SMP	structures monitoring program
SO	standing order
SOC	Statements of Consideration
SOER	Significant Operating Experience Report
SPCS	steam and power conversion systems
SRP-LR	Standard Review Plan for License Renewal
SSC	structures, systems, and components
SSEL	safe shutdown equipment list
SV	safety valve
T	thickness
TLAA	time-limited aging analysis
TR	Topical Report
UCS	Union of Concerned Scientists

ABBREVIATIONS (con't)

UFHA	updated fire hazards analysis
UGS	upper guide structure
USAR	Updated Safety Analysis Report
USAS	United States of America Standard
USE	upper shelf energy
UT	ultrasonic testing
VA	ventilating air
VAC	volt-alternating current
VCT	volume control tank
VDC	volt-direct current
VHP	vessel head penetration

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SECTION 1

INTRODUCTION AND GENERAL INFORMATION

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1 Introduction and General Discussion

1.1 Introduction

This document is an SER on the application for license renewal for the Fort Calhoun Station, Unit 1 (FCS), as filed by the Omaha Public Power District (OPPD or the applicant). By letters dated January 9 and April 5, 2002, OPPD submitted its application to the United States Nuclear Regulatory Commission (NRC or the Agency) for renewal of the FCS operating license for an additional 20 years. The NRC staff (the staff) prepared this report and summarizes the results of its safety review of the renewal application for compliance with the requirements of 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants." The NRC license renewal project manager for the FCS license renewal review is William F. Burton. Mr. Burton may be contacted by calling 301-415-2853, or by writing to the License Renewal and Environmental Impacts Project, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001.

In its January 9, 2002 submittal letter, the applicant requested renewal of the operating license issued under Section 104b of the Atomic Energy Act of 1954, as amended, for Unit 1 (license number DPR-40), for a period of 20 years beyond the current license expiration of midnight, August 9, 2013. The FCS site is located in Washington County, Nebraska on the west bank of the Missouri River, approximately 19 miles north of Omaha, Nebraska. Construction began on Unit 1 in June, 1968, and its operating license was issued on August 9, 1973. The unit consists of a Combustion Engineering (CE) pressurized-water reactor (PWR) nuclear steam supply system designed to generate 1500 MW-thermal, or approximately 475 MW-electric. Details concerning the plant and the site are found in the Updated Safety Analysis Reports (USAR) for the unit.

The license renewal process proceeds along two tracks: a technical review of safety issues and an environmental review. The requirements for these reviews are stated in NRC regulations 10 CFR Parts 54 and 51, respectively. The safety review for the FCS license renewal is based on the applicant's license renewal application (LRA) and on the answers to requests for additional information (RAIs) from the staff. In meetings and docketed correspondence, the applicant has also supplemented its answers to the RAIs. Unless otherwise noted, the staff reviewed and considered information submitted through April 4, 2003. Information received after that date was reviewed on a case-by-case basis, depending on the stage of the safety review. The LRA and all pertinent information and materials, including the USAR mentioned above, are available to the public for review at the NRC Public Document Room, 11555 Rockville Pike, Room 1-F21, Rockville, MD, 20852-2738 (301-415-4737/800-3974209); the W. Dale Clark Library, 215 South 15th Street, Omaha, NE 68102; and the Blair Public Library, 210 South 17th Street, Blair, NE 68008-2055. Material related to the LRA is also available through the NRC's website, at www.nrc.gov

This SER summarizes the results of the staff's safety review of the FCS LRA and delineates the scope of the technical details considered in evaluating the safety aspects of its proposed operation for an additional 20 years beyond the term of the current operating license. The LRA was reviewed in accordance with the NRC regulations and the guidance provided in the NRC NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, " dated July 2001(SRP-LR).

Sections 2 through 4 of the SER address the staff's review and evaluation of license renewal issues that have been considered during the review of the application. Section 5 is reserved for the report of the Advisory Committee on Reactor Safeguards (ACRS). The conclusions of this report are in Section 6.

Appendix A of this SER is a table that identifies the applicant's commitments associated with the renewal of the operating license. Appendix B contains the report of findings associated with the staff's scoping and screening inspection that was conducted from November 4 through November 8, 2002. Appendix C contains the report of findings associated with the staff's aging management review inspection that was conducted from January 6 through 10, 2003, and from January 20 through 23, 2003. Appendix D provides a chronology of NRC's and the applicant's principal correspondence related to the review of the application. Appendix E presents an index of the staff's RAIs and the applicant's responses. Appendix F is a bibliography of the references used during the course of the review. Appendix G is a list of principal contributors to the SER.

In accordance with 10 CFR Part 51, the staff will prepare a draft for comment, and a final plant-specific supplement to the generic environmental impact statement (GEIS) that discusses the environmental considerations related to renewing the license for FCS. NUREG-1437, Supplement 12, the plant-specific draft supplement to the GEIS, was issued on January, 2003.

1.2 License Renewal Background

Pursuant to the Atomic Energy Act of 1954, as amended, and NRC regulations, licenses for commercial power reactors to operate 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 -- not by technical limitations. However, some individual plant and equipment designs may have been engineered on the basis of an expected 40-year service life.

In 1982, the NRC held a workshop on nuclear power plant aging, in anticipation of the interest in license renewal. That led the NRC to establish a comprehensive program plan 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 pose technical issues that would preclude life extension for nuclear power plants. In 1986, the NRC published a request for comment on 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, and industry sponsored, demonstration programs 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 initial license. In addition, the NRC found that the scope of the review did not allow sufficient credit for existing programs, particularly the implementation of the maintenance rule, which also manages plant aging phenomena. As a result, in 1995 the NRC amended the license renewal rule. The amended 10 CFR Part 54 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 identification of all aging mechanisms. The rule changes were intended to ensure that important systems, structures, and components (SSCs) will continue to perform their intended function in 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, 10 CFR Part 51, to focus the scope of the review of environmental impacts of license renewal, in fulfilling NRC's responsibilities under the National Environmental Policy Act of 1969 (NEPA).

1.2.1 Safety Review

License renewal requirements for power reactors are based on two key principals:

(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 plant SSCs in the period of extended operation, and possibly a few other issues related to safety 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 (the Rule) defines the scope of license renewal as those 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 NRC's regulations for fire protection, environmental qualification, pressurized thermal shock, anticipated transients without scram, and station blackout.

Pursuant to 10 CFR 54.21(a), an applicant for a renewed license must review all SSCs within the scope of the Rule to identify SCs subject to an aging management review (AMR). SCs 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 qualified life or specified time period. As required by 10 CFR 54.21(a), an applicant for a renewed license must demonstrate that the effects of aging will be managed in such a way that the intended function or functions of those SCs will be maintained, consistent with the current licensing basis, 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 aging effects that may occur for active equipment are more readily detectable and will be identified and corrected through routine surveillance, performance indicators, and maintenance. The surveillance and maintenance programs for active equipment, as well as other aspects of maintaining the plant design and licensing basis, are required throughout the period of extended operation. Section 54.21(d) requires that a supplement to the FSAR 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 length of time the plant will be operated and these assumptions are incorporated into design

calculations for several of the plant's SSCs. Under 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 on these SSCs will be adequately managed for the period of extended operation.

In 2001, the NRC developed and issued Regulatory Guide (RG) 1.188, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses." This guide endorses an implementation guideline prepared by the Nuclear Energy Institute (NEI) as an acceptable method of implementing the license renewal rule. The NEI 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 2001. The NRC prepared the SRP-LR. The RG was used, along with the SRP-LR, to review this application and to assess technical issue reports involved in license renewal as submitted by industry groups.

The Omaha Public Power District (OPPD) is the first license renewal applicant to fully utilize the process defined in NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," dated July 2001. The purpose of GALL is to provide the staff with a summary of staff-approved aging management programs (AMPs) for the aging of most structures and components that are subject to an aging management review (AMR). If an applicant commits to implementing these staff-approved AMPs, the time, effort, and resources used to review an applicant's LRA will be greatly reduced, thereby improving the efficiency and effectiveness of the license renewal review process. The GALL Report is a compilation of existing programs and activities used by commercial nuclear power plants to manage the aging of structures and components within the scope of license renewal and which are subject to an AMR. The GALL Report summarizes the aging management evaluations, programs, and activities credited for managing aging for most of the structures and components used throughout the industry, and serves as a reference for both applicants and staff reviewers to quickly identify those aging management programs and activities that the staff has determined will provide adequate aging management during the period of extended operation.

1.2.2 Environmental Review

The environmental protection regulations, 10 CFR Part 51, were revised in December 1996 to facilitate the environmental review for license renewal. The staff prepared a GEIS, in which the staff 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. Analyses of those environmental impacts that must be evaluated on a plant-specific basis, Category 2 issues, must be included in the 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 staff performed a plant-specific review of the environmental impacts of license renewal, including whether there was new and significant information not considered in the GEIS. A public meeting was held on June 18, 2002, in Omaha, Nebraska, as part of the NRC scoping process, to identify environmental issues specific to the plant. Results of the environmental review and a

preliminary recommendation with respect to the license renewal action were documented in NRC's draft plant-specific supplement to the GEIS, which was issued by the NRC on January 6, 2003, and which was discussed at a separate public meeting held on February 26, 2003, in Omaha, NE. After consideration of comments on the draft, NRC will prepare and publish a final plant-specific supplement to the GEIS. These documents are published separate from this report.

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 FCS LRA in accordance with Commission guidance and the requirements of 10 CFR Part 54. The standards for renewing a license are contained in 10 CFR 54.29. This SER describes the results of the staff's safety review.

In 10 CFR 54.19(a), the Commission requires a license renewal applicant to submit general information. The applicant provided this general information in Section 1 of its LRA for FCS, submitted by letter dated January 9, 2002. The staff finds that the applicant has submitted the information required by 10 CFR 54.19(a) in Section 1 of the LRA.

In 10 CFR 54.19(b), the Commission requires that license renewal applications 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." The applicant states the following in its LRA regarding this issue:

> "The current indemnity agreement for Fort Calhoun Station, Unit 1 does not contain a specific expiration term for the operating license. Therefore, conforming changes to account for the expiration term of the proposed renewed license are not necessary, unless the license number is changed upon issuance of the renewed license."

The staff intends to maintain the license number on issuance of 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 renewal license for a nuclear facility must contain the following information: (a) an IPA, (b) a description of current licensing basis (CLB) changes during staff review of the application, (c) an evaluation of TLAAs, and (d) an FSAR supplement. Sections 3 and 4, and Sections A and B of the LRA, address the license renewal requirements of 10 CFR 54.21(a), (c), and (d), respectively.

In 10 CFR 54.21(b), the Commission requires that each year following submittal of the application and at least 3 months before the scheduled completion of the staff's review, an amendment to the renewal application must be submitted that identifies any change to the current licensing basis of the facility that materially affects the contents of the license renewal application, including the FSAR Supplement.

In 10 CFR 54.22, the Commission states requirements regarding technical specifications. In Appendix D of the LRA, the applicant states that no changes to the FCS technical specifications are necessary. This adequately addresses the requirements of 10 CFR 54.22.

The staff evaluated the technical information required by 10 CFR 54.21 and 10 CFR 54.22 in accordance with the NRC's regulations and the guidance provided by the SRP-LR. The staff's evaluation of the LRA in accordance with 10 CFR 54.21 and 54.22 is contained in Sections 2, 3, and 4 of this report.

The staff's evaluation of the environmental information required by 10 CFR 54.23 will be found in the final plant-specific supplement to the GEIS that state the considerations related to renewing the license for FCS. This will be prepared by the staff separate from this report. When the report of the Advisory Committee on Reactor Safeguards (ACRS), required by 10 CFR 54.25, is issued, it will be incorporated into Section 5 of this SER. The findings required by 10 CFR 54.29 will be made in Section 6 of this report.

1.4 Interim Staff Guidance

The license renewal program is a living program. The NRC staff, industry, and other interested stakeholders gain experience and develop lessons learned with each renewed license. The lessons learned address the NRC's performance goals of maintaining safety, improving effectiveness and efficiency, reducing regulatory burden, and increasing public confidence. The lessons learned are captured in interim staff guidance (ISG) for use by the staff and interested stakeholders until the improved license renewal guidance documents are revised.

The current set of relevant ISGs that have been issued by the staff, and the SER sections where the issues are addressed by the staff, is provided below:

ISG Issue (Approved ISG No.)	**Purpose**	**SER Section**
Station Blackout (SBO) Scoping **(ISG-02)**	The license renewal rule 10 CFR 54.4(a)(3) includes 10 CFR 50.63(a)(1)-SBO. The SBO rule requires that a plant must withstand and recover from an SBO event. The recovery time for offsite power is much faster than that of EDGs. The offsite power system should be included within the scope of license renewal.	2.5.2 3.6.2.4.4
Concrete Aging Management Program **(ISG-03)**	Lessons learned from the GALL Demonstration project indicated that GALL is not clear whether concrete needs any AMPs.	3.5.2.2.1 3.5.2.2.2 3.5.2.4.1 3.5.2.4.2
Fire Protection (FP) System Piping **(ISG-4)**	To clarify staff position for wall thinning of FP piping system in GALL AMPs (XI.M26 and XI.M27). New position is that there is no need to disassemble FP piping, as oxygen can be introduced in the FP piping which can accelerate corrosion. Instead, use non-intrusive method such as volumetric inspection. Testing of sprinkler heads should be performed every 50 years and 10 years after initial service. Eliminated Halon/carbon dioxide system inspections for charging pressure, valve line ups, and automatic mode of operation test from GALL, as the staff considers these test verifications to be operational activities.	3.0.3.9

Identification and Treatment of Electrical Fuse Holder **(ISG-5)**	To include fuse holder AMR and AMP (i.e., same as terminal blocks and other electrical connections). The position includes only fuse holders that are not inside the enclosure of active components (e.g., inside of switchgears, and inverters). Operating experience finds that metallic clamps (spring-loaded clips) have a history of age-related failures from aging stressors such as vibration, thermal cycling, mechanical stress, corrosion, and chemical contamination. The staff finds that visual inspection of fuse clips is not sufficient to detect the aging effects from fatigue, mechanical stress, and vibration.	3.6.2.4.5

1.5 Summary of Open Items

As a result of its review of the LRA for FCS, including additional information submitted to the NRC through April 4, 2003, the staff identified the following issues that remained open at the time this report was prepared. An issue was open if the applicant had not presented a sufficient basis for resolution, or where information provided to the staff in recent applicant submittals in response to potential open items (POIs) has yet to be reviewed by the staff. Each open item has been assigned a unique identifying number.

Item	Description
2.2-1	During the AMR inspection and audit, the team reviewed the on-site engineering analysis (EA)-FC-00-149, “NSR Steam and Water Systems Impacting SSC Within Scope For License Renewal.” The applicant identified piping systems and associated reference drawings for those systems that have met the 54.4(a)(2) criteria for spatial interaction. The applicant indicated that some of these systems are already within the scope of license renewal but some are not. The applicant also stated that flow-accelerated corrosion (FAC), chemistry, general corrosion of external surfaces, and structure monitoring programs are the applicable AMPs to manage aging effects for components in these systems. On the basis of its review, the staff determined that the information as provided by the applicant is

not sufficient for the staff's scoping and aging management reviews for these 10 CFR 54.4(a)(2) SSCs. For the additional SSCs that have been brought into scope to meet the 10 CFR 54.4(a)(2) criterion, the applicant should provide scoping information to the component level equivalent to that of the original license renewal application. This information is necessary for the staff to be able to determine, with reasonable assurance, that all the components required by 10 CFR 54.4(a)(2) to be within the scope of license renewal and subject to an AMR have been correctly identified. Also, the applicant should provide revised and/or new Section 2 tables, including links to Section 3 tables, so that the staff may perform an aging management review to determine whether the applicant has identified the proper aging effects for the combination of the material and environment, and has provided an adequate AMP for managing the corresponding aging effect for these SSCs. By letter dated February 20, 2003, the staff issued POI-1(a), requesting that the applicant provide the above information. By letter dated March 14, 2003, the applicant provided the requested information. The staff has reviewed the information and finds that the applicant has adequately identified the structures and components within the scope of license renewal as a result of the meeting the 54.4(a)(2) scoping criterion. POI-1(a) is resolved. The staff must review the AMR results for the additional components brought into scope and subject to an AMR. The results of the staff's review will be provided in the final SER.

2.2-2 The EA stated that the compressed air, demineralized water, and steam generator feedwater blowdown systems contained components that were functionally realigned. The team noted that this was inconsistent with LRA Table 2.2-1 and LRA Section 2.3.2.2. LRA Table 2.2-1 states that containment isolation and/or pressure boundary components in the compressed air, demineralized water, and blowpipe systems were functionally realigned to the commodity group "Containment Penetration and System Interface Components for Non-CQE Related System". However, LRA Section 2.3.2.2, which describes this commodity group, states that the group contains CIVs from the feedwater blowdown, compressed air, blowpipe, and demineralized water systems, as well as the piping between the containment penetrations and the CIVs. It also states that the demineralized water heat exchangers are included in the commodity group to maintain the component cooling water system pressure boundary. LRA Table 2.2-1 and the description in LRA Section 2.3.2.2 are inconsistent, in that the blowdown system is not identified in LRA Table 2.2-1 as having components that were functionally realigned. By letter dated February 20, 2003, the staff issued POI-1(d), requesting the applicant to resolve this discrepancy between LRA Table 2.2-1 and the description in LRA Section 2.3.2.2 and provide revised Section 2 tables and, if necessary, revised Section 3 tables to accurately describe which systems and/or components have been functionally realigned and how the components will be managed.

By letter dated March 14, 2003, the applicant responded to POI-1(d), providing revisions to LRA Table 2.2-1 and LRA Section 2.3.2.2, and providing an additional drawing to clearly identify the blowpipe system. On the basis of the applicant's response, POI-1(d) is resolved. However, the staff must review the

information provided to ensure that all components within scope and subject to an AMR have been identified.

2.3.3.15-1 Section 2.3.3.15 of the LRA describes that the raw water discharge from the CCW system heat exchangers and the discharge from the direct cooling raw water header flow into the circulating water discharge tunnel. Table 2.2-1 of the LRA designated the circulating water system as outside of license renewal scope without specific justification, but failure of the pressure boundary of buried piping or tunnels creates the potential for a loss of flow. Therefore, the location of the license renewal boundary at the discharge pipes for the RW system rather than at the outlet from the circulating water discharge tunnel has not been adequately justified. By letter dated February 20, 2003, the staff issued POI-3(a), requesting the applicant to justify the location of the license renewal boundary.

By letter dated March 14, 2003, the applicant responded to this POI, stating that the location for the raw water discharge license renewal boundary at check valves CW-188 and CW-189, upstream of the circulating water discharge tunnel, has been revised. The applicant included the circulating water discharge tunnel within the scope for license renewal as part of the intake structure. The applicant referenced a separate letter dated March 14, 2003, which included revised boundary drawing 11405-M-100 and new boundary drawing 11405-M-257, Sh. 2, as attachments. These drawings show that a continuous flow path from the raw water system to the river outfall has been included within scope for license renewal. This resolves the scoping issues associated with POI-3(a), but the expansion of scope introduces the need for evaluation of the applicant's aging management review for the discharge tunnel.

In its POI response, the applicant provided the following discussion regarding the aging management review of the discharge tunnel:

a. The circulating water discharge tunnel is constructed of reinforced concrete with a nominal wall thickness of 2' or greater and nominal floor/ceiling thicknesses of 2'-6" or greater throughout. The concrete circulating water discharge tunnel walls, floor and ceiling are constructed of Type B concrete in accordance with ACI 201.2R as specified in NUREG-1557.

b. The concrete is not exposed to aggressive river water or groundwater. The concrete that surrounds the embedded steel has a pH greater than or equal to 12.5. The concrete mix design specified a water-to-cement ratio of 0.44 and air entrainment of 5.00% + 1.00% for Class B concrete. The concrete at FCS was designed in accordance with ACI 318-63 (per USAR Section 5.3.1 Revision 0 and USAR Section 5.11.3.1 Revision 2).

c. The maximum flow rate in the circulating water tunnel is well below the velocity of 25 fps required to initiate abrasion. The calculated highest water velocity for a closed conduit is in the warm water recirculating tunnel at 12.6 fps. Therefore, this aging effect is not credible.

d. Per NUREG-1557, corrosion of embedded steel is not significant for concrete structures above or below grade that are exposed to a non-aggressive environment. A non-aggressive environment, as defined by NUREG-1557, is one with a pH greater than 11.5 or chlorides less than 500 ppm. NUREG-1557 also concludes that corrosion of embedded steel is not significant for concrete structures exposed to an aggressive environment but have a low water-to-cement ratio, adequate air entrainment, and designed in accordance with ACI 318-63 or ACI 349-85. A low water-to-cement ratio is defined as 0.35 to 0.45 and adequate air entrainment is defined as 3 to 6 percent. Therefore, corrosion of embedded steel is not credible.

e. The freeze/thaw exposure category is “Severe” since the concrete of concern is in direct contact with the soil. Based on recent analyses, the groundwater and river water contain minimal amounts of chlorides (8.0 ppm and 14.0 ppm respectively), sulfates (79 ppm and 229 ppm respectively), and the pH is slightly alkaline (7.48 and 8.39 respectively); therefore, the exposure category for sulfates, chlorides, and acids is “Mild”, and concrete degradation is not credible for the circulating water discharge tunnel.

f. The total flow of the raw water equates to less than 5% of the total volume of the circulating water discharge tunnel.

Based on the installation conditions enumerated above, the conditions specified in NUREG-1557 have been satisfied; therefore, minimal or no aging effects will be realized in the circulating water discharge tunnel. Tunnel failure will not occur to the point that the raw water intended function would be impacted or jeopardized during the period of extended operation. To verify this assumption, the applicant will perform a one-time inspection of the circulating water discharge tunnel per the one-time inspection program (B.3.5).

The staff evaluated the information provided in response to POI-3(a) and finds it acceptable because the applicant has brought the circulating water discharge tunnel within scope. Therefore, POI-3(a) is resolved. The staff’s review of the aging management review associated with the expanded scope will be provided in the final SER.

3.0-1 In its letter dated March 14, 2003, the applicant provided revisions to many tables in LRA Sections 2 and 3. The staff will review the revised information to determine whether the revisions change the staff’s conclusions as documented in the SER.

3.3.2.4.1.2-1 For the regenerative heat exchanger, which is constructed of stainless steel and exposed to chemically treated borated water, LRA Table 2.3.3.1-1 cites link 3.3.1.08 for aging management of cracking due to SCC, consistent with the

GALL. This link states that the aging management will consist of the chemistry program, with the effectiveness of the chemistry program verified by inspections performed using either the one-time inspection program, cooling water corrosion program, or periodic surveillance and preventative maintenance program. In discussions during the AMR inspection and audit, the applicant stated that the regenerative heat exchanger is welded such that the internals are not accessible. Due to the construction of the regenerative heat exchanger, the applicant stated that the aging management of the regenerative heat exchanger would consist of the chemistry program with further evaluation of cracking due to SCC provided by inspection of the welds via the ISI program. The applicant considered this adequate aging management to support the pressure boundary intended function of the heat exchanger shell. Though the staff agrees that this is acceptable for the external pressure boundary, the staff notes that it would not detect degradation of the regenerative heat exchanger tubes which could allow inventory to flow from the charging to the letdown side of the chemical and volume control system. This would reduce the effectiveness of the CVCS for managing reactor coolant system chemistry, and may also reduce the ability of the system to inject borated water during an event; therefore, the proposed aging management may not be adequate to ensure that this intended function of the heat exchanger is maintained. By letter dated February 20, the staff issued POI-10(b) and POI-10(i), requesting the applicant to describe inspections of the regenerative heat exchanger internals that would verify the absence of the identified aging effects, or justify that degradation of the internals would not result in loss of function. By letter dated March 14, 2003, the applicant responded to POI-10(b) and POI-10(i), stating that a potential failure of the internal boundary between the two sides of the regenerative heat exchanger would not affect the inventory available for injection during an accident. The only function of the boundary is to provide for heat transfer during normal letdown operation. This function is not required during an accident. On the basis of its review of the information in the POI responses, the staff finds that the applicant's response does not explain how the plant can withstand the regulated events if the pressure boundary fails.

This pressure boundary function is important for at least two reasons over and above the normal CVCS function of maintaining RCS water chemistry. The first is getting adequate boron injection during an event. The second is isolating a letdown line break, which is a containment bypass LOCA (note that the CVCS injection path is the normally used path for the controlled cooldown during Appendix R events).

With regard to injection during an event, letdown is designed to isolate during any event in which there is need for injection. If the letdown heat exchanger tubes leak sufficiently, there could be a continued loss of inventory via the letdown flowpath because one of the two letdown isolation valves is upstream of the heat exchanger, meaning it would be bypassed. This would rely on a single valve to isolate letdown and support injection.

Letdown is also designed to isolate during any breaks in the system to stop containment bypass. Again, if the letdown heat exchanger tubes leak sufficiently, the inboard isolation valve is bypassed and a single train/single valve is now relied on to stop the containment bypass LOCA.

On the basis of this information, the staff requests the applicant to provide additional information to demonstrate how degradation of the heat exchanger internals will not adversely impact on the injection function, or provide information on how the internals will be managed during the period of extended operation to ensure that the injection function is maintained.

3.6.2.3.1.2-1 The staff reviewed the USAR Supplement for the non-EQ cable AMP and found that the supplement does not provide an adequate description of the revised program, as required by 10 CFR 54.21(d). The applicant should submit to the staff a revised USAR Supplement that is consistent with the descriptions for GALL AMPs XI.E1, XI.E2, and XI.E3 to satisfy 10 CFR 54.21(d).

3.6.2.4.3.2-1 LRA Table 2.5.20-1 states that electrical bus bars and bus bar standoffs have no aging effects that require management. The basis for the applicant's conclusion was unclear to the staff. By letter dated February 20, 2003, the staff issued POI-6(b), requesting the applicant to provide information on the components' materials and environments, along with the basis for concluding that these components have no plausible aging effects. By letter dated March 14, 2003, the applicant responded to POI-6(b), stating that:

> The bus bar materials are copper and aluminum; their environment is in indoor air and outdoor air. In accordance with EPRI TR-114882, Non-Class1 Mechanical Implementation Guideline and Mechanical Tools, Revision 2, 1999, no aging effects were identified for aluminum, aluminum alloys, copper, or copper alloys (brass, bronze) in an indoor or outdoor air environment.
>
> The stand offs include fiberglass reinforced polyester resin and porcelain materials that are in ambient air external environment and are not continuously wetted. Internal environments are not applicable.
>
> Table 7-17 of EPRI NP-1558, A Review of Equipment Aging Theory and Technology lists the continuous use temperature of plastics. The continuous use temperature[(a)] listed for polyester with 40% glass content is 266 °F[(b)] (compared with the bounding temperature value of 122 °F). Applying the Arrhenius methodology, it is clear that fiberglass reinforced polyester is acceptable. Figure C-2 of EPRI NP-1558 contains the relative radiation stability of thermosetting resins. The threshold for gamma radiation for polyester (glass filled) is 1,000,000,000 Rads (compared with the bounding 60-year radiation dose of less than 1,000 Rads).
>
> a. Continuous use temperatures were determined as the temperatures corresponding to 100,000 hours (11.4 years) on the Arrhenius curve of the material for an endpoint of 50% reduction in tensile strength.
>
> b. Based on retention of tensile strength taken at 500 degrees F.

On the basis of its review of the applicant's response the POI-6(b), the staff was concerned that the applicant may not have considered all the aging effects of the bus bars/ducts. The staff discussed this issue with the applicant, pointing out

that the industry experience has indicated several problems with the bus bar/duct, such as loosening of splice plate bolts, degradation of Noryl insulation, presence of moisture or debris, oxidation of aluminum electrical connections, and corrosion of metallic components. The staff requests the applicant to provide a description of the aging management program used to detect the above aging effects, or provide justification why such a program is not needed.

3.6.2.4.4.2-1 The aging effect for the transmission ACSR conductor is loss of conductor strength and vibration. The applicant has addressed the vibration and the aluminum portion of the conductor, but did not address the steel portion. The most prevalent mechanism contributing to loss of conductor strength is corrosion, which includes corrosion of steel core and aluminum strand pitting. The staff requests the applicant to provide a description of its aging management programs used to manage the aging effects in high voltage conductors, or provide justification for why such programs are not needed.

3.6.2.4.5.2-1 In LRA Section 2.5.1, "Cables and Connectors," the applicant identifies fuse blocks as components within the scope of license renewal and subject to an AMR. The staff was unsure whether fuse holders were included within the component type "Fuse Block." By letter dated February 20, 2003, the staff issued POI-1(c), requesting the applicant to clarify whether fuse holders are within the scope of license renewal and subject to an AMR, and, if fuse holders are brought in and require aging management, to provide the associated aging management information.

By letter dated March 14, 2003, the applicant provided the requested information, stating that:

> Fuse holders are in the scope of license renewal as part of the cable and connector scoping and screening analysis. There are no fuse holders attached to electrical penetrations at FCS. Fuse holders at FCS that are within active enclosures such as power supplies, switchgear, and Motor Control Centers are considered outside the scope for license renewal. There are no fuse holders at FCS exposed to vibration or environments that would cause corrosion, chemical contamination, or oxidation of the connecting surfaces. Fuse holders within enclosures that are not considered active and subject to mechanical stress, fatigue and electrical transients will be included in the Fatigue-Monitoring Program(B.2.4).

The staff reviewed the applicant's response to POI-1(c) regarding whether fuse holders within the enclosures are considered active and whether they are subject to stress and fatigue. The staff discussed this issue with the applicant. The applicant believed that there are no fuse holders that would fall within the definition of being in an outside environment that would need aging management review, but was not sure. The staff is still unclear regarding the aging management of fuse holders. ISG-5, which discusses scoping, screening, and aging management of fuse holders, states that fuse holders inside the enclosure of an active component, such as switchgear, power supplies, power inverters, battery chargers, and circuit boards, are considered to be piece parts of the larger assembly, and thus 10 CFR 54.21 considers them outside the scope for license renewal. The staff requests the applicant to make a positive statement

that all fuse holders are within active enclosures and hence need not be subject to an AMR. If the applicant cannot make this statement, the applicant should clarify how fuse holders within the scope of license renewal and subject to an AMR will be managed during the period of extended operation. The staff was also concerned that the applicant may have missed fuse holders which are used in circuits to isolate safety loads from non-safety loads. The staff requests the applicant to investigate and confirm whether there are no fuse holders which fall into this category.

4.7.2.2-1 The staff has evaluated the information provided by the applicant in its LRA and in response to RAI 4.7.2-1. The staff has concluded that the applicant appropriately identified those TLAAs (fatigue crack growth, aging of CASS RCS piping and components, and primary water stress corrosion cracking (PWSCC) of Inconel 82/182 RCS welds), which may impact the extension of the applicant's existing LBB analysis through the period of extended operation. The applicant has committed to perform a plant-specific LBB analysis prior to entering the period of extended operation which will address these TLAAs and project the analysis to the end of the period of extended operation. However, the applicant's commitment does not appear to meet 10 CFR 54.21(c)(1), which requires the applicant to demonstrate that (i) the analysis remains valid for the period of extended operation, (ii) the analysis has 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 applicant should provide the information needed for the staff to determine whether (i) the applicant's LBB analysis remains valid for the period of extended operation, (ii) the applicant's LBB analysis has been projected to the end of the period of extended operation, or (iii) the effects of aging on the intended function(s) of the components within the scope of the LBB analysis will be adequately managed for the period of extended operation.

1.6 Summary of Confirmatory Items

Confirmatory Items are items for which the staff and the applicant have reached a satisfactory resolution, but the resolution has not yet been formally submitted to the staff.

As a result of its review of the LRA for FCS, including additional information submitted to the NRC through April 4, 2003, the staff identified the following issue that remained confirmatory at the time this report was prepared.

Item	Description
2.1.3.1.2-1	As part of its review of the implementation and results of the applicant's scoping activities, the staff performed a license renewal scoping and screening inspection at the FCS site during the week of November 8, 2002, and an inspection of the applicant's aging management programs (AMPs) during the weeks of January 6 and January 20, 2003. The inspectors reviewed the applicant's engineering evaluations, documentation of the portions of the systems added to scope, and selected layout markup drawings and discussed

the process with the cognizant individuals responsible for the evaluations. Additionally, the NRC inspectors performed walkdowns of selected areas of the plant containing SSCs of interest. The inspection team identified one item which should be considered by the applicant for inclusion within scope based on the 10 CFR 54.4(a)(1) criterion. Inspection Open Item 50-285/02-07-02 identified unqualified safety injection tank level and pressure indicators that should be considered in the scope of license renewal. These indicators are used to ensure that assumptions are met for the mitigation of a loss of coolant accident analysis. The applicant reviewed this issue and committed to include these components within scope.

3.0.3.12.2-1 During the staff's AMR inspection, the applicant committed to revise the general corrosion of external surfaces program to include the spent fuel pool cooling system.

4.3.2-1 Section 4.3.2 of the LRA contains a discussion of the proposed aging management program to address fatigue of the FCS pressurizer surge line. The discussion indicates the aging management program will consist of an inspection program. The LRA also indicates that the results of the surge line inspections will be used to assess the appropriate approach for addressing environmentally-assisted fatigue of the surge lines. However, Section 4.3.3 of the LRA indicates that a reevaluation of the fatigue usage of critical areas of the surge line will be performed prior to the period of extended operation and that the bounding locations will be included in the FMP. In RAI 4.3.2-3, the staff requested that the applicant describe how the effect of the reactor water environment will be considered in the reevaluation of the critical areas of the surge line and how the results of this evaluation will be monitored by the FMP.

The applicant's December 19, 2002, response indicated that the limiting surge line welds would be inspected prior to the period of extended operation. The applicant further indicated the results of these inspections will be used to assess the appropriate approach for addressing environmentally-assisted fatigue of the surge lines. The applicant indicated that the approach would include one or more of the following options:

1. further refinement of the fatigue analysis to lower the CUF(s) to below 1.0
2. repair of the affected locations
3. replacement of the affected locations
4. management of the effects of fatigue by an inspection program that has been reviewed and approved by the NRC (e.g., periodic nondestructive examination of the affected locations at inspection intervals to be determined by a method accepted by the NRC)

The applicant committed that, if Option 4 is selected, it will provide the inspection details, including scope, qualification method, and frequency to the NRC staff for review and approval prior to the period of extended operation. An AMP under this option would be a departure from the design basis CUF evaluation described in the USAR Supplement, and therefore would require a license amendment

pursuant to 10 CFR 50.59.

4.3.2-2 Section 4.3.4 of the LRA contains a discussion of the analysis of Class II and III components at FCS. American National Standards Institute (ANSI) B31.1 requires that a reduction factor be applied to the allowable bending stress range if the number of full-range thermal cycles exceeds 7000. The LRA indicates that the USAS B31.1 limit of 7000 equivalent full-range cycles may be exceeded during the period of extended operation for the nuclear steam supply system (NSSS) sampling system and that the affected portions of the NSSS sampling system would be tracked by the FMP. In RAI 4.3.4-1, the staff requested that the applicant provide the calculated thermal stress range for these affected portions of the NSSS sampling system.

The applicant's December 12, 2002, response indicated that the small-bore piping at FCS was designed and supported based on nomographs developed in accordance with the USAS B31.1 code. As a consequence, there were no specific stress calculations. The applicant committed that, as part of the FMP, the sampling piping will be analyzed and a stress calculation performed to determine the thermal stress range for the line. The applicant should confirm that the results, when completed, will meet USAS B31.1.

1.7 Summary of Proposed License Conditions

As a result of the staff's review of the FCS application for license renewal, including the additional information and clarifications submitted subsequently, the staff identified two proposed license conditions. The first license condition requires the applicant to include the USAR Supplement in the next USAR update required by 10 CFR 50.71(e) following issuance of the renewed license. The second license condition requires that the future inspection activities identified in the USAR Supplement be completed prior to the period of extended operation.