Safety Evaluation Report

with Open Items Related to
the License Renewal of
St. Lucie Nuclear Plant, Units 1 and 2

Docket Nos. 50-355 and 50-389

Florida Power & Light Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

February 2003

ABSTRACT

This document is a safety evaluation report regarding the application to renew the operating licenses for St. Lucie Nuclear Plant, Units 1 and 2, which was filed by the Florida Power and Light Company by letter dated November 29, 2001, and received by the NRC on November 30, 2001. The Office of Nuclear Reactor Regulation has reviewed the St. Lucie Nuclear Plant, Units 1 and 2, 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 November 29 2002, the Florida Power and Light Company requested renewal of the St. Lucie , Units 1 and 2, operating licenses (License Nos. DPR-67 and NFP-16, respectively), which were issued under Section 104b of the (Atomic Energy Act of 1954, as amended, for a period of 20 years beyond the current license expiration dates of March 1, 2016, and April 6, 2023, respectively. The St. Lucie Nuclear Plant, Units 1 and 2, are located on Hutchison Island in St. Lucie County, Florida. Each unit consists of Combustion Engineering pressurized-water reactor nuclear steam supply system designed to produce a core thermal power output of 2,700 megawatts or approximately 890 megawatts electric.

The NRC license renewal project manager for St. Lucie Nuclear Plant, Units 1 and 2, is Noel Dudley. Mr. Dudley may be contacted by calling 301-415-1154 or by writing to the License Renewal and Environmental Impacts Program, U.S. Nuclear Regulatory Commission, Washington, DC 20555-001.

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ABBREVIATIONS

AAC	Alternate AC
AC	Alternating Current
ABHVS	Auxiliary Building Heating and Ventilation System
ACI	American Concrete Institute
ACRS	Advisory Committee on Reactor Safeguards
AMP	Aging Management Program
AMR	Aging Management Review
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	Anticipated Transient Without Scram
A600IP	Alloy 600 Inspection Program
B&W	Babcock and Wilcox
BAWSP	Boric Acid Wastage Surveillance Program
BL	Bulletin
BWST	Borated Water Storage Tank
CASS	Cast Austenitic Stainless Steel
CBA	Core barrel assembly
CEOG	Combustion Engineering Owners' Group
CEA	Control Element Assembly
cfm	Cubic Feet per Minute
CFR	U. S. Code of Federal Regulations
CLB	Current Licensing Basis
CRDM	Control Rod Drive Mechanism
CRVS	Control Room Ventilation System
CSB	Core Support Barrel
CST	Condensate Storage Tank
CUF	Cumulative Usage Factor
C_vUSE	Charpy V-Notch Upper Shelf Energy
DBA	Design Basis Accident
DBD	Design Basis Document
DBE	Design Basis Events
DC	Direct Current
DOE	Department of Energy
DW	Demineralized Water
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EFPY	Effective Full Power Years
EFS	Emergency Feedwater System
EFW	Emergency Feedwater
EIC	Electrical and Instrumentation Controls
EOL	End of Live
EPRI	Electric Power Research Institute\
EQ	Environmental Qualification
ESF	Engineered Safety Features

FAC	Flow-accelerated Corrosion
FERC	Federal Energy Regulatory Commission
FHA	Fuel Handling Accident
FMP	Fatigue Monitoring Program
FP	Fire Protection
FPL	Florida Power and Light Company
ft-lb	Foot Pound
GEIS	Generic Environmental Impact Statement
GL	Generic Letter
GSI	Generic Safety Issue
HELB	High-energy Line Breaks
HEPA	High-efficiency Particulate Air (Filter)
HPI	High-pressure Injection
HPSI	High Pressure Safety Injection
HVAC	Heating, Ventilation, and Air Conditioning
IASCC	Irradiation-assisted Stress-corrosion Cracking
ICW	Intake Cooling Water
IEB	Inspection and Enforcement Bulletin
IEEE	Institue of Electrical and Electronics Engineers
IGA	Intergranular Attack
IGSCC	Intergranular Stress Corrosion Cracking
IN	Information Notice
INPO	Institute of Nuclear Power Operations
IPA	Integrated Plant Assessment
ISI	Inservice Inspection
IST	Inservice Testing
J	Joule
KPa	Kilo Pascal
ksi	Kilograms per Square Inch
LBB	Leak-before-break
LEFM	Linear Elastic Fracture Mechanics
LOCA	Loss-of-coolant Accident
LOOP	Loss of Offsite Power
LPI	Low-pressure Injection
LRA	License Renewal Application
MBGS	Miscellaneous Bulk Gas Supply
MCR	Main Control Room
MCRE	Main Control Room Environment
Mev	Million Electron Volts
MFS	Main Feedwater System
MIC	Microbiologically Influenced Corrosion
MRP	Material Research Program

NDE	Nondestructive Examination
NEI	Nuclear Energy Institute
NEPA	National Environmental Policy Act
NFPA	National Fire Protection Association
NPRDS	Nuclear Plant Reliability Data System (INPO)
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSAC	Nuclear Safety Analysis Center
NUMARC	Nuclear Management and Resource Council
NUREG	NRC Technical Report Designation
n/cm^2	Neutron per Suare Centimeter
ODSCC	Outside-diameter Stress-corrosion Cracking
ppm	Parts per Million
PRVS	Penetration Room Ventilation System
PTS	Pressurized Thermal Shock
PWR	Pressurized-water Reactor
PWSCC	Primary Water Stress-corrosion Cracking
P-T	Pressure - Temperature
QA	Quality Assurance
RAI	Request for Additional Information
RAB	Reactor Auxiliary Building
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RG	Regulatory Guide
RPV	Reactor Pressure Vessel
RI-ISS	Risk-Informed Inservice Inspection
RTD	Resistance Temperature Detector
RTE	Resistance Temperature Element
RT_{NDT}	Reference Temperature (for a reactor vessel material)
RT_{PTS}	Reference Temperature (RT_{NDT} evaluated for end of life fluence)
RV	Reactor Vessel
RVH	Reactor Vessel Head
RVI	Reactor Vessel Internals
SAE	Society of Automotive Engineers
SAR	Safety Analysis Report
SBVS	Shield Building Ventilation System
SC	Structures and Components
SCC	Stress Corrosion Cracking
SER	Safety Evaluation Report
SFP	Spent Fuel Pool
SFPC	Spent Fuel Pool Cooling
SG	Steam Generator
SOC	Statement of Considerations
SOER	Significant Operating Experience Report
SRP-LR	Standard Review Plan for License Renewal

SSC	Structures, Systems, and Components
TLAA	Time-Limited Aging Analysis
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
UHS	Ultimate Heat Sink
USAS	United States of America Standards
USE	Upper-Shelf Energy
UT	Ultrasonic Testing
VDIL	Vents, Drains, and Instrument Lines
VHP	Vessel Head Penetration
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 licenses for St. Lucie Nuclear Plant, Units 1 and 2, filed by Florida Power and Light Company (hereafter referred to as FPL or the applicant).

By letter dated November 29, 2001, FPL submitted its application to the U.S. Nuclear Regulatory Commission (NRC) for renewal of the operating licenses for St. Lucie Nuclear Plant, Units 1 and 2, for an additional 20 years. The NRC received the application on November 30, 2001. The NRC staff reviewed the St. Lucie 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. The NRC's license renewal project manager for St. Lucie Nuclear Plant, Units 1 and 2, is Noel Dudley. Mr. Dudley may be contacted by calling 301-415-1154, or by writing to the License Renewal and Environmental Impacts Program, U.S. Nuclear Regulatory Commission, Washington, DC 20555-001.

In its application, FPL requested renewal of the operating licenses issued under Section 104(b) of the Atomic Energy Act of 1954, as amended, for St. Lucie Nuclear Plant, Units 1 and 2 (License Nos. DPR-67 and DPR-16, respectively), for a period of 20 years beyond the current license expiration dates of March 1, 2016, and April 6, 2023, respectively. St. Lucie Nuclear Plant, Units 1 and 2, are located on Hutchinson Island in St. Lucie County, Florida. Each unit consists of a Combustion Engineering pressurized-water reactor (PWR) nuclear steam supply system (NSSS) designed to produce a core thermal power output of 2,700 megawatts or approximately 890 megawatts electric. Details concerning the plant and the site are found in the updated final safety analysis report (UFSAR) for each unit.

The license renewal process proceeds along two tracks including a technical review of safety issues and an environmental review. The requirements for these two reviews are stated in NRC regulations 10 CFR Parts 54 and 51, respectively. The safety review is based on FPL's application for license renewal and on the applicant's answers to requests for additional information (RAIs) from the NRC staff. In meetings and docketed correspondence, FPL has also supplemented its answers to the RAIs. The public can review the license renewal application (LRA) and all pertinent information and material, including the UFSARs, at the NRC Public Document Room, 11555 Rockville Pike, Rockville, MD 20852-2738. In addition, the LRA for the St. Lucie Nuclear Plant, Units 1 and 2, and significant information and material related to the license renewal review, are available on the NRC's Web site at www.nrc.gov.

This SER summarizes the findings of the staff's safety review of the LRA for the St. Lucie Nuclear Plant, Units 1 and 2, and describes 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 staff reviewed the LRA in accordance with NRC regulations and the guidance presented in the NRC's draft "Standard Review Plan [SRP] for the Review of License Renewal Applications for Nuclear Power Plants," dated August 2000. The draft SRP was finalized and issued as NUREG-1800 in July 2001.

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 an additional 20 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 anticipated interest in license renewal and held a workshop on nuclear power plant aging. The results of the workshop led the NRC to establish a comprehensive program plan for nuclear plant aging research. 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 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 an industry-sponsored demonstration program to apply the rule to pilot plants and to 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, 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 identifying all aging mechanisms. The rule changes were intended to ensure that important structures, systems, 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.

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 of 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 based on two key principles:

1. The regulatory process is adequate to ensure that the licensing bases of 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, including those plant SSCs (1) that are safety-related, (2) whose failure could affect safety-related functions, and (3) that are relied on to demonstrate compliance with the Commission's regulations for fire protection (FP), environmental qualification (EQ), pressurized thermal shock, anticipated transients without scram (ATWS), and station blackout (SBO).

Pursuant to 10 CFR 54.21(a), the applicant must review all SSCs that are within the scope of the rule to identify structures and components (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), the applicant must demonstrate that the effects of aging will be managed in such a way that the intended function or functions 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 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 and activities for active equipment, as well as other aspects of maintaining the plant design and licensing basis, are required to continue throughout the period of extended operation.

Pursuant to 10 CFR 54.21(b), each applicant is required to submit each year following the LRA, and at least 3 months before the scheduled completion of the NRC's review of the application, an amendment to the LRA that identifies any changes to the CLB for its facilities that materially affect the contents of the LRA, including the FSAR supplements.

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 on these SSCs will be adequately managed for the period of extended operation. Pursuant to 10 CFR 54.21(c)(2), each application must provide a list of exemptions granted pursuant to 10 CFR 50.12, and that are in effect, based on the TLAAs as defined in 10 CFR 54.3. Pursuant to CFR 54.21(c)(2), each application must also provide an evaluation that justifies the continuation of these exemptions for the period of extended operation.

Pursuant to 10 CFR 54.21(d), each application is required to include a supplement to the FSAR. This supplement must contain a summary description of the programs and activities for managing the effects of aging.

In July 2001, the NRC issued Regulatory Guide (RG) 1.188, “Standard Format and Content for Applications to Renew Nuclear Power Plant Operating License”; NUREG-1800, “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants” (SRP-LR); and NUREG-1801, “Generic Aging Lessons Learned (GALL) Report.” These documents describe methods acceptable to the NRC staff for implementing the license renewal rule, as well as techniques used by the NRC staff in evaluating applications for license renewals. The draft versions of these documents were issued for public comment on August 31, 2000 (65 FR 53047). The staff assessment of public comments is being issued as NUREG-1739, “Analysis of Public Comments on the Improved License Renewal Guidance Documents.” The regulatory guide endorsed 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,” revision 3 issued in March 2001. The staff used the regulatory guide, along with the SRP, to review this application and to assess topical reports involved in license renewal as submitted by industry groups.

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. Analyses of the environmental impacts of renewing this license that must be evaluated 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 performed a plant-specific review of the environmental impacts of license renewal, including whether there is new and significant information not considered in the GEIS for St. Lucie, Units 1 and 2. A public meeting was held on April 3, 2002, near St. Lucie, Units 1 and 2, as part of the NRC’s scoping process to identify environmental issues specific to the plant. The results of the environmental review process and a preliminary recommendation on the license renewal action were documented in NRC’s draft plant-specific Supplement 11 to the GEIS, issued in October 2002.

On December 3, 2002, during the 75-day comment period for the draft plant-specific supplement to the GEIS another public meeting was held near the site. At this meeting, the staff described the environmental review process and answered questions from members of the public to assist them in formulating any comments they might have regarding the review.

Draft Supplement 11 presents the NRC’s preliminary environmental analysis associated with renewal of the St. Lucie, Units 1 and 2, operating licenses for an additional 20 years that considers and weighs the environmental effects, and alternatives available for avoiding 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 received during the scoping period, the staff made a preliminary recommendation in draft Supplement 11 to NUREG-1437 that the Commission should determine whether the adverse environmental impacts are not so great that preserving the option of license renewal for energy planning would be unreasonable.

1.3 Summary of the 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 St. Lucie Nuclear Plant, Units 1 and 2, LRA in accordance with Commission guidance and the requirements of 10 CFR 54.4, 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 a license renewal applicant to submit general information. FPL submitted this general information in an enclosure to its November 29, 2000, letter regarding the application for renewed operating licenses for St. Lucie Nuclear Plant, Units 1 and 2. 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." The applicant stated the following in its renewal application regarding this issue:

> The current indemnity agreement for St. Lucie Units 1 and 2 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 agreement, which is the last to expire. Item 3 of the Attachment to the indemnity agreement, as revised by Amendment No. 10, lists four license numbers. Should the license numbers be changed upon issuance of the renewed licenses, FPL requests that the conforming changes be made to Item 3 of the Attachment, and to any other sections of the 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 must contain (1) an IPA, (2) CLB changes during the NRC review of the application, (3) an evaluation TLAAs, and (4) an FSAR supplement. On November 29, 2001, the applicant submitted the information required by 10 CFR 54.21(a) and (c) in the enclosure of its LRA. This enclosure is entitled "Application for Renewed Operating Licenses, St. Lucie Units 1 and 2."

In 10 CFR 54.22, the Commission states requirements regarding technical specifications. The applicant did not request any changes to the plant technical specifications in 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 5), that states the considerations related to renewing the licenses for St. Lucie Units 1 and 2.

1.4 Differences in the Designs of St. Lucie Units 1 and 2

St. Lucie Unit 1 was licensed approximately 7 years before St. Lucie Unit 2. During these 7 years, significant industry events occurred including the Three Mile Island Unit 2 event and the Browns Ferry fire event. The lessons learned from these events and other activities resulted in differences between St. Lucie Units 1 and 2. Even though the units are of the same design and the systems fulfill the same functional design requirements, some of the component design features are different.

For design-basis accidents (DBA), the Unit 1 spent fuel pool (SFP) is designed to remove decay heat by means of SFP boiling. The associated Unit 1 SFP makeup systems are comprised of seismically qualified piping from the discharge headers of the two intake cooling water system loops. Other non-safety-related makeup systems are available for normal makeup to the pool. The Unit 2 spent fuel pool cooling system, which consists of two pumps and a redundant set of heat exchangers, is designed to remove decay heat from the spent fuel during DBAs. The Unit 2 SFP makeup systems are similar to the Unit 1 makeup systems.

The Unit 1 fuel handling equipment is not within the scope of license renewal, since the results of the Unit 1 UFSAR analysis of a fuel handling accident indicated that offsite exposures would be less than those referenced in 10 CFR 50.34(a)(1), 10 CFR 50.67(b)(2) and 10 CFR 100.11. Because of the predicted radiological consequences of a fuel element drop accident, Unit 2 fuel handling equipment is within the scope of license renewal.

Unit 1 was designed to protect against single missiles. To meet this design requirement, the licensee provided for redundancy and separation of SCs, or provided missile barriers around safety-related components. Unit 2 was designed to protect against multiple missiles, including vertical missiles. To meet this design requirement, the licensee enclosed the Unit 2 component cooling water area, condensate storage tank, and emergency diesel generator fuel oil storage tanks in buildings. These buildings and the Unit 1 missile barriers are within the scope of license renewal.

The Unit 1 turbine building is within the scope of license renewal since it contains two safety-related motor-operated valves and their associated power cables. The Unit 2 turbine building contains no safety-related equipment, however, the building is within the scope of license renewal because of non-safety-related equipment whose failure could prevent satisfactory accomplishment of safety-related functions.

The Unit 1 and 2 condensate storage tanks are within the scope of license renewal because they are safety-related components. The Unit 1 condensate storage tank is in an outdoor environment and is protected by a missile shield comprised of a concrete wall around the tank. The Unit 2 condensate storage tank is in an indoor - not air conditioned environment. The condensate storage tank cross-connect piping for Unit 1 is within scope of license renewal because it is a non-safety-related component whose failure could prevent satisfactory accomplishment of safety-related functions. The cross-connect piping allows operators to line up the Unit 1 auxiliary feedwater system to take a suction from the Unit 2 condensate storage tank.

The Unit 1 demineralized water system piping in the diesel generator building was not designed as seismic Category 1, however, the piping is within the scope of license renewal because the failure of the piping could prevent satisfactory accomplishment of safety-related functions. The Unit 2 demineralized water system piping in the diesel generator building was designed as seismic Category 1 and is within the scope of license renewal.

The fire protection system is common to both Units 1 and 2. The system consists of two fire pumps powered from the Unit 1 electrical system. The Halon suppression system for the cable spreading room is unique to Unit 1. The use of primary water for the hose station water supply in the containment is unique to Unit 2.

For station blackout (SBO) considerations, Unit 1 credits the Unit 2 emergency diesel generators as the alternate alternating current (AC) sources. Unit 2 is a 4-hour direct current (DC) coping plant. For Unit 1, instrument air is required to operate valves used to remove decay heat during an SBO. Therefore, the instrument air system and a portion of the turbine cooling water system are within the scope of license renewal. For Unit 2, the similar decay heat removal valves are operated by DC power and, therefore, the Unit 2 instrument air system is not within the scope of license renewal.

The Unit 1 refueling water tank is aluminum and has experienced aging degradation. The applicant identified three different programs for managing the aging effects. The Unit 2 refueling water tank is stainless steel and the applicant identified a single program for managing the aging effects. The Unit 1 spent fuel racks contain Boraflex inserts. The applicant identified a program for managing the aging of these inserts. The Unit 2 fuel racks do not contain Boraflex inserts and, therefore, the applicant did not identify any aging management programs for Unit 2.

Significant maintenance activities are listed below:

- The licensee replaced the Unit 1 steam generators in 1997.
- The licensee removed the thermal shield and repaired damage to the Unit 1 core barrel support plate in 1983.

1.5 Open Items and Confirmatory Items

1.5.1 Open Items

Open Item 3.0.2.2-1: The staff conducted an on-site AMR inspection, which included verification of the applicant’s claim that some aging management programs are consistent with the GALL Report. The inspection also verified information concerning the scoping and screening results. The inspection was completed on January 31, 2003, and a report documenting the inspection findings is pending. The inspection findings are necessary to determine the acceptability of the aging management programs that are claimed to be consistent with the GALL report. The staff is in the process of reviewing the results of the AMR inspection findings and will complete its evaluations of these aging management programs and scoping and screening results when the inspection report is issued.

Open Item 3.0.5.7-1. This item concerns the detection of wall thinning of FP piping due to internal corrosion. The applicant stated that the internal loss of material can be detected by changes in flow or pressure, leakage, or by evidence of excessive corrosion products during flushing of the system. The applicant also stated that St. Lucie plant-specific operating experience has shown that the current methods of monitoring internal conditions are adequate and reliable. In accordance with Interim Staff Guidance (ISG)-4, "Aging Management of Fire Protection Systems for License Renewal," the applicant should perform a baseline pipe wall thickness evaluation of the FP piping using a nonintrusive means, such as a volumetric inspection, before the current license term expires. Alternatively, the applicant should provide assurance that adequate wall thickness evaluations on representative piping exist such that a baseline wall thickness evaluation is not necessary.

Open Item 3.0.5.10-1. Several components in the intake cooling water system credit the Systems and Structures Monitoring Program for managing loss of material in the raw water environment. In RAI B.2.10-2, the staff asked the applicant to justify the adequacy of this program for managing the aging effects on specific components in the intake cooling water system. The staff finds the applicant's response does not adequately address the aging management of the small valves, piping/tubing/fittings, thermowells, and orifices. The applicant, in a letter dated November 27, 2002, provided additional information concerning the materials, operating history, and repair history of the small valves, piping/tubing/fittings, thermowells, and orifices in the intake cooling water system. However, the applicant also relies on leakage detection for aging management of some components. It is the staff's position that leakage detection does not provide adequate aging management because leakage indicates a loss of component intended function.

Open Item 3.1.0.1-1: A commitment is requested to implement any recommended inspection methods, inspection frequencies, and acceptance criteria that result from industry initiatives by the CEOG, NEI, or EPRI MRP Integrated Task Group on Inconel materials (including Alloy 600 and Alloy 182/82 materials) that are recommended for managing stress corrosion cracking (including PWSCC) of Inconel components, and are found acceptable by the NRC. A commitment is also requested to implement any further requirements that may result from the staff's resolution of the issue of PWSCC in nickel-based alloy components (including those that may result from the staff's resolution of the industry's responses to NRC Bulletin 2002-02, and/or resolution of the V.C. Summer issue).

Open Item 3.1.0.1-2: In Florida Power and Light (FPL) Company's response to RAI 3.2.1-1, the applicant states that the Alloy 600 Inspection Program (A600IP) includes commitments made in the applicant's responses to NRC Bulletin 2002-01 (FPL letters L-2002-061 and L-2002-116 dated April 2, 2002, and June 27, 2002, respectively) and NRC Bulletin 2002-02 (FPL letter L-2002-185 dated September 11, 2002). The responses to these Bulletins are specific to degradation that may occur in the St. Lucie reactor vessel heads (RVHs) and associated penetration nozzles and attachment welds. The responses to these Bulletins do not address degradation that may occur in nickel-based alloy components of other Class 1 RCS subsystems (such as those in the St. Lucie pressurizers, steam generators, hot legs, and reactor vessel internals). The applicant should discuss and clarify the inspection programs for the remaining Class 1 nickel-based alloy base metal and weld components (other than RVH penetration nozzles and their attachment welds), taking into account the similarities and differences between susceptibility ranking and inspection methods proposed for the components when compared with those proposed for the RVH penetration nozzles and their associated attachment welds.

Open Item 3.1.0.3-1. If the risk-informed methodologies for Small Bore Class 1 Piping Inspection AMP are part of a RI-ISI program that is required to be approved under the provisions of 10 CFR 50.55a(a)(3), the potential exists for methodologies to "screen out" the volumetric examinations of the small bore piping based on risk information and therefore eliminate the volumetric examinations proposed for the small bore Class 1 piping components. In LRA Section 18.1.5 of Appendix A1 for St. Lucie 1 and LRA Section 18.1.14 of Appendix A2 for St. Lucie 2, Florida Power and Light Company (FPL) commits to submitting the inspection plan for Class 1 small-bore piping prior to the end of the initial licensing periods for the units. When this inspection plan is submitted to the staff, the staff requests:

- The applicant confirm that the risk-informed methodologies for the Small Bore Class 1 Piping Inspection will be used only to establish the minimum number and locations of the small bore Class 1 piping full-penetration butt welds to be volumetrically examined and will not be used as a basis to eliminate the volumetric examinations for the welds.

- The applicant provide a discussion in the inspection plan describing the risk-informed methodology and addressing how the methodology has been applied to determine the locations and number of small bore piping components for inspection. Confirm that the inspection plan for the small bore piping will include this information when submitted to the staff as part of the FSAR supplements summary descriptions for the Small Bore Class 1 Piping Inspection AMP.

Open Item 3.1.0.5-1. The applicant described the Reactor Vessel Surveillance Capsule Removal and Evaluation Subprogram. In accordance with ASTM E185, for current 40-year practice, it is recommended that the last capsule to be removed should receive the same or higher fluence than the peak EOL fluence. Therefore, the applicant should provide updated capsule removal schedules that reflect a capsule to be withdrawn with a predicted fluence equal to or greater than the peak EOL fluence for the extended period of operation for St. Lucie Units 1 and 2.

Open Item 3.1.1.2-1. The applicant has not identified in Table 3.1-1 and Section 3.1.1.2 of the LRA that loss of mechanical closure integrity is an applicable effect for the stainless steel or carbon steel non-Class 1 bolting materials as a result of stress relaxation. The applicant should provide the basis for not considering stress relaxation to be an applicable aging effect mechanism for the stainless steel and carbon steel non-Class 1 bolting materials. If loss of mechanical closure integrity due to stress relaxation is considered to be an applicable effect for the stainless steel and carbon steel non-Class 1 bolting materials, the applicant should provide revised AMRs for these bolting materials to reflect that loss of mechanical closure integrity is an applicable effect for these bolting materials and propose an applicable inspection-based AMP to manage loosening of the bolts during the extended periods of operation for the St. Lucie units.

Open Item 3.1.2.2-1. The pressurizer surge and spray nozzle thermal sleeves are fabricated from Alloy 600 materials and are welded to the low-alloy steel pressurizer surge and spray nozzles using Alloy 182/82 weld metals. Industry experience has demonstrated that these weld materials are susceptible to PWSCC. In its AMR provided October 3, 2002, the applicant concluded that there are no applicable aging effects for the pressurizer surge and spray nozzle thermal sleeves because the applied loads on the thermal sleeves are low. The attachment welds for the pressurizer surge and spray nozzle thermal sleeves may contain high residual stresses that result from solidification of the weld metal from the molten state. Therefore, the staff concludes that the attachment weld for the pressurizer surge and spray nozzle thermal sleeves may be susceptible to cracking as a result of PWSCC, and that the applicant's supplemental AMR for the pressurizer thermal sleeves needs to be revised to include cracking as an applicable effect for the components.

Open Item 3.6.2.1-1. Operating experience, as discussed in NUREG-1760, "Aging Assessment of Safety-Related Fuses Used in Low- and Medium-Voltage Applications in Nuclear Power Plants," identified that aging stressors such as vibration, thermal cycling, electrical transients, mechanical stress, fatigue, corrosion, chemical contamination, or oxidation of the connections surfaces can result in fuse holder failure. On this basis, fuse holders, including both the insulation material and the metallic clamps are subject to both an AMR and AMP for license renewal. Typical plant effects observed from fuse holder failure due to aging have resulted in challenges to safety systems, cable insulation failure due to over-temperature, failure of the containment spray pump to start, a reactor trip, etc. Therefore, managing age-related failure of fuse holders would have a positive effect on the safety performance of a plant. Information Notices 91-78, 87-42, and 86-87 are examples that underscore the safety significance of fuse holder and the potential problems that can arise from age-related fuse holder failure. The staff disagreed with the applicant that there were no aging effects requiring management for fuse holders.

Open Item 4.6.4-1. The staff is in the process of reviewing Topical Report WCAP-15973-P; Class 2 Proprietary Calculation CN-CI-02-60; and the applicant's January 8, 2003, relief request for the St. Lucie half nozzle designs. These documents represent the most up-to-date current licensing basis (CLB) for the TLAA on the St. Lucie alloy 600 half-nozzle repairs. The acceptability of TLAA 4.6.4 is pending acceptable approval of these documents. The FSAR Supplement summary descriptions for TLAA 4.6.4, "Alloy 600 Instrument Nozzle Repairs," as given in Sections 18.3.8 of LRA Appendix A1 and 18.3.7 of LRA Appendix A2, do not currently reflect that these documents are part of the CLB for the TLAA on the alloy 600 instrument nozzle repairs. In order to ensure that the FSAR Supplement summary descriptions for this TLAA are up to date, the applicant should supplement the FSAR Supplement summary descriptions, as given in Section 18.3.8 of Appendix A1 and Section 18.3.7 of Appendix A2 to the LRA, to include a reference to Topical Report WCAP-15973-P; Class 2 Proprietary Calculation CN-CI-02-60; and the January 8, 2003, relief request for St. Lucie half nozzle designs.

1.5.2 Confirmatory Items

Confirmatory Item 2.3.3.7-1. On October 3, 2002, the applicant provided a response to RAI 2.3.3-4 concerning the spent fuel pool makeup lines from the intake cooling water system. At the request of the staff, the applicant agreed to remove the paragraphs in its response that assessed the plant design for the spent fuel pool makeup lines from the intake cooling water system and to state that the makeup lines meet the scoping requirement of 10 CFR 54.4(a)(1).

Confirmatory Item 3.0.2.2-1. The applicant claims that several of its aging management programs (AMPs) are consistent with specific AMPs in the Generic Aging Lessons Learned (GALL) Report. In Appendix B of the LRA, the applicant describes the AMPs that are consistent with the GALL Report and identifies the specific GALL Report AMPs. However, the information concerning the specific GALL Report AMPs is not included the FSAR supplements in Appendix A of the LRA. The applicant agreed to include a reference to specific GALL Report AMPs in the FSAR supplements concerning the AMPs that are consistent with the GALL Report.

Confirmatory Item 3.0.5.1-1. Section 18.1.2 of Appendix A1 and Section 18.1.1 of Appendix A2 of the LRA provide the applicant's FSAR supplement for the Galvanic Corrosion Susceptibility Inspection Program at St. Lucie. The program descriptions are consistent with the material contained in Section 3.1.2 of Appendix B of the LRA, with the exception of the areas of Acceptance Criteria and Inspection Technique. The applicant needs to revise the FSAR supplements to describe these two attributes consistent with the SER.

Confirmatory Item 3.0.5.4-1. Section 18.2.4 of Appendix A1 and Section 18.2.3 of Appendix A2 of the LRA provide the applicant's FSAR supplement for the Boric Acid Wastage Surveillance Programs at St. Lucie. The staff reviewed the sections to verify that the information in the FSAR supplement provides an adequate summary of the program activities required by 10 CFR 54.21(d). The staff identified that the applicant needs to modify the FSAR supplement descriptions of the Boric Acid Wastage Surveillance Program to include portions of the waste management system within the scope of license renewal. The applicant needs to revise the FSAR supplements to describe these changes consistent with the SER.

Confirmatory Item 3.1.0.1-1. Sections 18.2.1 of Appendices A1 and A2 of the LRA provide the applicant's FSAR supplement for the A600IP. The program descriptions are consistent with the material contained in Section 3.2.1 of Appendix B to the LRA, with possible exceptions in the areas of Detection of Aging Effects, Monitoring and Trending, and Acceptance Criteria. These may be revised by the applicant's responses to Open Items 3.1.0.1-1 Parts 1 and 2. The applicant needs to revise the FSAR supplements to describe these attributes consistent with the SER.

Confirmatory Item 3.1.0.3-1. Section 18.1.5 of Appendix A1 and Section 18.1.4 of Appendix A2 of the LRA provide the applicant's FSAR supplement for the Small Bore Class 1 Piping Inspection AMP. The program descriptions are consistent with the material contained in Section 3.1.5 of the appendix to the LRA, with possible exceptions in the areas of Detection of Aging Management Effects and Monitoring and Trending. These program attributes may be revised by the applicant's responses to Open Item 3.1.0.3-1 and 3.1.0.3-2. The applicant needs to revise the FSAR supplements to describe these attributes consistent with the SER.

Confirmatory Item 3.6.2.1-1. The applicant committed to provide a description of Non-EQ Cables and Connections AMP to be added in the FSAR supplements in Appendix A of the LRA.

Confirmatory Item 4.3.1-1. The applicant stated that the Inservice Inspection Program would be used to manage the aging of the pressurizer surge line during the period of extended operation. The applicant plans to use the results of the Inservice Inspection Program to develop an approach for addressing environmental assisted fatigue of the surge line. If the applicant selects the approach of using an inspection program, the inspection details including scope, qualification, method, and frequency shall be provided to the NRC for review prior to the period of extended operation. The staff finds that the applicant's proposed options are acceptable to address environmentally assisted fatigue of the pressurizer surge lines during the period of extended operation in accordance with 10 CFR 54.21(c)(1). However, in accordance with 10 CFR 54.21(d), these options need to be included in the FSAR supplements.