18.0 LICENSE RENEWAL - AGING MANAGEMENT PROGRAMS AND ACTIVITIES

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18.1 INTRODUCTION

18.1.1 BACKGROUND

Renewed operating licenses for Joseph M. Farley Nuclear Plant (FNP) Units 1 and 2 were issued on May 12, 2005, extending the original licensed operating term by 20 years. FNP Units 1 and 2 will enter the period of extended operation on June 26, 2017 and April 1, 2021 for Units 1 and 2, respectively.

18.1.1.1 License Renewal Rule and Process

10 CFR Part 54, the license renewal rule, establishes the procedures, criteria, and standards governing nuclear plant license renewal.

Plant systems, structures, and components (SSCs) within the scope of license renewal are defined in 10 CFR 54.4(a) as:

- Safety-related SSCs (i.e., perform a safety-related function as defined in 10 CFR 54.4(a) (1)).
- Nonsafety-related SSCs whose failure could prevent satisfactory accomplishment of safety-related functions.
- All SSCs relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commission's regulations for fire protection (10 CFR 50.48), environmental qualification (10 CFR 50.49), pressurized thermal shock (10 CFR 50.61), anticipated transients without scram (10 CFR 50.62), and station blackout (10 CFR 50.63).

The license renewal rule focuses on managing the effects of aging on the passive intended functions of long-lived structures and components, and on evaluation of time-limited aging analyses (TLAA), as defined in 10 CFR 54.21. (See paragraph 18.1.1.3 for a discussion of the definition of a TLAA.)

The license renewal rule generically excludes structures and components associated only with active functions from an aging management review. Functional degradation resulting from the effects of aging on active functions is more readily determinable and detectable, and existing programs and regulatory requirements are expected to directly detect the effects of aging. The license renewal rule credits the continued applicability of existing programs and regulatory requirements, and the maintenance rule requirements (10 CFR 50.65), to monitor the performance and condition of systems, structures, and components that perform active functions.

The license renewal process includes the identification of systems, structures, and components within the scope of the license renewal rule, determining the in-scope structures and components subject to aging management review (i.e., are passive and long-lived), and assuring the effects of aging on the intended functions are adequately managed through the identification and/or development of various aging management programs and activities. The process also includes the identification and evaluation of TLAAs, including any exemptions containing TLAAs.

The license renewal rule and the renewed operating licenses require that a summary description of the aging management programs and activities and the TLAA evaluations become part of the FSAR. To meet this requirement, sections 18.2 through 18.5 are incorporated into the FSAR. After issuance of the renewed license, 10 CFR 54.37(b) requires that, for newly identified systems, structures, and components that would have been subject to aging management review or evaluation of TLAAs in accordance with 10 CFR 54.21, the FSAR be updated to describe how the effects of aging will be managed such that the intended functions(s) in 10 CFR 54.4(b) will be effectively maintained during the period of extended operation.

18.1.1.2 Aging Management Programs

The NRC, in the Standard Review Plan for License Renewal (NUREG-1800), Appendix A.1, “Aging Management Review – Generic (Branch Technical Position RLSB-1),” describes the elements of an acceptable aging management program to the NRC Staff. Additionally, NUREG-1801, “Generic Aging Lessons Learned Report,” describes aging management programs that have been found acceptable to the NRC Staff to manage the aging effects of SSCs for license renewal.

In support of the NRC’s license renewal application review process, the FNP aging management programs are evaluated for consistency with the corresponding programs described in NUREG-1801, when applicable. A program is considered reasonably and materially consistent with NUREG-1801 when it meets the key elements of the attributes described for that program. The FNP programs are identified as being consistent with, or consistent with exceptions to, the corresponding program(s) described in NUREG-1801 or as plant specific. Program consistency with NUREG-1801 means the program is consistent with a program described in Revision 0 of NUREG-1801, unless otherwise specified.

In many cases, programs and activities existing at the time of the license renewal application were found adequate for managing aging for the period of extended operation. In some cases,

the existing programs or activities required some degree of enhancement. Also, some new programs and activities were identified. It is important to note that only a portion of certain programs or activities may be relied upon for managing the effects of aging under the license renewal rule.

More than one program or activity may be credited to manage aging in a single system, structure, or component. Conversely, in other cases, one program or activity may manage the effects of aging in multiple systems.

18.1.1.3 Time-Limited Aging Analyses

The license renewal rule requires that TLAA be evaluated to capture certain plant-specific aging analyses explicitly based on the original 40-year operating life of the plant. In addition, the Rule requires that any exemptions based on TLAAs be identified and analyzed to justify extension of those exemptions through the renewal term.

TLAA evaluations are defined by the license renewal rule in 10 CFR 54.3 as those calculations and analyses that meet all of the following six criteria:

- Involve SSCs within the scope of license renewal.
- Consider the effects of aging.
- Involve time-limited assumptions defined by the operating term, e.g., 40 years.
- Were determined to be relevant in making a safety determination.
- Involve conclusions or provide the bases for conclusions related to the capability of the SSC to perform its intended functions, as delineated in the Rule.
- Are contained or incorporated by reference in the current licensing basis.

Once a TLAA has been identified, the Rule in 10 CFR 54.21 (c) requires it to be dispositioned by one of the following three specific criteria:

- The analyses remain valid for the period of extended operation.
- The analyses have been acceptably projected to the end of the period of extended operation.
- The effects of aging on the intended functions(s) will be adequately managed (e.g., programs or activities are in place) for the period of extended operation.

After the renewed license has been issued, 10 CFR 54.37 (b) requires that any newly identified calculations or analyses that would have been a TLAA be evaluated and a summary description placed in the FSAR.

18.1.2 AGING MANAGEMENT PROGRAMS

The following programs are credited to manage the effects of aging during the period of extended operation for license renewal and are described in section 18.2 as listed below:

- Inservice Inspection Program (Including Subsections IWB, IWC, IWD, IWE, IWL, and IWF) (18.2.2).
- Water Chemistry Control Program (18.2.3).
- Service Water Pond Dam Inspection Program (18.2.4).
- Reactor Vessel Surveillance Program (18.2.5).
- Boric Acid Corrosion Control Program (18.2.6).
- Overhead and Refueling Crane Inspection Program (18.2.7).
- Steam Generator Program (18.2.8).
- Flow Accelerated Corrosion Program (18.2.9).
- Fuel Oil Chemistry Control Program (18.2.10).
- Structural Monitoring Program (18.2.11).
- Service Water Program (18.2.12).
- Fire Protection Program (18.2.13).
- Reactor Vessel Internals Program (18.2.14).
- Flux Detector Thimble Inspection Program (18.2.15).
- External Surfaces Monitoring Program (18.2.16).
- Buried Piping and Tank Inspection Program (18.2.17).
- One-Time Inspection Program (18.2.18).

- Nickel Alloy Management Program (18.2.19).
- Non-EQ Cables Program (18.2.20).
- Periodic Surveillance and Preventive Maintenance Activities (18.2.21).
- Selective Leaching Program (18.2.22).

18.1.3 AGING MANAGEMENT PROGRAMS – TIME LIMITED AGING ANALYSES (TLAA)

The aging management programs credited for managing the associated TLAAs during the period of extended operation are described in section 18.3 as listed below:

- Environmental Qualification Program (18.3.1).
- Fatigue Monitoring Program (18.3.2).

18.1.4 TLAA EVALUATIONS

The evaluation of TLAAs for the period of extended operation is provided in section 18.4. The TLAAs evaluated for the period of extended operation are listed below:

- Reactor Vessel Neutron Embrittlement Analyses (18.4.1).
- Metal Fatigue Analysis (18.4.2).
- Containment Tendon Pre-Stress Analysis (18.4.3).
- Environmental Qualification calculations (18.4.4).
- Ultimate Heat Sink Silting calculations (18.4.5).
- Leak-Before-Break Analysis (18.4.6).
- RHR Relief Valve Capacity Verification calculations (18.4.7).

REFERENCES

1. NUREG-1800, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, U.S. Nuclear Regulatory Commission, Initial Report, July 2001.

2. NUREG-1801, Generic Aging Lessons Learned (GALL) Report, U.S. Nuclear Regulatory Commission, Initial Report, July 2001.

3. Joseph M. Farley Technical Specifications, Units 1 and 2.

4. NRC Interim Staff Guidance (ISG)-04, Aging Management of Fire Protection Systems for License Renewal, December 3, 2002.

5. NRC Interim Staff Guidance (ISG)-15, Interim Staff Guidance on the Identification and Treatment of Electrical Fuse Holders for License Renewal, March 10, 2003.

18.2 AGING MANAGEMENT PROGRAM DESCRIPTIONS

18.2.1 QUALITY ASSURANCE REQUIREMENTS

The FNP Operations Quality Assurance Program will apply the quality assurance criteria of 10 CFR 50, Appendix B to the program elements of corrective actions, confirmation process, and administrative controls for the license renewal aging management program activities (described in sections 18.2 and 18.3) and their implementing documents during the period of extended operation. These criteria will be applied to the license renewal aging management activities for all safety-related and nonsafety-related structures and components that perform an intended function for license renewal.

18.2.1.1 Corrective Action

The FNP Corrective Actions Program is initiated following the identification of conditions adverse to quality, and documented as required by appropriate procedures. Various processes are used to identify problems requiring corrective action. The primary vehicle for initiating corrective action is the condition reporting process described in the Corrective Action Program.

18.2.1.2 Confirmation Process

Condition reports are reviewed to determine regulatory reportability and significance. Those items determined to be significant conditions adverse to quality are also reviewed by site management. Corrective actions taken for significant items are reviewed for assurance that appropriate action has been taken.

18.2.1.3 Administrative Controls

Activities affecting quality are prescribed by documented instructions, procedures, or drawings of a type appropriate to the condition and are accomplished in accordance with these instructions, procedures, or drawings. They contain appropriate acceptance criteria and documentation requirements for determining whether important activities have been satisfactorily accomplished. Procedures establish review and approval requirements.

18.2.2 INSERVICE INSPECTION PROGRAM

The Inservice Inspection Program will be implemented during the period of extended operation in accordance with 10 CFR 50.55a, which imposes the inservice inspection (ISI) requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, for Class 1, 2, and 3 (Subsections IWB/IWC/IWD) pressure-retaining components and their integral attachments, containment and integral attachments (Subsections IWE/IWL), and the applicable component supports (Subsection IWF). In addition, Farley Class 1 and 2

piping weld examinations are performed per an NRC staff-approved risk-informed ISI program (Examination Categories B-F, B-J, C-F-1, and C-F-2).

The continued implementation of applicable 10 CFR 50.55a requirements, with approved alternatives and relief requests, provides reasonable assurance that the aging effects are managed such that the systems and components within the scope of the program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

This program is consistent with 10 attributes of the collection of acceptable programs described in NUREG-1801 Sections XI.M1, XI.M3, XI.M12, XI.S1, XI.S2, XI.S3, and XI.S4 with the clarification that exceptions to ASME Code requirements granted by approved alternatives or relief requests are not considered to be exceptions to the NUREG-1801 aging management program criteria.

18.2.3 WATER CHEMISTRY CONTROL PROGRAM

The Water Chemistry Control Program will manage aging during the period of extended operation through maintenance of low levels of detrimental impurities and the use of chemical additives.

The Primary Water Chemistry Control Program is based upon the guidance provided in the EPRI PWR Primary Water Chemistry Guidelines (Volumes 1 & 2).

The Secondary Water Chemistry Control Program is based upon the guidance provided in the EPRI PWR Secondary Water Chemistry Guidelines.

The Closed Cooling Water Chemistry Control Strategic Plan is based upon the guidance contained in the EPRI Closed Cooling Water Chemistry Guideline.

This program is consistent with the 10 attributes of the aging management program described in NUREG-1801, Section XI.M2. It is also consistent with the 10 attributes of the aging management program described in Section XI.M21A.

18.2.4 SERVICE WATER POND DAM INSPECTION PROGRAM

The service water pond dam and spillway will be inspected during the period of extended operation on a periodic basis in accordance with Nuclear Regulatory Commission (NRC) Regulatory Guide 1.127, Rev. 1, “Inspection of Water-Control Structures Associated with Nuclear Power Plants.” The service water pond dam inspection performed in accordance with Regulatory Guide 1.127 is an acceptable basis for inservice inspection and surveillance of the dam, its slopes, and associated spillway. The service water pond dam inspection(s) include the earthen dam, the service water pond embankments, and the spillway slopes.

This program is consistent with the 10 attributes of the aging management program described in NUREG-1801, Section XI.S7.

18.2.5 REACTOR VESSEL SURVEILLANCE PROGRAM

The Reactor Vessel Surveillance Program will be used to predict changes in reactor vessel beltline material fracture toughness during the period of extended operation. The program will be used to evaluate neutron embrittlement through surveillance capsule testing and evaluation, fluence calculations and benchmarking, and monitoring of effective full power years (EFPYs). For fluence calculations, FNP uses Regulatory Guide 1.190, which provides for a “best estimate” fluence calculation.

All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of ASTM E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage must be maintained for future insertion.

For each unit, FNP has removed all surveillance capsules and installed alternative dosimetry to monitor neutron fluence on the reactor vessel.

This program is consistent with the attributes of the aging management program described in NUREG-1801, Section XI.M31, with the exception of the surveillance capsule removal schedule.

18.2.6 BORIC ACID CORROSION CONTROL PROGRAM

The Boric Acid Corrosion Control Program implements the plant-specific commitments made in response to NRC Generic Letter 88-05 and subsequent NRC communications on boric acid corrosion and leakage detection which include NRC Bulletins 2001-01, 2002-01, 2002-02, 2003-02, and NRC Order EA-03-009 (as revised). The program is applicable to areas where there are carbon steel and low-alloy steel structures or components, or electrical components, on which borated reactor water might leak.

This program is consistent with the 10 attributes of the aging management program described in NUREG-1801, Section XI.M10.

18.2.7 OVERHEAD AND REFUELING CRANE INSPECTION PROGRAM

The Overhead and Refueling Crane Inspection Program will be used during the period of extended operation to manage the effects of general corrosion of the crane bridge and trolley structural girders and beams and the crane rails and support girders for the reactor cavity manipulator, spent-fuel bridge, spent-fuel cask, and the containment polar cranes. The contacting surfaces of the steel rails of these components will be periodically inspected in accordance with plant procedures.

This program is consistent with the 10 attributes of the aging management program described in NUREG-1801, Section XI.M23.

18.2.8 STEAM GENERATOR PROGRAM

The Steam Generator Program used to perform replacement steam generator tube surveillance in accordance with the Technical Specifications will be continued during the period of extended operation. The program includes monitoring of steam generator secondary side internal components the failure of which could prevent the steam generator from fulfilling its intended safety-related function. The program is based upon NEI 97-06, “Steam Generator Program Guidelines” or its successors.

This program is consistent with the 10 attributes of the aging management program described in NUREG-1801, Section XI.M19.

18.2.9 FLOW ACCELERATED CORROSION PROGRAM

Flow Accelerated Corrosion Program activities include, but are not limited to, analysis to determine susceptible locations, license renewal in-scope susceptible locations, baseline inspections of wall thickness, follow-up inspections, and predictive modeling techniques. These activities will provide reasonable assurance that systems will perform their intended safety function(s) during the period of extended operation.

The Flow Accelerated Corrosion Program was enhanced prior to entering the period of extended operation by adding the auxiliary feedwater pump turbine exhaust piping to the scope of the program.

This program is consistent with the 10 attributes of the aging management program described in NUREG-1801, Section XI.M17.

18.2.10 FUEL OIL CHEMISTRY CONTROL PROGRAM

The Fuel Oil Chemistry Program is governed by Technical Specifications (emergency diesel generators fuel oil systems) and the approved fire protection program (diesel-driven fire pumps fuel oil systems). It includes surveillance and maintenance procedures to mitigate corrosion as well as measures to verify the effectiveness of this aging management program and confirm the absence of an aging effect. Fuel oil quality is maintained by monitoring and controlling fuel oil contamination in accordance with the guidelines contained in selected American Society for Testing Materials (ASTM) standards.

The specific ASTM standards that FNP uses as guidelines for sampling and sample analysis are governed by the plant Technical Specifications and the approved fire protection program. Parameters important to corrosion are monitored by the FNP program, and no significant differences exist in the ability of the program to manage aging effects.

SNC has evaluated the scope of the program and improved procedural guidance for maintaining and monitoring the diesel-driven fire pump fuel oil system such that there is reasonable assurance that the system will perform its intended function during the period of extended operation.

The FNP Fuel Oil Chemistry Program is consistent with the 10 attributes of the aging management program described in NUREG-1801, Section XI.M30.

18.2.11 STRUCTURAL MONITORING PROGRAM

The FNP Structural Monitoring Program (SMP) is based upon the requirements and guidance set forth in 10 CFR 50.65 and Regulatory Guide 1.160. SNC will continue to use the SMP to monitor the condition of structures and structural components within the scope of the Maintenance Rule, thereby providing reasonable assurance that there is no loss of structure or structural component intended function during the period of extended operation. The SMP also addresses the masonry wall considerations identified in NRC IE Bulletin 80-11 and NRC Information Notice 87-67.

The FNP SMP was enhanced to include provisions to monitor structures and components during the period of extended operation which are in-scope for license renewal but are not currently monitored under the program.

This program is consistent with the 10 attributes of the aging management programs described in NUREG-1801, Sections XI.S5 and S6.

18.2.12 SERVICE WATER PROGRAM

The Service Water (SW) Program activities implement the recommendations of NRC Generic Letter 89-13. Mitigation, as well as performance and condition monitoring techniques, are used to manage fouling and loss of material in the SW system and components it serves. Collectively, these activities provide reasonable assurance that the SW system will perform its intended safety function(s) during the period of extended operation.

The scope of the SW Program was enhanced prior to the period of extended operation to include inspection of piping from the main service water header to the air compressor and the service water pump columns.

This program is consistent with the 10 attributes of the aging management program described in NUREG-1801, Section XI.M20.

18.2.13 FIRE PROTECTION PROGRAM

The Fire Protection Program will provide inspections, performance testing, monitoring, and aging management activities during the period of extended operation for water- and gas-based fire protection systems, fire dampers, fire doors, fire penetration seals, cable wrap, and fire

pump diesels (including the external surfaces of exposed fuel oil piping) requiring aging management of license renewal.

SNC implemented the following enhancements to the FNP Fire Protection Program prior to entering the period of extended operation through the use of administrative controls and procedures.

- The fire protection sprinkler system piping is subjected to wall thickness evaluations (e.g., nonintrusive volumetric testing and/or visual internal inspections during plant maintenance) at specific intervals. The plant-specific inspection interval was established from the initial inspection results and is revised as appropriate for subsequent inspection results.

- A sample of sprinkler heads are tested by using the guidance of National Fire Protection Association (NFPA) 25 (2002), Section 5.3.1.1.1, at or before 50 years service and every 10 years thereafter.

- Diesel-driven fire pump surveillance procedures have been upgraded to provide more detailed instructions related to inspection of the fuel oil supply piping.

- The current practice of replacing CO_2 hoses at 5-year intervals has been formalized in fire protection procedures.

This program is consistent with the 10 attributes of the aging management programs described in NUREG-1801, Sections XI.M26 and M.27, as amended by Interim Staff Guidance ISG-04.

18.2.14 REACTOR VESSEL INTERNALS PROGRAM

The FNP Reactor Vessel Internals Program was implemented prior to entering the period of extended operation and provides an integrated inspection program that addresses the reactor internals. It is governed by administrative controls and procedures to supplement the inspection requirements of ASME Section XI, IWB Category B-N-3 to ensure that aging effects do not result in a loss of intended function of internal components during the period of extended operation.

The program manages the effects of crack initiation and growth due to irradiation-assisted stress corrosion cracking; loss of fracture toughness due to irradiation embrittlement, thermal embrittlement, or void swelling; or changes in material properties (dimension) due to void swelling.

SNC will continue to participate in industry initiatives intended to clarify the nature and extent of aging mechanisms potentially affecting reactor vessel internals in accordance with the SNC commitment to the Nuclear Energy Institute (NEI) 03-08 Materials Initiative.[6] These initiatives resulted in industry guidance promulgated under reference 6 for operating PWR designs. SNC incorporated the results of these initiatives (to the extent that they are applicable to the FNP reactor internals) into the scope, inspection requirements (inspection locations, methods, qualifications, and frequencies), acceptance criteria, and corrective actions of the Reactor Vessel Internals Program.

SNC submitted an aging management program/plan documenting the plant-specific consistency with industry guidance and interim staff guidance LR-ISG-2011-04 for the FNP Reactor Vessel Internals for NRC review and approval at least 22 months prior to entering the period of extended operation for the FNP units.[12] Via reference 8, industry guidance was incorporated into NUREG-1801 aging management program X1.M16A "PWR Vessel Internals". Consistent with a living program, future revision to industry guidance will be incorporated as applicable into the Reactor Vessel Internals Program when issued under NEI 03-08.[6]

The FNP Reactor Vessel Internals Program is consistent with the 10 attributes of the aging management program described in LR-ISG-2011-04 which revised NUREG-1801, Section X1.M16A.

18.2.15 FLUX DETECTOR THIMBLE INSPECTION PROGRAM

The Flux Detector Thimble Inspection Program was implemented prior to entering the period of extended operation to formalize examinations already being performed. The program is administratively controlled by plant procedures. It is used to identify loss of material due to fretting/wear in the detector thimble tubes during the period of extended operation.

18.2.16 EXTERNAL SURFACES MONITORING PROGRAM

The External Surfaces Monitoring Program is a plant-specific condition monitoring program implemented prior to entering the period of extended operation. It includes periodic visual inspections of external surfaces of carbon steel, low alloy steel, and other susceptible materials in components requiring aging management for license renewal.

Plant procedures and administrative controls have been developed to provide for surface condition monitoring of selected equipment and components for signs of corrosion or wear. Periodic inspections of accessible portions of piping and tubing are performed to detect signs of loss of material, flange leakage, missing or damaged insulation, damaged coatings, and fretting of tubing.

Accessible in-scope polymers or elastomers are also inspected for loss of material, cracking, and change in material properties. Susceptible materials or components include accessible fasteners, ventilation systems seals and collars, other polymers and elastomers, copper, aluminum, and coated steel structural components which are not within the scope of the Structural Monitoring Program.

18.2.17 BURIED PIPING AND TANK INSPECTION PROGRAM

The Buried Piping and Tank Inspection Program will be used to manage the loss of material from external surfaces of in-scope pressure-retaining buried carbon steel piping and tanks and buried stainless steel and copper alloy piping during the period of extended operation. Administrative controls and procedures are put in place to ensure that buried piping and tanks are inspected when they are excavated for maintenance or when those components are exposed for any reason. This program was implemented prior to the period of extended operation.

SNC will perform an inspection of buried piping and tanks within 10 years after entering the period of extended operation, unless an opportunistic inspection has occurred within this 10-year period. Before the tenth year, SNC will perform an engineering evaluation to determine if sufficient inspections have been conducted to draw a conclusion regarding the ability of the underground coatings to protect the underground piping and tanks from degradation. If not, SNC will conduct a focused inspection to allow that conclusion to be reached.

This program is consistent with the 10 attributes of the aging management program described in NUREG-1801, Section XI.M34, with the exception that it also includes provisions for inspection of buried stainless steel and copper alloy piping.

18.2.18 ONE-TIME INSPECTION PROGRAM

The One-Time Inspection Program was implemented prior to the period of extended operation. The One-Time Inspection Program includes measures to verify the effectiveness of various other aging management programs and confirms the absence of aging effects requiring management. These measures include use of examinations of reactor coolant system small-bore (< 4-in. NPS) ASME Class 1 piping components for cracking as an indicator of the potential for stress corrosion cracking in other stainless steel components exposed to a borated water environment. Also included is a thickness measurement of the bottom of the condensate storage tank. One-time inspections are performed within a window of up to 10 years immediately preceding the period of extended operations.

The program will be administratively controlled by plant procedures. Administrative controls and procedures will be developed to identify the specific components which must be included, as well as the systems from which the remaining sample set will be collected.

This program is consistent with the 10 attributes of the aging management program described in NUREG-1801, Section XI.M32.

18.2.19 NICKEL ALLOY MANAGEMENT PROGRAM

The plant-specific Nickel Alloy Management Program was implemented prior to the period of extended operation to address the potential for primary water stress corrosion cracking (PWSCC) in nickel alloy components exposed to the reactor coolant environment. This program assessed nickel base alloy component susceptibility to PWSCC and provided for any required augmented inspection requirements to ensure that the susceptible components will be maintained within ASME acceptance criteria during the period of extended operation. Administrative controls and procedures were developed to implement the program in accordance with industry initiatives. Subsequent to this initial assessment, ASME code cases were developed and mandated by rulemaking under §10 CFR 50.55a which has superseded direct industry guidance and is applicable to the original and extended periods of operation.[(9)(10)]

The scope includes nickel base alloy reactor coolant pressure boundary components with known or potential susceptibility to PWSCC, excluding steam generator tubes, which are specifically addressed by the Steam Generator Program, and reactor internals which are addressed by the Reactor Internals Inspection Program. The scope and frequency of examinations and acceptance criteria for detected degradation not already covered by ASME Code Section XI have been incorporated into ASME Code Section XI code cases as conditioned in §10 CFR 50.55a.

FNP will continue to participate in industry initiatives (such as the PWR Owners Group and the EPRI Materials Reliability Program) in accordance with the SNC commitment to the NEI 03-08 Materials Initiative.[(6)] Susceptibility rankings and program inspection requirements are consistent with the latest version of the EPRI Materials Reliability Program safety assessment regarding Alloy 82/182 pipe butt welds or successor code cases as conditioned in §10 CFR 50.55a.

SNC submitted correspondence indicating that all plant components under the scope of the Nickel Alloy Management Program have been incorporated into the ISI Program 24 months prior to entering the period of extended operation for the FNP units.[(11)]

18.2.20 NON-EQ CABLES PROGRAM

The Non-EQ Cables Program is an inspection and testing program implemented prior to the period of extended operation. It is used to maintain the function of electrical cables and connections which are not subject to the environmental qualification requirements of 10 CFR 50.49, but are exposed to adverse localized environments caused by heat, radiation, or moisture.

The program is administratively controlled by procedures. The scope includes: 1) accessible electrical cables and connections (connectors, splices, terminal blocks, fuse holders, and electrical penetration assembly pigtails) installed in adverse localized environments caused by heat or radiation, coupled with the presence of oxygen; 2) electrical cables used in circuits with sensitive, high voltage, low-level signals such as radiation monitoring and nuclear instrumentation and; 3) inaccessible medium voltage cables that are exposed to significant moisture and voltage at the same time.

This program is consistent with the 10 attributes of the aging management programs described in NUREG-1801, Sections XI.E1 and E3, and for Section XI.32 as amended by Interim Staff Guidance ISG-15 and the alternate program drafted by the License Renewal Working Group.

18.2.21 PERIODIC SURVEILLANCE AND PREVENTIVE MAINTENANCE ACTIVITIES

The periodic surveillance and preventive maintenance activities are plant-specific periodic inspections and tests that are relied upon for license renewal to manage the aging effects applicable to the components included in the program that are not managed by other aging management programs. The periodic surveillance and preventive maintenance activities are implemented through repetitive tasks and surveillances.

The periodic surveillance and preventive maintenance activities credited for license renewal were implemented prior to the period of extended operation.

The specific items included in this program are as follows:

- Periodic visual inspection of a sample set of tank diaphragms for the boric acid tanks, reactor makeup water storage tanks, and condensate storage tanks.

18.2.22 SELECTIVE LEACHING PROGRAM

The Selective Leaching Program was implemented prior to the period of extended operation. The Selective Leaching Program includes measures to verify the integrity of components made of cast iron, bronze, brass, and other alloys exposed to raw water, treated water, or a groundwater environment that may lead to selective leaching of one of the metal components. Insofar as practical with respect to scheduled outages, the inspections for selective leaching are performed within a window of 5 years immediately preceding the period of extended operations and continue on a opportunistic basis into the period of extended operations. The aging management program includes a one-time visual inspection and hardness test to determine whether the loss of material due to selective leaching is occurring.

The program is administratively controlled by plant procedures. Administrative controls and procedures exist to identify the specific components which must be included, as well as the systems from which the remaining sample set will be collected.

The program is consistent with the 10 attributes of the aging management programs described in NUREG-1801, Section XI.M33.

REFERENCES

1. NUREG-1800, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, U.S. Nuclear Regulatory Commission, Initial Report, July 2001.

2. NUREG-1801, Generic Aging Lessons Learned (GALL) Report, U.S. Nuclear Regulatory Commission, Initial Report, July 2001.

3. Joseph M. Farley Technical Specifications, Units 1 and 2.

4. NRC Interim Staff Guidance (ISG)-04, Aging Management of Fire Protection Systems for License Renewal, December 3, 2002.

5. NRC Interim Staff Guidance (ISG)-15, Interim Staff Guidance on the Identification and Treatment of Electrical Fuse Holders for License Renewal, March 10, 2003.

6. Nuclear Energy Institute (NEI), NEI 03-08 Revision 2, “Guideline for the Management of Materials Issues,” January 2010.

7. Regulatory Issue Summary (RIS) 2011-07, “License Renewal Submittal Information for Pressurized Water Reactor Aging Management,” July 21, 2011.

8. NRC Final License Renewal Interim Staff Guidance LR-ISG-2011-04, “Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors,” noticed in Federal Register June 3, 2013.

9. Federal Register Volume 73, No. 176, September 10, 2008, “NRC Rulemaking amending §10 CFR 50.55a to incorporate by reference ASME Code Cases N-729-1 and N-722.

10. Federal Register Volume 76, No. 119, June 21, 2011, “NRC Rulemaking amending §10 CFR 50.55a to incorporate by reference ASME Code Cases N-770-1 and N-722-1.

11. Southern Nuclear letter NL-15-0336 dated June 22, 2015.

12. Southern Nuclear letter NL-15-1507 dated August 12, 2015.

18.3 AGING MANAGEMENT PROGRAMS – TIME LIMITED AGING ANALYSES (TLAA)

18.3.1 ENVIRONMENTAL QUALIFICATION PROGRAM

The Environmental Qualification (EQ) Program manages component thermal, radiation, and cyclical aging, as applicable, through the use of aging evaluations based on 10 CFR 50.49 (f) qualification methods. As required by 10 CFR 50.49, EQ components whose qualified lives expire before the end of the applicable license term are to be refurbished, replaced, or have their qualification extended prior to reaching the aging limits established in the evaluation.

This program is consistent with the attributes of the aging management program described in NUREG-1801, Section X.E1.

18.3.2 FATIGUE MONITORING PROGRAM

The design basis metal fatigue analyses for the FNP reactor coolant pressure boundary are TLAAs. The Fatigue Monitoring Program is used to monitor plant transients that are significant contributors to the fatigue cumulative usage factor. Demonstration that plant cycles have not exceeded design assumptions during the period of extended operation ensures that the design limit on fatigue usage is not exceeded. If projected plant cycles exceed a design assumption, SNC will take corrective action which may include a more refined analysis, replacement, or an inspection program approved by the NRC. As an alternative to monitoring the number of certain transients, stress-based monitoring of the plant transient may be used to compute the actual fatigue usage of each transient, at a bounding location.

The program includes monitoring for thermal stratification at susceptible locations in addition to the current transient counting required by Technical Specifications. SNC has evaluated the effects of environmentally assisted fatigue on piping and components comparable to the locations evaluated in Section 5.4 of NUREG/CR-6260. The results of that evaluation are given in paragraph 18.4.2.1.

This program is consistent with the attributes of the aging management program described in NUREG-1801, Section X.M1.

REFERENCES

1. NUREG-1800, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, U.S. Nuclear Regulatory Commission, Initial Report, July 2001.

2. NUREG-1801, Generic Aging Lessons Learned (GALL) Report, U.S. Nuclear Regulatory Commission, Initial Report, July 2001.

3. Joseph M. Farley Technical Specifications, Units 1 and 2.

4. NRC Interim Staff Guidance (ISG)-04, Aging Management of Fire Protection Systems for License Renewal, December 3, 2002.

5. NRC Interim Staff Guidance (ISG)-15, Interim Staff Guidance on the Identification and Treatment of Electrical Fuse Holders for License Renewal, March 10, 2003.

18.4 EVALUATION OF TIME LIMITED AGING ANALYSES (TLAA)

18.4.1 REACTOR VESSEL NEUTRON EMBRITTLEMENT ANALYSES

The reactor vessels are subjected to neutron irradiation from the core. This irradiation results in embrittlement of the reactor vessel materials. The following FNP analyses address the effects of neutron embrittlement of the reactor vessels for both units.

- Upper-Shelf energy (USE).
- Pressurized thermal shock (PTS).
- Pressure-Temperature (P-T) limits.
- Adjusted reference temperature (ART).
- Neutron fluence.

18.4.1.1 USE Calculation

Appendix G of 10 CFR Part 50 requires that the reactor vessel beltline materials must maintain a Charpy USE of no less than 50 ft-lb throughout the life of the reactor vessel.

SNC has projected the FNP analyses to the end of the period of extended operation for the limiting component of the beltline region materials. The limiting Unit 1 location has a projected end-of-life (EOL) USE of 52.8 ft-lb. For Unit 2, the limiting USE location has a projected EOL USE of 58 ft-lb. These TLAAs have been shown to be acceptable for the period of extended operation in accordance with 10 CFR 54.21 (c) (1) (ii).

18.4.1.2 PTS Calculation

The requirements of 10 CFR 50.61 provide for protection against PTS events in pressurized water reactors. The screening criterion in 10 CFR 50.61 is 270°F for plates, forgings, and axial welds and 300°F for circumferential welds. According to this regulation, if the calculated RT_{PTS} for the limiting reactor beltline materials is less than the specified screening criterion, then the vessel is acceptable with regard to the risk of vessel failure during postulated pressurized thermal shock transients.

SNC has updated the RT_{PTS} calculation for FNP Units 1 and 2 to include the period of extended operation, and has determined that the screening criteria are met for both units. The limiting material for FNP Unit 1 has a 54 EFPY RT_{PTS} value of 202.0°F. The limiting material for FNP Unit 2 has a 54 EFPY RT_{PTS} value of 214.4°F. These TLAAs have been shown to be acceptable for the period of extended operation in accordance with 10 CFR 54.21 (c) (1) (ii).

18.4.1.3 P-T Limits Calculation

Appendix G of 10 CFR Part 50 requires heatup and cooldown of the reactor pressure vessel be accomplished within established limits for P-T. Plant-specific calculations establish these limits. The calculations utilize materials and fluence data obtained through plant-specific reactor surveillance capsule programs.

The P-T limit curves that apply for the current operating conditions at FNP are included in the Pressure and Temperature Limits Report (PTLR) for each unit. When the operating conditions of each unit merit the use of a different curve, the PTLR for that unit is updated to include P-T limit curves that bound the current level of neutron embrittlement for the unit. SNC has updated the FNP P-T calculations, including the ART values, in accordance with 10 CFR 54.21 (c) (1) (ii).

18.4.1.4 ART Calculation

SNC updated the calculations to determine the ART for the critical components of the reactor vessel for 54 EFPY in accordance with 10 CFR 54.21 (c) (1) (ii). The ART values that apply for the current operating conditions at FNP are included in the PTLR for each unit. When the PTLR is updated to include P-T limit curves that bound the current level of neutron embrittlement for the unit, updated ART values are included.

18.4.1.5 Neutron Fluence Calculation

SNC updated the reactor vessel neutron embrittlement calculations including the neutron fluence calculations for the critical components of the reactor vessel for 54 EFPY in accordance with 10 CFR 54.21 (c) (1) (ii). The neutron fluence values that apply for the current operating conditions at FNP are summarized in the PTLR for each unit. When the PTLR is updated to include P-T limit curves that bound the current level of neutron embrittlement for the unit, changes in neutron fluence values are included.

18.4.2 METAL FATIGUE ANALYSIS

The thermal fatigue analyses of the FNP mechanical components have been identified as TLAAs.

18.4.2.1 ASME Section III, Class 1 Component Fatigue Analysis

Section III of the ASME Code requires a discrete analysis of the thermal and dynamic stress cycles on components that make up the reactor coolant pressure boundary. The required analysis completed for FNP incorporated a set of design transients. SNC reviewed the transient cycle assumptions and determined that the assumed transient cycles are conservative for 40 years and bounding for the period of extended operation, except in the specific cases described below.

The design basis for the FNP pressurizer surge line includes a stress analysis to ensure that cumulative fatigue usage will remain below the ASME Code allowable. SNC has evaluated the cumulative fatigue on the pressurizer surge line and will manage it during the period of extended operation using the Fatigue Monitoring Program in accordance with 10 CFR 54.21 (c) (1) (iii).

The design basis for the FNP RHR suction lines includes an analysis of the impact of thermal stratification on certain portions of these lines. The analysis meets the definition of a TLAA, pursuant to 10 CFR 54.3. In accordance with 10 CFR 54.21 (c) (1) (iii), SNC will monitor the actual transients on these lines using the Fatigue Monitoring Program described in subsection 18.3.2 to show that the assumptions used in the analysis will not be exceeded during 60 years of operation.

Thermal stratification of the pressurizer surge line and the resultant fatigue effects are similarly treated for FNP. As part of the FNP response to NRC IE Bulletin 88-11, SNC prepared an evaluation of the impact of thermal stratification on the surge line. The fatigue usage value calculated using the Fatigue Monitoring Program includes the impact of thermal stratification upon the cumulative fatigue of the surge line (demonstration in accordance with 10 CFR 54.21 (c) (1) (iii)).

SNC has evaluated the effect of environmentally assisted fatigue (EAF) for locations equivalent to those presented in Section 5.4 of NUREG/CR-6260. The application of the appropriate environmental factors from NUREG/CR-6583 resulted in an acceptable environmentally assisted fatigue adjusted value $<$ 1.0.

- Reactor vessel shell and lower head.
- Reactor vessel inlet and outlet nozzles.
- Surge line hot leg nozzle.
- Safety injection nozzle.
- Charging nozzles and alternate charging nozzles.
- Residual heat removal 6-in. RHR/SI nozzles to the RCS cold leg.

18.4.2.2 ASME Section III, Non-Class 1 Component Fatigue Analysis

For cracking due to thermal fatigue for in-scope FNP components outside the reactor coolant pressure boundary (non-Class 1), thermal stresses on piping will bound thermal stresses on other components in a system. The design of ASME Code Section III Class 2 and 3 piping systems at FNP incorporates the Code stress reduction factor for determining the acceptability of the piping design with respect to thermal stresses. Those in-scope components that are designed in accordance with ASME B31.1 Code requirements also incorporate the stress reduction factor based upon an assumed number of thermal cycles. In general, 7000 thermal cycles are assumed, leading to a stress reduction factor of 1.0 in the stress analyses. SNC evaluated the validity of this assumption for 60 years of plant operation. The results of this evaluation indicate that the 7000 thermal cycle assumption is valid and bounding for 60 years of operation. Therefore, the existing pipe stress calculations are valid for the period of extended operation in accordance with 10 CFR 54.21 (c) (1) (i).

SNC has determined that 22000 thermal cycles are assumed in the design for FNP small bore piping systems that receive a blowdown from the main steam or reactor coolant systems. This assumption has also been evaluated and determined to be bounding for 60 years of operation. For the air start system of the emergency diesel generators, 60 years of operation will produce more than 7000 thermal cycles. This piping has also been evaluated and found to be acceptable as designed. Therefore, the TLAA for these components is adequate for the period of extended operation, in accordance with 10 CFR 54.21 (c) (1) (i).

18.4.2.3 Reactor Coolant Pump Flywheel Fatigue

Westinghouse has generically analyzed the potential for cracking due to fatigue in reactor coolant pump (RCP) flywheels in WCAP-14535A and WCAP-15666. These two Westinghouse analyses are applicable to FNP. The evaluations of the growth of an assumed crack in the flywheel uses the assumption that the RCPs will experience 6000 start/stop cycles over 60 years of operation. The evaluations show that the crack growth is negligible for the flywheel model that bounds those in the RCPs at FNP. The number of start/stop cycles for the FNP RCPs is estimated to be significantly < 6000 through the period of extended operation. Therefore, these analyses are valid for FNP through the period of extended operation (demonstration in accordance with 10 CFR 54.21 (c) (1) (ii)).

18.4.3 CONTAINMENT TENDON PRESTRESS ANALYSIS

To meet the requirements on 10 CFR 50.55a (b) (2) (ix) (B), SNC used an analysis to predict the amount of residual prestress in the containment tendons for FNP. This analysis meets the definition of a TLAA. SNC performed a new analysis to estimate the amount of residual prestress on the tendons after 60 years of operation (demonstration in accordance with 10 CFR 54.21 (c) (1) (ii)).

The new calculation includes the latest measurements of containment tendon prestress taken since the plant began commercial operation. The calculation indicates that acceptable containment tendon prestress will continue to exist throughout the period of extended operation.

The minimum required prestressing forces for the vertical, hoop, and dome tendons (kip/wire) are 6.81, 6.01, and 6.35, respectively.

18.4.4 ENVIRONMENTAL QUALIFICATION CALCULATIONS

The FNP EQ program described in subsection 18.3.1 meets the requirements of 10 CFR 50.49. Aging evaluations which meet the definition of a TLAA can be found in test reports, test report evaluations (10 CFR 50.49 checklists), and calculations. Qualified service lives for the EQ components have already been determined. EQ components are tracked to determine when a component is nearing the end of its service life.

For those components that are nearing the end of their qualified service life, the EQ program has provisions for the components to be reevaluated for longer service, refurbished, requalified, or replaced. The EQ program described in subsection 18.3.1 will be continued through the extended term of operation as an aging management program in accordance with 10 CFR 54.21 (c) (1) (iii).

18.4.5 ULTIMATE HEAT SINK SILTING CALCULATIONS

The FNP ultimate heat sink (UHS) is a pond in which excessive silting could reduce the total volume of water available to maintain long-term shutdown cooling following a design basis accident. SNC conducts a regular surveillance to confirm water volume in the pond. The acceptance criteria for this surveillance involves a volume versus pond level curve that is calculated with 40-year assumptions as to the amount of silting that could occur without adversely impacting the volume of the pond and, therefore, meets the definition of a TLAA.

SNC has updated the calculations to include the pertinent depth-sounding data and to address the period of extended operation in accordance with 10 CFR 54.21 (c) (1) (ii).

18.4.6 LEAK-BEFORE-BREAK ANALYSIS

A leak-before-break (LBB) analysis has been performed for the FNP primary coolant loop and the pressurizer surge line. LBB analyses evaluate postulated flaw growth in the piping for the reactor coolant loops and the surge line. These analyses meet the definition of a TLAA.

For the primary coolant loop, SNC has updated the LBB analysis to account for the period of extended operation in accordance with 10 CFR 54.21 (c) (1) (ii). For the LBB analysis of the pressurizer surge line, SNC has determined that the current analysis is bounding for 60 years, in accordance with 10 CFR 54.21 (c) (1) (i).

18.4.7 RHR RELIEF VALVE CAPACITY VERIFICATION CALCULATION

SNC takes credit for the relief capacity of the RHR relief valves in the cold overpressure mitigation analysis for FNP. SNC performed a calculation that verifies relief valve capacity

given the safe operating P-T limit curves. The calculation adjusts the P-T limit curves to account for the flow-induced pressure drop from the beltline of the reactor vessel to the RHR relief valves. The calculation meets the definition of a TLAA. Pursuant to 10 CFR 54.21 (c) (1) (ii), SNC updated this calculation to include the calculated End-of-life extension limit curves prior to entering the period of extended operation.

REFERENCES

1. NUREG-1800, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, U.S. Nuclear Regulatory Commission, Initial Report, July 2001.

2. NUREG-1801, Generic Aging Lessons Learned (GALL) Report, U.S. Nuclear Regulatory Commission, Initial Report, July 2001.

3. Joseph M. Farley Technical Specifications, Units 1 and 2.

4. NRC Interim Staff Guidance (ISG)-04, Aging Management of Fire Protection Systems for License Renewal, December 3, 2002.

5. NRC Interim Staff Guidance (ISG)-15, Interim Staff Guidance on the Identification and Treatment of Electrical Fuse Holders for License Renewal, March 10, 2003.

6. WCAP-14535A, Topical Report On Reactor Coolant Pump Flywheel Inspection Elimination, November 1, 1996.

7. WCAP-15666, Extension of Reactor Coolant Pump Motor Flywheel Examination, October 2003.