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Chapter 18
PROGRAMS AND ACTIVITIES THAT MANAGE THE EFFECTS OF AGING

The integrated plant assessment for license renewal identified new and existing aging management programs and activities necessary to provide reasonable assurance that components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis (CLB) for the period of extended operation. The period of extended operation is defined as 20 years from the end of each units original operating license expiration date. This chapter describes these programs and activities and their planned implementation.

This chapter also discusses the evaluation results for each of the plant-specific time-limited aging analyses (TLAAs) performed for license renewal. The evaluations have demonstrated that the analyses remain valid for the period of extended operation; the analyses have been projected to the end of the period of extended operation; or that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

The NRC Safety Evaluation Report, NUREG-1766, (Reference 24) for the North Anna renewed operation licenses, identified commitments associated with the future development and enhancement of various aging management programs and activities. These commitments are compiled and listed in Appendix D of the NUREG and are provided herein as Table 18-1.

18.1 NEW AGING MANAGEMENT ACTIVITIES

The following sections provide a description of aging management programs and activities that were not in-place when the renewed operating licenses were issued for North Anna. These programs and activities were not part of the licensing basis for the original operating license period but were identified as necessary to manage aging of various station systems, structures, and components during the period of extended operation.

18.1.1 Buried Piping and Valve Inspection Activities

Prior to the period of extended operation, buried piping and valves will be inspected for the existence of aging effects (Item 1, Table 18-1). The Buried Piping and Valve Inspection Activities will include a one-time inspection of representative samples of piping and valves for different combinations of buried material and burial condition. Visual inspections will be used to detect cracking of protective coatings and loss of material from protective coatings or the substrate material.

The inspection will be completed in accordance with the schedule provided in Item 1, Table 18-1, and will include representative valves and sample lengths (i.e., several feet) of piping for each of the following combinations of material and burial conditions:

- Carbon steel, concrete encased
- Carbon steel, coated
- Carbon steel, coated, wrapped
- Carbon steel, coated, and wrapped with cathodic protection
- Stainless steel, coated, and wrapped

An engineering evaluation of the results of the buried piping and valves inspections will be performed to determine future actions. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.1.2 Infrequently Accessed Area Inspection Activities

The purpose of the Infrequently Accessed Area Inspection Activities is to provide reasonable assurance that equipment and components within the scope of License Renewal, which are not readily accessible, will continue to fulfill their intended functions during the period of extended operation (Item 9, Table 18-1). A one-time inspection will be performed in accordance with the schedule provided in Item 9, Table 18-1, to assess the aging of components and structures located in areas not routinely accessed due to high-radiation, high-temperature, confined spaces, location behind security or missile barriers, or normally flooded. The external condition of structures, supports, piping, and equipment will be determined by visual inspection. These inspections would detect the aging effect of loss of material. In addition, concrete will be inspected to detect the aging effects of loss of material, cracking, and change in material properties (Item 17, Table 18-1).

Infrequently accessed areas determined to be within the scope of license renewal and the focus of inspections within these area include:

- Reactor containment - Sump areas, cabling and supports*
- Reactor containment keyway - Leakage, structural support provided by the neutron shield tank
- Subsurface drains - Access shaft and component supports
- Cover for Containment dome plug - Structural condition
- Volume control tank cubicle - Structure, supports, and equipment

- Emergency diesel generator (EDG) exhaust bunkers - Structural condition
- Cable spreading rooms, Cable tunnels, Upper areas of emergency switchgear rooms - Cable raceways and supports*
- New fuel storage area - Supports and structure affecting spent fuel pool cooling*
- Auxiliary Building filter and ion exchanger cubicles - Structure, supports, and equipment*
- Tunnel from Turbine Building to Auxiliary Building - Structure, supports, and piping*
- Service water (SW) expansion joint vault - Supports and piping
- SW tie-in vault - Supports and piping
- Auxiliary SW valve pit - Supports and piping
- Turbine building SW valve pit - Structures, supports and piping
- SW valve house lower level - Supports, piping, and equipment
- SW pump house lower level - Supports, piping, and equipment
- Spray array structure in SW reservoir - Underwater supports*
- Auxiliary SW expansion joint vault - Supports and piping*
- Charging pump pipe chase - Structure, supports and piping*
- Auxiliary feedwater piping tunnel - Structure, supports and piping*

Note: Representative samples will be inspected in areas denoted with an asterisk.

Inspection results will be documented for evaluation and retention. Engineering evaluation assesses the severity of the visual inspection results and determines the extent of required actions or future inspections. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.1.3 Tank Inspection Activities

The purpose of the Tank Inspection Activities is to perform inspections of above ground and underground tanks to provide reasonable assurance that the tanks will perform their intended function through the period of extended operation (Item 10, Table 18-1).

A one-time inspection will be performed in accordance with the schedule provided in Item 10, Table 18-1, for specified tanks that are within the scope of license renewal and could experience aging effects. The aging effect of concern for tanks is loss of material. A representative sample of tanks will be designated for the one-time inspections in order to assess the condition of tanks that require aging management. The choice of representative tanks to be inspected is dependent on the material of construction for the tank, its contents, the foundation upon which the tank is based, and the type of coating. Visual inspections of internal and external surfaces will be performed. Volumetric examinations will be performed to look for indications of wall thinning on tanks that are founded on soil or buried. Indications of degradation will be referred for evaluation by engineering.

The following tanks will be inspected or represented by suitable replacement samples:

- EDG tanks (fuel oil, coolant, and starting air)
- AAC diesel generator tanks (fuel oil, coolant and starting air)
- Security diesel generator tank (fuel oil)
- Underground fuel oil storage tanks
- Diesel-driven fire pump fuel oil storage tanks
- Refueling water storage tanks
- Chemical addition tanks
- Emergency condensate storage tanks
- Casing cooling tanks
- SW pump house air receiver

An engineering evaluation may determine that the observed condition is acceptable or requires repair; or, in the case of degraded coatings, may direct removal of the coating, non-destructive examination of the substrate material, and replacement of the coating. Re-inspections will be dependent upon the observed surface condition, and the results of this engineering evaluation. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.1.4 Non-EQ Cable Monitoring

The purpose of the Non-EQ Cable Monitoring activities will be to perform inspections on a limited, but representative, number or accessible cable jackets and connector coverings that are utilized in non-EQ applications (Item 19, Table 18-1). In order to confirm that ambient conditions are not changing sufficiently to lead to age-related degradation of the in-scope cable jackets and connector coverings, initial visual inspections for the non-EQ application insulated power cables, instrumentation cables, and control cables (including low-voltage instrumentation and control cables that are sensitive to a reduction in insulation resistance) will be performed in accordance with the schedule provided in Item 19, Table 18-1. Visual inspection of the representative samples of non-EQ power, instrumentation, and control cable jackets and connector coverings will detect the presence of cracking, discoloration, or bulging, which could indicate aging effects requiring management. These effects could be due to high radiation, high temperature, or wetted condition environments. Subsequent inspections to confirm ambient conditions will be performed at least once per 10 years following the initial inspection. Additionally, upon issuance of final staff guidance regarding the aging management of fuse holders, this program will be revised to address the requirements of the final staff guidance (Item 26, Table 18-1).

The potentially adverse localized environment due to moisture which could lead to water-treeing in high- or medium-voltage cables that are within the scope of license renewal, is also detected by visually monitoring for the presence of water around cables. Programs utilizing periodic inspections and design features such as drains or sump pumps are used to control the cable localized environment. Cable found to be wetted for any significant period of time will be tested using an appropriate test method which has been proven to accurately assess the cable condition with regards to water treeing.

The source, intermediate, and power range neutron detector operate with high-voltage power supply in conjunction with low-voltage signal cables. The routine calibration of these detectors will be used to identify the potential existence of aging degradation in the associated cables.

Any anomalies resulting from the inspections will be dispositioned by Engineering and will consider the cable environment including the potential for moisture in the areas of the anomalies. Occurrence of an anomaly that is adverse to quality will be entered into the Corrective Action System. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable. Although age-related degradation is not expected for power, instrumentation, and control cables and connectors in their normal environments, visual inspections will provide reasonable assurance that the intended functions will be maintained.

18.2 EXISTING AGING MANAGEMENT ACTIVITIES

The following sections provide a description of aging management programs and activities that were essentially in-place when the renewed operating licenses were issued for North Anna. These programs and activities were part of the licensing basis for the original operating license period. For some programs, however, enhancements were identified during the license renewal process as necessary to manage aging of various station systems, structures, and components during the period of extended operation.

18.2.1 Augmented Inspection Activities

The purpose of the Augmented Inspection Activities is to perform examinations of selected components and supports in accordance with requirements identified in the Technical Specifications, UFSAR, license commitments, industry operating experience, and good practices for the station. Augmented inspections are outside the required scope of ASME Section XI. The scope of Augmented Inspection Activities to be performed during each refueling outage is identified by Engineering in accordance with controlled procedures. Component conditions are monitored to detect degradation due to loss of material and cracking. Inspections include visual, surface, and volumetric examinations. The extent of each component inspection is defined within the Augmented Inspection Activities program description.

Augmented Inspection Activities include:

- High Energy Lines Outside of Containment (Main Steam and Feedwater)
- Reactor vessel incore detector thimble tubes
- Component supports
- Steam generator feedwater nozzles
- Reactor vessel head
- Turbine throttle valves
- Steam generator supports

The scope of augmented inspection has been revised to include the core barrel hold-down spring (Item 3, Table 18-1). The inspection addresses the aging effect of loss of pre-load. Additionally, the scope for augmented inspections has been revised to include an inspection of the pressurizer surge line connection (two welds) to the reactor coolant system hot-leg loop piping (Item 2, Table 18-1). The inspections address the aging effect of thermal fatigue failure of the weld due to environmental effects, as described in NRC Generic Safety Issue (GSI)-190. The scope, frequency, qualifications, and methods of inspection are consistent with those utilized for the Inservice Inspection Program in accordance with ASME Section XI. The initial baseline inspection will occur during the fourth inspection interval. Additional inspections will be performed during each subsequent 40-month inspection period. Industry efforts to study the environmental effects on weld thermal fatigue failure will continue to be evaluated by Dominion. If warranted, alternatives to this planned inspection (re-evaluation, replacement, or repair) will be submitted to the NRC for review.

The acceptance standards for non-destructive examinations for the Augmented Inspection Activities are consistent with guidance provided in ASME Section XI or are provided within applicable examination procedures. Evidence of loss of material, loss of pre-load, or cracking requires engineering evaluation for determination of corrective action. Occurrence of significant degradation that is adverse to quality will be entered into the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.2.2 Battery Rack Inspections

The purpose of the Battery Rack Inspections is to provide reasonable assurance of the integrity of the supports for various station batteries. Loss of material due to corrosion is the aging effect. Periodic checks of the rack integrity are performed, coincident with periodic battery inspections, to determine the physical condition of the battery support racks. The condition and mechanical integrity of the battery support racks are visually inspected to provide reasonable assurance that their function to adequately support the batteries is maintained. Visual inspections are adequate to identify degradation of the physical condition of the support racks. These inspections check for corrosion of the support rack structural members.

If any material condition deterioration is sufficiently extensive to interfere with integrity of the racks, the Corrective Action System will determine the cause and appropriate action to repair and prevent recurrence of the degradation. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.2.3 Boric Acid Corrosion Surveillance

Leakage from borated systems inside Containment creates the potential for degradation of components. Inspections are performed to provide reasonable assurance that borated water leakage does not lead to undetected loss of material from the reactor coolant pressure boundary and surrounding components. Carbon steel is particularly susceptible, but copper also can be damaged.

In Generic Letter 88-05 (Reference 2), the NRC identified concerns with boric acid corrosion of carbon-steel reactor pressure boundary components inside Containment. In response to this generic letter, activities were developed to examine primary coolant components for evidence of borated water leakage that could degrade the external surfaces of nearby structures or components, and to implement corrective actions to address coolant leakage.

Primary coolant systems inside Containment are examined for evidence of borated water leakage. An overall visual inspection of coolant system piping is performed, with particular interest in potential leakage locations. Insulated portions of the coolant systems are examined for signs of borated water leakage through the insulation by examining accessible joints and exposed surfaces of piping and equipment. Vertical components are examined at the lowest elevation. Components and connections that are not accessible are examined by looking for borated water leakage on the surrounding area of the floor or adjacent equipment and insulation. The inspection scope includes connections to the reactor coolant system from the normal coolant letdown and makeup piping, and from the emergency core cooling systems. Components that are in the vicinity of borated water leakage are also examined for damage resulting from the leakage.

When visual inspections indicate evidence of borated water leakage, an evaluation is performed to determine if degradation of the leaking component or nearby affected components has occurred; and whether the observed condition is acceptable without repair. Occurrence of significant degradation that is adverse to quality is entered into the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.2.4 Chemistry Control Program for Primary Systems

The purpose of the Chemistry Control Program for Primary Systems is to provide reasonable assurance that water quality is compatible with the materials of construction in the plant systems and equipment in order to minimize the loss of material and cracking. The Chemistry Control Program for Primary Systems creates an environment in which material degradation is minimized, therefore, maintaining material integrity and reducing the amount of corrosion product that could accumulate and interfere with equipment operation or heat transfer.

Chemistry sampling is performed and the results are monitored and trended by maintaining logs of measured parameters. Acceptability of the measurements is determined by comparison with the limits established in the Chemistry Control Program for Primary Systems. Acceptance criteria for the measured primary chemistry parameters are listed in the Chemistry Control Program for Primary Systems. The acceptance criteria reflect EPRI guidelines for parameters that have been shown to contribute to component degradation. Adherence to the guidelines minimizes the aging effects of loss of material and cracking.

Action levels are established to initiate corrective action when the established limits are approached or exceeded. Depending on the magnitude of the out-of-limit condition, plant shutdown may be performed to minimize aging effects while plant actions are being taken. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.2.5 Chemistry Control Program for Secondary Systems

The purpose of the Chemistry Control Program for Secondary Systems is to provide reasonable assurance that water quality is compatible with the materials of construction in the plant systems and equipment in order to minimize the loss of material and cracking. The Chemistry Control Program for Secondary Systems creates an environment in which material degradation is minimized, therefore, maintaining material integrity and reducing the amount of corrosion product that could accumulate and interfere with equipment operation or heat transfer.

Chemistry results are monitored and trended by maintaining logs of measured parameters. Acceptability of the measurements is determined by comparison with limits established by the Chemistry Control Program for Secondary Systems. Acceptance criteria for the measured secondary chemistry parameters are listed in the Chemistry Control Program for Secondary Systems. The acceptance criteria reflect EPRI guidelines for parameters that have been shown to contribute to component degradation. Adherence to the guidelines minimizes the aging effects of loss of material and cracking.

Action levels are established to initiate corrective action when the established limits are exceeded. Depending on the magnitude of the out-of-limit condition, power is reduced or the plant is shut down to minimize aging effects while plant actions are being taken. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.2.6 Civil Engineering Structural Inspection

The maintenance rule, 10 CFR 50.65, requires licensees to monitor the condition of structures against established goals. During the period of extended operation, the provisions of the Maintenance Rule Program will be utilized to provide reasonable assurance of the continuing capability of civil engineering structures to fulfill their intended functions. The scope of Civil Engineering Structural Inspections has been expanded to include inspections required for license renewal (Item 4, Table 18-1). The expanded scope is summarized in a Dominion report.

Annual monitoring of groundwater chemistry, including surveillance scheduling that accommodates seasonal chemistry variations (Items 16 & 28, Table 18-1), is a commitment that will be implemented in accordance with the schedule provided in Table 18-1. Fulfilling the requirements to perform annual monitoring of groundwater chemistry and account for seasonal variations is accomplished by the implementation of a periodic testing procedure that is performed quarterly. Having this Chemistry Department periodic test procedure already in place prior to the period of extended operation complies with the schedule listed in Table 18-1.

Structural monitoring inspections are visual inspections that are performed to assess the overall physical condition of the structure. For concrete structures, this includes elastomer sealant materials.

Inspections are performed by trained inspectors and include representative samples of both the interior and exterior accessible surfaces of structures. Documentation of inspection results includes a general description of observed conditions, location and size of anomalies, and the noted effects of environmental conditions. If an inaccessible area becomes accessible by such means as dewatering, excavation or installation of radiation shielding, an opportunity will exist for additional inspections. The application for the renewed operating licenses included a commitment for guidance to be provided in plant procedures in accordance with the schedule provided in Item 5, Table 18-1, to take advantage of such inspection opportunities when they arise for inaccessible areas. This commitment has been fulfilled through revised Station procedures.

A visual indication of: 1) loss of material for concrete and structural steel, 2) significant cracking for concrete and masonry walls, 3) cracking or change in material properties for elastomers, 4) loss of material or loss of form for soil, and 5) gross indications of change in material properties of concrete, each requires an engineering evaluation (Item 17, Table 18-1).

Inspections of masonry walls are included in this program. The inspections check for cracks of joints and missing or broken blocks.

The engineering evaluation of inspection results, including groundwater chemistry results, determines whether analysis, repair, or additional inspections or testing is required to provide reasonable assurance that structures will continue to fulfill their intended functions. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.2.7 Fire Protection Program

Regulatory requirements associated with fire protection systems and implementation plans are provided in 10 CFR 50.48 and 10 CFR 50, Appendix R. The Fire Protection Plan includes applicable National Fire Protection Association (NFPA) commitments and maintains compliance with NRC Branch Technical Position (BTP) 9.5-1 from the Standard Review Plan (Reference 3). Aging management concerns related to fire protection involve visual inspections of fire protection equipment and barriers, including doors, walls, floors, ceilings, penetration seals, fire-retardant coatings, fire dampers, cable-tray covers, and fire stops.

Applicable aging effects that are found by visual examination include loss of material, separation and cracking/delamination, heat transfer degradation, and change in material properties. Aging effects on piping systems (including valve bodies and pump casings) that are dry or that carry water are evaluated in the same manner as for any other mechanical system. Testing of the fire protection pumps provides indication of heat transfer degradation, and inspections of the pumps provide indication of loss of material. Verification of piping integrity to maintain a pressure boundary for the fire protection system, and the availability of water are addressed by routine plant walkdowns, by pressure/flow tests that are conducted periodically, and by the Work Control Process (Item 30, Table 18-1). Visual inspections are performed periodically for hose stations, hydrants, and sprinklers.

Provisions to replace sprinklers or test a representative sample of sprinklers that have been in service for 50 years will be incorporated into the Fire Protection Program (Item 6, Table 18-1). This task conforms to the requirements of NFPA-25, Section 2-3.1.1. If testing is performed, re-testing will be performed at 10-year intervals per NFPA-25.

Fire protection equipment is examined for indications of visible damage. Acceptable sizes for breaks, holes, cracks, gaps, or clearances in fire barriers, and acceptable amounts of sealant in penetrations are established in the inspection procedures. Any questions regarding the ability of the barrier to fulfill its fire protection function are addressed by engineering evaluation. Acceptance criteria for fire protection equipment performance tests (i.e., flow and pressure tests) are provided in the appropriate test procedures. Occurrence of significant degradation that is adverse to quality is entered into the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.2.8 Fuel Oil Chemistry

The Fuel Oil Chemistry program manages the loss of material by requiring that oil quality is compatible with the materials of construction in plant systems and equipment. Poor fuel oil quality could lead either to degradation of storage tanks or accumulations of particulates or biological growth in the tanks. The purpose of the Fuel Oil Chemistry program is to minimize the existence of contaminants such as water, sediment, and bacteria which could degrade fuel oil quality and damage the fuel oil system and interfere with the operation of safety-related equipment.

The Fuel Oil Chemistry program is an aging effects mitigation activity which does not directly detect aging effects. The Fuel Oil Chemistry guidelines address the parameters to be monitored and the acceptance limit for each parameter. The acceptance criteria reflect ASTM guidelines for parameters that have been shown to contribute to component degradation. Adherence to the guidelines mitigates the aging effect of loss of material. Parameters analyzed and found to be outside established limits will be reported to Engineering, an evaluation will be performed, and appropriate corrective actions will be taken. Occurrence of significant deviations that are adverse to quality is entered into the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.2.9 General Condition Monitoring Activities

General Condition Monitoring Activities are performed for the assessment and management of aging for components that are located in normally accessible areas. The results of this monitoring are the basis for initiating required corrective action in a timely manner. This monitoring is based on the observations that are made during focused inspections that are performed on a periodic basis. Guidance is implemented in procedures for engineers and health physics technicians regarding inspection criteria that focus on detection of aging effects during General Condition Monitoring Activities (Item 8, Table 18-1). An engineering document provides direction for performing plant walkdowns to monitor equipment conditions. The walkdowns include surveillance activities and observations of maintenance tasks. Indications of age-related degradation are monitored during these walkdowns. Additional inspection information regarding the integrity of components is provided by inspections that support the implementation of the Boric Acid Corrosion Control Program. For the health physics technicians, procedural guidance exists for performing walkdowns within the Radiological Control Area to monitor potential pressure-boundary degradation.

The external condition of supports, piping, doors, and equipment will be determined by visual inspection. General Condition Monitoring Activities are performed in three different ways:

- Inspections of radiologically controlled areas for borated water leakage
- Periodic focused inspections such as system walkdowns
- Area inspections for condition of structural supports and doors

Inspection criteria for non-ASME Section XI component supports and doors, as part of General Condition Monitoring are procedurally implemented using guidance provided in an Engineering document. Doors that require inspection for age-related degradation also are designated as EQ doors. Monitoring of the EQ doors occurs as directed by the Technical Requirements Manual (Item 7, Table 18-1). Initial inspections will be completed, using the criteria, in accordance with the schedule provided in Item 7, Table 18-1.

These inspections provide information to manage the aging effects of loss of material, change in material properties, and cracking.

The acceptance criteria for visual inspections are identified in procedures that direct the various monitoring activities. Responsibility for the evaluation of identified visual indications of aging effects is assigned to Engineering personnel. Evaluations of anomalies found during General Condition Monitoring Activities determine whether analysis, repair, or further inspection is required. Occurrence of significant degradation that is adverse to quality is entered into the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.2.10 Inspection Activities - Load Handling Cranes and Devices

The load handling cranes within the scope of license renewal are listed below:

- Containment polar cranes
- Containment jib cranes
- Containment annulus monorails
- Refueling manipulator cranes
- Fuel handling bridge crane
- New fuel transfer elevator

- Spent fuel crane
- Auxiliary Building monorails

The long-lived passive components of these cranes that are subject to aging management review include rails, towers, load trolley steel, fasteners, base plates, and anchorage. An internal inspection of representative sections of the box girders for the polar cranes will be implemented as a one-time only inspection (Item 13, Table 18-1). This inspection will be performed in accordance with the schedule provided in Item 13, Table 18-1. An engineering evaluation will determine whether subsequent inspections are required.

The Inspection Activities - Load Handling Cranes and Devices has been developed in accordance with ASME B30.2 (Reference 13) and the inspection activities for monorails are developed in accordance with ASME B30.11 (Reference 14).

The Work Control Process directs structural integrity inspections of applicable cranes which include specific steps to check (visually inspect) the condition of structural members and fasteners on the cranes, the runways along which the cranes move, and the baseplates and anchorages for the runways. The applicable aging effect is identified as loss of material. If the nature of any identified discrepancies is such that corrective action can be completed within the scope of the procedure performing the inspection, no additional corrective action may be necessary. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.2.11 Inservice Inspection (ISI) Program - Component and Component Support Inspections

The ISI Program - Component and Component Support Inspections are performed in accordance with the requirements of Subsections IWB, IWC, and IWF of ASME XI, Rules for Inservice Inspection of Nuclear Power Plant Components. For this program, the license renewal concerns with respect to Subsection IWC include only the carbon steel piping that is susceptible to high energy line breaks in the feedwater and main steam systems. Inservice Inspection requirements may be modified by applicable Relief Requests and Code Cases, which are approved by the NRC specifically for each unit. The scope and details of the inspections to be performed are contained in the individualized Inservice Inspection Plan for each unit. Each Inservice Inspection Plan is developed for a 120-month inspection interval and submitted to the NRC. The examinations required by ASME Section XI utilize visual, surface, and volumetric inspections to detect loss of material, cracking, gross indications of loss of pre-load, and gross indications of reduction in fracture toughness (which presents itself as cracking of cast-austenitic stainless steel valve bodies due to thermal embrittlement).

Dominion actively participates in the EPRI-sponsored Materials Reliability Project Industry Task Group on thermal fatigue which currently is developing industry guidance for the management of fatigue caused by cyclic thermal stratification and environmental effects. Dominion is committed to following industry activities related to failure mechanisms for small-bore piping and will evaluate changes to inspection activities based on industry recommendations (Item 11, Table 18-1). This commitment is closed based on an engineering evaluation indicating that no new industry initiatives have occurred regarding inspections of small-bore piping. The current ASME requirements remain in effect to perform visual inspections of small-bore socket welds, and visual and volumetric inspections of small-bore butt welds.

Acceptance standards for inservice inspections are identified in Subsection IWB for Class 1 components, Subsection IWC for included Class 2 components, and in Subsection IWF for component supports. Table IWB 2500-1 refers to acceptance standards listed in paragraph IWB 3500. Anomalous indications beyond the criteria set forth in the Code acceptance standards that are revealed by the inservice inspections of Class 1 components may require additional inspections of similar components in accordance with Section XI. Evidence of loss of material, cracking, and gross indications of either loss of pre-load or reduction of fracture toughness requires engineering evaluation for determination of corrective action. Occurrence of significant degradation that is adverse to quality will be entered into the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.2.12 ISI Program - Containment Inspection

The ISI Program - Containment Inspection for concrete containments and containment steel liners implements the requirements in 10 CFR 50.55a and Subsections IWE and IWL of ASME Section XI. The program incorporates applicable code cases and approved relief requests. The provisions of 10 CFR 50.55a are invoked for inaccessible areas within the Containment structure.

Loss of material is the aging effect for the containment steel liner. Surface degradation and wall thinning are checked by visual and volumetric examinations. The frequency and scope of examination requirements are specified in 10 CFR 50.55a and Subsection IWE. Loss of material, cracking and change in material properties are the aging effects for the containment concrete and are checked by visual examinations. The frequency and scope of examination requirements are specified in 10 CFR 50.55a and Subsections IWL. These inspections provide reasonable assurance that aging effects associated with the containment liner and concrete are detected prior to compromising design basis requirements. The evaluations of accessible areas provide the basis for extrapolation to the expected condition of inaccessible areas, and an assessment of degradation in such areas.

During the course of containment inspections, anomalous indications are recorded on inspection reports that are kept in Station Records. Acceptance standards for the IWE inspections are identified in ASME Section XI Table IWE 2500-1 and refer to 10 CFR 50, Appendix J. For the IWL inspections, acceptance standards are identified in ASME Section XI Table IWL 2500-1. Engineering evaluations are performed for inspection results that do not meet established acceptance standards. Occurrence of significant degradation that is adverse to quality will be entered into the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.2.13 ISI Program - Reactor Vessel

The ISI Program - Reactor Vessel is performed in accordance with the requirements of Subsection IWB of ASME XI, Rules for Inservice Inspection of Nuclear Power Plant Components. Inservice Inspection requirements may be modified by applicable Relief Requests and Code Cases, which are approved by the NRC specifically for each unit. The scope and details of the inspections to be performed are contained in the individualized Inservice Inspection Plan for each unit. Each Inservice Inspection Plan is developed and submitted to the NRC for a 120-month inspection interval. Dominion will follow industry efforts (in addition to the existing reliance on chemistry control and ASME Section XI inspections) regarding inspection of core support lugs (Item 12, Table 18-1). Industry recommendations will be considered to determine the need for enhanced inspections.

In accordance with ASME Section XI, reactor vessel components are inspected using a combination of surface examinations, volumetric examinations, and visual examinations to detect the aging effects of loss of material, cracking, gross indications of loss of pre-load, and gross indications of reduction in fracture toughness. Acceptance standards for inservice inspections are identified in Subsection IWB for Class 1 components. Table IWB 2500-1 refers to acceptance standards listed in paragraph IWB 3500. Anomalous indications that are revealed by the inservice inspections may require additional inspections of similar components, in accordance with Section XI. Evidence of aging effects requires engineering evaluation for determination of corrective action. Occurrence of significant degradation that is adverse to quality is entered into the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.2.14 Reactor Vessel Integrity Management

The scope of the Reactor Vessel Integrity Management activities is focused on ensuring adequate fracture toughness of the reactor vessel beltline plate and weld materials. Neutron dosimetry and material properties data derived from the reactor vessel materials irradiation surveillance program are used in calculations and evaluations that demonstrate compliance with applicable regulations. The Reactor Vessel Integrity Management activities includes the following aspects:

- Irradiated sample (capsule) surveillance.

- Vessel fast neutron fluence calculations.
- Measurements and calculations of nil-ductility transition temperature (RT_{NDT}) for vessel
- beltline materials.
- Measurements and calculations of Charpy Upper Shelf Energy (C_vUSE).
- Calculation of reactor coolant system pressure/temperature (P-T) operating limits, and Low Temperature Overpressure Protection System (LTOPS) setpoints.
- Pressurized thermal shock (PTS) screening calculations.

Specimen capsules were placed in each of the reactors prior to initial irradiation and contain reactor vessel plate and weld material samples. The baseline mechanical properties of reactor vessel steels are determined from pre-irradiation testing of Charpy V-notch and tensile specimens. Post-irradiation testing of similar specimens provides a measure of radiation damage. Refer to Section 5.4.3.6.

Fast neutron irradiation is the cause of radiation damage to the reactor vessel beltline. The results of surveillance capsule dosimetry analyses are used as benchmarks for calculations of neutron fluence to the surveillance capsules and to the reactor vessel beltline.

Measured values of Charpy transition temperature and C_vUSE are obtained from mechanical testing of irradiation surveillance program specimens. Measured values of transition temperature are used to determine the reference temperature for nil-ductility transition (RT_{NDT}) for the limiting reactor vessel beltline material. RT_{NDT} is a key analysis input for the determination of reactor coolant system P-T operating limits and LTOPS setpoints. Measured values of transition temperature shift are similarly utilized in PTS screening calculations required by 10 CFR 50.61. Measured values of C_vUSE are used to verify compliance with the upper shelf energy requirements of 10 CFR 50 Appendix G.

Acceptable values are established for the following parameters:

- Heatup and cooldown limits, as implemented by Technical Specifications, to ensure reactor vessel integrity.
- A PTS reference temperature that is within the screening criteria of 10 CFR 50.61.
- A fast fluence value for the surveillance capsule that bounds the expected fluence at the affected vessel beltline material through the period of extended operation.
- C_vUSE greater than limits set forth in 10 CFR 50, Appendix G.

Based on established parameters, calculations are performed to ensure that the units will remain within the acceptable values.

18.2.15 Reactor Vessel Internals Inspection

Visual inservice inspections are implemented in accordance with Category B-N-3 (Removable Core Support Structures) of ASME Section XI, Subsection IWB, to determine the possible occurrence of age-related degradation. These inspections are performed at 10-year intervals in accordance with the inspection plans submitted to the NRC. The scope of components that comprise the reactor internals includes the upper and lower core internals assemblies. This includes core support and hold-down spring components, as well as, the baffle/former bolting and barrel/former bolting. Additionally, a one-time focused inspection of the reactor vessel internals will be performed in accordance with the schedule provided in Item 14, Table 18-1. The one-time inspection will look for indications of the presence of aging effects identified in the aging management review for the reactor vessel internals. The inspection will be performed on one reactor (at either Surry or North Anna) and an engineering evaluation of results will determine the need for inspections of the other units. Dominion will remain active in industry groups, including the EPRI-sponsored Materials Reliability Project Industry Task Group, to stay aware of any new industry recommendations regarding such aging management issues as neutron embrittlement, void swelling, and the synergistic effect of thermal and neutron embrittlement of internals sub-components (Item 14, Table 18-1). If future industry developments suggest the need for an alternate inspection plan during the period of extended operation, or negate the need for a one-time inspection, then Dominion will modify the proposed inspection plan.

Visual inspections are utilized to detect loss of material and cracking; as well as, gross indications of loss of pre-load and/or reduced fracture toughness. The acceptance standards for the visual examinations are summarized in ASME Subsection IWB-3520.2, Visual Examination, VT-3. These inspections are directed to be performed with the internals assemblies removed from the reactor vessel.

Acceptance standards for Reactor Vessel Internals Inspection activities are identified in ASME Section XI, Subsection IWB. Table IWB 2500-1 identifies references to the acceptance standards listed in Paragraph IWB 3500. Anomalous indications, that are revealed to be beyond the criteria in the acceptance standards by the inservice inspections, may require additional inspections. Evidence of any component degradation requires engineering evaluation for determination of corrective action. Occurrence of significant degradation that is adverse to quality is entered into the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.2.16 Secondary Piping and Component Inspection

The purpose of the Secondary Piping and Component Inspection program is to identify, inspect, and trend components that are susceptible to the aging effect of loss of material as a result of Flow Accelerated Corrosion (FAC) in either single or two-phase flow conditions. This program has been implemented in accordance with NRC Generic Letter 89-08, Erosion/Corrosion-Induced Pipe Wall Thinning (Reference 15), and NUREG-1344, Erosion/Corrosion-Induced Pipe Wall

Thinning in U.S. Nuclear Power Plants (Reference 16), and EPRI Guideline NSAC-202L, Recommendations for an Effective Flow Accelerated Corrosion Program (Reference 17).

The scope of the Secondary Piping and Component Inspection program includes portions of the feedwater systems, the main and auxiliary steam systems, and the steam generator blowdown lines.

The Secondary Piping and Component Inspection program also includes susceptible vent and drain lines.

The identification of components and piping segments to be included in each Secondary Piping and Component Inspection effort is performed by Engineering using plant chemistry data, past inspection data, predictions from FAC-monitoring computer codes, and industry experience. Determination of whether a piping component has experienced FAC degradation is made by measuring the current wall thickness using the UT method and comparing against previous baseline thickness measurement, if available. Visual inspections of the internals of non-piping components, such as pumps and valves, are performed as the equipment is opened for other repairs and/or maintenance, to determine whether flow-accelerated degradation is occurring.

The decision to repair or replace a component is made by Engineering. For the internal surface examinations, engineering evaluations are utilized to determine whether the results of visual inspections indicate conditions that require corrective action. Occurrences of significant degradation that are adverse to quality are entered into the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.2.17 Service Water System Inspections

Compliance with Generic Letter 89-13 (Reference 18) requires a variety of inspections, non-destructive examinations, and heat transfer testing for components cooled by service water. Generic Letter 89-13 directed utilities to assess the following aspects of operational problems with service water cooling systems:

- Biofouling
- Heat Transfer Testing
- Routine Inspection and Maintenance
- Single-failure Walkdown
- Procedure Review

The SW System Inspections program provides reasonable assurance that corrosion (including microbiologically-influenced corrosion, MIC), erosion, protective coating failure, silting, and biofouling of service water piping and components will not cause a loss of intended function. The primary objectives of this program are to (1) remove excessive accumulations of biofouling agents, corrosion products, and silt; and (2) repair defective protective coatings and degraded SW system piping and components that could adversely affect performance. Preventive maintenance, inspection, and repair procedures have been developed to provide reasonable assurance that any adverse effects of exposure to service water are adequately addressed. The addition of biocide to the SW system reduces biological growth (including MIC) that could lead to degradation of components exposed to the service water. Additionally, a one-time measurement of sludge buildup in the SW reservoir will be performed (Item 23, Table 18-1). This measurement will be completed in accordance with the schedule provided in Item 23, Table 18-1.

SW System Inspections are performed to check for biofouling, damaged coatings, and degraded material condition. Heat transfer parameters for components cooled by service water are monitored. Visual inspections are performed to check for loss of material and changes in material properties. Heat transfer testing is performed to identify the aging effects of loss of material and heat transfer degradation.

Volumetric inspections are also performed to check for loss of material due to MIC.

The acceptance criteria for visual inspections are identified in the procedures that perform the individual inspections. The procedures identify the type and degree of anomalous conditions that are signs of degradation. In the case of service water, degradation includes biofouling as well as material degradation. Engineering evaluations determine whether observed deterioration of material condition is sufficiently extensive to lead to loss of intended function for components exposed to the service water. An engineering evaluation will also determine if additional sludge measurements in the SW reservoir are needed. The degraded condition of material or of heat transfer capability may require prompt remediation. Occurrence of significant degradation that is adverse to quality is entered into the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.2.18 Steam Generator Inspections

Steam Generator Inspections are performed in accordance with Technical Specifications and Inservice Inspection requirements of ASME Section XI. Steam Generator Inspections plans are based upon the guidelines established by Nuclear Energy Institute document, NEI 97-06 (Reference 4) and the Electric Power Research Institute steam generator inspection guidelines (Reference 5). Steam generator tubing inspections are performed on a sampling basis. The sample population inspected meets or exceeds the requirements of Technical Specifications. Qualified techniques, equipment and personnel are used for inspections in accordance with site-specific eddy current analysis guidelines.

Examination of steam generator sub-components other than tubes are performed as required by the governing edition and addenda of ASME Section XI, as imposed by 10 CFR 50.55a. In some cases the specific inspection requirements of ASME Section XI are modified by regulatory commitments and approved Relief Requests. Inspections of the steam generators to check for loss of material, cracking, and gross indications of loss of pre-load include a combination of visual inspections, surface examinations, and volumetric examinations. Tubing inspections are performed in accordance with ASME Section XI, Subsection IWB.

Acceptance standards for steam generator inspections are provided in ASME Section XI, Subsections IWB-3500 and IWC-3500. Evidence of component degradation requires engineering evaluation for determination of corrective action. Occurrence of significant degradation that is adverse to quality will be entered into the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.2.19 Work Control Process

Performance testing and maintenance activities, both preventive and corrective, are planned and conducted in accordance with the station's Work Control Process. The Work Control Process integrates and coordinates the combined efforts of Maintenance, Engineering, Operations, and other support organizations to manage maintenance and testing activities. Performance testing on heat exchangers evaluates the heat transfer capability of the components to determine if heat transfer degradation is occurring. Maintenance activities provide opportunities for inspectors who are Quality Maintenance Team (QMT) or Visual Test (VT) qualified to visually inspect the surfaces (internal and external) of plant components and adjacent piping (Item 21, Table 18-1). Adjacent piping is primarily the internal piping surfaces immediately adjacent to a system component that is accessible through the component for visual inspection. Visual inspections performed through the Work Control Process provide data that can be used to determine the effectiveness of the Chemistry Control Program for Primary Systems and Chemistry Control Program for Secondary Systems to mitigate the aging effects of cracking, loss of material, and change of material properties.

The application for the renewed operating licenses included a commitment to have changes made in procedures to reasonably assure that consistent inspections of components are completed during the process of performing work control process activities (Item 15, Table 18-1). Implementation of consistent inspections is accomplished using automated inspection instructions for work orders involving components and structures that have been identified as requiring aging management. The instructions consistently require inspections to identify a variety of aging mechanisms required for the renewed operating licenses.

The Work Control Process also provides opportunities through preventive maintenance sampling (predictive analysis) to collect lubricating oil and engine coolant samples for subsequent analysis of contaminants that would provide early indication of an adverse environment that can lead to material degradation.

The inspections, testing, and sampling performed under the Work Control Process provide reasonable assurance that the following aging effects will be detected:

- loss of material
- cracking
- heat transfer degradation
- separation and cracking/delamination
- change in material properties (Item 18, Table 18-1). The change in material properties is specifically required to be monitored for elastomeric sealants. This monitoring occurs during the periodic Maintenance Rule visual inspections of structures, and during routine inspections performed for maintenance activities. These inspection and maintenance activities are scheduled through the work control process.

The acceptance criteria for visual inspections, testing, or sampling are currently identified in the procedures that perform the individual maintenance, testing, or sampling activity. The procedures identify the type and degree of anomalous conditions that are signs of degradation.

Whenever evidence of aging effects exists, an engineering evaluation is performed to determine whether the observed condition is acceptable without repair. Occurrence of significant aging effects that is adverse to quality is entered into the Corrective Action System. If the evaluation of an anomalous condition indicates that the occurrence was unexpected for the operational conditions involved, the Work Control Process will be used to ensure that locations with similar material and environmental conditions are inspected as directed by a Station procedure (Item 29, Table 18-1).

As confirmation that the Work Control Process has inspected representative components from each component group for which the Work Control Process is credited to manage the effects of aging, periodic audits of inspections actually performed will be performed and, if Work Control Process activities are found not to be representative, supplemental inspections will be performed (Item 22, Table 18-1). Two audits of the Work Control Process are anticipated, and each will consist of a review of the previous 10 years of historical data. These audits will be performed in accordance with the schedule provided in Item 22, Table 18-1. Any required supplemental inspections would be completed within 5 years after the audits are performed.

18.2.20 Corrective Action System

The Corrective Action System is a required element of the Quality Assurance Program outlined in the Quality Assurance Topical Report (Chapter 17 of the Updated Final Safety Analysis Report). The Quality Assurance Program implements the requirements of 10 CFR 50, Appendix B, and is consistent with the summary in Section A.2 of NUREG-1800, Standard

Review Plan for License Renewal. The Corrective Action System activities include the elements of corrective action, confirmation process, and administrative controls; and is applicable to the safety-related and non-safety-related structures, systems, and components that are within the scope of license renewal.

18.3 TIME-LIMITED AGING ANALYSIS

As part of the application for a renewed license, 10 CFR 54.21(c) requires that an evaluation of TLAAs for the period of extended operation be provided. The following TLAAs have been identified and evaluated to meet this requirement.

18.3.1 Reactor Vessel Neutron Embrittlement

The reactor vessel is subjected to neutron irradiation from the core. This irradiation results in the embrittlement of the reactor vessel materials. Analyses have been performed that address the following:

- Upper shelf energy
- PTS
- RCS P-T operating limits

18.3.1.1 Upper Shelf Energy

The Charpy V-notch test provides information about the fracture toughness of reactor vessel materials. 10 CFR 50 requires the C_VUSE of reactor vessel beltline materials to meet Appendix G requirements. If the USE of a reactor vessel beltline material is predicted to not meet Appendix G requirements, then licensees must submit an analysis that demonstrates an equivalent margin of safety at least three years prior to the time the material is predicted to not meet those requirements.

Reactor vessel calculations have been performed which demonstrated that the upper shelf energy values of limiting reactor vessel beltline materials at the end of the period of extended operation meet Appendix G requirements. Thus, the TLAA has been projected to the end of the period of extended operation and is found to be adequate.

18.3.1.2 Pressurized Thermal Shock

A limiting condition on reactor vessel integrity, known as PTS, may occur during postulated system transients, such as a loss-of-coolant accident (LOCA) or a steam line break. Such transients may challenge the integrity of the reactor vessel under the following conditions: severe overcooling of the inside surface of the vessel wall followed by high re-pressurization, significant degradation of vessel material toughness caused by radiation embrittlement, and the presence of a critical-size defect in the vessel wall.

The reference temperature for pressurized thermal shock (RT_{PTS}) is defined in 10 CFR 50.61. RT_{PTS} values for the limiting reactor vessel materials at the end of the period of extended operation have been recalculated by Dominion. At the end of the period of extended operation, the calculated RT_{PTS} values for the beltline materials are less than the applicable screening criteria established in 10 CFR 50.61. Thus, the TLAA has been projected to the end of the period of extended operation and is found to be adequate.

18.3.1.3 **Pressure-Temperature Limits**

Atomic Energy Commission (AEC) General Design Criterion (GDC) 14 of Appendix A of 10 CFR 50, *Reactor Coolant Pressure Boundary,* requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage (or rapid failure) and of gross rupture. AEC GDC 31, *Fracture Prevention of Reactor Coolant Pressure Boundary,* requires that the reactor coolant pressure boundary be designed with sufficient margin to ensure that when stressed under operating, maintenance, and testing conditions the boundary behaves in a non-brittle manner and the probability of rapidly propagating fracture is minimized.

Reactor vessel neutron fluence values corresponding to the end of the period of extended operation and reactor vessel beltline material properties were used to determine the limiting value of reference nil ductility (RT_{NDT}), and to calculate RCS P-T operating limits valid through the end of a period of extended operation. Maximum allowable LTOPS power operated relief valve lift setpoints have been developed on the basis of the P-T limits applicable to the period of extended operation. Revised RCS P-T limit curves and LTOPS setpoints will be submitted for review and approval prior to the expiration of the existing technical specification limits in order to maintain compliance with the governing requirements of 10 CFR 50 Appendix G.

The TLAA has been projected to the end of the period of extended operation and is found to be adequate.

18.3.2 Metal Fatigue

The thermal fatigue analyses of the station’s mechanical components have been identified as TLAA.

18.3.2.1 **ASME Boiler and Pressure Vessel Code, Section III, Class 1**

The steam generators, pressurizers, reactor vessels, loop stop valves, reactor coolant pumps, control rod drive mechanisms (CRDMs), and all reactor coolant system pressure boundary piping have been analyzed using the methodology of the ASME Boiler and Pressure Vessel Code, Section III, Class 1.

The ASME Boiler and Pressure Vessel Code, Section III, Class 1, requires a design analysis to address fatigue and establish limits such that the initiation of fatigue cracks is precluded.

Experience has shown that the transients used to analyze the ASME III requirements are often very conservative. Design transient magnitude and frequency are more severe than those occurring during plant operation. The magnitude and number of the actual transients are monitored. This monitoring assures that the existing frequency and magnitude of transients are conservative and bounding for the period of extended operation, and that the existing ASME III equipment will perform its intended functions for the period of extended operation. A cycle counting program (Section 18.4.2) is in place to provide reasonable assurance that the actual transients are smaller in magnitude and within number of the transients used in the design.

Fatigue analyses for the steam generators, pressurizers, reactor vessels, reactor coolant pumps, CRDMs, and all RCS pressure boundary piping have been evaluated and determined to remain valid for the period of extended operation.

Fatigue analyses for the reactor vessel closure studs and the loop stop valves have been re-analyzed. The analyses for these components have been projected to be valid for the period of extended operation.

18.3.2.2 **Reactor Vessel Underclad Cracking**

In early 1971, an anomaly was identified in the heat-affected zone of the base metal in a European-manufactured reactor vessel. A generic fracture mechanics evaluation by Westinghouse demonstrated that the growth of underclad cracks during a 40-year plant life would be insignificant.

The evaluation was extended to 60 years using fracture mechanics evaluation based on a representative set of design transients. The occurrences were extrapolated to cover 60 years of service life. This 60-year evaluation shows insignificant growth of the underclad cracks and is documented in WCAP-15338 (Reference 21). The plant-specific design transients are bounded by the representative set used in the evaluation.

The analysis associated with reactor vessel underclad crack growth has been projected to the end of the period of extended operation and has been found to be acceptable.

18.3.2.3 **ANSI B31.1 Piping**

The balance-of-plant piping is designed to the requirements of ANSI B31.1, *Power Piping.*

ANSI B31.1 design requirements assume a stress range reduction factor in order to provide conservatism in the piping design while accounting for fatigue due to thermal cyclic operation. This reduction factor is 1.0, provided the number of anticipated cycles is limited to 7000 equivalent full-temperature cycles. A piping system would have to be thermally cycled approximately once every three days over a plant life of 60 years to reach 7000 cycles. Considering this limitation, a review of the ANSI B31.1 piping within the scope of license renewal has been performed to identify those systems that operate at elevated temperature and to establish their cyclic operating practices. Under current plant operating practices, piping systems within the scope of license renewal are only occasionally subject to cyclic operation. Typically, these systems are subjected to continuous steady-state operation. Significant variation in operating temperatures occur only during plant heatup and cooldown, during plant transients, or during periodic testing.

The analyses associated with ANSI B31.1 piping fatigue have been evaluated and determined to remain valid for the period of extended operation except for sample lines for the hot and cold legs. The analyses associated with sample lines for the hot and cold legs have been projected to be valid to the end of the period of extended operation.

18.3.2.4 **Environmentally Assisted Fatigue**

GSI-190 (Reference 6) identifies a NRC staff concern about the effects of reactor water environments on reactor coolant system component fatigue life during the period of extended operation. The reactor water's environmental effects as described in GSI-190, are not included in the CLB. As a result, the criterion specified in 10 CFR 54.3(a)(6) is not satisfied. Hence, environmental effects are not TLAAs. GSI-190, which was closed in December 1999, has concluded that environmental effects have a negligible impact on core damage frequency, and as such, no generic regulatory action is required (Reference 7). However, as part of the closure of GSI-190, the NRC has concluded that licensees who apply for license renewal should address the effects of coolant environment on component fatigue life as part of their aging management programs. As demonstrated in the preceding sections, fatigue evaluation in the original transient design limits remain valid for the period of extended operation. Confirmation by transient cycle counting will ensure that these transient design limits are not exceeded. Secondly, the reactor water's environmental effects on fatigue life were evaluated using the most recent data from laboratory simulation of the reactor coolant environment.

As a part of the industry effort to address environmental effects for operating nuclear power plants during the current 40-year licensing term, Idaho National Engineering Laboratories evaluated, in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components" (Reference 8), fatigue-sensitive component locations at plants designed by all four U. S. Nuclear Steam Supply System vendors. The pressurized water reactor calculations, especially the early-vintage Westinghouse PWR calculations, are directly relevant to the Dominion stations. The description of the "Older Vintage Westinghouse Plant" evaluated in NUREG/CR-6260 applies to the North Anna station. In addition, the transient cycles considered in the evaluation match or bound the design. The results of NUREG/CR-6260 analyses, and additional data from NUREG/CR-6583 (Reference 9) and NUREG/CR-5704 (Reference 10), were then utilized to scale up the plant-specific cumulative usage factors (CUF) for the fatigue-sensitive locations to account for environmental effects.

Based on these adjusted CUFs (using the environmental fatigue penalty factor), it has been determined that the surge line connection at the reactor coolant system's hot leg pipe exceeds the design threshold of 1.0. As a consequence, management of environmentally assisted fatigue is required. Additionally, the CUFs that were adjusted for environmental effects for the safety injection (SI) and charging line nozzles initially were determined to exceed the design threshold of 1.0. Subsequent fatigue evaluations (Reference 35), using the methodology of ASME Section III, NB-3200, confirm that the CUFs for the SI and charging line nozzles do not exceed the ASME Code allowable value of 1.0, including the effects of the reactor water environment. Therefore, the SI and charging line nozzles will not require enhanced inspections for the management of environmentally assisted fatigue.

The approach to manage environmentally assisted fatigue for the surge line will be developed from one or more of the following options and submitted to the NRC for review prior to the period of extended operation (Items 24 & 25, Table 18-1):

1. Further refinement of the fatigue analysis (e.g., NB-3200 analysis) to lower the CUFs to below 1.0, or
2. Repair of the affected locations, or
3. Replacement of the affected locations, or
4. Inspection of the affected locations.

The surge line weld at the hot leg pipe connection will be included (Item 2, Table 18-1) in an Augmented Inspection Activities (Section 18.2.1). Baseline inspections of the surge line welds are planned prior to entry into the period of extended operation. Inspections will also occur once per 40-month ISI period, thereafter. The results of these inspections and the results of planned research by the EPRI-sponsored Materials Reliability Program will be utilized to assess the appropriate approach for addressing environmentally assisted fatigue of the surge lines during the period of extended operation.

The use of inspections (Option 4) to manage environmentally assisted fatigue during the period of extended operation, requires inspection details such as scope, qualification, method, and frequency be provided to the NRC for review prior to entering the period of extended operation. The NRC review ensures that the inspection intervals for the periodic inspection of the affected locations would be determined by a method accepted by the NRC.

Implementation of one or more of the above listed options will ensure that the potential effects of the reactor water environment have been addressed for the period of extended operation as required by GSI-190.

18.3.3 Environmental Qualification of Electric Equipment

10 CFR 54.49 requires that each holder of a nuclear power plant operating license establish a program for qualifying safety-related electric equipment. Such a program has been implemented at the station and is invoked by Administrative Procedure. Analyses and tests that qualify safety-related equipment for the period of extended operation are considered TLAAs.

The Environmental Qualification (EQ) Program (Section 18.4.1) requires that all electrical equipment important to safety located in a harsh environment shall be managed through the period of extended operation.

18.3.4 Containment Liner Plate

The accumulated fatigue effects of applicable liner loading conditions were evaluated in accordance with Paragraph N-415 of the ASME Boiler and Pressure Vessel Code, Section III, 1968. The evaluation was based on 1000 cycles of operating pressure variations, 4000 cycles of operating temperature variations, and 20 design earthquake cycles. The operating pressure variations are anticipated to be less than 100 and temperature variations are anticipated to be less than 400 for forty years of operation. Extrapolating these anticipated values for sixty years of operation results in 150 pressure variations and 600 temperature variations (Reference Table 3.8-7). The number of design cycles was conservatively increased to 1500 cycles of operating pressure variations, 6000 cycles of operating temperature variation, and 30 design earthquake cycles by using a multiplication factor of 1.5, to account for the period of extended operation.

A review of the identified calculations has determined that the increase in the number of cycles due to the period of extended operation is acceptable. Effects of the Containment Type A pressure tests on fatigue of the Containment liner plate have been included in the evaluation. Therefore, the Containment liner is adequate for a 60-year operating period as currently designed. The analyses associated with the Containment liner plate have been revised and projected to be valid for the period of extended operation.

18.3.5 Plant-Specific Time-Limited Aging Analyses

18.3.5.1 Crane Load Cycle Limit

The following are cranes included in license renewal scope and in NUREG-0612 (Reference 11):

- Containment polar cranes
- Containment annulus monorails
- Fuel handling bridge crane
- Spent fuel crane

- Auxiliary Building monorails

NUREG-0612 requires that the design of heavy load overhead handling systems meet the intent of Crane Manufacturers Association of America, Inc. (CMAA) Specification #70. The crane load cycle provided in CMAA-70 has been identified as a TLAA, with the most limiting number of loading cycles being 100,000.

The most frequently used cranes are spent fuel cranes. Each of these cranes will experience approximately 25,000 cycles of half-load lifts to support the refueling of both units over a 60-year period. In addition, the crane is used to load new fuel into the fuel pool, to perform the various rearrangements required by operations support, to accommodate inspections by fuel vendors, and to load spent fuel casks. In such service, the crane is conservatively expected to make a total of 50,000 half-load lifts in a 60-year period.

Therefore, the analyses associated with crane design, including fatigue, are valid for the period of extended operation.

18.3.5.2 **Reactor Coolant Pump Flywheel**

During normal operation, the reactor coolant pump flywheel possesses sufficient kinetic energy to produce high-energy missiles in the unlikely event of failure.

The aging effect of concern is fatigue crack initiation in the flywheel bore keyway. An evaluation of a failure over the period of extended operation has been performed. It demonstrates that the flywheel design has a high structural reliability with a very high flaw tolerance and negligible flaw crack growth over a 60-year service life (Reference 12).

The analysis associated with the structural integrity of the reactor coolant pump flywheel has been evaluated and determined to be valid for the period of extended operation.

18.3.5.3 **Leak-Before-Break**

Westinghouse (Westinghouse Owners Group) tested and analyzed crack growth with the goal of eliminating reactor coolant system primary loop pipe breaks from plant design bases. The objective of the investigation was to examine mechanistically, under realistic yet conservative assumptions - whether a postulated crack causing a leak, will grow to become unstable and lead to a full circumferential break when subjected to the worst possible combinations of plant loading.

The detailed evaluation has shown that double-ended breaks of reactor coolant pipes are not credible, and as a result, large LOCA loads on primary system components will not occur. The overall conclusion of the evaluation was, that, under the worst combination of loading, including the effects of safe shutdown earthquake, the crack will not propagate around the circumference and cause a guillotine break. The plant has leakage detection systems that can identify a leak with margin, and provide adequate warning before the crack can grow.

The concept of eliminating piping breaks in reactor coolant system primary loop piping has been termed “leak-before-break” (LBB).

In 1986, Westinghouse performed an LBB analysis of the primary loop piping. Two TLAAs related to LBB have been identified: fatigue crack growth and thermal aging of cast austenitic stainless steel. The original fatigue crack growth analysis has been performed for the design transient cycles and with consideration of thermal aging effect for forty years. The steam generator primary nozzles to safe-end welds in the primary loop piping that have been analyzed for LBB are the only components fabricated with Alloy 82/182-weld material for NAPS 1 and 2. Dominion will continue to participate in the ongoing NRC/industry program on Alloy 82/182-weld material and will implement the findings/resolution from this effort, as appropriate (Item 20, Table 18-1).

To maintain the plant’s LBB design basis, the thermal aging effect for 60 years has been revalidated. The change in the material property has been found to be insignificant. Since the number of design transient cycles will not be exceeded during 60 years of operation, the LBB analysis is projected to be valid for the period of extended operation.

18.3.5.4 **Spent Fuel Pool Liner**

The spent fuel pool liner located in the Fuel Building is needed to prevent a leak to the environment. A design calculation has been identified which documents that the spent fuel pool design meets the general industry criteria. The calculation includes a fatigue analysis to add a further degree of confidence.

The normal thermal cycles occur at each refueling, resulting in 80 cycles for both units in 60 years. Total number of thermal cycles is expected to be 90, which includes normal, upset, emergency, and faulted conditions.

The calculations show that the allowable thermal cycles for spent fuel pool liner for the most severe thermal condition, which includes a loss of cooling, is 100.

Therefore, the existing calculations remain valid for the period of extended operation.

18.3.5.5 **Piping Subsurface Indications**

Calculations have been identified that addressed piping subsurface indications detected by inspections, performed in accordance with ASME Section XI. Section XI provides the acceptance criteria for various flaw orientations, locations and sizes. The calculations determined the number of thermal cycles required for the flaws to reach unacceptable size.

Required cycles for the flaws to reach an unacceptable size are 20,700 or higher.

Since it is expected that the number of the cycles experienced by the piping will not exceed these values for sixty years of operation, the analyses have been determined to remain valid for the period of extended operation.

18.3.5.6 **Reactor Coolant Pump and ASME Code Case N-481**

Periodic volumetric inspections of the welds in the primary loop pump casings in commercial nuclear power plants are required by Section XI of the ASME Boiler and Pressure Vessel Code. Since the reactor coolant pump casings are inspected prior to being placed in service, and no significant mechanisms exist for crack initiation and propagation; it has been concluded that the inservice volumetric inspection could be replaced with an acceptable alternate inspection. In recognition of this conclusion, ASME Code Case N-481, *Alternative Examination Requirements for Cast Austenitic Pump Casings,* provides an alternative to the volumetric inspection requirement. The code case allows the replacement of volumetric examinations of primary loop pump casings with fracture mechanics based integrity evaluations - Item (d) of the code case - supplemented by specific visual examinations. The analysis has been performed on the reactor coolant pump casing integrity in accordance with the ASME Code Case N-481 requirements. The analysis has been projected to be valid for 60 years.

18.3.6 Exemptions

The requirements of 10 CFR 54.21(c) stipulate that the application for a renewed license should include a list of plant-specific exemptions granted pursuant to 10 CFR 50.12 and that are based on TLAA, as defined in 10 CFR 54.3. Each active 10 CFR 50.12 exemption has been reviewed to determine whether the exemption is based on a TLAA. No plant-specific exemptions granted pursuant to 10 CFR 50.12 and based on a TLAA as defined in 10 CFR 54.3 have been identified.

18.4 TLAA SUPPORTING ACTIVITIES

18.4.1 Environmental Qualification Program

The EQ Program activities are in compliance with the requirements of 10 CFR 50.49. The EQ Program will be continued throughout the period of extended operation. Electrical equipment located in a harsh environment is evaluated for environmental qualification if they are required to function in the conditions that will exist post-accident after being subjected to the normal effects of aging. A harsh environment results from a LOCA or main steam line break inside Containment, high radiation levels due to the post-LOCA effects outside Containment, or high energy line breaks outside Containment.

The EQ Program is applicable to the following groups of components:

- Safety-related electrical equipment that is relied upon to remain functional during and following a design-basis event (DBE)
- Non-safety-related electrical equipment whose failure, under postulated environmental conditions, could prevent accomplishment of safety functions
- Certain post-accident monitoring equipment as described in Regulatory Guide 1.97 (Reference 19).

Guidance regarding environmental qualification was given in NRC Bulletin 79-01B (Reference 20) for Unit 1 and in NUREG-0588 (Category II) (Reference 23) for Unit 2.

The Equipment Qualification Master List provides a listing of electrical equipment that is important to safety and is located in a potentially harsh environment.

Based on the definitions of 10 CFR 54, certain EQ calculations are considered to be TLAA. As stated in 10 CFR 54.21(c) and in NEI 95-10 (Reference 22), analyses for TLAAs utilize one of the following three options:

i. The analyses remain valid for the period of extended operation,

ii. The analyses have been projected to the end of extended operation, or

iii. The effects of aging will be adequately managed during the period of extended operation.

For purposes of license renewal, EQ components will be evaluated utilizing Option iii in accordance with the EQ Program. EQ concerns for license renewal will consider only those in-scope components that have a qualified lifetime greater than 40 years. Components with a qualified lifetime of less than 40 years already are included in a program of periodic replacement and are not considered TLAAs.

10 CFR 50.49(j) requires that a qualification record be maintained for all equipment covered by the EQ Rule. The qualification process verifies that the equipment is capable of performing its safety function when subjected to various postulated environmental conditions. These conditions include expected ranges of temperature, pressure, humidity, radiation, and accident conditions such as chemical spray and submergence.

The process of qualifying EQ equipment includes analysis, data collection, and data reduction with appropriate assumptions, acceptance criteria and corrective actions.

Qualification Document Reviews (QDRs) provide the basis for qualifying EQ components. The QDRs provide the following information for each piece of equipment that is qualified:

- The performance characteristics required under normal, DBE, and post-DBE conditions.

- The voltage, frequency, load, and other electrical characteristics for which equipment performance can be provided with reasonable assurance.
- The environmental conditions, including temperature, pressure, humidity, radiation, chemical spray, and submergence, at the location where the equipment must function.

18.4.2 Transient Cycle Counting

During normal, upset, and test conditions; reactor coolant system pressure boundary components are subjected to transient temperatures, pressures, and flows, resulting in cyclic changes in internal stresses in the equipment. The cyclic changes in internal stresses cause metal fatigue. Class 1 reactor coolant system components have been designed to withstand a number of design transients without experiencing fatigue failures during their operating life. The purpose of the Transient Cycle Counting is to record the number of normal, upset, and test events, and their sequence that the station experiences during operation. Design transients are counted to provide reasonable assurance that plant operation does not occur outside the design assumptions.

The Transient Cycle Counting activities are applicable to the reactor coolant system pressure boundary components for which the design analysis assumes a specific number of design transients. A summary of reactor coolant system design transients for which transient cycle counting is performed is listed below:

- Heatups/Cooldowns < 100°F/Hr.
- Step load increase/decrease of 10%
- Large load reduction of 50%
- Loss of load > 15%
- Loss of AC power
- Loss of flow in one loop
- Full power reactor trip
- Inadvertent auxiliary pressurizer spray
- Inadvertent SI
- Normal charging and letdown return to service
- Charging trip with delayed return to service

The aging effect that is managed by counting transient cycles is cracking due to metal fatigue. The Transient Cycle Counting activities monitor transient cycles that have been experienced by each unit and compare the actual number of cycles to a design assumption. Any concerns related to fatigue are mitigated, as long as the number and magnitude of transient cycles are less than the design assumptions. Approaching a design limit may indicate a situation that is adverse to quality, and would initiate the Corrective Action System. Subsequently, an engineering analysis will determine the design margin remaining, taking credit for the actual magnitude of transients and their sequence to confirm that the allowable factor has not been exceeded. If warranted, component repair or replacement would be initiated.

18.5 REFERENCES

1. Working Draft of the NRC Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants.
2. Generic Letter 88-05, *Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants*, March 17, 1988.
3. NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition*, US Nuclear Regulatory Commission. (Formerly NUREG-75/087)
4. NEI 97-06, *Steam Generator Program Guidelines*, Revision 2, Nuclear Energy Institute.
5. PWR Steam Generator Examination Guidelines, Revision 7, Electric Power Research Institute.
6. Generic Safety Issue (GSI)-190, *Fatigue Evaluation for Metal Components for 60-year Plant Life*, U.S. Nuclear Regulatory Commission, August 1996.
7. Memorandum from Ashok C. Thadani, to William D. Travers, U.S. Nuclear Regulatory Commission, *Closeout of Generic Safety Issue 190*, December 26, 1999.
8. NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components*, U.S. Nuclear Regulatory Commission, March 1995.
9. NUREG/CR-6583, *Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels*, U.S. Nuclear Regulatory Commission, March 1998.
10. NUREG/CR-5704, *Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels*, U.S. Nuclear Regulatory Commission, April 1999.
11. NUREG-0612, *Control of Heavy Loads at Nuclear Power Plants*, U.S. Nuclear Regulatory Commission, July 1980.
12. WCAP-14535A, *Topical Report On Reactor Coolant Pump Flywheel Inspection Elimination*, Westinghouse Electric Corporation, November 1996.
13. American National Standards Institute: ANSI B30.2-1976, *Overhead and Gantry Cranes.*

14. American National Standards Institute: ANSI B30.11-1973, *Monorail Systems and Underhung Cranes.*

15. Generic Letter 89-08, *Erosion/Corrosion-Induced Pipe Wall Thinning*, May 2, 1989.

16. NUREG-1344, *Erosion/Corrosion-Induced Pipe Wall Thinning in US Nuclear Power Plants*, April 1, 1989.

17. NSAC-202L, *Recommendation for an Effective Flow Accelerated Corrosion Program*, Electric power Research Institute, April 8, 1999.

18. Generic Letter 89-13, *Service Water System Problems Affecting Safety-Related Equipment*, July 18, 1989 (Supplement 1 dated 4/4/90).

19. U.S. Nuclear Regulatory Commission, *Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident*, Regulatory Guide 1.97, December 1980.

20. IE Bulletin 79-01B, *Environmental Qualification of Class 1E Equipment*, Office of Inspection and Enforcement, January 14, 1980 (Supplement 1 dated 2/29/80; Supplement 2 dated 9/30/80; and Supplement 3 dated 10/24/80).

21. WCAP-15338, *A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants*, Westinghouse Electric Corporation, March 2000.

22. NEI 95-10, *Industry Guidance for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule*, Revision 2, August 2000.

23. NUREG-0588 (Category II), *Interim Staff Position on Environmental Qualification of Safety-related Electrical Equipment*, August 1, 1979, (Revision 1 dated 11/1/79).

24. NUREG-1766, Safety Evaluation Report Related to the License Renewal of North Anna Power Station, Units 1 and 2, and Surry Power Station Units 1 and 2, December 2002

25. Letter from Leslie N. Hartz (Dominion) to NRC, Dominion Position Regard Fuse Holders, Serial No. 02-691, November 4, 2002.

26. Letter from Eugene S. Grecheck (Dominion) to NRC, Supplemental Information to Support License Renewal, Serial No. 02-706, December 2, 2002.

27. Letter from David A. Christian (Dominion) to NRC, Request for Additional Information License Renewal Applications, Serial No. 01-686, January 16, 2002.

28. Letter from David A. Christian (Dominion) to NRC, Request for Additional Information License Renewal Applications, Serial No. 02-163, May 22, 2002.

29. Letter from David A. Christian (Dominion) to NRC, Request for Additional Information License Renewal Applications, Serial No. 01-685, January 4, 2002.

30. Letter from David A. Christian (Dominion) to NRC, Request for Additional Information License Renewal Applications, Serial No. 01-647, November 30, 2001.

31. Letter from David A. Christian (Dominion) to NRC, Request for Additional Information License Renewal Applications, Serial No. 01-514, September 27, 2001.

32. Letter from David A. Christian (Dominion) to NRC, Request for Additional Information License Renewal Applications, Serial No. 01-732, February 5, 2002.

33. Letter from David A. Christian (Dominion) to NRC, License Renewal Applications - Submittal, Serial No. 01-282, May 29, 2001.

34. Letter from Leslie N. Hartz (Dominion) to NRC, Request for Additional Information License Renewal Applications, Serial No. 02-332A, October 1, 2002.

35. Letter from Leslie N. Hartz (Dominion) to NRC, Engineering Evaluation Results and Closure of Commitments for Management of Environmentally-Assisted Fatigue in Accumulator and Charging Nozzles, Serial No. 03-616, January 6, 2004.

Table 18-1
LICENSE RENEWAL COMMITMENTS

Item	Commitment	Schedule[a]	Source	Ref.
1	Develop and implement inspection program for buried piping and valves.	One-time between years 30-40. Additional inspections based on results.	Table B4.0[b], RAI B2.1.1-1	33 31
2	Add pressurizer surge line to Augmented Inspection Program.	Prior to Period of Extended Operation (Complete)	Table B4.0, RAI 4.3-7	33 27
3	Add core barrel hold-down spring to Augmented Inspection Program.	Prior to Period of Extended Operation (Complete)	Table B4.0	33
4	Expand scope of Civil Engineering Structural Inspection to cover License Renewal requirements.	Prior to Period of Extended Operation (Complete)	Table B4.0	33
5	Revise plant documents to use inspection opportunities when inaccessible areas become accessible during work activities.	Prior to Period of Extended Operation (Complete)	Table B4.0	33
6	Incorporate NFPA-25, Section 2-3.1.1 for sprinklers.	Prior to year 50. If testing used, repeat every 10 years.	Table B4.0	33
7	Develop inspection criteria for non-ASME supports and doors.	Prior to Period of Extended Operation (Complete)	Table B4.0	33
8	Develop procedural guidance for inspection criteria that puts focus on aging effects.	Prior to Period of Extended Operation (Complete)	Table B4.0	33
9	Develop and implement inspection program for infrequently accessed areas.	One-time between years 30-40. Additional inspections based on results.	Table B4.0, RAI 3.5-1, RAI 3.5.8-1	33 30 32

a. The Period of Extended operation is the period of 20 years beyond the expiration date of each unit's original operating license. For North Anna Unit 1, the Period of Extended Operation is from April 2, 2018 to April 1, 2038 and for North Anna Unit 2, from August 22, 2020 to August 21, 2040.

b. Table B4.0 is the table of Licensee Followup Actions located in the License Renewal Application for North Anna (Reference 33).

Table 18-1 (continued)
LICENSE RENEWAL COMMITMENTS

Item	Commitment	Schedule[a]	Source	Ref.
10	Develop and implement inspection program for tanks.	One-time between years 30-40. Additional inspections based on results.	Table B4.0	33
11	Follow industry activities related to failure mechanisms for small-bore piping. Evaluate changes to inspection activities based on industry recommendations.	On-going activity (Complete)	Table B4.0, RAI 3.1.1.2-2	33 30
12	Follow industry activities related to core support lugs. Evaluate need to enhance inspection activities based on industry recommendations.	On-going activity	Table B4.0	33
13	Inspect representative sections of polar crane box girders.	One-time between years 30-40. Additional inspections based on results.	Table B4.0	33
14	Follow industry activities related to reactor vessel internals issues such as void swelling, thermal and neutron embrittlement, etc. Evaluate industry recommendations.	One-time inspection between years 30-40 on most susceptible single unit (Surry or North Anna). Additional inspections based on results.	Table B4.0	33
15	Implement changes into procedures to assure consistent inspection of components for aging effects during work activities.	Prior to Period of Extended Operation (Complete)	Table B4.0	33

a. The Period of Extended operation is the period of 20 years beyond the expiration date of each unit's original operating license. For North Anna Unit 1, the Period of Extended Operation is from April 2, 2018 to April 1, 2038 and for North Anna Unit 2, from August 22, 2020 to August 21, 2040.

Table 18-1 (continued)
LICENSE RENEWAL COMMITMENTS

Item	Commitment	Schedule[a]	Source	Ref.
16	Incorporate groundwater monitoring into the civil engineering structural monitoring program. Consider groundwater chemistry in engineering evaluations of deficiencies.	Prior to Period of Extended Operation (Complete)	RAI 3.5-2	30
17	Incorporate management of concrete aging into the civil structural monitoring program and the infrequently accessed area inspection programs.	Prior to Period of Extended Operation	RAI 3.5-7	30
18	Incorporate management of elastomers into the work control activities.	Prior to Period of Extended Operation (Complete)	RAI 3.5.6-4, RAI B2.2.19-3	32 30
19	Develop and implement inspection program for Non-EQ cables.	One-time between years 30-40. Additional inspections every 10 years thereafter.	RAI 3.6.2-1	30
20	Follow industry activities related to Alloy 82/182 weld material. Implement activities based on industry recommendations, as appropriate.	On-going activity	RAI 4.7.3-1	29
21	Inspectors credited in the Work Control Process will be QMT or VT qualified.	Prior to Period of Extended Operation	RAI B2.2.19-1	30
22	Perform audit of work control inspections to ensure representation by all in-scope license renewal systems and to determine need for supplemental inspections.	Prior to Period of Extended Operation and every 10 years thereafter. Supplemental inspections within 5 years of audit.	RAI B2.2.19-3	30

a. The Period of Extended operation is the period of 20 years beyond the expiration date of each unit's original operating license. For North Anna Unit 1, the Period of Extended Operation is from April 2, 2018 to April 1, 2038 and for North Anna Unit 2, from August 22, 2020 to August 21, 2040.

Table 18-1 (continued)
LICENSE RENEWAL COMMITMENTS

Item	Commitment	Schedule[a]	Source	Ref.
23	Measure the sludge buildup in the SW reservoir at North Anna.	One-time between years 35 and 40	RAI 3.5.8-2	28, 32
24	Provide inspection details for pressurizer surge line inspections to the NRC for review and approval.	Prior to Period of Extended Operation	RAI 4.3-7, RAI 4.3-6	27
25	Provide inspection details for SI and charging line inspections to the NRC for review and approval if analysis is not successful in reducing the CUF below 1.0. The analysis described in Reference 35 confirms the CUFs to be below 1.0, and no inspections are required.	Prior to Period of Extended Operation (Complete)	RAI 4.3-6,	27, 34
26	Address NRC staff final guidance regarding fuse holders when issued.	When issued or prior to Period of Extended Operation, whichever is later.	See Reference	25
27	Not applicable to North Anna	N/A	N/A	N/A
28	Revise procedures for groundwater testing to account for possible seasonal variations.	Prior to Period of Extended Operation (Complete)	See Reference	26

a. The Period of Extended operation is the period of 20 years beyond the expiration date of each unit's original operating license. For North Anna Unit 1, the Period of Extended Operation is from April 2, 2018 to April 1, 2038 and for North Anna Unit 2, from August 22, 2020 to August 21, 2040.

Table 18-1 (continued)
LICENSE RENEWAL COMMITMENTS

Item	Commitment	Schedule[a]	Source	Ref.
29	Inspect similar material/environment components, both within the system and outside the system, if aging identified in a location within a system cannot be explained by environmental/operational conditions at that specific location.	Prior to Period of Extended Operation (Complete)	RAI B2.2.19-3	30
30	Supplement the NFPA pressure and flowrate testing credited in each LRA as part of the fire protection program activity with the work control process activity in order to manage aging effects for the fire protection system piping.	Prior to Period of Extended Operation	RAI B2.2.7-2	28, 30

a. The Period of Extended operation is the period of 20 years beyond the expiration date of each unit's original operating license. For North Anna Unit 1, the Period of Extended Operation is from April 2, 2018 to April 1, 2038 and for North Anna Unit 2, from August 22, 2020 to August 21, 2040.