APPENDIX A
UPDATED FINAL SAFETY ANALYSIS REPORT SUPPLEMENT

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A.0 INTRODUCTION

This appendix provides the information to be submitted in an Updated Final Safety Analysis Report Supplement as required by 10 CFR 54.21(d) for the Vermont Yankee Nuclear Power Station (VYNPS) License Renewal Application (LRA). The LRA contains the technical information required by 10 CFR 54.21(a) and (c). Appendix B provides descriptions of the programs and activities that manage the effects of aging for the period of extended operation. Section 4.0 documents the evaluations of time-limited aging analyses for the period of extended operation. These LRA sections have been used to prepare the program and activity descriptions for the VYNPS Updated Final Safety Analysis Report (UFSAR) Supplement information in this appendix.

This appendix is divided into two parts. The first part identifies changes to the existing sections of the UFSAR related to license renewal. The second part provides new information to be incorporated into the UFSAR. The information presented in both parts will be incorporated into the UFSAR following issuance of the renewed operating license. Upon inclusion of the UFSAR supplement in the VYNPS UFSAR, future changes to the descriptions of the programs and activities will be made in accordance with 10 CFR 50.59.

A.1 CHANGES TO EXISTING UFSAR INFORMATION

This section identifies changes to existing sections of the UFSAR that reflect a renewed operating license. Proposed text deletions are indicated by a strikethrough and proposed text additions are indicated by an underline.

A.1.1 UFSAR Chapter 4 Changes

Section 4.2.2 – Power Generation Design Bases

(1st paragraph)

1. The original reactor vessel design lifetime was ~~shall be~~ 40 years. The vessel is acceptable for 60 years of operation.

Section 4.2.4.1 – Reactor Vessel

(1st paragraph)

The reactor vessel is a welded vertical cylindrical pressure vessel with hemispherical heads. The reactor vessel was originally ~~is~~ designed and fabricated for a useful life of 40 years based upon the specified design and operating conditions. The vessel is acceptable for 60 years of operation. The vessel is designed, fabricated, inspected, tested, and stamped in accordance with the ASME Boiler and Pressure Vessel Code, Section III, its interpretations, and applicable requirements for Class A vessels as defined therein. The reactor vessel and its supports are designed in accordance with the loading criteria of Appendix C, "Structural Loading Criteria." The materials used in the design and fabrication of the reactor pressure vessel are shown in Table 4.2.1.

(5th paragraph)

Another way of minimizing the NDT temperature is by reducing the integrated neutron exposure at the inner surface of the reactor vessel. The coolant annulus between the vessel and core shroud and the core location in the vessel limit the integrated neutron exposure of reactor vessel material to less than 1 x 10^{19} nvt from neutrons with energy levels greater than 1 MeV, within the original 40-year design lifetime of the vessel. This is not the expected exposure, nor is it the absolute limit of safe exposure; it is an exposure value that can be demonstrated to be safe and is practical to maintain. The estimated exposure for the 60~~40~~-year life is less than 5.39 x 10^{17} nvt for neutron energies greater than 1 MeV at the vessel inner surface ~~(Reference 17)~~.

Section 4.2.4.9 – Reactor Vessel Insulation

The reactor vessel insulation has an average maximum heat transfer rate of approximately 80 Btu/hr-ft2 at the operating conditions of 550°F for the vessel and 135°F for the outside air. The maximum insulation thicknesses are 4 inches for the upper head, 3-1/2 inches for the cylindrical shell and nozzles, and 3 inches for the bottom head. The upper head insulation is designed to permit complete submersion in water during shutdown without loss of insulating material, contamination of the water, or adverse effect on the insulation efficiency of the insulation assembly after draining and drying. The lower head and cylindrical shell insulation is permanently installed ~~for the 40-year design life of the vessel~~. The insulation panels for the cylindrical shell of the vessel are held in place by vessel insulation supports located at two elevations on the vessel. The support brackets for each support are full-penetration welded to the vessel at 12 evenly spaced locations around the circumference.

Section 4.2.5 – Safety Evaluation

(3rd paragraph)

The reactor vessel was originally ~~is~~ designed for a 40-year life and would ~~will~~ not be exposed to more than 1 x 10^{19} nvt of neutrons with energies exceeding 1 MeV. The reactor vessel was ~~is~~ also designed for the transients which could occur during the 40-year life. Vessel operation up to 60 years was reviewed and the maximum fluence to the vessel inner wall is 5.39 x 10^{17} n/cm^2, still well below the original design value.

Section 4.3.4 – Description

(17th paragraph, page 4.3-7)

The original design objective for the recirculation pump casing was ~~is~~ a useful life of 40 years, accounting for corrosion, erosion, and material fatigue. The pump drive motor, impeller, wear rings, and seals are designed for as long a life as is practical. The pump casing was reviewed for license renewal and loss of material due to corrosion, erosion, and cracking due to material fatigue are managed such that the casing will continue to perform its intended function consistent with the current licensing basis for the period of extended operation.

Section 4.6.3 – Description [MSIV]

(24th paragraph, page 4.6-7)

The isolation valve is designed to pass saturated steam at 1250 psig and 575°F with a moisture content of approximately 0.23%, oxygen content of 30 ppm, and a hydrogen content of 4 ppm. The design objective for the valve was ~~is~~ a minimum of 40 years' service at the specified operating conditions. The estimated operating cycles per year is 100 cycles during the first year and 50 cycles per year thereafter. In addition to minimum wall thickness required by applicable codes, a corrosion allowance of 0.120 inches minimum is added to provide for 40 years' services. Allowable operating cycles and corrosion allowance were reviewed and remain valid for the period of extended operation (60-years).

A.1.2 UFSAR Chapter 6 Changes

Section 6.4.1 – High Pressure Coolant Injection System

(14th paragraph, page 6.4-5)

The system was ~~is~~ designed for an original service life of 40 years, accounting for corrosion, erosion, and material fatigue. The HPCI system was reviewed for license renewal and loss of material due to corrosion, erosion, and cracking due to material fatigue are managed such that the system will continue to perform its intended function consistent with the current licensing basis for the period of extended operation.

A.1.3 UFSAR Appendix C Changes

Section C.2.2.2 – Allowable Limits

(2nd paragraph)

Generic Definition	$P_{60\cancel{40}}$ = 60~~40~~ year event encounter probability				
Upset (likely)	1.0	>	$P_{60\cancel{40}}$	>	10^{-1}
Emergency (low probability)	10^{-1}	>	$P_{60\cancel{40}}$	>	10^{-3}
Faulted (extremely low probability)	10^{-3}	>	$P_{60\cancel{40}}$	>	10^{-6}

(sixth paragraph)

SF_{min} is related to the event probability by the following equation:

$$SF_{min} = \frac{9}{3 - \log_{10} P_{60\cancel{40}}} \qquad \text{(Equation A)}$$

where:

$10^{-1} > P_{\underline{60}\cancel{40}} \geq 10^{-5}$ (Equation A applies)
$10^{-1} > P_{\underline{60}\cancel{40}} \geq 10^{-6}$ (SF_{min} = 1.125)
$1.0 > P_{\underline{60}\cancel{40}} \geq 10^{-1}$ (SF_{min} = 2.25)

(Page C.2-6 of 65, first paragraph)

These expressions show the probabilistic significance of the classical safety factor concept as applied to reactor safety. The SF_{min} values corresponding to the current governing accident event probabilities are summarized as follows:

Item	Governing Loading Conditions	$P_{\underline{60}\cancel{40}}$	SF_{min}

(Last paragraph of section C.2.2)

The minimum safety factor decreases as the event probability diminishes and if the event is too improbable (incredible: $P_{\underline{60}\cancel{40}} < 10^{-6}$) then no safety factor is appropriate or required.

Section C.2.4.1 – Criteria

(2nd paragraph)

Stress analysis requirements and load combinations for the reactor vessel were originally ~~have been~~ evaluated for the cyclic conditions expected throughout a ~~the~~ 40-year life, with the conclusion that ASME code limits are satisfied. The vessel is acceptable for 60 years of operation. The vessel design report contains the results of the detailed design stress analyses performed for the reactor vessel to meet the code requirements. Selected components, considered to possibly have higher than code design primary stresses as a result of rare events or a combination of rare events, have been analyzed in accordance with the requirements of the loading criteria in this appendix. Results of the most critical of those analyses are included in a following section. The conclusion is that the limits in the criteria have been met.

Section C.2 – Recirculation Loop Piping (pages C.2.-32 and C.2-33)

Limiting Structural Loading Conditions, Calculated Stresses, and Allowables for Selected Structures, Systems, and Components

Statement of Criteria	Method of Analysis	Results of Analysis
B. For load combinations that have a very low probability of occurrence, maintain primary stresses below the following limit: $\frac{2.25}{SF}$ times B31.1.0 allowable stresses, where SF - $\frac{9}{3 - \log_{10} P_{60\cancel{40}}}$ and $P_{60\cancel{40}}$ = Probability of load combination occurrence in 60~~40~~-year plant life.	B. Effects from the following loading combinations determined in accordance with rules of B31.1.0: 1. The sum of the longitudinal stresses due to pressure, dead weight, and inertia effects of maximum hypothetical earthquake must be less than 1.8 times the hot allowable stress. The probability of this load occurrence during the 60~~40~~-year plant life is 10^{-3} and SF = 1.5.	[no changes to this column]

Statement of Criteria	Method of Analysis	Results of Analysis
	2. The sum of the longitudinal stresses due to maximum pressure, dead weight and inertia effects of design basis earthquake must be less than 1.5 times the hot allowable stress. The probability of this load occurring during the 60~~40~~ year plant life is 10^{-2} and SF = 1.8.	
	3. The sum of the longitudinal stresses due to maximum pressure, dead weight and inertia effects of maximum hypothetical earthquake must be less than 2.0 times the hot allowable stress. The probability of this load combination occurring during the 60~~40~~ year plant life is .25 x 10^{-3} and SF = 1.36.	

40 10

Section C.2 – Main Steam Piping (pages C.2.-35 and C.2-36)

Limiting Structural Loading Conditions, Calculated Stresses, and Allowables for Selected Structures, Systems, and Components

Statement of Criteria

B. For load combinations that have a very low probability of occurrence, maintain primary stresses below the following limit:

$\frac{225}{SF}$ times B31.1.0 allowable stresses,

where

$$SF - \frac{9}{3 - \log_{10} P_{60\cancel{40}}}$$

and

$P_{60\cancel{40}}$ = Probability of load combination occurrence in 60~~40~~-year plant life.

Method of Analysis

B. Effects from the following loading combinations determined in accordance with rules of B31.1.0:

1. The sum of the longitudinal stresses due to pressure, dead weight, and inertia effects of maximum hypothetical earthquake must be less than 1.8 times the hot allowable stress. The probability of this load occurrence during the 60~~40~~-year plant life is 10^{-3} and SF = 1.5.

Results of Analysis

[no changes to this column]

Statement of Criteria	Method of Analysis	Results of Analysis
	2. The sum of the longitudinal stresses due to maximum pressure, dead weight and inertia effects of design basis earthquake must be less than 1.5 times the hot allowable stress. The probability of this load occurring during the 60~~40~~ year plant life is 10^{-2} and SF = 1.8.	
	3. The sum of the longitudinal stresses due to maximum pressure, dead weight and inertia effects of maximum hypothetical earthquake must be less than 2.0 times the hot allowable stress. The probability of this load combination occurring during the 60~~40~~ year plant life is $.25 \times 10^{-3}$ and SF = 1.36.	

A.1.4 UFSAR Appendix I Changes

Section I.4 – Probability of Line Break

(2nd paragraph)

A BWR typically has about 250 piping components of size 4 inches or larger which are located between the vessel and the first shutoff valve. Of this total, 100 components are associated with steam and 150 with water.If a detectable leak rate is 5 gpm, then (from Figure I-3) a crack in a steamline has a 3.7×10^{-4} probability leading to line break, and a crack in a waterline has 6×10^{-6} probability of line break. The probability of a steam line break in a 60~~40~~ year plant design life is:

$$5.3 \times 10^{-4} \frac{\text{Leaks}}{\text{Component-Year}} \times$$

$$\times\ 100 \text{ components}$$

$$\times\ 60\cancel{40} \text{ years}$$

$$\times\ 3.7 \times 10^{-4}$$

$$= 1.2 \times 10^{-3}\ \cancel{7.8 \times 10^{-4}}$$

This is equivalent to a system reliability of 0.9988 ~~0.9992~~.

For waterlines, the probability of a break is:

$$5.3 \times 10^{-4} \times 150 \times 60\cancel{40} \times 6 \times 10^{-6} = 2.9\ \cancel{1.9} \times 10^{-5},$$

or a reliability of 0.99997 ~~0.99998~~.

A.2 NEW UFSAR SECTION

The following information will be integrated into the UFSAR to document aging management programs and activities credited in the VYNPS license renewal review and time-limited aging analyses evaluated for the period of extended operation.

A.2.0 Supplement For Renewed Operating License

The Vermont Yankee Nuclear Power Station license renewal application (Reference A.2-1) and information in subsequent related correspondence provided sufficient basis for the NRC to make the findings required by 10CFR54.29 (Final Safety Evaluation Report) (Reference A.2-2). As required by 10 CFR 54.21(d), this UFSAR supplement contains a summary description of the programs and activities for managing the effects of aging (Section A.2.1) and a description of the evaluation of time-limited aging analyses for the period of extended operation (Section A.2.2). The period of extended operation is the 20 years after the expiration date of the original operating license.

A.2.1 Aging Management Programs and Activities

The integrated plant assessment for license renewal identified aging management programs 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. This section describes the aging management programs and activities that will be required during the period of extended operation. All aging management programs will be implemented prior to entering the period of extended operation.

VYNPS quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B. The Entergy Quality Assurance Program applies to safety-related structures and components. Corrective actions and administrative (document) control for both safety-related and nonsafety-related structures and components are accomplished per the existing VYNPS corrective action program and document control program and are applicable to all aging management programs and activities that will be required during the period of extended operation. The confirmation process is part of the corrective action program and includes reviews to assure that proposed actions are adequate, tracking and reporting of open corrective actions, and review of corrective action effectiveness. Any follow-up inspection required by the confirmation process is documented in accordance with the corrective action program.

A.2.1.1 Buried Piping Inspection Program

The Buried Piping Inspection Program includes (a) preventive measures to mitigate corrosion and (b) inspections to manage the effects of corrosion on the pressure-retaining capability of buried carbon steel, stainless steel, and gray cast iron components. Preventive measures are in accordance with standard industry practice for maintaining external coatings and wrappings. Buried components are inspected when excavated during maintenance. If trending within the corrective action program identifies susceptible locations, the areas with a history of corrosion problems are evaluated for the need for additional inspection, alternate coating, or replacement.

A focused inspection will be performed within the first 10 years of the period of extended operation, unless an opportunistic inspection (or an inspection via a method that allows an assessment of pipe condition without excavation) occurs within this ten-year period.

A.2.1.2 BWR CRD Return Line Nozzle Program

Under the BWR CRD Return Line Nozzle Program, VYNPS has rerouted the CRD return flow to the reactor water cleanup (RWCU) system, with the rerouted line flow valved open, and capped the CRD return line vessel nozzle to mitigate cracking. Continuing inservice inspection (ISI) examinations monitor the effects of crack initiation and growth on the intended function of the control rod drive return line nozzle and cap.

A.2.1.3 BWR Feedwater Nozzle Program

Under the BWR Feedwater Nozzle Program, VYNPS has replaced the original low flow control valve with a drag valve having improved flow characteristics, replaced the feedwater spargers with interference-fit thermal sleeve spargers, and installed a thermal sleeve bypass leak detection system to mitigate cracking. This program continues enhanced inservice inspection (ISI) of the feedwater nozzles in accordance with the requirements of ASME Section XI, Subsection IWB and the recommendation of General Electric (GE) NE-523-A71-0594 to monitor the effects of cracking on the intended function of the feedwater nozzles.

A.2.1.4 BWR Penetrations Program

The BWR Penetrations Program includes (a) inspection and flaw evaluation in conformance with the guidelines of staff-approved boiling water reactor vessel and internals project (BWRVIP) documents BWRVIP-27 and BWRVIP-49 and (b) monitoring and control of reactor coolant water chemistry in accordance with the guidelines of BWRVIP-130 to ensure the long-term integrity of vessel penetrations and nozzles.

A.2.1.5 BWR Stress Corrosion Cracking Program

The BWR Stress Corrosion Cracking Program includes (a) preventive measures to mitigate intergranular stress corrosion cracking (IGSCC), and (b) inspection and flaw evaluation to monitor IGSCC and its effects on reactor coolant pressure boundary components made of stainless steel, CASS, or nickel alloy.

VYNPS has taken actions to prevent IGSCC and will continue to use materials resistant to IGSCC for component replacements and repairs following the recommendations delineated in NUREG-0313, Generic Letter 88-01, and the staff-approved BWRVIP-75 report. Inspection of piping identified in NRC Generic Letter 88-01 to detect and size cracks is performed in accordance with the staff positions on schedule, method, personnel qualification and sample expansion included in the generic letter and the staff-approved BWRVIP-75 report.

A.2.1.6 BWR Vessel ID Attachment Welds Program

The BWR Vessel ID Attachment Welds Program includes (a) inspection and flaw evaluation in accordance with the guidelines of staff-approved BWR Vessel and Internals Project (BWRVIP) BWRVIP-48 and (b) monitoring and control of reactor coolant water chemistry in accordance with the guidelines of BWRVIP-130 (EPRI Report 1008192) to ensure the long-term integrity and safe operation of reactor vessel inside diameter (ID) attachment welds and support pads.

A.2.1.7 BWR Vessel Internals Program

The BWR Vessel Internals Program includes (a) inspection, flaw evaluation, and repair in conformance with the applicable, staff-approved, industry BWR Vessel and Internals Project (BWRVIP) documents and (b) monitoring and control of reactor coolant water chemistry in accordance with the guidelines of BWRVIP-130 to ensure the long-term integrity of vessel internal components.

A.2.1.8 Containment Leak Rate Program

As described in 10 CFR Part 50, Appendix J, containment leak rate tests are required to assure that (a) leakage through primary reactor containment and systems and components penetrating primary containment shall not exceed allowable values specified in technical specifications or associated bases and (b) periodic surveillance of reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of containment, and systems and components penetrating primary containment. Corrective actions are taken if leakage rates exceed acceptance criteria.

A.2.1.9 Diesel Fuel Monitoring Program

The Diesel Fuel Monitoring Program entails sampling to ensure that adequate diesel fuel quality is maintained to prevent plugging of filters, fouling of injectors, and corrosion of fuel systems. Exposure to fuel oil contaminants such as water and microbiological organisms is minimized by periodic draining and cleaning of tanks and by verifying the quality of new oil before its introduction into storage tanks.

A.2.1.10 Environmental Qualification (EQ) of Electric Components Program

The EQ of Electric Components Program manages the effects of component thermal, radiation, and cyclic aging through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods. As required by 10 CFR 50.49, EQ components not qualified for the current license term are refurbished, replaced, or their qualification extended prior to reaching the aging limits established in the evaluations. Aging evaluations for EQ components are considered time-limited aging analyses (TLAAs) for license renewal.

A.2.1.11 Fatigue Monitoring Program

In order not to exceed design limits on fatigue usage, the Fatigue Monitoring Program tracks the number of critical thermal and pressure transients for selected reactor coolant system components. The program ensures the validity of analyses that explicitly assumed a fixed number of thermal and pressure transients by assuring that the actual effective number of transients does not exceed the assumed limit.

The transient cycles tracked by this program are referenced in Section 4.2.5.

A.2.1.12 Fire Protection Program

The Fire Protection Program includes a fire barrier inspection and a diesel-driven fire pump inspection. The fire barrier inspection requires periodic visual inspection of fire barrier penetration seals, fire barrier walls, ceilings, and floors, and periodic visual inspection and functional tests of fire rated doors to ensure that their operability is maintained. The diesel-driven fire pump inspection requires that the pump be periodically tested to ensure that the fuel supply line can perform its intended function.

Corrective actions, confirmation process, and administrative controls in accordance with the requirements of 10CFR Part 50 Appendix B are applied to the Fire Protection Program.

A.2.1.13 Fire Water System Program

The Fire Water System Program applies to water-based fire protection systems that consist of sprinklers, nozzles, fittings, valves, hydrants, hose stations, standpipes, and aboveground and underground piping and components that are tested in accordance with applicable National Fire Protection Association (NFPA) codes and standards. Such testing assures functionality of systems. Also, many of these systems are normally maintained at required operating pressure and monitored such that leakage resulting in loss of system pressure is immediately detected and corrective actions initiated.

In addition, wall thickness evaluations of fire protection piping are periodically performed on system components using non-intrusive techniques (e.g., volumetric testing) to identify evidence of loss of material due to corrosion.

A sample of sprinkler heads will be inspected using the guidance of NFPA 25 (2002 Edition) Section 5.3.1.1.1, which states, "Where sprinklers have been in place for 50 years, they shall be replaced or representative samples from one or more sample areas shall be submitted to a recognized testing laboratory for field service testing." This sampling will be repeated every 10 years after initial field service testing.

A.2.1.14 Flow-Accelerated Corrosion Program

The Flow-Accelerated Corrosion Program applies to safety-related and nonsafety-related carbon steel components carrying two-phase or single-phase high-energy fluid $\geq$ 2% of plant operating time.

The program, based on EPRI recommendations for an effective flow-accelerated corrosion program, predicts, detects, and monitors FAC in plant piping and other pressure retaining components. This program includes (a) an evaluation to determine critical locations, (b) initial operational inspections to determine the extent of thinning at these locations, and (c) follow-up inspections to confirm predictions. The program specifies repair or replacement of components as necessary.

A.2.1.15 Heat Exchanger Monitoring Program

The Heat Exchanger Monitoring Program inspects heat exchangers for degradation. If degradation is found, then an evaluation is performed to evaluate its effects on the heat exchanger's design functions including its ability to withstand a seismic event.

Representative tubes within the population of heat exchangers are eddy current tested at a frequency determined by internal and external operating experience to ensure that effects of aging are identified prior to loss of intended function. Along with each eddy current test, visual inspections are performed on accessible heat exchanger heads, covers and tube sheets to monitor surface condition for indications of loss of material. The population of heat exchangers includes the HPCI gland seal condenser, HPCI lube oil cooler, RCIC lube oil cooler, CST steam reheat coil, drywell atmospheric cooling units (RRU-1, 2, 3 & 4), reactor recirculation pump seal water coolers, reactor recirculation pump motor upper and lower bearing oil coolers, and reactor recirculation pump motor air coolers.

A.2.1.16 Inservice Inspection—Containment Inservice Inspection (CII) Program

The VYNPS Containment Inservice Inspection (CII) Program is a plant-specific program encompassing the requirements for the inspection of Class MC pressure-retaining components (primary containment) and their integral attachments in accordance with the requirements of 10CFR50.55a(b)(2) and the 1998 Edition of ASME Section XI with 2000 Addenda, Inspection Program B.

The primary inspection method for the primary containment and its integral attachments is visual examination. Visual examinations are performed either directly or remotely with sufficient illumination and resolution suitable for the local environment to assess general conditions that may affect either the containment structural integrity or leak tightness of the pressure retaining component. The program includes augmented ultrasonic exams to measure wall thickness of the containment drywell shell.

A.2.1.17 Inservice Inspection—Inservice Inspection (ISI) Program

The VYNPS Inservice Inspection (ISI) Program is based on ASME Inspection Program B (IWA-2432), which has 10-year inspection intervals. Every 10 years the program is updated to the latest ASME Section XI code edition and addendum approved by the Nuclear Regulatory Commission in 10CFR50.55a. On September 1, 2003 VYNPS entered the fourth ISI interval. The Code Edition and Addenda used for the fourth interval is the 1998 Edition with 2000 addenda.

The program consists of periodic volumetric, surface, and visual examination of components and their supports for assessment, signs of degradation, flaw evaluation and corrective actions.

A.2.1.18 Instrument Air Quality Program

The Instrument Air Quality Program ensures that instrument air supplied to components is maintained free of water and significant contaminants, thereby preserving an environment that is not conducive to loss of material. Dewpoint, particulate contamination, and hydrocarbon concentration are periodically checked to verify the instrument air quality is maintained.

A.2.1.19 Non-EQ Inaccessible Medium-Voltage Cable Program

In the Non-EQ Inaccessible Medium-Voltage Cable Program, in-scope medium-voltage cables, not designed for, but exposed to significant moisture and voltage are tested at least once every ten years to provide an indication of the condition of the conductor insulation. The specific test performed is a proven test for detecting deterioration of the insulation system due to wetting, such as power factor, partial discharge, polarization index, or other testing that is state-of-the-art at the time the test is performed. Significant moisture is defined as periodic exposures that last more than a few days. Significant voltage exposure is defined as being subjected to system voltage for more than 25% of the time.

Inspections for water collection in cable manholes and conduit occur at least once every two years.

A.2.1.20 Non-EQ Instrumentation Circuits Test Review Program

Under the Non-EQ Instrumentation Circuits Test Review Program, calibration or surveillance results for non-EQ electrical cables in circuits with sensitive, high voltage, low-level signals; (i.e., neutron flux monitoring instrumentation); are reviewed. Most neutron flux monitoring system cables and connections are calibrated as part of the instrumentation loop calibration at the normal calibration frequency, which provides sufficient indication of the need for corrective actions based on acceptance criteria related to instrumentation loop performance. The review of calibration results is performed once every 10 years.

For neutron flux monitoring system cables that are disconnected during instrument calibrations, testing is performed at least once every 10 years using a proven method for detecting deterioration for the insulation system (such as insulation resistance tests, or time domain reflectometry).

A.2.1.21 Non-EQ Insulated Cables and Connections Program

The Non-EQ Insulated Cables and Connections Program provides reasonable assurance that intended functions of insulated cables and connections exposed to adverse localized environments caused by heat, radiation and moisture can be maintained consistent with the current licensing basis through the period of extended operation. An adverse localized environment is significantly more severe than the specified service condition for the insulated cable or connection.

A representative sample of accessible insulated cables and connections in adverse localized environments is visually inspected at least once every ten years for cable and connection jacket surface anomalies such as embrittlement, discoloration, cracking or surface contamination.

A.2.1.22 Oil Analysis Program

The Oil Analysis Program maintains oil systems free of contaminants (primarily water and particulates) thereby preserving an environment that is not conducive to loss of material, cracking, or fouling. Activities include sampling and analysis of lubricating oil for detrimental contaminants, water, and particulates.

Sampling frequencies are based on vendor recommendations, accessibility during plant operation, equipment importance to plant operation, and previous test results.

A.2.1.23 One-Time Inspection Program

The elements of the One-Time Inspection Program include (a) determination of the sample size based on an assessment of materials of fabrication, environment, plausible aging effects, and operating experience; (b) identification of the inspection locations in the system or component based on the aging effect; (c) determination of the examination technique, including acceptance criteria that would be effective in managing the aging effect for which the component is examined; and (d) evaluation of the need for follow-up examinations to monitor the progression of any aging degradation.

A one-time inspection activity is used to verify the effectiveness of the water chemistry control programs by confirming that unacceptable cracking, loss of material, and fouling is not occurring on components within systems covered by water chemistry control programs [Sections A.2.1.34, A.2.1.35, and A.2.1.36].

One-time inspection activities on

- internal carbon steel surfaces exposed to indoor air in the standby gas treatment system,
- internal surfaces of carbon steel and copper alloy components in the potable water and radwaste systems containing untreated water,
- carbon steel retired in place (RIP) system components in the area around containment penetration X-21,
- small bore piping in the reactor coolant system and associated systems that form the reactor coolant pressure boundary,
- reactor vessel flange leak-off line, and
- main steam flow restrictors

are used to confirm that loss of material, cracking, and reduction of fracture toughness, as applicable, are not occurring or are so insignificant that an aging management program is not warranted.

When evidence of an aging effect is revealed by a one-time inspection, routine evaluation of the inspection results will identify appropriate corrective actions.

A.2.1.24 Periodic Surveillance And Preventive Maintenance Program

The Periodic Surveillance and Preventive Maintenance Program includes periodic inspections and tests that manage aging effects not managed by other aging management programs. The preventive maintenance and surveillance testing activities are generally implemented through repetitive tasks or routine monitoring of plant operations.

Temperatures are monitored during periodic emergency diesel generator (EDG), John Deere diesel, and control room chilled water condenser surveillance tests to verify that associated heat exchangers are capable of removing the required amount of heat, thereby managing fouling of the heat exchanger tubes.

Periodic surveillance demonstrates the ability of the standby gas treatment system to maintain a test vacuum confirming the absence of aging effects for the reactor building exterior concrete walls.

Periodic leakage testing on the reactor building railroad inner and outer doors verifies the ability of the rubber seals to perform their intended function.

Periodic inspections using visual or other non-destructive examination techniques verify that the following components are capable of performing their intended function.

- reactor building crane, rails, and girders
- refueling platform carbon steel components
- equipment lock sliding doors
- yard concrete handholes and manholes
- HPCI gland seal exhauster fan housing and piping
- standby gas treatment system demister and demister loop seal components
- hydrogen analyzer pre-coolers
- housings of the RHR corner room recirculation units
- EDG and John Deere diesel intake air, air start, and exhaust components
- EDG intake air cooler
- John Deere diesel lube oil coolers and radiators
- housings of ECCS corner room recirculation units, control room HVAC package heating and cooling coils, control room chiller, and control room chilled water condensers
- control room ventilation fan duct flexible connections
- nonsafety-related components of the circulating water, diesel generator air start, and instrument air supply systems

A.2.1.25 Reactor Head Closure Studs Program

The Reactor Head Closure Studs Program includes inservice inspection (ISI) in conformance with the requirements of ASME Section XI, Subsection IWB, and preventive measures (e.g. rust inhibitors, stable lubricants, appropriate materials) to mitigate cracking and loss of material of reactor head closure studs, nuts, washers, and bushings.

A.2.1.26 Reactor Vessel Surveillance Program

VYNPS is a participant in the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) as incorporated into the plant Technical Specifications by Amendment 218. The Reactor Vessel Surveillance Program monitors changes in the fracture toughness properties of ferritic materials in the reactor pressure vessel (RPV) beltline region. As BWRVIP-ISP capsule test reports become available for RPV materials representative of VYNPS, the actual shift in the reference temperature for nil-ductility transition of the vessel material may be updated. In accordance with Appendix H to 10CFR50, VYNPS reviews relevant test reports to assure compliance with fracture toughness requirements and P-T limits.

BWRVIP-116, "BWR Vessel and Internals Project Integrated Surveillance Program (ISP) Implementation for License Renewal," describes the design and implementation of the ISP during the period of extended operation. BWRVIP-116 identifies additional capsules, their withdrawal schedule, and contingencies to ensure that the requirements of 10CFR50 Appendix H are met for the period of extended operation.

A.2.1.27 Selective Leaching Program

The Selective Leaching Program ensures the integrity of components made of cast iron, bronze, brass, and other alloys exposed to a raw water, treated water, or groundwater environment that may lead to selective leaching of one of the metal components. The program includes a one-time visual inspection and hardness measurement of selected components that may be susceptible to selective leaching to determine whether loss of material due to selective leaching is occurring, and whether the process will affect the ability of the components to perform their intended function for the period of extended operation.

A.2.1.28 Service Water Integrity Program

The Service Water Integrity Program relies on implementation of the recommendations of GL 89-13 to ensure that the effects of aging on the service water systems (SWS) will be managed for the period of extended operation. The SWS include the service water, residual heat removal service water, and alternate cooling systems. The program includes component inspections for erosion, corrosion, and blockage and performance monitoring to verify the heat transfer capability of the safety-related heat exchangers cooled by SWS. Chemical treatment using biocides and chlorine and periodic cleaning and flushing of redundant or infrequently used loops are the methods used to control or prevent fouling within the heat exchangers and loss of material in SWS components.

A.2.1.29 Structures Monitoring—Masonry Wall Program

The objective of the Masonry Wall Program is to manage cracking so that the evaluation basis established for each masonry wall within the scope of license renewal remains valid through the period of extended operation.

The program includes all masonry walls identified as performing intended functions in accordance with 10 CFR 54.4. Included walls are the 10CFR50.48-required walls and masonry walls in the reactor building, intake structure, control room building, and turbine building.

Masonry walls are visually examined at a frequency selected to ensure there is no loss of intended function between inspections.

A.2.1.30 Structures Monitoring—Structures Monitoring Program

Structures monitoring is in accordance with 10 CFR 50.65 (Maintenance Rule) as addressed in Regulatory Guide (RG) 1.160 and NUMARC 93-01. Periodic inspections are used to monitor condition of structures and structural components to ensure there is no loss of structure or structural component intended function.

A.2.1.31 Structures Monitoring—Vernon Dam FERC Program

The Vernon dam is subject to the Federal Energy Regulatory Commission (FERC) 5-year inspection program. This program consists of a visual inspection by a qualified independent consultant approved by FERC and is in compliance with Title 18 of the Code of Federal Regulations, Conservation of Power and Water Resources, Part 12 (Safety of Water Power Projects and Project Works), Subpart D (Inspection by Independent Consultant). The NRC has found that mandated FERC 5-year inspection programs are acceptable for aging management.

A.2.1.32 System Walkdown Program

The System Walkdown Program entails inspections of external surfaces of components subject to aging management review. The program is also credited with managing loss of material from internal surfaces, for situations in which internal and external material and environment combinations are the same such that external surface condition is representative of internal surface condition.

Surfaces that are inaccessible during plant operations are inspected during refueling outages. Surfaces are inspected at frequencies to provide reasonable assurance that effect of aging will be managed such that applicable components will perform their intended function during the period of extended operation.

A.2.1.33 Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program

The purpose of the Thermal Aging and Neutron Irradiation Embrittlement of CASS Program is to assure that reduction of fracture toughness due to thermal aging and reduction of fracture toughness due to radiation embrittlement will not result in loss of the intended function. This program evaluates CASS components in the reactor vessel internals and requires non-destructive examinations as appropriate.

A.2.1.34 Water Chemistry Control—Auxiliary Systems Program

The purpose of the Water Chemistry Control - Auxiliary Systems Program is to manage aging effects for components exposed to treated water.

Program activities include sampling and analysis of stator cooling water and plant heating boiler systems, and flushing of the John Deere Diesel cooling water system.

A.2.1.35 Water Chemistry Control—BWR Program

The objective of the Water Chemistry Control-BWR Program is to manage aging effects caused by corrosion and cracking mechanisms. The program relies on monitoring and control of water chemistry based on EPRI Report 1008192 (BWRVIP-130). BWRVIP-130 has three sets of guidelines: one for primary water, one for condensate and feedwater, and one for control rod drive (CRD) mechanism cooling water. EPRI guidelines in BWRVIP-130 also include recommendations for controlling water chemistry in the torus, condensate storage tank, demineralized water storage tanks, and spent fuel pool.

The Water Chemistry Control - BWR Program optimizes the primary water chemistry to minimize the potential for loss of material and cracking. This is accomplished by limiting the levels of contaminants in the RCS that could cause loss of material and cracking. Additionally, VYNPS has instituted hydrogen water chemistry (HWC) with noble metals to limit the potential for intergranular SCC (IGSCC) through the reduction of dissolved oxygen in the treated water.

A.2.1.36 Water Chemistry Control—Closed Cooling Program

The Water Chemistry Control-Closed Cooling Program includes preventive measures that manage loss of material, cracking, and fouling for closed cooling water systems (reactor building closed cooling water, turbine building closed cooling water, AOG closed cooling water, emergency diesel generator closed cooling water, AOG refrigerant skid water, and chilled water). These chemistry activities provide for monitoring and controlling closed cooling water chemistry using VYNPS procedures and processes based on EPRI guidance for closed cooling water chemistry.

A.2.2 Evaluation of Time-Limited Aging Analyses

In accordance with 10 CFR 54.21(c), an application for a renewed license requires an evaluation of time-limited aging analyses (TLAA) for the period of extended operation. The following TLAA have been identified and evaluated to meet this requirement.

A.2.2.1 Reactor Vessel Neutron Embrittlement

The reactor vessel neutron embrittlement TLAA either remain valid for the period of extended operation (P-T limits) in accordance with 10 CFR 54.21(c)(1)(i) or have been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii). Fifty-four EFPY would be the effective full power years at the end of the period of extended operation assuming an average capacity factor of 90% for 60 years.

A.2.2.1.1 Reactor Vessel Fluence

Calculated fluence is based on a time-limited assumption defined by the operating term. As such, fluence is the time-limited assumption for the time-limited aging analyses that evaluate reactor vessel embrittlement.

GE's Licensing Topical Report NEDC-32983P-A, which was approved by the NRC for licensing applications in Reference A.2-6, documents the method used for the neutron flux calculation. The NRC found that, in general, this method adheres to the guidance in Regulatory Guide 1.190 for neutron flux evaluation.

A.2.2.1.2 Pressure-Temperature Limits

In March 2003, VYNPS submitted a license amendment request (Reference A.2-4) to change the P-T limits to incorporate data from analysis of the first VYNPS surveillance capsule and to extend the curves to 32 EFPY. The NRC approved this submittal as Amendment 218 to the VYNPS license (Reference A.2-5). As stated in the SER (Reference A.2-5), VYNPS used conservative values for determining the 32 EFPY P-T limits. The projected fluence and ARTs for 54 EFPY, including the extended power uprate, are still less than the conservative values on which the 32 EFPY P-T curves are based. As such the current 32 EFPY pressure-temperature limits do not require modification for the period of extended operation and the TLAA remains valid in accordance with 10 CFR 54.21(c)(1)(i).

A.2.2.1.3 Charpy Upper-Shelf Energy

The predictions for percent drop in C_VUSE at 54 EFPY are based on chemistry data and unirradiated C_VUSE data submitted to the NRC in support of the VYNPS power uprate (Reference A.2-7), and the ¼T fluence maximum value.

The 54 EFPY C_VUSE values were calculated using Regulatory Guide 1.99, Position 1, Figure 2; specifically, the formula for the lines was used to calculate the percent drop in C_VUSE.

Because VYNPS does not have complete unirradiated data for all beltline materials, equivalent margin analyses were done for the limiting plate and weld, using the technique in NEDO-32205 (Reference A.2-3). The results showed that the percent reductions in C_VUSE are less than the limiting decreases identified in the NRC SER for BWRVIP-74 (Reference A.2-8). A conservative assumption used in the calculation of C_VUSE reduction is that no credit is taken for axial or azimuthal lead factors to reduce the peak fluence. Instead, the maximum calculated ¼T fluence value is assumed for all plates and welds.

A.2.2.1.4 Adjusted Reference Temperature

VYNPS has projected values for ART at 54 EFPY using the methodology of Regulatory Guide 1.99. These values were calculated using the chemistry data, margin values, initial RT_{NDT} values, and chemistry factors (CFs) submitted to the NRC in support of the VYNPS power uprate (Reference A.2-7), and the ¼T fluence maximum value. New fluence factors (FFs) were calculated using the expression in Regulatory Guide 1.99, Revision 2, Equation 2 using 54 EFPY fluence values.

The RT_{NDT} TLAA has been projected through the period of extended operation, with acceptable results, in accordance with 10 CFR 54.21(c)(1)(ii).

A.2.2.1.5 Reactor Vessel Circumferential Weld Inspection Relief

Relief from reactor vessel circumferential weld examination requirements under Generic Letter 98-05 is based on assessments indicating an acceptable probability of failure per reactor operating year. The analysis is based on reactor vessel metallurgical conditions as well as flaw indication sizes and frequencies of occurrence that are expected at the end of a licensed operating period.

VYNPS requested NRC approval for this relief for the remainder of the original 40-year license term (Reference A.2-9). The basis for this relief request is an analysis that satisfied the limiting conditional failure probability for the circumferential welds at the expiration of the current license, based on the NRC SERs for BWRVIP-05 (Reference A.2-11) and BWRVIP-74 (Reference A.2-13) and the extent of neutron embrittlement.

The 54 EFPY fluence value for VYNPS is considerably lower than the corresponding 64 EFPY generic value. As a result, the shift in reference temperature is lower than the 64 EFPY shift in the NRC analysis. However, the unirradiated reference temperature of the VYNPS material is higher than the initial value assumed in the NRC analysis. This combination of opposing effects yields an adjusted reference temperature that is lower than the NRC mean analysis value. Therefore, this TLAA has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A.2.2.1.6 Reactor Vessel Axial Weld Failure Probability

The BWRVIP recommendations for inspection of reactor vessel shell welds (BWRVIP-05, Reference A.2-10) are based on generic analyses supporting an NRC SER (References A.2-11 and A.2-12) conclusion that the generic-plant axial weld failure rate is no more than 5 x 10^{-6} per reactor year as calculated in the BWRVIP-74 SER (Reference A.2-13). BWRVIP-05 showed that this axial weld failure rate is orders of magnitude greater than the 40-year end-of-life circumferential weld failure probability, and used this analysis to justify relief from inspection of the circumferential welds as described above.

The basis for this relief request was a plant-specific analysis that showed the limiting conditional failure probability for the VYNPS circumferential welds at the end of the original operating term were less than the values calculated in the BWRVIP-05 SER. The BWRVIP-05 SER concluded that the reactor vessel failure frequency due to failure of the limiting axial welds in the BWR fleet at the end of 40 years of operation is less than 5 x 10^{-6} per reactor year. This failure frequency is dependent upon given assumptions of flaw density, distribution, and location. The failure frequency also assumes that essentially 100% of the reactor vessel axial welds will be inspected.

The BWRVIP-74 SER states it is acceptable to show that the mean RT_{NDT} of the limiting beltline axial weld at the end of the period of extended operation is less than the limiting value given in the SERs for BWRVIP-74 and BWRVIP-05. The projected 54 EFPY mean RT_{NDT} values for VYNPS are less than the limiting 64 EFPY RT_{NDT} in the analysis performed by the NRC staff (Table 2.6-5 of the BWRVIP-05 SER). As such, this TLAA has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A.2.2.2 Metal Fatigue

A.2.2.2.1 Class 1 Metal Fatigue

Class 1 components evaluated for fatigue and flaw growth include the reactor pressure vessel (RPV) and appurtenances, certain reactor vessel internals, the reactor recirculation system (RRS), and the reactor coolant system (RCS) pressure boundary. The VYNPS Class 1 systems include components within the ASME Section XI, IWB inspection boundary.

The design of the reactor vessel internals is in accordance with the intent of ASME Section III. A review of design basis documents reveals that the only reactor vessel internals components for which there is a fatigue evaluation are the core shroud tie rods (stabilizers), the result of a repair to structurally replace circumferential shroud welds.

The VYNPS fatigue monitoring program will assure that the allowed number of transient cycles is not exceeded. The program requires corrective action if transient cycle limits are approached. Consequently, the TLAA (fatigue analyses) based on those transients will remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i) or the effects of aging on the intended function(s) will be adequately managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

A.2.2.2.2 Non-Class 1 Metal Fatigue

For non-Class 1 components identified as subject to cracking due to fatigue, a review of system operating characteristics was conducted to determine the approximate frequency of any significant thermal cycling. If the number of equivalent full temperature cycles for 60 years of operation is below the limit used for the original design (usually 7000 cycles), the component is acceptable for the period of extended operation. If the number of equivalent full temperature cycles exceeds the limit, the individual stress calculations require evaluation. No components were identified with projected cycles exceeding 7000. Therefore, the TLAA for non-Class 1 piping and components remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(i).

A.2.2.2.3 Environmental Effects on Fatigue

The effects of reactor water environment on fatigue were evaluated for license renewal. Projected cumulative usage factors (CUFs) were calculated for the limiting locations identified in NUREG/CR-6260. Several locations may exceed a CUF of 1.0 with consideration of environmental effects during the period of extended operation. For these locations, prior to the period of extended operation, VYNPS will (1) refine the fatigue analysis to lower the predicted CUF to less than 1.0; (2) manage fatigue at the affected locations with an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method acceptable to the NRC); or (3) repair or replace the affected locations.

A.2.2.3 Environmental Qualification of Electrical Components

The VYNPS EQ program implements the requirements of 10 CFR 50.49 (as further defined by the DOR Guidelines, NUREG-0588, and Reg. Guide 1.89). The program requires action before individual components exceed their qualified life. In accordance with 10 CFR 54.21(c)(1)(iii), implementation of the EQ program provides reasonable assurance that the effects of aging on components associated with EQ TLAA will be adequately managed such that the intended functions can be maintained for the period of extended operation.

A.2.2.4 Fatigue of Primary Containment, Attached Piping, and Components

The torus and attached piping systems were analyzed as part of the Mark I containment long-term program, using methods and assumptions consistent with NUREG-0661. The fatigue analyses performed included the torus, SRV piping and penetrations, and other torus attached piping.

The fatigue analysis of the torus during normal operation and upset conditions has been projected through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii). The Mark 1 generic fatigue analysis of the SRV piping is projected for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii). The fatigue analysis for the SRV torus penetration remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i). The Mark 1 generic analyses for torus attached piping have been projected through the period of extended operation in accordance with 10 CFR 50.21(c)(ii).

A.2.2.5 Core Plate Rim Hold-Down Bolt Loss of Preload

The calculation of loss of preload on the core plate rim hold-down bolts is a TLAA. BWRVIP-25 calculated the loss of preload for these bolts for forty years. Appendix B to BWRVIP-25 projected this calculation to 60 years, showing that the VYNPS bolts would experience only 5 to 19 percent loss of preload. This TLAA is thus projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A.2.2.6 Lower Plenum Fatigue Analysis

BWRVIP-47 identified fatigue analyses of lower plenum pressure boundary components as TLAA. The only lower plenum CUF identified for VYNPS was a CUF for the CRD penetrations equal to 0.13. This CUF is maintained by limiting the allowed number of transients and as such the associated analysis remains valid for the period of extended operation per 10 CFR 54.21(c)(1)(i).

A.2.2.7 Vessel ID Attachment Welds Fatigue Analysis

The BWRVIP-48 fatigue analyses for various configurations of different vessel ID bracket attachments are considered TLAA. The analyses addressed VYNPS bracket configurations. Analysis of fatigue for 60 years showed that no CUFs are above 0.4. This analysis remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.2.2.8 Instrument Penetrations Fatigue Analysis

The BWRVIP-49 fatigue analysis for several configurations of instrumentation penetrations, including the VYNPS configuration, is considered a TLAA. Analysis of fatigue for 60 years showed that all CUFs are below 0.4. This analysis remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.2.3 References

A.2-1 VYNPS License Renewal Application

A.2-2 [NRC SER for VYNPS License Renewal - later]

A.2-3 GE Report NEDO-33205-A, “10CFR50 Appendix G, Equivalent Margin Analysis for Low Upper Shelf Energy in BWR/2 through BWR/6 Vessels,” Revision 1, February, 1994.

A.2-4 Balduzzi, M. A. to USNRC Document Control Desk, “Technical Specifications Proposed Change No. 258, RPV Fracture Toughness and Material Surveillance Requirements,” letter BVY 03-29 dated March 26, 2003.

A.2-5 Ennis, R. B. (NRC) to M. Kansler, Vermont Yankee Nuclear Power Station, “Issuance of Amendment re: Reactor Pressure Vessel Fracture Toughness and Material Surveillance Requirements (TAC NOS. MB8119 and MB8379),” letter NVY 04-027 dated March 29, 2004.

A.2-6 Richards, S.A. (NRC), to J. F. Klapproth, Safety Evaluation for NEDC-32983P, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation (TAC No. MA9891), MFN 01-050, letter dated September 14, 2001.

A.2-7 Thayer, J.K., to USNRC Document Control Desk, Technical Specification Proposed Change No. 263, Extended Power Uprate, letter BVY 03-80 dated September 10, 2003.

A.2-8 Grimes, C.I. (NRC), to C. Terry, (BWRVIP Chairman), Acceptance for referencing of EPRI Proprietary Report TR-113596, "BWR Vessel and Internals Project, BWR Reactor Vessel Inspection and Flaw Evaluation Guidelines (BWRVIP-74) and Appendix A, 'Demonstration of Compliance with the Technical Information requirements of the License Renewal Rule (10CRF54.21)', letter dated October 18, 2001.

A.2-9 BVY 03-83, Supplement to Fourth-Interval Inservice Inspection (ISI) Program Plan - Submittal of Relief Request ISI-06, September 25, 2003.

A.2-10 BWRVIP-05, EPRI Report TR-105697, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," for the Boiling Water Reactor Owners Group (Proprietary), September 28, 1995, with supplementing letters of June 24 and October 29, 1996; May 16, June 4, June 13, and December 18, 1997; and January 13, 1998.

A.2-11 Lainas, G. C. (NRC), to C. Terry (Niagara Mohawk Power Company, BWRVIP Chairman), BWRVIP-05 SER (Final), Final Safety Evaluation of the BWRVIP Vessel and Internals Project BWRVIP-05 Report (TAC No. M93925), letter dated July 28, 1998.

A.2-12 Strosnider, J. R., Jr., (NRC) to C. Terry (BWRVIP Chairman), BWRVIP-05 SER, Supplement to Final Safety Evaluation of the BWRVIP Vessel and Internals Project BWRVIP-05 Report (TAC No. MA3395), letter dated March 7, 2000.

A.2-13 Grimes, C.I. (NRC), to C. Terry, (BWRVIP Chairman), “Acceptance for Referencing of EPRI Proprietary Report TR-113596, BWR Vessel and Internals Project, BWR Reactor Vessel Inspection and Flaw Evaluation Guidelines (BWRVIP-74) and Appendix A, Demonstration of Compliance with the Technical Information requirements of the License Renewal Rule (10CRF54.21),” letter dated October 18, 2001.

A.2-14 BWRVIP-74-A, (EPRI Report 1008872), "BWRVIP-74-A: BWR Vessel and Internals Project BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines for License Renewal," June 2003.