RAS M-287

Report No.: SIR-07-132-NPS
Revision No.: 1
Project No.: VY-16Q
File No.: VY-16Q-404
December 2007

Summary Report of Plant-Specific Environmental Fatigue Analyses for the Vermont Yankee Nuclear Power Station

Prepared for:
Entergy Nuclear Operations. Inc.
(Contract Order No. 10150394)

Prepared by:
Structural Integrity Associates, Inc.
Centennial, CO

Prepared by: _____ Date: 12/15/2007
T. J. Herrmann P.E.

Reviewed by: Gary L. Stevens Date: 12/15/2007
G. L. Stevens, P.E.

Approved by: _____ Date: 12/15/2007
T. J. Herrmann P.E.

DOCKETED
USNRC

August 12, 2008 (11:00am)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

Template Secy-028

U.S. NUCLEAR REGULATORY COMMISSION
In the Matter of Entergy Nuclear Vermont Yankee LLC
Docket No. 50-271 Official Exhibit No. E2-24-VY
OFFERED by (Applicant/Licensee) Intervenor
NRC Staff Other
IDENTIFIED on 7/21/08 Witness/Panel NEC2
Action Taken: (ADMITTED) REJECTED WITHDRAWN
Reporter/Clerk MAC

DS-03

REVISION CONTROL SHEET

Document Number: SIR-07-132-NPS

Title: Summary Report of Plant-Specific Environmental Fatigue Analyses for the Vermont Yankee Nuclear Power Station

Client: Entergy Nuclear Operations, Inc.

SI Project Number: VY-16Q

Section	Pages	Revision	Date	Comments
1.0 2.0 3.0 4.0 5.0	1-1 2-1 – 2-2 3-1 – 3-18 4-1 5-1 – 5-2	0	7/27/07	Initial issue.
1.0 2.0 3.0 4.0 5.0	1-1 2-1 – 2-2 3-1 – 3-18 4-1 5-1 – 5-2	1	12/15/07	Revised based on revision to VY-16Q-309 and VY-16Q-310 associated with CAR 07-25 and NCR 07-11. Editorial correction on page 3-5.

Table of Contents

LIST OF TABLES

1.0 INTRODUCTION

This report provides the results of plant-specific environmental fatigue calculations for the Vermont Yankee Nuclear Power Station (VYNPS). These calculations are performed to satisfy Nuclear Regulatory Commission (NRC) requirements for Entergy Nuclear Vermont Yankee's (ENVY's) License Renewal Application for VYNPS, submitted to the NRC in 2006.

Generic Safety Issue (GSI) 166 [1], later renumbered as GSI-190 [2], was identified by the NRC staff because of concerns about the effects of reactor water environments on fatigue life during the period of extended operation [3]. GSI-190 was closed in December 1999, based on a memorandum from NRC-RES to NRC-NRR [4]. Timing of issue closure required the first two license renewal applicants – Baltimore Gas & Electric Company for the Calvert Cliffs Nuclear Power Plant and Duke Energy for the Oconee Nuclear Station – to address GSI-190 in their applications prior to issue closure. Each of the applicants developed responses to the NRC staff without the benefit of information from GSI-190 closure. Subsequent license renewal applicants have had the benefit of this information that could be used to guide the resolution of the fatigue design basis and time limited aging analyses (TLAA) issues.

This report addresses VYNPS reactor water environmental effects on the fatigue life of selected fatigue-sensitive reactor coolant system (RCS) components, in accordance with the resolution of GSI-190, as required by Chapter X, "Time Limited Aging Analyses Evaluation of Aging Management Programs Under 10CFR54.21(c)(1)(iii), Section X.M1 "Metal Fatigue of Reactor Coolant Pressure Boundary", of the Generic Aging Lessons Learned (GALL) Report [5]. Consistent with the requirements of the GALL report, the method chosen for this environmentally-assisted fatigue (EAF) evaluation is based on evaluation of the locations identified in NUREG/CR-6260 [6] and the NRC-accepted EAF relationships generated from laboratory data, as documented in References [7] and [8].

2.0 BACKGROUND

As a part of the NRC's Fatigue Action Plan [3], incorporation of environmental fatigue effects originally involved a reduced set of fatigue design curves, such as those proposed by Argonne National Laboratory (ANL) in NUREG/CR-5999 [9]. As a part of the effort to close GSI-166 (later GSI-190) for operating nuclear power plants during the current 40-year licensing term, Idaho National Engineering Laboratory (INEL) evaluated fatigue-sensitive component locations at plants designed by all four U. S. nuclear steam supply system (NSSS) vendors. The ANL fatigue curves were used by INEL to recalculate the cumulative usage factors (CUFs) for fatigue-sensitive component locations in early and late vintage Combustion Engineering (CE) pressurized water reactors (PWRs), early and late vintage Westinghouse PWRs, early and late vintage General Electric (GE) boiling water reactors (BWRs), and Babcock & Wilcox Company (B&W) PWRs. The results of the INEL calculations were published in NUREG/CR-6260 [6]. The INEL calculations took advantage of conservatisms present in governing ASME Code fatigue calculations, including the numbers of actual plant transients relative to the numbers of design-basis transients, but did not recalculate stress ranges based on actual plant transient profiles. The BWR calculations, especially the early-vintage GE BWR calculations, are directly relevant to VYNPS.

The fatigue-sensitive component locations chosen for the older-vintage GE BWR plant were: (1) the reactor vessel shell and lower head, (2) the reactor vessel feedwater nozzle, (3) the reactor recirculation piping (including the reactor inlet and outlet nozzles), (4) the core spray line reactor vessel nozzle and associated Class 1 piping, (5) the residual heat removal (RHR) return line Class 1 piping, and (6) the feedwater line Class 1 piping. For the recirculation, RHR, and feedwater piping locations, INEL performed representative design-basis fatigue calculations. This is because no CUF calculations had originally been performed since the piping systems for the selected BWR plant were initially designed and analyzed in accordance with the criteria of USAS B31.1-1967 [10].

The six RCS component locations described above are evaluated for EAF effects for VYNPS in this report through separate plant-specific analyses of nine VY component locations (with report section numbers indicated): the reactor pressure vessel (RPV) shell and lower head (3.1); the RPV shell at the shroud support junction (3.1); the feedwater nozzle (3.2); the recirculation / residual heat removal Class 1 piping (3.3.1 and 3.5); the recirculation inlet nozzle forging (3.3.2); the recirculation inlet nozzle safe end (3.3.2); the recirculation outlet nozzle forging (3.3.3); the core spray nozzle, safe end, and Class 1 piping (3.4); and the feedwater Class 1 piping (3.6).

The calculations reported in NUREG/CR-6260 were based on the interim reduced fatigue design curves given in NUREG/CR-5999 [9]. Such an approach penalizes the component location fatigue analysis unnecessarily, because research has shown that a combination of environmental conditions is required before reactor water environmental effects become pronounced. The strain rate must be sufficiently low and the strain range must be sufficiently high to cause continuing rupture of the passivation layer that protects the exposed surface area. Temperature, dissolved oxygen content, metal sulfur content, and water flow rate are additional variables to be considered. In order to take these parameters into consideration, EPRI and GE jointly developed a method, called the F_{en} approach [11], which permits reactor water environmental effects to be applied selectively, as justified by parameter combinations.

In 1999, the NRC staff raised a number of issues relative to the use of the EPRI/GE methodology in various industry applications. Those issues, coupled with more recent laboratory fatigue data in simulated LWR reactor water environments generated by ANL for carbon and low-alloy steels and stainless steels, resulted in a revised F_{en} methodology, as published in NUREG/CR-6583 [7] for carbon and low alloy steels, and NUREG/CR-5704 [8] for stainless steels. The methodology documented in these reports was used to evaluate environmental effects for VYNPS components, as described in Section 3.0 of this report.

3.0 ENVIRONMENTAL FATIGUE CALCULATIONS

Section 2.0 identifies the locations evaluated in NUREG/CR-6260 for the older vintage GE plant, which corresponds to VYNPS. NUREG/CR-6260 provided an assessment of these six selected component locations with respect to environmental fatigue using the older reduced environmental fatigue curves. Potential reactor water environmental effects are evaluated using the updated F_{en} methodology on a plant-specific basis in this subsection, in order to address the associated effects on fatigue as required by the GALL Report [5].

For each of the components identified in Section 2.0, environmental fatigue calculations were performed. The details of these calculations are documented in the Reference [12, 17, 18, 21, 22 and 24] calculations. The calculations were carried out using the appropriate methodology contained in NUREG/CR-6583 for carbon/low alloy steel material, and in NUREG/CR-5704 for stainless steel material. This methodology is as follows:

For Carbon Steel [7]: $F_{en} = \exp(0.585 - 0.00124T' - 0.101\, S^* T^* O^* \dot{\varepsilon}^*)$
$= \exp(0.554 - 0.101\, S^* T^* O^* \dot{\varepsilon}^*)$

For Low Alloy Steel [7]: $F_{en} = \exp(0.929 - 0.00124T' - 0.101\, S^* T^* O^* \dot{\varepsilon}^*)$
$= \exp(0.898 - 0.101\, S^* T^* O^* \dot{\varepsilon}^*)$

Note that the above expressions have been corrected as summarized in Reference [23].

where:

F_{en}	=	fatigue life correction factor
T'	=	25°C (NUREG/CR-6583, Section 6, F_{en} relative to air)
S*	=	S for 0 < sulfur content, S ≤ 0.015 wt. %
	=	0.015 for S > 0.015 wt. %
T*	=	0 for T < 150°C
	=	(T - 150) for 150 ≤ T ≤ 350°C
T	=	fluid service temperature (°C)
O*	=	0 for dissolved oxygen, DO < 0.05 parts per million (ppm)
	=	ln(DO/0.04) for 0.05 ppm ≤ DO ≤ 0.5 ppm
	=	ln(12.5) for DO > 0.5 ppm

$\dot{\varepsilon}^*$ = 0 for strain rate, $\dot{\varepsilon} > 1\%/sec$

= $\ln(\dot{\varepsilon})$ for $0.001 \leq \dot{\varepsilon} \leq 1\%/sec$

= $\ln(0.001)$ for $\dot{\varepsilon} < 0.001\%/sec$

For Types 304 and 316 Stainless Steel [8]: $F_{en} = \exp(0.935 - T^* \dot{\varepsilon}^* O^*)$

where:

F_{en} = fatigue life correction factor

T = fluid service temperature (°C)

T^* = 0 for $T < 200°C$

= 1 for $T \geq 200°C$

$\dot{\varepsilon}^*$ = 0 for strain rate, $\dot{\varepsilon} > 0.4\%/sec$

= $\ln(\dot{\varepsilon}/0.4)$ for $0.0004 \leq \dot{\varepsilon} \leq 0.4\%/sec$

= $\ln(0.0004/0.4)$ for $\dot{\varepsilon} < 0.0004\%/sec$

O^* = 0.260 for dissolved oxygen, DO < 0.05 parts per million (ppm)

= 0.172 for DO ≥ 0.05 ppm

Bounding F_{en} values are determined or, where necessary, computed for each load pair in a detailed fatigue calculation. The environmental fatigue is then determined as $U_{env} = (U)(F_{en})$, where U is the original fatigue usage, and U_{env} is the EAF usage factor.

INFORMATION REDACTED

Since implementation of HWC in 2003, VYNPS's availability has exceeded 98.5% and the objective for future HWC system availability is a minimum of 99% [12]. With these considerations, the overall availability for HWC since implementation at VYNPS until the end of the 60-year operating period was estimated at 98.5%.

Some nozzles, (e.g., recirculation outlet nozzle) have three materials: a Ni-Cr-Fe dissimilar metal weld (DMW), a low alloy steel forging, and a stainless steel safe end. To ensure the maximum CUF considering environmental effects was identified, locations in both the safe end and nozzle forging were selected. This selection produces bounding environmental fatigue results for the entire nozzle assembly for the following reasons:

- The highest thermal stresses from the finite-element model (FEM) analysis occur in the stainless steel safe end. Stainless steel F_{en} multipliers at VYNPS are significantly higher than Ni-Cr-Fe multipliers (F_{en} values are 2.55 or higher for stainless steel [12] vs. a constant value of 1.49 for Ni-Cr-Fe [11]). Therefore, evaluation of the safe end bounds the Ni-Cr-Fe weld material.
- The highest pressure stresses from the FEM analysis occur in the low alloy steel nozzle forging. Low alloy steel F_{en} multipliers at VYNPS are higher than Ni-Cr-Fe multipliers (F_{en} values are 2.45 or higher for low alloy steel [12] vs. a constant value of 1.49 for Ni-Cr-Fe [11]). Therefore, evaluation of the nozzle forging bounds the Ni-Cr-Fe weld material.

The number of cycles for forty years was adjusted based on the number of cycles actually experienced by the plant, projected out to 60 years of operation [14]. In addition, VYNPS has implemented extended power uprate (EPU). These effects have been incorporated into the evaluations documented in this report. With the use of this information, the CUF values documented in this report are applicable for 60 years of operation.

The environmental fatigue calculations are shown in Tables 3-1 through 3-9 and summarized in Table 3-10. Component-specific details are provided in the subsections that follow.

3.1 Reactor Vessel Shell and Lower Head

The environmental fatigue calculations for the reactor vessel shell and lower head location are shown in Table 3-1. The limiting CUF value reported in the VY LRA for the RPV shell/bottom

head location corresponds to a point located on the outside surface of the RPV bottom head at the junction with the support skirt. Therefore, this location is not exposed to the reactor coolant, and EAF effects do not apply. Based on this, evaluation of the limiting location along the inside surface of the RPV bottom head was performed.

The calculations shown in Table 3-1 are for the RPV lower head at the area with the highest alternating stress, which represents the limiting RPV bottom head location [12]. Reference [15] is the governing stress report for this low alloy steel location. The design fatigue calculation for the limiting RPV lower head location is reproduced in Table 3-1. The effects of EPU as well as conservative cycle counts for 60 years of plant operation are incorporated in this table. The final results in Table 3-1 show an EAF adjusted CUF of 0.0809 for 60 years, which is acceptable (i.e., less than the allowable value of 1.0).

The calculations shown in Table 3-2 are for the RPV shell at the RPV shell junction to the shroud support plate, which represents the limiting RPV shell location exposed to the reactor coolant [12]. Reference [16] is the governing stress report for this low alloy steel location. The design fatigue calculation for the limiting RPV shell location is reproduced in Table 3-2, which considers the effects of EPU and conservative cycle counts were used for 60 years of plant operation. The final results in Table 3-2 show an EAF adjusted CUF of 0.7364 for 60 years, which is acceptable (i.e., less than the allowable value of 1.0).

3.2 Reactor Vessel Feedwater Nozzle

The environmental fatigue calculations for the reactor vessel feedwater nozzle location are summarized in Table 3-3. The calculations summarized in Table 3-3 show both the blend radius, which represents the limiting feedwater nozzle location, and the safe end. Reference [17] contains the governing fatigue calculation for this location. Upper RPV region chemistry was assumed for the feedwater nozzle blend radius location, since this location is exposed to the reactor water chemistry in this region, whereas feedwater line chemistry was assumed for the safe end location.

The governing fatigue calculation for the limiting feedwater nozzle locations includes the effects of EPU and cycle counts for 60 years of operation obtained from Attachment 1 of Reference [14]. The blend radius cumulative usage factor (CUF) from system cycling is 0.0636 for 60 years. The safe end CUF is 0.1471 for 60 years. Although the carbon steel safe end has a higher CUF prior to considering environmental effects, the environmental multiplier from Table 3-3 results in a higher CUF at the low alloy steel blend radius. For the safe end location, the EAF adjusted CUF is 0.2560 for 60 years. For the blend radius location, EAF adjusted CUF is 0.6392 for 60 years, which is acceptable (i.e., less than the allowable value of 1.0).

3.3 Reactor Recirculation Piping (Including the Reactor Inlet and Outlet Nozzles)

Three locations were identified for the reactor recirculation piping in NUREG/CR-6260: the reactor vessel nozzle (includes both the inlet and outlet nozzles), and the recirculation piping. The evaluations for each of these components are described in the following subsections.

3.3.1 Reactor Recirculation Piping

Two locations (both stainless steel) were identified for VY for the reactor recirculation/RHR piping that correspond to the equivalent locations to those identified in NUREG/CR-6260: the RHR return tee connection to the recirculation piping, and the valve to pipe weld at the RHR isolation valve. Reference [18] contains the governing fatigue calculations for these locations. These analyses determined the limiting location to be at the RHR return tee.

The environmental fatigue calculations for the limiting recirculation/RHR piping location is summarized in Table 3-4, which includes the effects of EPU and cycle counts for 60 years of plant operation.

A review of the shutdown cooling mode of operation since the time of recirculation piping replacement in 1986 was performed by VYNPS, and the number of cycles per loop was conservatively estimated to be 150 through Year 60 [14]. Based on this, the cycle counts for the

Recirculation piping were reduced by a factor of 150/300 (50%) for all transients with the exception of transients that have fewer than 10 transient cycles. To ensure this cycle reduction adequately considered the potential impact on the RHR piping, the full number of transient cycles listed in Attachment 1 of Reference [14] was initially applied to the PIPESTRESS model and the highest CUF for the RHR piping was lower than the value obtained for the recirculation piping with reduced cycles.

Due to replacement of the recirculation piping, HWC conditions exist for 39% of the time, and NWC conditions exist for 61% of the time. This is based on 17.5 years of operation with NWC between March 1986 when the piping was replaced and November 2003 when HWC was implemented, and 46 years of operation from March 1986 to the end of the period of extended operation in March 2032. Using the bounding EAF multipliers (8.36 for HWC and 15.35 for NWC) [12], the overall multiplier is 12.62. Applying this to the 60-Year CUF of 0.0590 results in a total environmentally assisted CUF of 0.7446.

3.3.2 Reactor Recirculation Inlet Nozzle

References [15, 19 and 20] are the applicable stress reports for this location. An evaluation was performed for both the inlet nozzle forging (low alloy steel) and the safe end (stainless steel).

The environmental fatigue calculations for the recirculation inlet nozzle forging location are shown in Table 3-5. The governing fatigue calculation for the recirculation inlet nozzle location is reproduced in Table 3-5 [12], which includes the effects of EPU and cycle counts for 60 years of plant operation from Attachment 1 of Reference [14]. The final results show an EAF adjusted CUF of 0.5034 for 60 years, which is acceptable (i.e., less than the allowable value of 1.0).

The environmental fatigue calculations for the recirculation inlet nozzle safe end location are shown in Table 3-6. The governing fatigue calculation for the recirculation inlet nozzle location is reproduced in Table 3-6 [12], which includes the effects of EPU and cycle counts for 60 years

of plant operation from Attachment 1 of Reference [14]. The final results show an EAF adjusted CUF of 0.0199 for 60 years, which is acceptable (i.e., less than the allowable value of 1.0).

3.3.3 Reactor Recirculation Outlet Nozzle

The recirculation outlet nozzle was evaluated for environmental fatigue effects. Reference [24] is the fatigue calculation for this location. An evaluation was performed for both the outlet nozzle safe end (stainless steel) and the nozzle inner corner blend radius (low alloy steel). The results for the limiting nozzle forging location are reported here.

The environmental fatigue calculations for the limiting recirculation outlet nozzle forging blend radius location are shown in Table 3-7 [24], which includes the effects of EPU and cycle counts for 60 years of plant operation from Attachment 1 of Reference [14]. The final results in Table 3-7 show an EAF adjusted CUF of 0.0836 for 60 years, which is acceptable (i.e., less than the allowable value of 1.0).

3.4 Core Spray Line Reactor Vessel Nozzle and Associated Class 1 Piping

Locations that were evaluated in NUREG/CR-6260 included the reactor vessel nozzle blend radius (low alloy steel), the reactor vessel nozzle safe end (Alloy 600) and the core spray piping (stainless steel).

Reference [21] is the applicable fatigue calculation for these locations, which shows the nozzle limiting location to be the blend radius. The design fatigue calculations for the limiting location at the core spray nozzle, safe end, and piping are summarized in Table 3-8 [21], which include the effects of EPU and cycle counts for 60 years of plant operation from Attachment 1 of Reference [14]. The cumulative fatigue usage, prior to considering environmental effects for the blend radius, is 0.0166. Factoring in the environmental multiplier from Table 3-8 [12], the EAF adjusted CUF is 0.1668 for 60 years, which is acceptable (i.e., less than the allowable value of 1.0).

3.5 RHR Return Line Class 1 Piping

The environmental fatigue calculations for the RHR return line Class 1 piping are covered by the calculations in Subsection 3.3.1 above.

3.6 Feedwater Line Class 1 Piping

The environmental fatigue calculation for the limiting feedwater Class 1 piping location (carbon steel) is summarized in Table 3-9. The calculations shown in Table 3-9 are for the limiting feedwater Class 1 piping location. Per Reference [22], the limiting total fatigue usage for the analyzed feedwater/high pressure coolant injection (HPCI) piping system occurs on the riser to the RPV feedwater nozzle N4B. The limiting fatigue usage value for the feedwater Class 1 piping location is 0.1661, which includes the effects of EPU and cycle counts for 60 years of plant operation from Attachment 1 of Reference [14]. The final results in Table 3-9 show the EAF adjusted CUF of 0.2890 for 60 years, which is acceptable (i.e., less than the allowable value of 1.0).

3.7 Summary of Results

The results of the calculations contained in Tables 3-1 through 3-9 are summarized in Table 3-10.

It is noteworthy that the CUF results presented in this section include uniformly applied environmental effects without consideration of threshold criteria that might indicate an absence of conditions that would lead to environmental fatigue effects. Furthermore, conservative values were applied for temperature, strain rate and metal sulfur content in calculating environmental multipliers. Therefore, the environmental adjustments to the CUF results are considered to be conservative.

Table 3-1. Environmental Fatigue Evaluation for the Reactor Vessel Shell

Component: RPV Shell/Bottom Head
NUREG/CR-6260 CUF: 0.032 *(for reference only)*
Reference: NUREG/CR-6260, p. 5-102
Stress Report CUF: 0.0057 *(for Point 14, see below)*
Material: Low Alloy Steel *(Material = A-533 Gr. B)*

Design Basis CUF Calculation for 40 years:

$E_{fatigue\ curve}/E_{analysis}$ =	1.149	*Conservatively used minimum E of 26.1 from Section S2 Appendix of RPV Stress Report.*
Power Uprate =	1.0067	*=(549 - 100) / (546 - 100) per 4.4.1.b of 26A6019, Rev. 1*
K_t =	1.000	*stress concentration factor*
m =	2.0	*NB-3228.5 of ASME Code, Section III*
n =	0.2	*NB-3228.5 of ASME Code, Section III*
S_m =	26,700	psi *(ASME Code, Section II, Part D)*

P_L+P_B+Q *(see Note 1)*	K_e *(see Note 2)*	S_{alt} *(see Note 3)*	n *(see Note 4)*	N *(see Note 5)*	U
44,526	1.00	25,762	200	35,300	0.0057
				Total, U_{40} =	**0.0057**

Notes:
1. *P_L+P_B+Q is obtained for Point 14 from p. A52 of VYC-378, Rev. 0.*
2. *K_e computed in accordance with NB-3228.5 of ASME Code, Section III.*
3. *$S_{alt} = 0.5 * K_e * K_t * E_{fatigue\ curve}/E_{analysis} *$ Power Uprate $* (P_L+P_B+Q)$.*
4. *n for 40 years is the number of Heatup-Cooldown cycles, per p. B8 of VYC-378, Rev. 0.*
5. *N obtained from Figure I-9.1 of Appendix I of ASME Code, Section III.*
6. *n for 60 years is the projected number of Heatup-Cooldown cycles.*

Revised CUF Calculation for 60 Years:

P_L+P_B+Q *(see Note 1)*	K_e *(see Note 2)*	S_{alt} *(see Note 3)*	n *(see Note 6)*	N *(see Note 4)*	U
44,526	1.00	25,762	300	35,300	0.0085
				Total, U_{60} =	**0.0085**

Environmental CUF Calculation for 60 Years:

Maximum $F_{en\text{-}HWC}$ Multiplier for HWC Conditions = 5.39
Maximum $F_{en\text{-}NWC}$ Multiplier for NWC Conditions = 13.17

$U_{env\text{-}60} = U_{60} \times F_{en\text{-}NWC} \times 0.53 + U_{60} \times F_{en\text{-}HWC} \times 0.47 =$ 0.0809
Overall Multiplier = $U_{env\text{-}60}/U_{60}$ = 9.51

Table 3-2. Environmental Fatigue Evaluation for the Reactor Vessel Shell at Shroud Support

Component:	RPV Shell at Shroud Support	
NUREG/CR-6260 CUF:	0.032	*(for reference only)*
Reference:	NUREG/CR-6260, p. 5-102	
Stress Report CUF:	0.0549	*(for Point 9, see below)*
Material:	Low Alloy Steel	*(Material = A-533 Gr. B)*

Design Basis CUF Calculation for 40 years:

Hydrotest H_H =	26,240	psi	*(p. S3-97 of RPV Stress Report)*
Hydrotest H_r =	-1,250	psi	*(p. S3-97 of RPV Stress Report)*
Stress Concentration Factor, K_t =	2.40		*(p. S3-99d of RPV Stress Report)*
Hydrotest K_tH_H =	62,976	psi	*(p. S3-97 of RPV Stress Report)*
Improper Startup H_H =	28,060	psi	*(p. S3-98 of RPV Stress Report)*
Improper Startup H_r =	-1,025	psi	*(p. S3-98 of RPV Stress Report)*
Improper Startup Skin Stress =	156,099	psi	*(p. S3-98 of RPV Stress Report)*
Improper Startup K_tH_H + Skin Stress =	223,443	psi	*(p. S3-98 of RPV Stress Report)*
Warmup H_H =	-5,707	psi	*(p. S3-99a of RPV Stress Report)*
Warmup H_r =	-102	psi	*(p. S3-99a of RPV Stress Report)*
Warmup K_tH_H =	-13,696	psi	*(p. S3-99a of RPV Stress Report)*
$E_{fatigue\ curve}/E_{analysis}$ =	1.0417		*30.0 / 28.8 per S3-99f of RPV Stress Report and ASME Code fatigue curve*
Power Uprate =	1.0067		*=(549 - 100) / (546 - 100) per 4.4.1.b of 26A6019, Rev. 1*
m =	2.0		*NB-3228.5 of ASME Code, Section III*
n =	0.2		*NB-3228.5 of ASME Code, Section III*
S_m =	26,700	psi	*(ASME Code, Section II, Part D)*

P_L+P_B+Q *(see Note 1)*	Events	K_e *(see Note 2)*	S_{alt} *(see Note 3)*	n *(see Note 4)*	N *(see Note 5)*	U
34,690	Improper Startup - Warmup	1.00	124,825	5	332	0.0151
33,095	Hydrotest - Warmup	1.00	40,804	322	8,095	0.0398
					Total, U_{40} =	**0.0549**

Notes:

1. *P_L+P_B+Q is computed for Point 9 based on the $[(H_H - H_r)_{Event\ 1} - (H_H - H_r)_{Event\ 2}]$ stress intensity.*
2. *K_e computed in accordance with NB-3228.5 of ASME Code, Section III.*
3. *$S_{alt} = 0.5 * K_e * E_{fatigue\ curve}/E_{analysis} *$ Power Uprate $* [(K_tH_H - H_r)_{Event\ 1} - (K_tH_H - H_r)_{Event\ 2}]$.*
4. *n for 40 years is the number of cycles as follows per p. S3-99e and S3-99f of the RPV Stress Report:*

Improper Startup =	*5*	*cycles*
Hydrotest =	*2*	*cycles*
Isothermal at 70°F and 1,000 psi =	*120*	*cycles (same as number of Startup events)*
Warmup-Cooldown =	*199*	*cycles*
Warmup-Blowdown =	*1*	*cycle*
TOTAL =	*327*	*cycles*

5. *N obtained from Figure I-9.1 of Appendix I of ASME Code, Section III.*
6. *n for 60 years is the projected number of cycles as follows:*

Improper Startup =	*1*	*cycles*
Hydrotest =	*1*	*cycles*
Isothermal at 70°F and 1,000 psi =	*300*	*cycles (same as number of Startup events)*
Warmup-Cooldown =	*300*	*cycles*
Warmup-Blowdown =	*1*	*cycle*
TOTAL =	*603*	*cycles*

Revised CUF Calculation for 60 Years:

P_L+P_B+Q *(see Note 1)*		K_e *(see Note 2)*	S_{alt} *(see Note 3)*	n *(see Note 6)*	N *(see Note 4)*	U
34,690	Improper Startup - Warmup	1.00	124,825	1	332	0.0030
33,095	Hydrotest - Warmup	1.00	40,804	602	8,095	0.0744
					Total, U_{60} =	**0.0774**

Environmental CUF Calculation for 60 Years:

Maximum $F_{en\text{-}HWC}$ Multiplier for HWC Conditions =	5.39
Maximum $F_{en\text{-}NWC}$ Multiplier for NWC Conditions =	13.17
$U_{env\text{-}60} = U_{60} \times F_{en\text{-}NWC} \times 0.53 + U_{60} \times F_{en\text{-}HWC} \times 0.47$ =	**0.7364**
Overall Multiplier = $U_{env\text{-}60}/U_{60}$ =	**9.51**

Table 3-3. Environmental Fatigue Evaluation for the Reactor Vessel Feedwater Nozzle Forging Blend Radius

Low Alloy Steel:

$F_{en} = \exp(0.898 - 0.101S^*T^*O^*\square^*)$

Assume S* = 0.015 (maximum)
Assume □N= ln(0.001) = -6.908 (minimum)

For a BWR with HWC environment (post-HWC implementation):
DO = 97 ppb = 0.097 ppm, so O* = ln(0.097/0.04) = 0.886

Thus:

T (°C)	T (°F)	F_{en}
0	32	2.45
50	122	2.45
100	212	2.45
150	302	2.45
200	392	3.90
250	482	6.20
288	550	8.82

Thus, maximum F_{en} = 8.82

For a BWR with NWC environment (pre-HWC implementation):
DO = 114 ppb = 0.114 ppm, so O* = ln(0.114/0.04) = 1.047

Thus:

T (°C)	T (°F)	F_{en}
0	32	2.45
50	122	2.45
100	212	2.45
150	302	2.45
200	392	4.25
250	482	7.35
288	550	11.14

Thus, maximum F_{en} = 11.14

[T*= (T-150) for T > 150°C]

Carbon Steel:

$F_{en} = \exp(0.554 - 0.101S^*T^*O^*\square^*)$

Assume S* = 0.015 (maximum)
Assume □ = ln(0.001) = -6.908 (minimum)

For a BWR with HWC environment (post-HWC implementation):
DO = 40 ppb = 0.040 ppm < 0.050 ppm so O* = 0
Thus:

T (°C)	T (°F)	F_{en}
0	32	1.74
50	122	1.74
100	212	1.74
150	302	1.74
200	392	1.74
250	482	1.74
288	550	1.74

Thus, maximum F_{en} = 1.74

For a BWR with NWC environment (pre-HWC implementation):
DO = 40 ppb = 0.040 ppm < 0.050 ppm so O* = 0
Thus:

T (°C)	T (°F)	F_{en}
0	32	1.74
50	122	1.74
100	212	1.74
150	302	1.74
200	392	1.74
250	482	1.74
288	550	1.74

Thus, maximum F_{en} = 1.74

[T*= (T-150) for T > 150°C]

No.	Component	Material	60-Year CUF	Overall Environmental Multiplier	60-Year Environmental CUF (1,2)
1	Feedwater Nozzle Forging Blend Radius	Low Alloy Steel	0.0636	10.05	0.6392
2	Feedwater Nozzle Forging Safe End	Carbon Steel	0.1471	1.74	0.2560

Notes:
1. An Fen Multiplier was used for each respective component with the following conditions:
   + 47% HWC conditions and 53% NWC conditions
2. Results using updated ASME Code fatigue calculations and actual cycles accumulated to-date and projected to 60 years.

Table 3-4. Environmental Fatigue Evaluation for the Recirculation/RHR Piping Tee

Stainless Steel:		$F_{en} = \exp(0.935 - T^*\varepsilon^*O^*)$		
For a BWR with HWC environment (post-HWC implementation): DO = 46 ppb = 0.046 ppm < 0.050 ppm, so $O^* = 0.260$ Conservatively use $T^* = 1$ for T > 200°C Thus:			**For a BWR with NWC environment (pre-HWC implementation):** DO = 123 ppb = 0.123 ppm > 0.05 ppm, so $O^* = 0.172$ Conservatively use $T^* = 1$ for T > 200°C Thus:	
$\varepsilon^* = 0$ for $\varepsilon > 0.4$%/sec	so F_{en} =	2.55	so F_{en} =	2.55
$\varepsilon^* = \ln(\varepsilon/0.4)$ for $0.0004 <= \varepsilon <= 0.4$%/sec	so F_{en} ranges from	2.55	so F_{en} ranges from	2.55
	to	15.35	to	8.36
$\varepsilon^* = \ln(0.0004/0.4)$ for $\varepsilon < 0.0004$%/sec	so F_{en} =	15.35	so F_{en} =	8.36
	Thus, maximum F_{en} =	15.35	Thus, maximum F_{en} =	8.36

No.	Component	Material	60-Year CUF	Overall Environmental Multiplier	60-Year Environmental CUF (1,2)
1	Recirculation /RHR Piping Return Tee	Stainless Steel	0.0590	12.62	0.7446

Notes: 1. An F_{en} multiplier was used for each respective component with the following conditions:
+ 39% HWC conditions and 61% NWC conditions
2. Results using updated ASME Code fatigue calculations and actual cycles accumulated to-date and projected to 60 years.

Table 3-5. Environmental Fatigue Evaluation for the Reactor Recirculation Inlet Nozzle Forging

Component: Recirculation Inlet Nozzle Forging
NUREG/CR-6260 CUF: 0.310 *(for reference only)*
Reference: NUREG/CR-6260, p. 5-105
Stress Report CUF: 0.0433 *(updated for Point 12, see below)*
Material: Low Alloy Steel *(Material = A-508 Cl. II per p. I-S8-4 of CBIN Stress Report Section S8)*

Design Basis CUF Calculation for 40 years:

$E_{fatigue\ curve}/E_{analysis}$ =	1.1278	*= 30.0 / 26.6 (per p. I-S8-24 of CBIN Stress Report Section S8 and ASME Code fatigue curve)*
Power Uprate =	1.0067	*=(549 - 100) / (546 - 100) per 4.4.1.b of 26A6019, Rev. 1*
K_t =	1.660	*stress concentration factor (p. A270 of VYC-378, Rev. 0)*
m =	2.0	*NB-3228.5 of ASME Code, Section III*
n =	0.2	*NB-3228.5 of ASME Code, Section III*
S_m =	26,700	psi *(ASME Code, Section II, Part D)*

P_L+P_B+Q *(see Note 1)*	Skin Stress *(see Note 2)*	K_e *(see Note 3)*	S_{alt} *(see Note 4)*	n *(see Note 5)*	N *(see Note 6)*	U
43,110	15,145	1.00	49,224	200	4,614	0.0433
					Total, U_{40} =	**0.0433**

Notes:
1. *P_L+P_B+Q is obtained for Point 12 from p. A270 of VYC-378, Rev. 0.*
2. *Skin Stress is obtained for Point 12 from p. A270 of VYC-378, Rev. 0.*
3. *K_e computed in accordance with NB-3228.5 of ASME Code, Section III.*
4. *$S_{alt} = 0.5 * K_e * E_{fatigue\ curve}/E_{analysis} *$ Power Uprate $* [(P_L+P_B+Q) K_t +$ Skin Stress].*
5. *n for 40 years is the number of Heatup-Cooldown cycles, per p. B28 of VYC-378, Rev. 0.*
6. *N obtained from Figure I-9.1 of Appendix I of ASME Code, Section III.*
7. *n for 60 years is the projected number of Heatup-Cooldown cycles.*

Revised CUF Calculation for 60 Years:

P_L+P_B+Q *(see Note 1)*	Skin Stress *(see Note 2)*	K_e *(see Note 3)*	S_{alt} *(see Note 4)*	n *(see Note 5)*	N *(see Note 7)*	U
43,110	15,145	1.00	49,224	300	4,614	0.0650
					Total, U_{60} =	**0.0650**

Environmental CUF Calculation for 60 Years:

Maximum $F_{en\text{-}HWC}$ Multiplier for HWC Conditions = 2.45
Maximum $F_{en\text{-}NWC}$ Multiplier for NWC Conditions = 12.43

$U_{env\text{-}60} = U_{60} \times F_{en\text{-}NWC} \times 0.53 + U_{60} \times F_{en\text{-}HWC} \times 0.47 =$ 0.5034
Overall Multiplier = $U_{env\text{-}60}/U_{60}$ = 7.74

Table 3-6. Environmental Fatigue Evaluation for Reactor Recirculation Inlet Nozzle Safe End

Component: Recirculation Inlet Nozzle Safe End
NUREG/CR-6260 CUF: 0.310 *(for reference only)*
Reference: NUREG/CR-6260, p. 5-105
Stress Report CUF: 0.0017 *(updated for Location 6-I, see below)*
Material: Stainless Steel *(316L per p. 8 of 23A4292, Rev. 4)*

Design Basis CUF Calculation for 40 years:

$E_{fatigue\ curve}/E_{analysis}$ =	1.1076	*= 28.3 / 25.55 (per p. 62 of Reference [18] and ASME Code fatigue curve)*
Power Uprate =	1.0067	*=(549 - 100) / (546 - 100) per 4.4.1.b of 26A6019, Rev. 1*
K_t =	1.280	*stress concentration factor (p. B27 of VYC-378, Rev. 0)*
m =	1.7	*NB-3228.5 of ASME Code, Section III*
n =	0.3	*NB-3228.5 of ASME Code, Section III*
S_m =	16,600	psi *(ASME Code, Section II, Part D)*

P_L+P_B+Q *(see Note 1)*	P+Q+F *(see Note 2)*	K_e *(see Note 3)*	S_{alt} *(see Note 4)*	n *(see Note 5)*	N *(see Note 6)*	U
47,183	36,972	1.00	26,385	2,076	1,242,266	0.0017
					Total, U_{40} =	**0.0017**

Notes:
1. *P_L+P_B+Q is obtained for Surface I (after weld overlay) from p. 117 of Reference [18].*
2. *P+Q+F is obtained for Point 6-I from p. 118 of Reference [18] (BEFORE weld overlay).*
3. *K_e computed in accordance with NB-3228.5 of ASME Code, Section III.*
4. *$S_{alt} = 0.5 * K_e * E_{fatigue\ curve}/E_{analysis} *$ Power Uprate $* [(P+Q+F) K_t]$.*
5. *n for 40 years is the number of cycles as follows per p. B26 of VYC-378, Rev. 0:*

Design Hydrotest =	*130*	
Loss of Feedpumps Composite:		
Startup/Shutdown =	*290*	
SRV Blowdown =	*8*	
Loss of Feedwater Pumps	*30*	*10 events x 3 up/down cycles per event*
SCRAM =	*270*	
Normal +/- Seismic =	*11*	*10 cycles of upset seismic, plus 1 Level C seismic event*
Normal =	*739*	*= Sum of all of above events*
Zeroload =	*598*	*= Startup/Shutdown + SRV Blowdown + Scram + LOFP*
Total number of cycles =	*2,076*	

6. *N obtained from Figure I-9.2 of Appendix I of ASME Code, Section III.*
7. *n for 60 years is the projected number of cycles as follows:*

Design Hydrotest =	*120*	
Loss of Feedpumps Composite:		
Startup/Shutdown =	*300*	
SRV Blowdown =	*1*	
Loss of Feedwater Pumps	*30*	*10 events x 3 up/down cycles per event*
SCRAM =	*289*	*All remaining scrams*
Normal +/- Seismic =	*11*	*Assume the same*
Normal =	*751*	*= Sum of all of above events*
Zeroload =	*620*	*= Startup/Shutdown + SRV Blowdown + Scram + LOFP*
Total number of cycles =	*2,122*	

Revised CUF Calculation for 60 Years:

P_L+P_B+Q *(see Note 1)*	P+Q+F *(see Note 2)*	K_e *(see Note 3)*	S_{alt} *(see Note 4)*	n *(see Note 5)*	N *(see Note 7)*	U
47,183	36,972	1.00	26,385	2,122	1,242,266	0.0017
					Total, U_{60} =	**0.0017**

Environmental CUF Calculation for 60 Years:

Maximum $F_{en\text{-}HWC}$ Multiplier for HWC Conditions =	15.35
Maximum $F_{en\text{-}NWC}$ Multiplier for NWC Conditions =	8.36
$U_{env\text{-}60} = U_{60} \times F_{en\text{-}NWC} \times 0.53 + U_{60} \times F_{en\text{-}HWC} \times 0.47$ =	**0.0199**
Overall Multiplier = $U_{env\text{-}60}/U_{60}$ =	**11.64**

Table 3-7. Environmental Fatigue Evaluation for Recirculation Outlet Nozzle Forging

Low Alloy Steel:

$F_{en} = \exp(0.898 - 0.101S^*T^*O^*\varepsilon^*)$

Assume $S^* = 0.015$ (maximum)
Assume $\varepsilon^* = \ln(0.001) = -6.908$ (minimum)

For a BWR with HWC environment (post-HWC implementation):
DO = 46 ppb = 0.046 ppm
DO < 0.050 ppm, so $O^* = 0$
Thus:

T (°C)	T (°F)	F_{en}
0	32	2.45
50	122	2.45
100	212	2.45
150	302	2.45
200	392	2.45
269.45	517.01	2.45
288	550	2.45

Thus, maximum F_{en} = 2.45

For a BWR with NWC environment (pre-HWC implementation):
DO = 123 ppb = 0.123 ppm, so $O^* = \ln(0.123/0.04) = 1.123$

Thus:

T (°C)	T (°F)	F_{en}
0	32	2.45
50	122	2.45
100	212	2.45
150	302	2.45
200	392	4.42
269.45	517.01	10.00
288	550	12.43

Thus, maximum F_{en} = 12.43

[$T^* = (T-150)$ for $T > 150°C$]

No.	Component	Material	60-Year CUF	Overall Environmental Multiplier	60-Year Environmental CUF (1,2)
1	Recirculation Outlet Nozzle Forging Blend Radius	Low Alloy Steel	0.0108	7.74	0.0836

Notes: 1. An F_{en} multiplier was used for each respective component with the following conditions:
+ 47% HWC conditions and 53% NWC conditions
2. Results using updated ASME Code fatigue calculations and actual cycles accumulated to-date and projected to 60 years.

Table 3-8. Environmental Fatigue Evaluation for Core Spray Reactor Vessel Nozzle Forging Blend Radius, Safe End, and Piping

Low Alloy Steel:

$F_{en} = \exp(0.898 - 0.101S^*T^*O^*\square^*)$

Assume S* = 0.015 (maximum)
Assume □N= ln(0.001) = -6.908 (minimum)

For a BWR with HWC environment (post-HWC implementation):
DO = 97 ppb = 0.097 ppm, so O* = ln(0.097/0.04) = 0.886

Thus:

T (°C)	T (°F)	F_{en}
0	32	2.45
50	122	2.45
100	212	2.45
150	302	2.45
200	392	3.90
250	482	6.20
288	550	8.82

Thus, maximum F_{en} = 8.82

For a BWR with NWC environment (pre-HWC implementation):
DO = 114 ppb = 0.114 ppm, so O* = ln(0.114/0.04) = 1.047

Thus:

T (°C)	T (°F)	F_{en}
0	32	2.45
50	122	2.45
100	212	2.45
150	302	2.45
200	392	4.25
250	482	7.35
288	550	11.14

Thus, maximum F_{en} = 11.14

[T* = (T-150) for T > 150°C]

Stainless Steel:

$F_{en} = \exp(0.935 - T^*{}^\circ{}^*O^*)$

For a BWR with HWC environment (post-HWC implementation):
DO = 97 ppb = 0.097 ppm > 0.050 ppm, so O* = 0.172
Conservatively use T* = 1 for T > 200°C
Thus:

°* = 0 for ° > 0.4%/sec	so F_{en} =	2.55
°* = ln(°/0.4) for 0.0004 <= ° <= 0.4%/sec	so F_{en} ranges from	2.55
	to	8.36
°* = ln(0.0004/0.4) for ° < 0.0004%/sec	so F_{en} =	8.36
	Thus, maximum F_{en} =	8.36

For a BWR with NWC environment (pre-HWC implementation):
DO = 114 ppb = 0.114 ppm > 0.05 ppm, so O* = 0.172
Conservatively use T* = 1 for T > 200°C
Thus:

so F_{en} =	2.55
so F_{en} ranges from	2.55
to	8.36
so F_{en} =	8.36
Thus, maximum F_{en} =	8.36

No.	Component	Material	60-Year CUF	Overall Environmental Multiplier	60-Year Environmental CUF (1,2)
1	Core Spray Nozzle Forging Blend Radius	Low Alloy Steel	0.0166	10.05	0.1668
2	Core Spray Nozzle Safe End	Ni-Cr-Fe	0.0398	1.49	0.0593
3	Core Spray Piping	Stainless Steel	0.0011	8.36	0.0092

Notes:
1. An Fen Multiplier was used for each respective component with the following conditions: + 47% HWC conditions and 53% NWC conditions
2. Results using updated ASME Code fatigue calculations and actual cycles accumulated to-date and projected to 60 years.

Table 3-9. Environmental Fatigue Evaluation for the Feedwater Line Class 1 Piping

Carbon Steel:

$F_{en} = \exp(0.554 - 0.101S^*T^*O^*\varepsilon^*)$

Assume $S^* = 0.015$ (maximum)
Assume $\varepsilon^* = \ln(0.001) = -6.908$ (minimum)

For a BWR with HWC environment (post-HWC implementation):
DO = 40 ppb = 0.040 ppm < 0.050 ppm so $O^* = 0$
Thus:

T (°C)	T (°F)	F_{en}
0	32	1.74
50	122	1.74
100	212	1.74
150	302	1.74
200	392	1.74
250	482	1.74
288	550	1.74

Thus, maximum F_{en} = 1.74

For a BWR with NWC environment (pre-HWC implementation):
DO = 40 ppb = 0.040 ppm < 0.050 ppm so $O^* = 0$
Thus:

T (°C)	T (°F)	F_{en}
0	32	1.74
50	122	1.74
100	212	1.74
150	302	1.74
200	392	1.74
250	482	1.74
288	550	1.74

Thus, maximum F_{en} = 1.74

[$T^* = (T-150)$ for $T > 150°C$]

No.	Component	Material	60-Year CUF	Overall Environmental Multiplier	60-Year Environmental CUF (1,2)
1	Feedwater Piping Riser to RPV Nozzle N4B	Carbon Steel	0.1661	1.74	0.2890

Notes: 1. An F_{en} multiplier was used for each respective component with the following conditions:
+ 47% HWC conditions and 53% NWC conditions
2. Results using updated ASME Code fatigue calculations and actual cycles accumulated to-date and projected to 60 years.

Table 3-10. Summary of Environmental Fatigue Calculations for VYNPS

No.	Component	Material	40-Year Design CUF (1)	60-Year CUF (2)	Overall Environmental Multiplier (3)	60-Year Environmental CUF
1	RPV Shell/Bottom Head	Low Alloy Steel	0.0057	0.0085	9.51	0.0809
2	RPV Shell at Shroud Support	Low Alloy Steel	0.0549	0.0774	9.51	0.7364
3	Feedwater Nozzle Blend Radius	Low Alloy Steel	(4)	0.0636	10.05	0.6392
4	Recirculation/RHR Class 1 Piping (Return Tee)	Stainless Steel	(4)	0.0590	12.62	0.7446
5	Recirculation Inlet Nozzle Forging	Low Alloy Steel	0.0433	0.0650	7.74	0.5034
6	Recirculation Inlet Nozzle Safe End	Stainless Steel	0.0017	0.0017	11.64	0.0199
7	Recirculation Outlet Nozzle Forging	Low Alloy Steel	(4)	0.0108	7.74	0.0836
8	Core Spray Nozzle Forging Blend Radius (5)	Low Alloy Steel	(4)	0.0166	10.05	0.1668
9	Feedwater Class 1 Piping	Carbon Steel	(4)	0.1661	1.74	0.2890

Notes:
1. Updated 40-year CUF calculation based on recent ASME Code methodology and design basis cycles.
2. CUF results using updated ASME Code methodology and actual cycles accumulated to-date and projected to 60 years.
3. An F_{en} multiplier was used for each respective component with the following conditions:
   + 47% HWC conditions and 53% NWC conditions
4. 40 year values were not calculated for these locations
5. Only the highest CUF from Table 3-8 is shown

4.0 SUMMARY AND CONCLUSIONS

The results of Tables 3-1 through 3-9, as summarized in Table 3-10, demonstrate that the fatigue usage factor, including environmental effects, remains within the allowable value of 1.0 for 60 years of operation for the following component locations:

- ✓ Reactor vessel shell, bottom head and shroud support
- ✓ Reactor vessel feedwater nozzle
- ✓ Reactor recirculation piping (including the reactor inlet and outlet nozzles)
- ✓ Core spray line reactor vessel nozzle and associated Class 1 piping
- ✓ Feedwater line Class 1 piping

Therefore, the environmental fatigue assessment results for all of the NUREG/CR-6260 locations associated with the older vintage BWR plant are acceptable for 60 years of operation for VYNPS.

5.0 REFERENCES

1. U. S. Nuclear Regulatory Commission, Generic Safety Issue 166, "Adequacy of Fatigue Life of Metal Components."

2. U. S. Nuclear Regulatory Commission, Generic Safety Issue 190, "Fatigue Evaluation of Metal Components for 60-Year Plant Life."

3. SECY-95-245, "Completion of the Fatigue Action Plan," James M. Taylor, Executive Director for Operations, U. S. Nuclear Regulatory Commission, Washington, DC, September 25, 1995.

4. Memorandum, Ashok C. Thadani, Director, Office of Nuclear Regulatory Research, to William D. Travers, Executive Director for Operations, Closeout of Generic Safety Issue 190, "Fatigue Evaluation of Metal Components for 60 Year Plant Life," U. S. Nuclear Regulatory Commission, Washington, DC, December 26, 1999.

5. NUREG-1801, Revision 1, "Generic Aging Lessons Learned (GALL) Report," U.S. Nuclear Regulatory Commission, September 2005.

6. NUREG/CR-6260 (INEL-95/0045), "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," March 1995.

7. NUREG/CR-6583 (ANL-97/18), "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," March 1998.

8. NUREG/CR-5704 (ANL-98/31), "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," April 1999.

9. NUREG/CR-5999 (ANL-93/3), "Interim Fatigue Design Curves for Carbon, Low-Alloy, and Austenitic Stainless Steels in LWR Environments," April 1993.

10. USAS B31.1 - 1967, USA Standard Code for Pressure Piping, "Power Piping," American Society of Mechanical Engineers, New York.

11. EPRI Report No. TR-105759, "An Environmental Factor Approach to Account for Reactor Water Effects in Light Water Reactor Pressure Vessel and Piping Fatigue Evaluations," December 1995.

12. Structural Integrity Associates Calculation No. VY-16Q-303, Revision 0, " Environmental Fatigue Evaluation of Reactor Recirculation Inlet Nozzle and Vessel Shell/Bottom Head."

13. INFORMATION REDACTED

14. Entergy Design Input Record (DIR) Rev. 1, EC No. 1773, Rev. 0, "Environmental Fatigue Analysis for Vermont Yankee Nuclear Power Station," 7/26/07, SI File No. VY-16Q-209.

15. VY Calculation No. VYC-378, Revision 0, "Vermont Yankee Reactor Cyclic Limits for Transient Events," 10/16/85, SI File No. VY-05Q-211.

16. Chicago Bridge & Iron RPV Stress Report, Section S3, Revision 4, "Stress Analysis, Shroud Support, Vermont Yankee Reactor Vessel, CB&I Contract 9-6201," 2/3/70, SI File No. VY-16Q-203.

17. Structural Integrity Associates Calculation No. VY-16Q-302, Revision 0, "Fatigue Analysis of Feedwater Nozzle."

18. Structural Integrity Associates Calculation No. VY-16Q-307, Revision 0, "Recirculation Class 1 Piping Fatigue and EAF Analysis."

19. CB&I RPV Stress Report, Section S8, Revision 4, "Stress Analysis, Recirculation Inlet Nozzle, Vermont Yankee Reactor Vessel, CB&I Contract 9-6201," 2/3/70, SI File No. VY-16Q-203.

20. GE Nuclear Energy Certified Stress Report No. 23A4292, Revision 0, "Reactor Vessel – Recirculation Inlet Safe End Nozzle," January 21, 1985, SI File No. VY-16Q-203.

21. Structural Integrity Associates Calculation No. VY-16Q-310, Revision 1, "Fatigue Analysis of Core Spray Nozzle."

22. Structural Integrity Associates Calculation No. VY-16Q-311, Revision 0, "Feedwater Class 1 Piping Fatigue Analysis."

23. EPRI/BWRVIP Memo. No. 2005-271, "Potential Error in Existing Fatigue Reactor Water Environmental Effects Analyses," July 1, 2005

24. Structural Integrity Associates Calculation No. VY-16Q-306, Revision 0, "Fatigue Analysis of Recirculation Outlet Nozzle."