RAS 13984

August 6, 2007

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)	
	)	
Entergy Nuclear Vermont Yankee, LLC	)	Docket No. 50-271-LR
and Entergy Nuclear Operations, Inc.	)	ASLBP No. 06-849-03-LR
	)	
(Vermont Yankee Nuclear Power Station)	)	

ENTERGY'S RESPONSE TO NEC'S MOTION TO FILE A NEW OR AMENDED CONTENTION

I. INTRODUCTION

Applicants Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (collectively "Entergy") submit this response, pursuant to 10 C.F.R. § 2.309(h)(1), to "New England Coalition, Inc.'s (NEC) Motion to File a Timely New or Amended Contention." For the reasons discussed below, Entergy does not oppose admission of the new contention proffered by NEC ("New Contention") but respectfully submits that (1) the original NEC Contention 2 in this proceeding ("NEC Contention 2") should be dismissed, and (2) all further proceedings on the New Contention should be held in abeyance pending review by NEC of the final fatigue calculations that were provided to NEC by Entergy on August 2, 2007 and the potential submittal by NEC of a contention based on those final calculations that may supersede the New Contention. This response is supported by the Declaration of Terry J. Herrmann ("Herrmann Decl."), filed simultaneously herewith.

Template=SECY-041

SECY-02

II. ORIGINAL CONTENTION 2 SHOULD BE DISMISSED AND CONSIDERATION OF THE NEW CONTENTION SHOULD BE HELD IN ABEYANCE PENDING THE OPPORTUNITY FOR NEC TO REVIEW THE FINAL FATIGUE CALCULATIONS AND SUBMIT, IF IT WISHES, A PROPOSED CONTENTION BASED ON THEM

A. BACKGROUND

NEC Contention 2 asserts that Entergy's license renewal application for the Vermont Yankee Nuclear Power Station ("VY") ("Application")[1] should be denied because it "does not include an adequate plan to monitor and manage the effects of aging [due to metal fatigue] on key reactor components that are subject to an aging management review, pursuant to 10 C.F.R. § 54.21(a) and an evaluation of time limited analysis, pursuant to 10 C.F.R. § 54.21(c)." Memorandum and Order (Ruling on Standing, Contentions, Hearing Procedures, State Statutory Claim, and Contention Adoption), LBP-06-20, 64 NRC 131, 183 (2006).

At the time NEC Contention 2 was admitted, Entergy had not yet performed detailed, plant-specific analyses of key reactor components that would establish the potential for their failure due to environmentally assisted fatigue during the extended operations period after license renewal. Preliminary versions of the plant-specific fatigue analyses were provided to NEC on June 7 and June 13, 2007 and NEC filed its New Contention on July 12, 2007, pursuant to the instructions of the Atomic Safety and Licensing Board ("Board") in its June 18, 2007 Order (Setting Deadline for any Motion to Dismiss NEC Contention 2 as Moot) ("June 18 Order").

Final VY-specific fatigue analyses have now been completed. Herrmann Decl., ¶ 10. They provide confidence that component failure due to environmentally assisted fatigue ("EAF") will not be a concern at VY during the period of extended operation. Id. Copies of the reports

[1] Vermont Yankee Nuclear Power Station, License Renewal Application (January 25, 2006), available in the NRC ADAMS system with Accession No. ML060300085.

describing those analyses were provided to NEC and the other parties to this proceeding on August 2, 2007. See Exhibit 1 hereto.

B. DISCUSSION

Section 4.3 of the Application evaluates the analysis of metal fatigue for Class 1 and selected non-Class 1 components for the period of extended operation. Class 1 components (reactor vessel and recirculation system piping) are subjected to fatigue analysis under Section III of the ASME Code. ASME Section III requires evaluation of fatigue by considering design thermal and loading cycles. Cumulative usage factors ("CUF") can be calculated for plant components that identify the proportion of the allowable fatigue cycles that have been, or are projected to be, experienced by the components. Table 4.3-1 of the Application shows the CUFs for Class 1 components based on the number of transients projected to occur over the operating life of VY. As reflected in Table 4.3-1, the ASME Code design basis CUFs are significantly below unity for all components.

Section 4.3.2 of the Application addresses the fatigue evaluation for components designed under ANSI Code B31.1, and demonstrates that the design-basis stress reduction factor used for these components also remains valid and bounding for the period of extended operation of the plant.

Section 4.3.3 of the Application assesses the effects of the reactor water environment on fatigue life, known as environmentally assisted fatigue. The component locations where EAF effects need to be evaluated are given in NUREG/CR-6260, which is endorsed by NUREG-1801 (Volume 2, Section X.M.1). They are: (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. Components in these six locations need to be analyzed for EAF effects. It is not disputed that these are the locations and components of interest from the standpoint of EAF.

Entergy originally evaluated limiting locations for EAF by multiplying the ASME Code CUFs by a factor that accounts for the effects of EAF.[2] See Application at 4.3-6 and Table 4.3-3. For locations in limiting Class 1 components that did not have specific CUFs because they were designed under ANSI Code B.31.1, CUFs were estimated based on generic values in NUREG/CR-6260. The values reported in NUREG/CR-6260 were in turn based on interim fatigue curves given in NUREG/CR-5999.[3]

There were several components for which the originally estimated EAF CUFs obtained using these generic values were greater than unity. See Application, Table 4.3.3. For those components, the Application commits Entergy to manage the effects of aging "[p]rior to entering the period of extended operation" by implementing one or more of the following:

1. "further refinement of the fatigue analysis to lower the predicted CUFs to less than 1.0";
2. "management of fatigue at the affected location by an inspection program that has been approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by method acceptable to the NRC)";
3. "repair or replacement of the affected locations."

Application at 4.3-7.

Entergy has implemented Option 1 of the three listed above, and has performed a more refined fatigue analysis that applies updated ASME Code methodology and uses actual cycles

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[2] See Application, Table 4.3-1 n.1.

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

accumulated to date by the VY components in question, which are then projected to sixty years of plant operations. Herrmann Decl., ¶ 8. The final plant-specific analyses show that the environmentally assisted CUFs for the critical locations for the sixty years encompassed by VY's original and extended license periods are in all cases less than unity, signifying that there is confidence that component failure due to EAF will not be a concern at VY during the period of extended operation. Id., ¶ 10. Since the environmentally-assisted CUFs for all of the NUREG/CR-6260 components are acceptable for 60 years of operation for VY, there is no anticipated need to implement Option 2 of those listed in the Application at 4.3.7 (develop a detailed inspection program for the components in question) or Option 3 (repair or replacement of the affected components).[4]

NEC's New Contention specifically challenges the preliminary results obtained by Entergy through the exercise of Option 1. Accordingly, it supersedes Contention 2, and the original contention should be dismissed.[5]

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[4] Entergy is reserving the option of voluntarily developing a detailed component inspection program for the period of extended plant operations. See Item 27, Amendment 27 to Application, dated July 3, 2007, ADAMS Accession No. ML 07900203. The commitment states in relevant part:

> During the period of extended operation, VY may also use one of the following options for fatigue management if ongoing monitoring indicates a potential for a condition outside the analysis bounds noted above:
> 1) Update and/or refine the affected analyses described above.
> 2) Implement an inspection program that has been reviewed and approved by the NRC (e.g., periodic nondestructive examination of the affected locations at inspection intervals to be determined by a method acceptable to the NRC).
> 3) Repair or replace the affected locations before exceeding a CUF of 1.0.

Amendment 27, Attachment 1, at p. 6.

[5] Entergy was unable to move to dismiss NEC Contention 2 as moot by the July 12, 2007 deadline set in the Board's June 18 Order because the fatigue analyses had not been finalized as of that date. However, the Commission has held that "it is well-recognized" that where a contention alleges the omission of particular information on an issue from an application, and the information is later supplied by the applicant, the contention is moot and must be dismissed. USEC (American Centrifuge Plant), CLI-06-9, 63 NRC 433, 444 (2006) (citing Duke Energy Corp. (McGuire Nuclear Station, Units 1 and 2; Catawba Nuclear Station, Units 1 and 2), CLI-02-28, 56 NRC 373, 383 (2002), citing Duke Energy Corp. (Catawba Nuclear Station, Units 1 and 2), CLI-83-19, 17 NRC 1041, 1050 (1983)) (footnote omitted). Here, the plant-specific fatigue analyses supply the information that NEC alleges was omitted from the Application and Contention 2 has become moot.

The New Contention, in turn, is directed at preliminary fatigue analyses, which are superseded by the final ones provided to NEC and the other parties on August 2, 2007. Consideration of the New Contention should therefore be held in abeyance until NEC has determined what action it wishes to take with respect to the contention, including leaving it unchanged, amending it, or replacing it altogether with another contention directed at the final analyses. In the interest of time, Entergy will not contest the admissibility of any such new contention provided the parties are given the opportunity to move for its summary disposition, if appropriate.

III. CONCLUSION

As demonstrated above, the alleged deficiency in the Application raised by NEC in Contention 2 has been rendered moot by the fatigue analyses recently performed by Entergy. Accordingly, NEC Contention 2 should be dismissed. Consideration of NEC's New Contention should be held in abeyance until NEC has determined what action it wishes to take with respect to it in light of the final analyses performed by Entergy.

Respectfully Submitted,

Matias F. Travieso-Diaz

David R. Lewis
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Counsel for Entergy

Dated: August 6, 2007

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)	
	)	
Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc.	)	Docket No. 50-271-LR ASLBP No. 06-849-03-LR
	)	
(Vermont Yankee Nuclear Power Station)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of "Entergy's Response to NEC's Motion to File a New or Amended Contention" and "Declaration of Terry J. Herrmann" were served on the persons listed below by deposit in the U.S. Mail, first class, postage prepaid, or with respect to Judge Elleman by overnight mail, and where indicated by an asterisk by electronic mail, this 6th[th] day of August, 2007.

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August 2, 2007

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In the Matter of
Entergy Nuclear Vermont Yankee, LLC, and Entergy Nuclear Operations, Inc.
(Vermont Yankee Nuclear Power Station)
Docket No. 50-271-LR; ASLBP No. 06-849-03-LR

Re: Structural Integrity Associates Final Fatigue Analysis Reports

Dear Mesdames Baty, Hofmann and Tyler:

On June 7 and 13, 2007, Entergy provided copies of several reports containing preliminary Vermont Yankee site-specific calculations of environmentally assisted fatigue of critical components relevant to New England Coalition's Contention 2. Those calculations have now been finalized and reports describing the calculations and their results are enclosed herewith in a compact disc. Listed in the Attachment to this letter are

the materials being provided. Entergy will supply in the future production numbers for these reports.

Please note that three of the documents included herewith contain proprietary information. They are Calculation No. VY-16Q-303 Rev. 0 and reports SIR-07-130-NPS Rev. 0 (File VY-16Q-401) and SIR-07-132-NPS Rev.0 (File-VY-16Q-404). Unredacted copies of those documents are contained in the compact discs being provided to the New England Coalition and the Department of Public Service. We request that they be treated in accordance with provisions of the Board's Protective Order (January 12, 2007), be protected from disclosure to unauthorized persons, and be made available for review to only those individuals who have executed a Non-Disclosure Agreement. The copies of these documents being provided to the NRC Staff have been redacted to delete the proprietary information.

Sincerely,

Matias F. Travieso-Diaz

Matias F. Travieso-Diaz
Counsel for Entergy

Enclosures (as noted)

ATTACHMENT

Final Structural Integrity Associates Calculations & Reports for VY Being Provided

Structural Integrity Calculation or Report No.	Title	.pdf file name
Calculation File No. VY-16Q-301, Rev. 0	Feedwater Nozzle Stress History Development for Green Functions	VY-16Q-301R0.pdf
Calculation File No. VY-16Q-302, Rev. 0	Fatigue Analysis of Feedwater Nozzle	VY-16Q-302R0.pdf
Calculation File No. VY-16Q-303, Rev. 0	Environmental Fatigue Evaluation of Reactor Recirculation Inlet Nozzle and Vessel Shell/Bottom Head	VY-16Q-303R0.pdf
Calculation File No. VY-16Q-304, Rev. 0	Recirculation Outlet Nozzle Finite Element Model	VY-16Q-304R0.pdf
Calculation File No. VY-16Q-305, Rev. 0	Recirculation Outlet Stress History Development for Nozzle Green Function	VY-16Q-305R0.pdf
Calculation File No. VY-16Q-306, Rev. 0	Fatigue Analysis of Recirculation Outlet Nozzle	VY-16Q-306R0.pdf
Calculation File No. VY-16Q-307, Rev. 0	Recirculation Class 1 Piping Fatigue and EAF Analysis	VY-16Q-307R0.pdf
Calculation File No. VY-16Q-308, Rev. 0	Core Spray Nozzle Finite Element Model	VY-16Q-308R0.pdf
Calculation File No. VY-16Q-309, Rev. 0	Core Spray Nozzle Green's Functions	VY-16Q-309R0.pdf
Calculation File No. VY-16Q-310, Rev. 0	Fatigue Analysis of Core Spray Nozzle	VY-16Q-310R0.pdf
Calculation File No. VY-16Q-311, Rev. 0	Feedwater Class 1 Piping Fatigue Analysis	VY-16Q-311R0.pdf

Structural Integrity Calculation or Report No.	**Title**	**.pdf file name**
Report No. SIR-07-130-NPS, Rev. 0 File No. VY-16Q-401	Environmental Fatigue Analysis for the Vermont Yankee Reactor Pressure Vessel Feedwater Nozzles	VY-16Q-401R0.pdf
Report No. SIR-07-141-NPS, Rev .0 File No. VY-16Q-402	Environmental Fatigue Analysis for the Vermont Yankee Reactor Pressure Vessel Reactor Recirculation Outlet Nozzle	VY-16Q-402R0.pdf
Report No. SIR-07-138-NPS, Rev. 0 File No. VY-16Q-403	Environmental Fatigue Analysis for the Vermont Yankee Reactor Pressure Vessel Core Spray Nozzle	VY-16Q-403R0.pdf
Report No. SIR-07-132-NPS, Rev.0 File No. VY-16Q-404	Summary Report of Plant Specific Environmental Fatigue Analyses for the Vermont Yankee Nuclear Power Station	VY-16Q-404R0.pdf

COPY

August 2, 2007

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)	
	)	
Entergy Nuclear Vermont Yankee, LLC	)	Docket No. 50-271-LR
and Entergy Nuclear Operations, Inc.	)	ASLBP No. 06-849-03-LR
	)	
(Vermont Yankee Nuclear Power Station)	)	

DECLARATION OF TERRY J. HERRMANN

Terry J. Herrmann states as follows under penalties of perjury:

I. PERSONAL BACKGROUND

1. My name is Terry J. Herrmann. I am a Senior Consulting Engineer with Structural Integrity Associates, Inc. ("SIA"), a consulting firm specializing in the prevention and control of structural and mechanical failures. My professional and educational experience is summarized in the *curriculum vitae* attached as Exhibit 1 to this Declaration. Briefly summarized, I have 30 years of experience related to the design, construction, testing, failure analysis, project management and probabilistic risk assessment of nuclear generating facilities. I was the station responsible engineer for submittal of the license amendment to renew the James A. Fitzpatrick Nuclear Power Plant ("JAFNPP") operating license. In that role, I became acquainted with Nuclear Regulatory Commission ("NRC") guidance related to time limiting aging analyses, such as the application of environmentally assisted fatigue multipliers to cumulative usage factor values.

2. As project manager, I have personal knowledge of the matters discussed in this Declaration that relate to the fatigue analyses performed by SIA for certain components at the Vermont Yankee Nuclear Power Station ("VY") at the request of Entergy Nuclear Operations, Inc. ("Entergy").

II. DISCUSSION

3. Fatigue is an age-related degradation mechanism caused by cyclic mechanical and thermal stresses on a component. The results of fatigue can be observed in the cracking of components subjected to cyclic stresses of sufficient magnitude and duration.

4. Design cyclic loadings and thermal conditions for ASME Code Section III Class 1 components are established in the design specifications applicable to those components. The design specifications define the number of mechanical and thermal cycles that a component is to be designed to withstand and still satisfy ASME Code Section III limits and safety factors.

5. At any point in time, the cumulative usage factor ("CUF") for a component represents the fraction of the allowable fatigue cycles that have been, or are projected to be, experienced by the component, including relevant safety factors imposed by ASME Code Section III. The ASME Code Section III criterion requires that the CUF for a Class 1 component not exceed unity.

6. The potential effects of the reactor coolant environment on component fatigue life, also referred to as environmentally assisted fatigue ("EAF"), were the subject of NRC Generic Safety Issue ("GSI") 190, "Fatigue Evaluation of Metal Components for 60-year Plant Life." While GSI 190 was closed out by the NRC Staff in 1999 without the imposition of additional requirements on current term (40-year) licensees, the NRC Staff does require that EAF effects be incorporated into the fatigue analyses performed by license renewal applicants. The criteria and methodology for performing EAF analyses are specified in 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, NUREG-1801 (Rev. 1).

7. SIA was contracted by Entergy to calculate the EAF multipliers and resulting CUFs for critical plant components in accordance with the approach described in the GALL report. SIA has performed fatigue analyses for nuclear power plant components for more than 20 years.

8. SIA personnel performed separate plant-specific analyses of nine VY component locations: (1) the reactor pressure vessel (RPV) shell and lower head; (2) the RPV shell at the shroud support junction; (3) the feedwater nozzle; (4) the recirculation / residual heat removal Class 1 piping; (5) the recirculation inlet nozzle forging; (6) the recirculation inlet nozzle safe end; (7) the recirculation outlet nozzle forging; (8) the core spray nozzle, safe end, and Class 1 piping; and (9) the feedwater Class 1 piping. The analyses applied updated ASME Code methodology using actual cycles accumulated to date by those components, which are then projected to sixty years of operation. In cases where zero cycles were projected, additional events were included in case any might occur.

9. SIA prepared technical reports containing the EAF calculations for the nine component locations listed above, and a summary report (SIR-07-132-NPS; SIA File Number VY-16Q-404) that presents the results of the various analyses. Exhibit 2 to this Declaration is a copy of this summary report, redacted to delete two proprietary items. The other documents associated with SIA's analyses are available separately.

10. As summarized in Table 3-10 of Exhibit 2, the results of the analyses, which have been finalized as of the date of this Declaration, show that the environmentally assisted CUFs for these critical locations for sixty years encompassed by VY's original and extended license periods are in all cases less than unity, signifying that there is confidence that component failure due to fatigue is not a concern at VY during the period of extended operation.

III. CONCLUSION

11. VY has made a commitment in its License Renewal Application to further refine its current fatigue analyses to include the effects of reactor water environment and to verify that the predicted cumulative usage factors (CUFs) are less than 1.0. In my opinion, the above described fatigue analyses performed by SIA satisfy this commitment.

I declare under penalty of perjury that the foregoing is true and correct.

Terry J. Herrmann
Executed on August 2, 2007

Terry J. Herrmann, PE

Senior Consulting Engineer

Education

MS, Engineering Management, Syracuse University (2003)
BS, Mechanical Engineering, Syracuse University (1977)

Professional Associations

Registered Professional Mechanical Engineer, State of New York: License # 060333-1

Professional Experience

2006 to Present	Structural Integrity Associates, San Jose, CA Senior Consulting Engineer
2001 to 2006	Entergy Nuclear Operations, Inc, Oswego, NY Senior Engineer (Nuclear)
1998 to 2001	New York Power Authority, Oswego, NY Senior Mechanical Design Engineer
1993 to 1998	New York Power Authority, Oswego, NY Technical Programs Consultant
1989 to 1993	New York Power Authority, Oswego, NY Systems Engineering Supervisor
1981 to 1989	New York Power Authority, Oswego, NY Plant Engineer / Senior Plant Engineer
1977 to 1981	Stone & Webster Engineering Corporation Boston, MA / Lycoming NY

Summary

Mr. Herrmann has nuclear power generation experience related to design, construction, testing, failure analysis, project management and probabilistic risk assessment (PRA). He has led multi-discipline teams in addressing complex problems within limited time constraints.

At Entergy, Mr. Herrmann developed the Root Cause Analysis (RCA) program at the Fitzpatrick station. Root Cause Analysis timeliness and quality both improved during his tenure.

Mr. Herrmann has performed a number of Root Cause Analyses, including a Boiling Water Reactor (BWR) Primary Containment Suppression Pool through-wall crack related to normal system operational loads. As a designated Entergy fleet role model in RCA, he mentored, provided support to and led RCA teams at facilities in Vermont, Nebraska and Louisiana.

In addition to leading and mentoring Root Cause Analysis teams, Mr. Herrmann was the station

responsible engineer for submittal of the license amendment to renew the Fitzpatrick plant operating license. He also performed risk assessments for online plant maintenance and outages and implemented the NRC Mitigating Systems Performance Index (MSPI) as the site PRA engineer as well as supporting Department of Homeland Security risk assessment efforts related to critical asset protection.

While pursuing his MS in Engineering Management, Mr. Herrmann commissioned a state-of-the-art full-scale thermal and air quality research chamber for the Building Energy & Environmental Systems Laboratory at Syracuse University as part of the Syracuse Center of Excellence in Environmental and Energy Systems. He developed test procedures, conducted testing, performed analyses and presented testing results at the 2003 ASHRAE summer meeting.

At the New York Power Authority, Mr. Herrmann held a number of positions of responsibility, including Systems Engineering Supervisor, Maintenance Rule Coordinator, and Surveillance Testing Program Coordinator. He was involved with the development and implementation of programs in Finite Element Analysis, Design Basis Reconstitution and Preventive Maintenance. As a Kepner-Tregoe® Problem-Solving/Decision-Making program leader Mr. Herrmann helped improve station skills to resolve longstanding equipment deficiencies. Fitzpatrick was a Grand Winner of the Kepner-Tregoe® International Rational Process Achievement Award in 2002.

Mr. Herrmann has been involved with PRA model development and application for nearly 20 years. He performed reviews of the Fitzpatrick PRA model to validate conformance with plant configuration and accurate representation of systems interactions.

At Stone & Webster Engineering Corporation, Mr. Herrmann began his career as a construction field engineer. He provided oversight of subcontractors for compliance with quality standards and schedule adherence. Also during this time, he was involved with initial construction testing, construction tagging, and turnover of systems to the utility for completion of pre-operational testing.

Over the years, Mr. Herrmann has become acknowledged as a Subject Matter Expert in the areas of operability evaluations, 10CFR50.59 evaluations and design calculations in addition to Root Cause Analysis and troubleshooting.

Publications / Presentations

- Presentation to the 2006 Materials Science and Technology Conference (www.matscitech.org), "Failure Analysis of Relays Used in a Nuclear Reactor Application", Cincinnati, OH
- ASHRAE Technical Paper KC-03-4-1, "Performance Test Results for a Large Coupled Indoor/Outdoor Environmental Simulator (C-I/O-ES)", ASHRAE Summer Meeting, 2003
- Presentation to the 2003 Human Performance, Root Cause & Trending Conference (www.hprct.org), "Systematic Approach to Corrective Action Improvement", Groton, CT
- Technical Paper, "Development of a Unique Ultra-Clean Full-Scale Thermal and Air Quality Research Facility", Indoor Air 2002 (www.indoorair2002.org), The 9th International Conference on Indoor Air Quality and Climate, Monterey, CA

EXHIBIT 2

Report No.: SIR-07-132-NPS
Revision No.: 0
Project No.: VY-16Q
File No.: VY-16Q-404
July 2007

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

NOTE

This document references vendor proprietary information. Such information is identified with -2xxP SI Project File numbers in the list of references. Any such references and the associated information in this document where those references are used are identified so that this information can be treated in accordance with applicable vendor proprietary agreements.

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

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

Prepared by: Terry J. Herrmann
Terry J. Herrmann, P.E.
Date: 7/27/2007

Reviewed by: Gary L. Stevens
Gary L. Stevens, P.E.
Date: 7/27/2007

Approved by: Terry J. Herrmann
Terry J. Herrmann, P.E.
Date: 7/27/2007

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	1-1	0	7/27/07	Initial issue.
2.0	2-1 – 2-2			
3.0	3-1 – 3-18			
4.0	4-1			
5.0	5-1 – 5-2			

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:

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, $\dot{\varepsilon} > 1\%/\text{sec}$

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

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

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 $T \ge 200°C$

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

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

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

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.

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 were identified for the reactor recirculation/RHR piping in NUREG/CR-6260 (both stainless steel): the RHR return tee connection to the recirculation piping, and a tapered transition on the RHR line just upstream of the RHR return tee. 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, which has not been replaced, 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 0.0043. Factoring in the environmental multiplier from Table 3-8 [12], the EAF adjusted CUF is 0.0432 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-HWC} Multiplier for HWC Conditions = 5.39
Maximum F_{en-NWC} Multiplier for NWC Conditions = 13.17

$U_{env-60} = U_{60} \times F_{en-NWC} \times 0.53 + U_{60} \times F_{en-HWC} \times 0.47$ = 0.0809
Overall Multiplier = U_{env-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_{II} =	26,240	psi *(p. S3-97 of RPV Stress Report)*
Hydrotest H_I =	-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_{II} =	62,976	psi *(p. S3-97 of RPV Stress Report)*
Improper Startup H_{II} =	28,060	psi *(p. S3-98 of RPV Stress Report)*
Improper Startup H_I =	-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_{II} + Skin Stress =	223,443	psi *(p. S3-98 of RPV Stress Report)*
Warmup H_{II} =	-5,707	psi *(p. S3-99a of RPV Stress Report)*
Warmup H_I =	-102	psi *(p. S3-99a of RPV Stress Report)*
Warmup K_tH_{II} =	-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 $[(H_{II} - H_I)_{Event\ 1} - (H_{II} - H_I)_{Event\ 2}]$ 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 * $[(K_tH_{II} - H_I)_{Event\ 1} - (K_tH_{II} - H_I)_{Event\ 2}]$.*
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:

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

Assume $S^* = 0.015$ (maximum)
Assume $\dot{\varepsilon}^* = \ln(0.001) = -6.908$ (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:

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

Assume $S^* = 0.015$ (maximum)
Assume $\dot{\varepsilon}^* = \ln(0.001) = -6.908$ (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 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:		
ε* = 0 for ε > 0.4%/sec	so F_{en} =	2.55		so F_{en} =	2.55
ε* = ln(ε/0.4) for 0.0004 <= ε <= 0.4%/sec	so F_{en} ranges from	2.55		so F_{en} ranges from	2.55
	to	15.35		to	8.36
ε* = ln(0.0004/0.4) for ε < 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-HWC} Multiplier for HWC Conditions = 2.45
Maximum F_{en-NWC} Multiplier for NWC Conditions = 12.43

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

$[T^* = (T-150)$ for $T > 150°C]$

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

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*□)

Assume S* = 0.015 (maximum)
Assume □* = 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

[T*= (T-150) for T > 150°C]

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

Stainless Steel:

F_{en} = exp(0.935 - T*°*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	0.0043	10.05	0.0432
2	Core Spray Nozzle Safe End	Ni-Cr-Fe	0.0184	1.49	0.0274
3	Core Spray Piping	Stainless Steel	0.0005	8.36	0.0042

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

[T^*= (T-150) for T > 150°C]

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

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 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 Forging 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)	0.0043	10.05	0.0432
9	Feedwater Piping Riser to RPV Nozzle N4B	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 (with the exception of Recirculation piping that uses 61% HWC conditions and 39% NWC conditions).
4. 40 year CUF 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

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

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