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August 25, 2008

**UNITED STATES OF AMERICA**

**NUCLEAR REGULATORY COMMISSION**

Before the Atomic Safety and Licensing Board

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

# ENTERGY'S PROPOSED FINDINGS OF FACT AND CONCLUSIONS OF LAW ON NEW ENGLAND COALITION CONTENTIONS

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## TABLE OF CONTENTS

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# ENTERGY'S PROPOSED FINDINGS OF FACT AND CONCLUSIONS OF LAW ON NEW ENGLAND COALITION CONTENTIONS

Pursuant to 10 C.F.R. § 2.1209, the Atomic Safety and Licensing Board's ("Licensing Board" or "Board") Memorandum and Order (Regarding Corrections to the Transcript and Proposed Findings of Fact and Conclusions of Law) dated August 5, 2008, and its directive at the evidentiary hearing held in Newfane, Vermont on the above captioned matter (see Tr. at 1739-40),[1] Applicants Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (collectively "Entergy") submit their proposed findings of fact and conclusions of law concerning each of the contentions by the New England Coalition ("NEC") which were the subject of the evidentiary hearing in this proceeding. This submittal is presented in the form of a decision by the Board that includes findings of fact and conclusions of law with respect to each contention.

[1] In this filing, the notation "Tr. at xxxx" denotes a reference to page xxxx of the transcript of the evidentiary hearing. The hearing transcript begins at page 694. All citations to the transcript herein refer to the original transcript supplied by the court reporter, without incorporating any pagination changes that might result from the transcript corrections subsequently identified by the Board or the parties.

# I. PROCEDURAL HISTORY

On January 25, 2006, Entergy filed an application ("Application") pursuant to 10 C.F.R. Part 54 to renew Operating License No. DPR-28 for the Vermont Yankee Station ("VY"). Entergy seeks to extend the license, which expires on March 21, 2012, for an additional twenty years. On March 27, 2006, the Commission published a notice of opportunity to request a hearing on the application. 71 Fed. Reg. 15,220 (Mar. 27, 2006). Among the parties that filed hearing requests challenging Entergy's proposed license renewal was the New England Coalition ("NEC").[2] The Board granted NEC's hearing request and admitted several of its proposed contentions. Entergy Nuclear Vermont Yankee, L.L.C., & Entergy Nuclear Operations, Inc. (Vermont Yankee Nuclear Power Station) LBP-06-20, 64 N.R.C. 131, 143 (2006). The Board ruled that the 10 C.F.R. Part 2, Subpart L procedures were appropriate for each of the contentions. Id. at 204.

Three of the contentions proposed by NEC were set for hearing: Contention 2A/2B, which criticizes Entergy's analyses of environmentally assisted fatigue ("EAF") of critical reactor piping and components;[3] Contention 3, which challenges Entergy's plan to monitor and manage aging of the steam dryer during the period of extended operation;[4] and Contention 4,

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2 Petition for Leave to Intervene, Request for Hearing, and Contentions (May 26, 2006) ("NEC Petition to Intervene").

3 As admitted by the Board, NEC Contention 2A states: ". . . [T]he analytical methods employed in Entergy's [environmentally corrected CUF, or] CUFen Reanalysis were flawed by numerous uncertainties, unjustified assumptions, and insufficient conservatism, and produced unrealistically optimistic results. Entergy has not, by this flawed reanalysis, demonstrated that the reactor components assessed will not fail due to metal fatigue during the period of extended operation." Memorandum and Order (Ruling on NEC Motions to File and Admit New Contention), LBP-07-15, 66 N.R.C. 261, 270 (2007).

Subsequently, the Board admitted new NEC Contention 2B, which challenges the confirmatory analysis performed by Entergy in 2008. In admitting the contention, however, the Board ruled that it is really a subset of Contention 2A and does not merit a separate statement. Order (Granting Motion to Amend NEC Contention 2A) (Apr. 24, 2008) ("April 24, 2008 Order") at 2.

4 As recast by the Board, Contention 3 raises two issues: "1. Whether Entergy has established sound evaluation and implementation procedures to assure that the integrity of the steam dryer is not jeopardized. Specifically, NEC contends that the status of the dryer cracks must be continuously monitored and assessed by a competent engineer. While Entergy has established that it will continuously monitor plant parameters indicative of steam dryer cracking, it has not provided information on its assessment program for the monitoring data or the qualifi-

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which alleges that Entergy's program for managing flow-accelerated corrosion ("FAC") of piping and components at VY during the license renewal period is inadequate.[5] The facts, testimony and evidence relating to these contentions are described below.

On October 10, 2007, the Board, accompanied by representatives of the parties, conducted a visit of the VY site in order to view areas of the plant relevant to the contentions at issue. On October 11, 2007, the Board held two limited appearance sessions in Brattleboro, Vermont where members of the public made oral limited appearance statements.

On February 27, 2008, the NRC Staff issued its "Safety Evaluation Report Related to the License Renewal of Vermont Yankee Nuclear Power Station" ("FSER"). The FSER has been introduced in evidence as Staff Exhibit 1.

On March 20, 2008, the Advisory Committee on Reactor Safeguards ("ACRS") issued a letter to the Commission Chairman reporting the results of its review of the proposed renewal of the VY license. The ACRS addressed, inter alia, the issues raised in the contentions which are the subject of this proceeding. The ACRS found:

> We agree with the staff that there are no issues related to the matters described in 10 CFR 54.29(a)(1) and (a)(2) that preclude renewal of the operating license for VYNPS. The programs established and committed to by ENO provide reasonable assurance that VYNPS can be operated in accordance with its current licensing basis for the period of extended operation without undue risk to the health and safety of the public. The ENO application for renewal of the operating license for VYNPS should be approved.

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cations of the personnel evaluating this information. 2. Whether a steam dryer aging management program that does not provide a means to estimate and predict stress loads on the dryer during operation for comparison to established fatigue limits is valid." Memorandum and Order (Ruling on Motion for Summary Disposition of NEC Contention 3) (Sept. 11, 2007), slip op. at 11-12.

5 Contention 4 states: "Entergy's License Renewal Application does not include an adequate plan to monitor and manage aging of plant piping due to flow-accelerated corrosion during the period of extended operation." LBP-06-20, 64 N.R.C. at 192.

Entergy Exh. E2-37 at 4.

On July 9, 2008 the parties entered into a Joint Stipulation ("Joint Stipulation") as to certain facts relating to the three admitted contentions. The stipulated facts will be referenced where appropriate in the discussion of each contention.

The evidentiary hearing on the NEC Contentions was held in Newfane, Vermont, on July 21 through 24, 2008. The hearing, which was conducted under the procedures of Subpart L of 10 C.F.R. Part 2, included incorporation into the record of the direct and rebuttal testimony on the NEC Contentions submitted earlier by the parties and the admission into evidence of numerous exhibits proffered by each party.[6] Examination of the witnesses was conducted by the Board, but

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[6] As to NEC Contentions 2A and 2B, the following testimony was admitted into evidence: Testimony of James C. Fitzpatrick and Gary L. Stevens on NEC Contentions 2A/2B, Environmentally-Assisted Fatigue (May 12, 2008) ("Entergy NEC 2 Dir."), Post Tr.7 at 63; Supplemental Testimony of James C. Fitzpatrick and Gary L. Stevens on NEC Contention 2A/2B, Environmentally-Assisted Fatigue ("Entergy NEC 2 Reb."), Post Tr. at 763; Affidavit of John R. Fair Concerning NEC Contentions 2A and 2B (Metal Fatigue) (May 13, 2008) ("Staff NEC 2 Dir."), Post Tr. at 768; Pre-filed Direct Testimony of Dr. Joram Hopenfeld Regarding NEC Contentions 2A, 2B, 3 and 4 (dated June 20, 2006 [sic], filed April 30, 2008) ("Hopenfeld Dir."), Post Tr. at 778; Pre-filed Rebuttal Testimony of Dr. Joram Hopenfeld Regarding NEC Contentions 2A, 2B, 3 and 4 (dated June 20, 2006 [sic], filed June 2, 2008) ("Hopenfeld Reb."), Post Tr. at 778. Entergy introduced into evidence thirty-seven exhibits relating to NEC Contentions 2A and 2B, numbered Entergy Exhibits ("Exh.") E2-02 through E2-37 and E2-39 (Tr. at 764, 1457). The Staff introduced into evidence sixteen exhibits, Staff Exh. 1-13, 22, 23, and D (Tr. at 771, 773, 774, 776 and 1462). NEC introduced forty-four exhibits (NEC-JH02 through NEC-JH35, NEC-JH62 through NEC-JH66 and NEC JH68-NEC-JH72) (Tr. at 779).

As to NEC Contention 3, the following testimony was admitted into evidence: Joint Declaration of John R. Hoffman and Larry D. Lukens in NRC Contention 3 (May 9, 2009) ("Entergy NEC 3 Dir."), Post Tr. at 1186; Affidavit of Kaihwa Robert Hsu, Jonathan G. Rowley, and Thomas G. Scarbrough Concerning NEC Contention 3 (Steam Dryer) (May 13, 2008) ("Staff NEC 3 Dir."), Post Tr. at 1188; Pre-filed Direct Testimony of Dr. Joram Hopenfeld Regarding NEC Contentions 2a, 2b, 3 and 4 (dated June 20, 2006 [sic], filed April 30, 2008) ("Hopenfeld Dir."), Post Tr. at 778; Pre-filed Rebuttal Testimony of Dr. Joram Hopenfeld Regarding NEC Contentions 2A, 2B, 3 and 4 (dated June 20, 2006 [sic], filed June 2, 2008) ("Hopenfeld Reb."), Post Tr. at 778. Entergy introduced into evidence fifteen exhibits relating to NEC Contention 3, numbered Entergy Exh. E3-02 through E3-16 (Tr. at 1187). The Staff introduced into evidence four exhibits, Staff Exh. 14, 15, and 19 (Tr. at 1190). NEC introduced ten exhibits, NEC Exh. JH-54 through NEC JH-61, NEC JH-68, and NEC JH-69 (Tr. at 1190).

As to NEC Contention 4, the following testimony was admitted into evidence: Testimony of Jeffrey S. Horowitz and James C. Fitzpatrick on NEC Contention 4, Flow Accelerated Corrosion (May 12, 2008) ("Entergy NEC 4 Dir."), Post Tr. at 1425; Affidavit of Kaihwa R. Hsu and Jonathan G. Rowley Concerning NEC Contention 4 (Flow Accelerated Corrosion (May 13, 2008) ("Staff NEC 4 Dir."), Post Tr. at 1430; NRC Staff Rebuttal Testimony Concerning NEC Contention 4 (June 2, 2008) ("Staff NEC 4 Reb."), Post Tr. at 1430; Pre-filed Direct Testimony of Dr. Joram Hopenfeld Regarding NEC Contentions 2a, 2b, 3 and 4 (dated June 20, 2006 [sic], filed April 30, 2008) ("Hopenfeld Dir."), Post Tr. at 778; Pre-filed Rebuttal Testimony of Dr. Joram Hopenfeld Regarding NEC Contentions 2A, 2B, 3 and 4 (dated June 20, 2006 [sic], filed June 2, 2008) ("Hopenfeld Reb."), Post Tr. at 778; Pre-filed Direct Testimony of Dr. Rudolf Hausler regarding NEC's Contention 4 (dated June 20, 2006 [sic], filed April 30,

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the parties had an opportunity before the hearing to submit proposed questions for the Board to propound to the witnesses pursuant to 10 C.F.R. §§ 2.1207(a)(3)(i) and (ii). At the end of the examination of all witnesses on each contention, the parties were given an opportunity to propose additional questions for the Board to consider asking the witnesses to follow up on the examination answers. Tr. at 1157, 1406, 1711. Questions were proposed by the parties and, after consideration, the Board posed such additional questions to the witnesses as it deemed necessary. Id. [7]

Through the prefiled written testimony of the witnesses and their extensive examination by the Board at the evidentiary hearing, a thorough record was compiled that provides a sufficient basis for the findings of fact and conclusions of law set forth herein.

## II. NEC CONTENTIONS 2A AND 2B

### A. Background

Contention 2, as propounded by NEC, asserted that "Entergy's License Renewal Application 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

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2008) ("Hausler NEC 4 Dir."), Post Tr. at 1434; Pre-filed Rebuttal Testimony of Dr. Rudolf Hausler Regarding NEC's Contention 4 ("Hausler NEC 4 Reb.") (dated June 20, 2006 [sic], filed June 2, 2008), Post Tr. at 1435; Pre-filed Direct Testimony of Ulrich Witte Regarding NEC Contention 4 (filed April 30, 2008) ("Witte NEC 4 Dir."), Post Tr. at 1436; Pre-filed Rebuttal Testimony of Ulrich Witte Regarding NEC Contention 4 (filed June 6, 2008, as marked up in Attachment 4 to the Board's July 16, 2008 Order (Rulings on Motions to Strike and Motions in Limine) ("Witte NEC 4 Reb."), Post Tr. at 1435. Entergy introduced into evidence forty-two exhibits relating to NEC Contention 4, numbered Entergy Exh. E4-02 through E4-43 (Tr. at 1426). The Staff introduced into evidence eight exhibits (16, 17, 18, 20, 21, A, B, and C) (Tr. at 1431). NEC introduced forty-four exhibits, NEC Exh. NEC JH-36 through NEC JH-53 and Exh. NEC JH70 (Tr. at 1431-32), NEC Exh. NEC RH-02 through NEC RH-05 (Tr. at 1434), and NEC Exh. NEC W02-W22 (Tr. at 1436).

Several of the exhibits and testimony proffered by NEC, and one exhibit submitted by Entergy, were proprietary and were segregated from the rest of the record. NEC provided redacted versions of those exhibits. No reference was made during the course of the hearing to any proprietary information in the testimony or exhibits.

7 Pursuant to 10 C.F.R. § 2.1207(a)(3)(iii), the Board, by separate order, is providing to the Commission's Secretary all questions submitted by the parties under 10 C.F.R. § 2.1207(a)(3)(i)-(ii). See Tr. at 1421-22.

10 C.F.R. § 54.21(a) and an evaluation of time limited aging analysis, pursuant to 10 C.F.R. § 54.21(c)." LBP-06-20, 64 N.R.C. at 183 (footnote omitted). The Board admitted the contention, finding that NEC had raised a litigable issue "whether Entergy's 'plan to develop a plan' to manage environmentally assisted metal fatigue is sufficient to meet the license renewal requirements of 10 C.F.R. § 54.21(c)(i)-(iii)." Id., 64 N.R.C. at 186. The Board ruled that, while Entergy had stated that it was relying upon 10 C.F.R. § 54.21(c)(1)(iii) (i.e., demonstrating that the effects of aging will be adequately managed), the Application contained only a summary of "options for future plans rather than demonstrating compliance" with § 54.21(c)(1)(iii). Id.

At issue in NEC Contentions 2A and 2B is Entergy's assessment of EAF effects on nine reactor component locations, chosen in accordance with NRC Staff guidance. The initial assessment of environmentally assisted fatigue ("EAF") contained in the VY Application consisted of a screening evaluation of EAF effects for all nine locations by determining cumulative usage factors ("CUFs") at each location (using generic values for certain components that had been designed under the ANSI B.31.1 Code), performing calculations of the environmentally adjusted cumulative usage factor "$CUF_{en}$s" at each location, and determining whether the total $CUF_{en}$s for 60 years of plant operation remain less than unity.

Subsequent to the admission of NEC Contention 2 for adjudication, Entergy informed the Board that it was performing a reanalysis to recalculate the $CUF_{en}$s at the nine locations of interest to demonstrate that the $CUF_{en}$s will remain less than unity throughout the period of extended operations. LBP-07-15, 66 N.R.C. 261, 265 (2007). Entergy performed the reanalysis, disclosing to the parties the preliminary results on June 7 and 13, 2007, and the final results on August 2, 2007. Id.[8] Entergy formally docketed its reanalysis on September 17, 2007. Application Amendment 31 (Staff Exh. 22, Attachment).

[8] Entergy has often referred to its 2007 reanalysis as "refined analysis." See, e.g., Entergy NEC 2 Dir. at A25.

On July 12 and September 4, 2007, NEC filed motions to file a new or amended contention regarding Entergy's program to manage the aging effects of metal fatigue, claiming that Entergy's reanalysis was flawed. In LBP-07-15, the Board admitted NEC's new contention (identifying it as NEC 2A) as alleging that the "analytical methods employed in [environmentally corrected CUF or] $CUF_{en}$ Reanalysis were flawed by numerous uncertainties, unjustified assumptions, and insufficient conservatism, and produced unrealistically optimistic results. Entergy has not, by this flawed reanalysis, demonstrated that the reactor components assessed will not fail due to metal fatigue during the period of extended operation." LBP-07-15, 66 N.R.C. at 270. The Board ordered that the parties litigate NEC's new contention, NEC Contention 2A, holding NEC Contention 2 in abeyance. Id. at 271.

In response to requests for additional information from the Staff, Entergy performed additional, confirmatory analyses of the $CUF_{en}$s for two locations at the feedwater nozzle. Letter, Entergy to NRC, "License Renewal Application, Amendment 34," (NEC Exh. NEC-JH_34 at Attachment 1); Letter Entergy to NRC, "License Renewal Application Amendment 36," (Staff Exh. 23 at Attachment 2).[9] On March 17, 2008, NEC filed a motion to file a new or amended contention challenging these confirmatory analyses. On April 24, 2008, the Board admitted a new contention based on NEC's March 17 motion. The Board found the new contention to be "a subset of Contention 2A" and designated it as NEC Contention 2B. April 24, 2008 Order at 2.

Contentions 2A and 2B were set for hearing, and Contention 2 remains in abeyance. The parties were not to litigate Contention 2 unless and until Entergy returned to reliance on a metal fatigue management program. If Entergy were to propose a new metal fatigue management program (e.g., if it failed to prevail on Contentions 2A and 2B), then NEC might amend NEC Con-

9 At the hearing, the Board defined the EAF analysis included in the Application as the "initial analysis"; the "refined analysis" performed by Entergy in 2007 was termed the "reanalysis"; and the confirmatory analysis Entergy performed in early 2008 was designated as the "confirmatory analysis." Tr. at 803. That nomenclature will be used hereinafter.

tention 2 to address and support its challenges to the revised program. LBP-07-15, 66 N.R.C. at 271.[10]

### B. Witnesses

#### 1. Entergy Witnesses

Entergy's testimony on NEC Contentions 2A and 2B was presented by a panel of two witnesses, each with extensive experience in the evaluation of fatigue in boiling water reactor ("BWR") components and first-hand knowledge of how the fatigue evaluations for critical VY reactor components were performed. The first witness on the panel, Mr. James C. Fitzpatrick, has thirty years of experience in design, construction, and modification of nuclear power plant structures, piping systems, pressure vessels, and other equipment. Entergy NEC 2 Dir. at A3.

Mr. Fitzpatrick worked for over twenty years at VY, and in the last six years of his employment he was Senior Lead Engineer, Design Engineering. Id. In that capacity, he was responsible for overseeing the analyses used to predict the long-term performance of critical VY components, including the potential fatigue of metal piping and equipment exposed to the reactor coolant environment. Id. at A13. In particular, he was responsible for the development of Entergy's proposed program to manage the effects of fatigue on critical reactor pressure boundary components during the proposed VY license renewal period, and therefore has first-hand knowledge of the program and the analyses that support it. Id.

The second Entergy witness, Mr. Gary L. Stevens, is an expert in the application of finite element analysis, fracture mechanics, and structural and fatigue analyses for nuclear components. Id. at A16. He has extensive experience in the application of American Society of Mechanical Engineers ("ASME") Code Sections III and XI methodology to fatigue analyses of reactor vessels and internals components, was the Chairman of former ASME Section XI Task Group on

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[10] As discussed below, since Entergy has prevailed on Contentions 2A and 2B, it need not propose a new fatigue management program, and NEC Contention 2 is therefore dismissed.

Operating Plant Fatigue Assessments, is the Secretary of the ASME Section XI Working Group on Operating Plant Criteria, is the Secretary of the ASME Section XI Subgroup on Evaluation Standards, and is a member of the ASME Section XI Subcommittee on Nuclear Inservice Inspection. Id. He supervised the Structural Integrity Associates ("SIA") technical staff involved in performing the EAF calculations for VY and provided expert technical consultation and review to all aspects of the work. Id. at A18. Mr. Stevens prepared the confirmatory analysis calculation for the feedwater nozzle. Id.

Entergy witness Mr. Fitzpatrick is qualified as an expert in the design and operating history of critical reactor components at VY, the performance of site-specific EAF analyses at VY, and the prediction of the potential fatigue of critical reactor components during the period of extended operations.

Entergy witness Mr. Stevens is qualified as an expert in the analysis and prediction of the performance of BWR components with respect to EAF, the industry standards and guidance relating to the performance of EAF analyses, the development, characteristics and use of computer codes to analyze the potential for EAF failure of such components, and the potential fatigue of critical VY reactor components during the period of extended operations.

The testimony and opinions of the Entergy witnesses on NEC Contentions 2A and 2B are based on both their technical expertise and experience and their first hand knowledge of the factual issues raised in the contentions.

2. NRC Staff Witnesses

The Staff's testimony on NEC Contentions 2A and 2B was presented by Mr. John R. Fair.[11] Mr. Fair has over 35 years experience in the nuclear power industry, including 31 years at

[11] The Staff had designated another witness, Dr. Kenneth Chang, to testify with respect to NEC Contentions 2A and 2B, and had submitted pre-filed direct testimony by Dr. Chang on those contentions. Affidavit of Kenneth C. Chang Concerning NEC Contentions 2A & 2B (Metal Fatigue), Staff Exh. 2 (May 13, 2008). On the eve of the hearing, however, the Staff advised that, for medical reasons, Dr. Chang would not be able to testify and would be not appear as a witness. Letter from Mary C. Baty to the Board, dated July 11, 2008. At the hearing,

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the NRC. John R. Fair; Statement of Professional Qualifications (Staff Exh. 3, following page 6). During his career at the NRC, Mr. Fair has acquired significant experience developing staff technical positions regarding fatigue evaluation of ASME Code components. Id.

Mr. Fair has served as a member of ASME Code working groups on fatigue analysis. Id. He has significant experience reviewing topics related to the mechanical design of ASME Code components and fatigue evaluations for license renewal applications. Id. He has reviewed over a dozen license renewal applications. Tr. at 1144 (Fair).

Mr. Fair also has significant experience with design analysis of ASME Code and ANSI B31.1 piping systems. Staff Exh. 3. In connection with Entergy's Application, Mr. Fair has advised his colleagues in the Division of License Renewal on Entergy's metal fatigue submissions and supported them at meetings with Entergy and the Advisory Committee on Reactor Safeguards ("ACRS"). Mr. Fair was involved in preparation of "Draft Regulatory Issue Summary 2008-XX Fatigue Analysis of Nuclear Power Plant Components" dated April 11, 2008. Id.

Staff witness Mr. Fair is qualified as an expert in the staff technical positions and regulatory guidance regarding fatigue evaluation of ASME Code components, the industry standards and guidance relating to the performance of EAF analyses, and the criteria for evaluating component fatigue in air and reactor water environments.

3. NEC Witness

NEC's witness on these contentions, Dr. Joram Hopenfeld, received a Bachelor of Science degree, a Master of Science degree, and a Ph.D. in Engineering from the University of California, Los Angeles. NEC Exh. NEC-JH_02. Dr. Hopenfeld lists as his relevant areas of expertise thermal/hydraulics, materials, environmental interaction, radioactivity transport, industrial

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the Staff moved to withdraw Dr. Chang's testimony, but the Board ruled that the testimony would be admitted as an exhibit. Tr. at 1462. Since Dr. Chang was unavailable to testify, the opinions expressed in his affidavit cannot be given weight as those of an expert, but the facts attested to in his affidavit are admissible as evidence.

instrumentation and environmental monitoring. Id. At the hearing, however, Dr. Hopenfeld acknowledged inter alia that he is not an expert on stress numerical analysis (Tr. at 831), that he does not know whether environmentally assisted fatigue evaluations are commonly performed at nuclear power plants (Tr. at 1014), that he does not know how certain factors that may affect component fatigue such as oxygen content in the feedwater, surface finish, and heat transfer coefficient are computed (Tr. at 964-65, 1082,1095), and that he does not believe in the accuracy of the results of his own calculations of environmentally assisted fatigue (Tr. at 1130-31). These acknowledged limitations demonstrate that Dr. Hopenfeld has little experience in the analysis and evaluation of environmentally assisted fatigue in reactor piping and components at BWRs and is not well qualified to provide expert opinions on environmentally assisted fatigue issues.

### C. Applicable Legal Standards

Time-limited aging analyses ("TLAAs") are defined in § 54.3 as follows:

> Time-limited aging analyses, for the purposes of this part, are those licensee calculations and analyses that:
>
> (1) Involve systems, structures, and components within the scope of license renewal, as delineated in § 54.4(a);
>
> (2) Consider the effects of aging;
>
> (3) Involve time-limited assumptions defined by the current operating term, for example, 40 years;
>
> (4) Were determined to be relevant by the licensee in making a safety determination;
>
> (5) Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, and component to perform its intended functions, as delineated in § 54.4(b); and
>
> (6) Are contained or incorporated by reference in the CLB.

10 C.F.R. § 54.3. As the reference to "those licensee calculations and analyses" and items (3) and (6) make clear, TLAAs are time-limited aging analyses that are part of a plant's CLB.

A license renewal application must include in the application "[a]n evaluation of time-limited aging analyses." 10 C.F.R. § 54.21(c). 10 C.F.R. § 54.21(c)(1) defines the showing that must be made in this evaluation: The application must demonstrate that any one of three conditions is met with respect to each TLAA:

> (i) The analyses remain valid for the period of extended operation;
>
> (ii) The analyses have been projected to the end of the period of extended operation; or
>
> (iii) The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

10 C.F.R. § 54.21(c)(1).

The Commission explained these three options as requiring the applicant to:

> (1) Justify that these analyses are valid for the period of extended operation;
>
> (2) Extend the period of evaluation of the analyses such that they are valid for the period of extended operation, for example, 60 years; or
>
> (3) Justify that the effects of aging will be adequately managed for the period of extended operation if an applicant cannot or chooses not to justify or extend an existing time-limited aging analysis.

60 Fed. Reg. 22,461, 22,480 (May 8, 1995) (emphasis added). The Commission has thus indicated that 10 C.F.R. § 54.21(c)(1) allows the applicant to choose how to evaluate an existing TLAA. None of the methods are superfluous; each method is a separate and independent means of evaluating TLAAs. Under the express terms of the regulation, justifying the validity of, or extending a TLAA is not required before the issuance of a license renewal. To the contrary, the applicant may choose "not to justify or extend an existing time-limited aging analysis" and instead justify that the effects of aging will be adequately managed. Id.

The fatigue-related TLAAs that must be evaluated are those fatigue analyses that involve time-limited assumptions and are part of the CLB, as discussed above. An analysis of EAF is not part of the current licensing basis and therefore is not, per se, a TLAA.[12] However, EAF is considered in an applicant's aging management program for fatigue pursuant to the resolution of Generic Safety Issue ("GSI") 190,[13] which was closed out after the license renewal rule was promulgated. In closing out GSI-190, the NRC Staff stated:

> The conclusion to close out this issue is based upon low core damage frequencies from fatigue failures of metal components estimated by technical studies making use of recent fatigue data developed on test specimens. The results of these probabilistic analyses and associated sensitivity studies led the staff to conclude that no generic regulatory action is required. However, calculations including environmental effects, that were performed to support resolution of this issue, and the nature of age-related degradation indicate the potential for an increase in the frequency of pipe leaks as plants continue to operate. Thus, the staff concludes that, consistent with the existing requirements in 10 CFR 54.21, licensees should address the effects of the coolant environment on component fatigue life as aging management programs are formulated in support of license renewal.

Memorandum from A. Thadani to W. Travers, "Closeout of Generic Safety Issue 190, 'Fatigue Evaluation of Metal Components for 60-Year Plant Life'" (Dec. 26, 1999) (Exh. E2-04-VY) (emphasis added).[14]

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[12] Section 4.3.1 of the Application states in relevant part:

> A review of the fatigue evaluations reveals the maximum cumulative usage factors (CUFs) for applicable VYNPS Class 1 components. The documents reviewed are current design basis fatigue evaluations that do not consider the effects of reactor water environment on fatigue life. The maximum cumulative usage factors (CUF) for Class 1 components are summarized in Table 4.3-1.

Application, § 4.3.1 at 4.3-2.

[13] In promulgating the license renewal rule, the Commission indicated that designation of an issue as a GSI would not exclude the issue from the scope of the aging management review or time-limited aging analysis evaluation. 60 Fed. Reg. at 22,484. Rather, the Commission identified several options for addressing such an issue, including "develop[ing] an aging management program which, for that plant, incorporates resolution of the aging effects issue." Id. at 22,485.

[14] In connection with a predecessor issue (GSI-166), the NRC had determined that requiring a backfit of the environmental fatigue data to current operating licenses (i.e., to the initial operating terms) could not be justified. See SECY-95-245, Completion of the Fatigue Action Plan (Sept. 25, 1995) (ADAMS Accession No. ML031480210). Accordingly, the CLB of plants in their initial license term do not include EAF analyses.

The evaluation of TLAAs related to metal fatigue is addressed in section 4.3 of the Vermont Yankee Nuclear Power Station ("VYNPS") License Renewal Application ("LRA"). As indicated therein, the effects of aging due to fatigue of Class 1 components are managed by the VYNPS Fatigue Monitoring Program, which tracks and evaluates the plant's operational cycles to ensure those cycles remain within allowable numbers and requires corrective actions if specified limits are approached. LRA at 4.3-3. This program is further described in Section B.1.11 of the LRA, which incorporates the aging management program recommended in Section X.M1 (Metal Fatigue of Reactor Coolant Pressure Boundary) of the Generic Aging Lessons Learned ("GALL") Report, NUREG-1801.[15] As part of this program, the effects of reactor water environment on fatigue life are considered as described in Section 4.3.3 of the LRA and the amplifying commitments in LRA Amendment 35 (Exh. E2-09-VY). Thus, the actions addressing EAF are part of a broader aging management program to be implemented under 10 C.F.R. § 54.21(c)(1)(iii).

Amendment 35 to the Application states:

> At least 2 years prior to entering the period of extended operation, for the locations identified in NUREG/CR-6260 for BWRs of the VY vintage, VY will refine our current fatigue analyses to include the effects of reactor water environment and verify that the cumulative usage factors (CUFs) are less than 1. This includes applying the appropriate Fen factors to valid CUFs determined in accordance with one of the following:
>
> > 1. For locations, including NUREG/CR-6260 locations, with existing fatigue analysis valid for the period of extended operation, use the existing CUF to determine the environmentally adjusted CUF.

[15] LRA Amendment 31 (Staff Exh. 22) eliminated the exceptions that were identified in Section B.1.11, making the Fatigue Management Program consistent with Section X.M1 of the GALL Report.

> 2. More limiting VY-specific locations with a valid CUF may be added in addition to the NUREG/CR-6260 locations.
>
> 3. Representative CUF values from other plants, adjusted to or enveloping the VY plant specific external loads may be used if demonstrated applicable to VY.
>
> 4. An analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case) may be performed to determine a valid CUF.
>
> 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.

Entergy Exh. E2-09, Attachment 3, Commitment 27.

Therefore, Entergy has committed to implementing one of three options for managing a location where a the environmentally adjusted CUF of 1.0 may be exceeded: "(1) further refinement of the fatigue analyses . . . ; (2) management of fatigue at the affected locations by an inspection program that has been reviewed and approved by the NRC . . . ; (3) repair or replacement of the affected locations." For purposes of its license renewal application, Entergy has pursued option (1), i.e., it has performed "refined analyses" beyond those included in the Application (i.e., the 2007 reanalysis and the 2008 confirmatory analysis), and has sought to demonstrate that they adequately manage the effects of EAF for the affected locations.

Contention 2A challenges Entergy's 2007 reanalysis, and Contention 2B challenges the confirmatory analysis Entergy performed in early 2008. Therefore, the legal issue to be addressed with respect to NEC Contentions 2A and 2B is whether the 2007 EAF reanalysis and the 2008 confirmatory analysis provide reasonable assurance that the effects of aging of the component locations at issue will be adequately managed for the period of extended operation. 10 C.F.R. § 54.21(c)(1); 10 C.F.R. § 54.29(a). See also Nuclear Power Plant License Renewal Final Rule, 60 Fed. Reg. at 22,479 (". . . the [license renewal] process is not intended to demonstrate absolute assurance that structures or components will not fail, but rather that there is reasonable assurance that they will perform such that the intended functions . . . are maintained consistent with the CLB").

A collateral legal issue has been raised by the fact that the Staff has imposed a licensing condition requiring that Entergy perform additional confirmatory analyses for two other reactor components, the recirculation outlet nozzle and the core spray nozzle. FSER, Section 4.3.3.2 at 4–41 through 4–43. Entergy is to submit these analyses to the Staff no later than two years prior to the start of the period of extended operation, in March 2012. Id. at 4–43. The question raised by this requirement is whether it is legally permissible under 10 C.F.R. § 54.29 to issue a license renewal when certain time limited aging analyses ("TLAAs") have not been performed. The Board asked the parties to brief this issue, and a related one – whether such a licensing condition requiring the performance of environmentally adjusted fatigue analyses after the license renewal is issued complies with the Part 54 requirement that the license application contain . . . an evaluation of TLAAs pursuant to 10 C.F.R. § 54.21(c). Order (Regarding the Briefing of Certain Legal Issues) (June 27, 2008).

Having received and reviewed briefs by the parties on these legal issues, the Board concludes that it is lawful to require $CUF_{en}$ analyses to be performed after a license renewal is issued, because these analyses are not TLAAs to which the requirements of 10 C.F.R. § 54.21(c) applies. Rather, EAF is an aging effect that is being managed as part of the broader fatigue monitoring program implemented pursuant to 10 C.F.R. § 54.21(c)(1)(iii). That program includes tracking transient cycles during the period of extended operation to ensure that fatigue analyses remain valid, and taking corrective action if they do not.

In sum, Entergy has chosen the approach specified in 10 C.F.R. § 54.21(c)(1)(iii) with respect to fatigue of critical reactor components. Entergy has addressed TLAAs related to metal fatigue by establishing a Fatigue Monitoring Program consistent with Section X.M1 of the GALL Report, as discussed in the Section 4.3 and Section B.1.11 of the LRA and the amendments thereto. EAF is addressed in this aging management program.

Section X.M1 of the GALL Report provides a program acceptable to the NRC Staff for managing metal fatigue of the reactor coolant pressure boundary, including the effects of the coolant environment on component fatigue life:

> The AMP [aging management program] addresses the effects of the coolant environment on component fatigue life by assessing the impact of the reactor coolant environment on a sample of critical, components for the plant. Examples of critical components are identified in NUREG/CR-6260. The sample of critical components can be evaluated by applying environmental life correction factors to the existing ASME Code fatigue analyses. Formulae for calculating the environmental life correction factors are contained in NUREG/CR-6583 for carbon and low-alloy steels and in NUREG/CR-5704 for austenitic stainless steels.

GALL Report, Vol. 2, Rev. 1 at X M-1. The program provides for corrective actions to prevent the usage factor from exceeding the code design limit, which may include repair, replacement or "a more rigorous analysis of the component to demonstrate that the design code limit will not be exceeded during the period of extended operations." Id. at X M-1 to X M-2. The GALL Report states, "this is an acceptable option for managing metal fatigue for the reactor coolant pressure boundary, considering environmental effects," and thus "no further evaluation is recommended for license renewal if the applicant selects this option under 10 CFR § 54.21(c)(1)(iii) to evaluate metal fatigue for the reactor coolant pressure boundary." Id.

Entergy's commitments, which are amplified in Amendment 35 to the LRA (Exh. E2-09-VY, Att. 3, Commitment 27), follow this guidance. Such commitments are a permissible means of satisfying 10 C.F.R. §§ 54.21(c)(1)(iii) and 54.29(a).

### D. Summary of the Evidence on NEC Contentions 2A and 2B

#### 1. Overview

Metal fatigue is an age-related degradation mechanism caused by cyclic mechanical and thermal stresses at a location on a metallic component. The results of fatigue can be observed in the cracking of components subjected to cyclic stresses of sufficient magnitude and duration. Finding 1.[16]

The design specifications for a given safety-related component specify the number of mechanical and thermal cycles that the component is expected to experience during its design life, and define the safety limits and applicable codes that must be satisfied. For components exposed to the primary reactor coolant pressure boundary, the specified requirements for evaluation of cyclic loading and thermal conditions are contained in Section III of the ASME Code for Class 1 components. Finding 2.

For a Class 1 component, stress cycles from the loadings specified in the governing design specification will produce total stresses of several different magnitudes. The number of times these stress magnitudes occur also varies. The allowable number of cycles for a given alternating stress range is determined from the ASME Code design fatigue curve for the material being evaluated. The fatigue usage for that stress cycle is the ratio of the number of applied stress cycles (n) to the allowable number of stress cycles (N) from the ASME Code design fa-

[16] "Finding xx" refers to finding of fact number xx, set forth in Section V below.

tigue curve. The cumulative usage factor ("CUF") for the component is the sum of the individual usage factors for all of the various stress magnitudes. At any point in time, the CUF for a component represents the fraction of the allowable fatigue cycles that the component has experienced up to that time. Finding 3. ASME Code Section III requires that the CUF for a Class 1 component not exceed unity; that is, the total number of applied stress cycles is not to exceed the allowable number of stress cycles. Finding 4. This is a criterion for acceptability established by the Code, but exceeding the criterion does not mean the component will fail, given the number of factors of conservatism included in the analytical process. Finding 31. A CUF of unity means that there is a 1 to 5 percent probability that a small crack, 3 millimeters deep, may have formed on the component. Id.

For components (equipment and piping) exposed to reactor coolant water, the fatigue life, as measured by the allowable number of stress cycles, may be reduced compared to the components' fatigue life when exposed to an air environment. The ASME Code design fatigue curves were developed based on laboratory testing of specimens in an air environment, with safety factors incorporated into the curves to account for several factors including atmosphere, surface finish, size effects, etc. Laboratory testing of specimens in water under reactor operating conditions indicate that, under certain situations, additional environmental factors may need to be included in the calculated CUF to fully accommodate reactor coolant environmental conditions. Accounting for the effects of operating in a reactor coolant environment in the fatigue analysis is called environmentally assisted fatigue ("EAF") analysis. Finding 5.

To quantify the effects of the reactor coolant environment on component fatigue, the CUF for a component exposed to reactor coolant may be multiplied by an adjustment factor, or "EAF multiplier" ("$F_{en}$"), when appropriate environmental conditions exist. This results in an environmentally adjusted CUF or $CUF_{en}$. The resulting $CUF_{en}$ must still not exceed unity. Finding 6.

Section 4.3.3 of the Application presents Entergy's initial assessment of the effects of the reactor coolant environment on fatigue life for nine plant-specific locations of six reactor components at VY, selected in accordance with NUREG/CR-6260 and the NRC Staff's "GALL Report." Finding 7. The component locations identified in NUREG/CR-6260 and endorsed by the GALL Report 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. Due to the inclusion of both piping and nozzles, as well as the different materials for the nozzle forgings and nozzle safe ends, a total of nine locations for the six components identified in the NUREG/CR-6260 list above were evaluated for EAF at VY. Finding 8.

The initial $CUF_{en}$s computed by Entergy for VY are presented in Table 4.3.3 of the Application. Seven of the nine locations specified in NUREG/CR-6260 had $CUF_{en}$s greater than unity, and therefore greater than the specified criterion of the ASME Code. Finding 9. To address these results, Amendment 35 to the Application commits Entergy, inter alia, to refine its fatigue analyses and verify that the environmentally adjusted CUFs are less than 1. Finding 11.

Entergy engaged Structural Integrity Associates ("SIA") in 2007 to perform reanalyses to calculate the CUFs, $F_{en}$s and $CUF_{en}$s for all nine locations of interest in accordance with the approach described in the GALL Report. Finding 12. Final versions of the reanalyses were issued in December 2007. Finding 13.

The results of the 2007 reanalysis indicate that the $CUF_{en}$s for the nine limiting piping and vessel locations for the sixty years throughout VY's extended license period are in all cases less than unity, signifying that component failure due to fatigue will not be a concern at VY during the period of extended operation. Finding 14.

Since performance of the 2007 reanalysis has demonstrated that environmentally assisted fatigue will not be a concern during the period of extended operation, it is Entergy's position that no further actions regarding metal fatigue are currently necessary. Finding 15. Nonetheless, the condition of piping and components will continue to be monitored under the plant's in-service inspection program through the period of extended plant operation. Id. In addition, the VY Fatigue Monitoring Program will continue to track plant cycles and transients to ensure that the numbers of transient cycles experienced by the plant remain within the analyzed numbers of cycles for all transients. Id.

If, at some future time, the results of continued monitoring suggest that the evaluations no longer encompass 60 years of plant operation, further refined analysis, submittal of an inspection program for NRC review, or replacement of the component in question may become necessary or desirable. Finding 16.

2. Entergy's EAF Analyses

The initial assessment of environmentally assisted fatigue contained in the VY Application consisted of the evaluation of EAF effects for all nine locations in accordance with the provisions of Section X.M1 of the GALL Report by performing $CUF_{en}$ calculations for the nine locations of interest. Finding 17. The initial $CUF_{en}$s computed by Entergy for VY found that seven of the nine locations had $CUF_{en}$s greater than unity, and therefore greater than the specified criterion of the ASME Code. Finding 18. Entergy obtained these results by using CUFs for the reactor components derived from either the original design reports for the plant or updated analyses prepared in 2003 by General Electric, the plant's vendor, to account for the effects of the proposed power uprate on the CUFs. Finding 19. For piping components of interest, CUFs were reported in the Application utilizing generic values provided in NUREG/CR-6260 because the VY design basis for piping uses ANSI B31.1 Power Piping Code methodology, which does not require explicit fatigue analysis. Finding 20.

To address these results, Entergy engaged SIA in 2007 to perform a reanalysis of the reactor and piping component locations specified by NUREG/CR-6260. SIA was to perform a more refined analysis of the locations that showed $CUF_{en}$s greater than unity, and provide plant-specific analyses for the reactor piping, for which the Application analysis had used generic CUF values. Finding 21.

The reanalysis sought to demonstrate the acceptability of the $CUF_{en}$s for the locations at issue, not to calculate the available margin. Finding 22. Accordingly, once the analysis had shown the $CUF_{en}$ is less than one, acceptability had been demonstrated and there would be no need to proceed to calculate what the actual margin is. Id.

The methodology used in the reanalyses to compute CUFs starts by constructing a finite element "mesh" model of the component, which was then used to determine the stresses to which each element in the model is subjected, using ANSYS, a well established computer code. Finding 24. Next, the pressure and temperature transients to which each component will be subjected during VY's operation are identified. Finding 25. The magnitude and frequency of each transient are given by the design specifications, and have been proved to be conservative on the high side (i.e., overestimate the severity of the pressure and temperature fluctuations and the frequency of occurrence of the transients). Id.

For each type of transient, the analysis develops a time history of the temperature and pressure stresses imparted on the component location during the transient. The magnitudes of those stresses are added linearly in order to develop a stress time history for that type of transient. Finding 26.

Since the fatigue of a component is due to stress fluctuations and the order of occurrence of the transients is not known a priori, the ASME Code conservatively requires that transients be "paired" so as to create the largest possible stress fluctuations, and thus a transient resulting in the highest stresses is paired with (followed by) a transient with the lowest stresses. Finding 27.

This is done until all the available transient pairs have been exhausted, taking into account the relative frequency of occurrence of each transient. Id. That process results in the most conservative estimate of the stress fluctuations to which a component will be subjected by effectively assuming the worst possible ordering of the transients. Id.

Each set of stress fluctuations results in a number of fatigue cycles being imparted on the component at a given location. This number of applied cycles is used to compute a fraction of the component's fatigue life using the number of cycles that are allowed by the ASME Code for the material at each stress level. The number of allowable cycles at each stress level is determined using an ASME Code "S-N" fatigue curve. Finding 28. The addition of all fractions for all stress levels experienced by the component yields the cumulative usage factor or "CUF" for the component at the location of interest. Id. The general overall methodology is standard in the industry and has been in place for over 30 years, so there is no disagreement as to its appropriateness. Finding 29.

The S-N curves provided in the ASME Code are established based on strain testing of various material specimens in air. Finding 32. To translate the results obtained from those curves to the fatigue limits for the materials in a reactor environment, it is necessary to apply $F_{en}$ correction factors to compute the environmentally adjusted fatigue cumulative usage factors when appropriate. Id.

The general methodology for computing $CUF_{en}$s incorporates a number of conservatisms. In addition to those, the VY EAF calculations in the 2007 reanalysis incorporated a number of VY-specific conservatisms, including:

a. The number of transient cycles for 60 years used in the calculations is conservative relative to the numbers of transients expected to occur through 60 years of operation;

b. The calculations use conservative, design basis transient pressure and temperature fluctuation definitions, as opposed to the less abrupt changes experienced during actual plant transients;

c. The calculations use bounding values for pressure, temperature and flow rate at extended power uprate ("EPU") conditions for the entire 60-year period of plant operation, instead of using the actual conditions that existed during the first thirty-four years of plant operation before the EPU implementation; and

d. The calculations obtain bounding $F_{en}$ multipliers through the use of temperature, strain rate and sulfur content values selected to maximize the $F_{en}$ multipliers. Finding 33.

The results of the 2007 reanalysis show that the environmentally adjusted fatigue usage factors for all locations and components analyzed remain within the allowable value of 1.0 through 60 years of VY operation. Finding 34.

Upon review of Entergy's 2007 reanalysis, the NRC Staff determined that it had no concerns with the reanalysis calculations prepared by Entergy for the six piping and component locations that did not involve nozzle corners. Finding 36. For those six locations, Entergy's reanalysis had consisted of performing conservative calculations based on ASME Code methodology. Those locations could be shown to have acceptable $CUF_{en}$s without using many analytical refinements. Finding 37. The Staff, however, took the position that Entergy's methodology could lead to non-conservative results for locations with significant geometric discontinuity, such as locations at the blend radius (nozzle corner) regions of three reactor pressure vessel components: the feedwater nozzle, the recirculation outlet nozzle, and the core spray nozzle. Finding 39.

At the three nozzle corner locations, the reanalysis had used a single stress difference component as output from a Green's Function analysis to generate stress difference histories for all transients. Finding 40. The Staff felt this simplification might introduce potential non-conservatisms in the computation of CUFs, specifically in the nozzle corner regions, compared to the use of all six stress components of the stress tensor.

To resolve the Staff's concerns, Entergy proposed, and the NRC Staff accepted, that Entergy perform a confirmatory $CUF_{en}$ analysis of the feedwater nozzle using methods that would be acceptable to the NRC. Finding 41. Entergy selected, with the agreement of the Staff, the feedwater nozzle for the confirmatory analysis because (1) it is the limiting nozzle (i.e., has the highest $CUF_{en}$) in the VY refined calculations among the three nozzles regarding which the Staff had questions, (2) it is subjected to more transients and cycles than the other two nozzles, and (3) the transients it experiences are more severe than the transients experienced by the other two nozzles. Finding 42.

The confirmatory analysis used the same finite element model, thermal transient definitions, numbers of transient cycles, and water chemistry inputs as the 2007 reanalysis performed by SIA for Entergy, but differed from the reanalysis in several respects. In particular: (1) when the thermal transient stress histories were determined, the confirmatory analysis computed 6-component stress histories for each transient, whereas the refined analysis used a simplified single stress component difference to obtain the stress time history for all of the transients; (2) in the confirmatory analysis, six stress components are combined to obtain a maximum stress intensity history for all evaluated transients. In the reanalysis, on the other hand, only the maximum stress difference, which is essentially equal to the stress intensity computed from the finite element program, is used; (3) in the confirmatory calculation, a maximum $F_{en}$ is computed for each incremental stress load-pair, which constitute the paired transient stress state points into which the applied loading history is separated for calculating the CUF. Each $F_{en}$ value is based on the maximum transient temperature unique to each load pair, and the contributions of all load pairs are added to produce a composite $CUF_{en}$. In the reanalysis, on the other hand, a single, maximum $F_{en}$ is applied to the total CUF resulting from all load pairs, and is based on the maximum transient temperature for all load pairs. Finding 45.

The confirmatory feedwater nozzle EAF evaluation was performed for the two controlling locations on the nozzle, the inside surface of the nozzle blend radius (nozzle corner) and at the inside surface of the nozzle safe end. Finding 46. The confirmatory calculation yielded a CUF (before application of $F_{en}$ factors) of 0.089 at the nozzle corner, versus a CUF of 0.064 at the same location using the refined analysis methodology. Finding 47. The corresponding results for the safe end location of the nozzle showed a reduction in the computed CUF when using the confirmatory analysis methodology versus the results of the reanalysis for that location. Id. The confirmatory calculation yielded an environmentally adjusted $CUF_{en}$ of 0.353 for the nozzle corner, significantly below the acceptable ASME limit. Finding 48. The reanalysis calculation, on the other hand, yielded an environmentally adjusted CUF of 0.639 for the nozzle corner, which was higher than the confirmatory analysis result but still significantly below the acceptable limit of 1.0. Id.

The Staff requested that Entergy compute the $CUF_{en}$s using the confirmatory analysis methodology and the single, limiting value of $F_{en}$ correction factor used in the reanalysis (instead of using individual, applicable $F_{en}$s for each transient). Using the bounding $F_{en}$ value of 10.05 used in the reanalysis, the confirmatory analysis results increased to 0.893, still below unity. Finding 49.

After review of the confirmatory analysis results for the feedwater nozzle locations, the Staff found that the methodology and results of the confirmatory analysis were appropriate and acceptable. Finding 50. The Staff then imposed a license condition requiring similar confirmatory analyses for two other nozzles, the recirculation outlet nozzle and the core spray nozzle. Finding 51. Entergy is to submit these analyses to the Staff no later than two years prior to the start of the period of extended operation, in March 2012. Id.

The Board finds that it is acceptable to perform these two confirmatory analyses after the license renewal decision is reached for several reasons. First, there is no requirement to address the effects of coolant environment on component fatigue life prior to license renewal. Finding 52. Second, the methods for performing the calculations have been defined, and the required performance of the calculations two years prior to the license renewal period ensures that the environmental effects are addressed prior to entering the period of extended operation. Id. Third, because the $CUF_{en}$s at those two nozzle corner locations obtained in Entergy's reanalysis are very small (0.084 for the recirculation outlet nozzle and 0.167 for the core spray nozzle) and based on comparisons to the feedwater nozzle, which is a limiting component, it is extremely improbable that the results of the confirmatory analyses of these two nozzles would yield $CUF_{en}$s greater than unity. Id.

The Board finds that there is no practical difference between the approaches in the reanalysis and confirmatory analyses because they both yield conservatively calculated $CUF_{en}$s for all nine limiting piping and vessel locations that are well within the acceptable limit. Thus, regardless of what method one chooses to apply, the conclusion is the same – the critical reactor components will not experience failure due to fatigue during the period of extended operation. Finding 53.

### 3. Issues Raised by NEC in Contentions 2A and 2B

NEC asserts the following "errors" in Entergy's 2007 reanalysis and confirmatory analysis: (1) Entergy used "outdated" statistical equations to calculate the $F_{en}$ parameters, and should have used instead the results in the 2007 guidance document NUREG/CR-6909; (2) Entergy failed to account for factors that affect the values of the $F_{en}$ parameters; (3) Entergy has not provided proof that the base metal of the feedwater nozzles is not cracked; (4) Entergy used inappropriate heat transfer equations to calculate the thermal stress for each transient; (5) that the number of plant transients estimated to occur during the operating life of VY is not sufficiently conservative; (6) Entergy's calculation of the $F_{en}$ parameters does not appropriately account for oxygen concentrations and resulting changes in water chemistry; and (7) Entergy failed to perform an error analysis on its calculations. In addition, NEC criticizes the 2007 reanalysis because it uses a simplified Green's Function methodology, which allegedly results in "the underestimation of CUF values by approximately 40%."

### a. Use of NUREG/CR-6909 Methodology

Entergy performed its reanalysis and confirmatory analysis following the criteria and methodology for performing EAF analyses specified in Section X.M1 of the GALL Report. The methodology is comprised of three steps: (1) the CUF for a component is calculated in accordance with ASME Code guidance; (2) the environmental multiplier, $F_{en}$, is calculated in accordance with the guidance in NUREG/CR-6583 for carbon and low alloy steels, and in NUREG/CR-5704 for stainless steels; and (3) the $CUF_{en}$ is calculated as the product of the CUF for the component and the corresponding $F_{en}$. Finding 55.

NUREG/CR-6909, issued in February 2007, incorporates additional information from an expanded database with respect to the fatigue behavior of stainless steels, and presents revised fatigue corves for calculating CUFs in air that are somewhat different from those contained in the ASME Code. Finding 56. The NRC Staff has approved the use of NUREG/CR-6909 in licensing of new reactors, but has not required its use in fatigue analyses for operating plants or plants undergoing license renewal. Id. One reason that the Staff has not required utilization of the NUREG/CR-6909 methodology for operating plants and those plants seeking renewal of their licenses is that application of the NUREG/CR-6909 methodology would produce less conservative results (i.e., lower $CUF_{en}$ estimates) than use of existing ASME Code fatigue curves coupled with the methodology in NUREG/CR-6583 and NUREG/CR-5704. Finding 57.

A set of calculations performed by Entergy prior to the hearing confirmed the Staff's assessment that use of the NUREG/CR-6909 methodology would produce less conservative EAF estimates than those obtained using the ASME fatigue curves and the methodology in NUREG/CR-6583 and NUREG/CR-5704, as employed in Entergy's reanalysis and confirmatory analysis. Finding 58. Entergy's $CUF_{en}$ recomputation started with the stress results from the 2007 reanalysis, replaced the fatigue curves and the $F_{en}$s used in the reanalyses with ones computed following NUREG/CR-6909, and obtained new $CUF_{en}$s that incorporated all the NUREG/CR-6909 methodology. Finding 59.

Entergy recomputed the $CUF_{en}$s for all nine locations specified in NUREG/CR-6260 by applying both the revised air curves set forth in NUREG/CR-6909 and the methodology for computing $F_{en}$ factors for carbon, low alloy and stainless steel provided in that NUREG. Finding 60. The values of $CUF_{en}$ computed following NUREG/CR-6909 guidance are in every case (i.e., at all nine locations) less than unity and lower than the corresponding values obtained in Entergy's reanalysis using the ASME air curves and the methodology in NUREG/CR-6583 and NUREG/CR-5704. Id.

#### b. Factors Affecting $F_{en}$ Parameter Values

While a number of environmental factors may affect the potential fatigue of components subjected to a reactor coolant environment, the most significant ones as reflected in laboratory data are the strain rate, the presence of dissolved oxygen in the coolant, the fluid temperature, and the sulfur content in the material. Finding 62. All of those factors were taken into account in the Entergy reanalysis and confirmatory analysis. Id. The Staff confirmed that Entergy properly considered dissolved oxygen, strain rate, temperature and sulfur content in calculating Fens. Id. The Staff also verified that values of strain rate, temperature, and sulfur content used in the calculation of the $F_{en}$s would remain valid for the period of extended operations. Id.

NEC, on the other hand, contends that as many as thirteen factors need to be considered, and Entergy has failed to properly (or at all) account for them. Those factors are as follows.

(1) Strain Rate

When a component is subjected to cyclic strains, the oxide layer that protects the base material may be cracked, thereby exposing the base metal to the environment. If the rate of strain application is sufficiently high, the underlying material may not be affected. However, if the rate of strain application is low, the material is exposed to the environment for a longer period of time. Once above a threshold value, the environmental effect no longer depends on the magnitude of the strain but on how long the base material is exposed to the environment. Finding 64. Entergy used a value of strain rate in its $F_{en}$ computations that maximized the $F_{en}$ values. Id. NEC did not disagree with Entergy's treatment of the strain rate. Id.

(2) Dissolved Oxygen in Feedwater

Entergy investigated the variability in dissolved oxygen in the feedwater from plant data, including water chemistry excursions over a thirteen year period, and obtained a mean plus one sigma value for dissolved oxygen to which each component of interest is exposed. Finding 66. Entergy then provided those bounding values to SIA for use in the $CUF_{en}$ analyses. Id. These values range from 40 to 128 parts per billion ("ppb"). Id.

NEC claims that, in calculating the $F_{en}$ parameters, Entergy did not properly account for unanticipated increases in oxygen levels during transients. Finding 67. Dr. Hopenfeld's opinion that dissolved oxygen levels increase during transients is based in part on a figure in an EPRI Report that he interprets as showing that $F_{en}$s can rise to levels of 80 or higher due to high oxygen levels in the feedwater during plant transients. Id. However, Mr. Stevens (the author of the EPRI Report in question) explained that the curve from which the values Dr. Hopenfeld cites was intended to show the range of possible variations in $F_{en}$ and the higher values of $F_{en}$ apply to conditions of high oxygen levels and low strain rates that do not exist at VY. Id. Moreover, the report recommends that bulk oxygen levels should be time-averaged before they are used as inputs to the fatigue analysis, which is the approach that Entergy took in its EAF analyses. Id.

Plant data indicate that the oxygen concentration does not vary significantly during transients. Finding 68. In addition, the transients where increased oxygen concentration could be observed (startup and shutdown) are very small contributors to the total CUF. Id.

Another factor cited by Dr. Hopenfeld in support of his conclusion that high levels of dissolved oxygen occur during transients is "plain physics," by which he means that when the temperature of the coolant goes down, the solubility of gases like oxygen increases. Finding 69. However, for the transients of interest, if the temperature decreases, the oxygen will remain in solution because the vessel is still pressurized and the feedwater is not in a saturated condition. Id.

Finally, Dr. Hopenfeld cites the recommendation in NUREG/CR-6909 at A5 that "A value of 0.4 ppm [400 ppb] for carbon and low-alloy steels and 0.05 ppm [50 ppb] for austenitic stainless steels can be used for the [dissolved oxygen] content to perform a conservative evaluation." Finding 70. Mr. Fair, however, clarified that the 400 ppb value was used in NUREG/CR-6909 as a default value if no plant-specific data are available. Id. There is no need to apply this default value at VY, since site-specific oxygen data exist for the plant.

NEC also claims that fatigue is sensitive to electrochemical potential, but its witness Dr. Hopenfeld acknowledges that electrochemical potential cannot be measured at a plant in all applicable locations and one must resort to dissolved oxygen measurement as an alternative. Finding 73. The $F_{en}$ expressions used by Entergy, and documented in NUREG/CR-6583 and NUREG/CR-5704, are not dependent on electrochemical potential. Id.

(3) Surface Finish

In the development of the ASME design curves, an adjustment factor of 4 was applied to account for the difference between the specimens tested in the laboratory, which were mirror polished specimens, and the components actually installed in the plant. Finding 74. In developing the revised fatigue air curve in NUREG/CR-6909, this adjustment factor was scaled down to a range between 2.0 and 3.5. Id. In its $CUF_{en}$ calculations, Entergy used the more conservative 4.0 adjustment factor since the ASME Code fatigue curves were utilized to calculate CUF values. Id.

The surface finish adjustment is included in the ASME design curve, so it does not need to be duplicated when computing the $F_{en}$. Finding 75.

(4) Coolant Temperature Below 150°C

When the temperature of the coolant drops below 150°C, the value of $F_{en}$ drops to a constant value, becoming a number close to two for carbon steel and just over two for stainless steel. Finding 76.

(5) Other Environmental Factors

NEC provided a list of environmental factors that it alleges affect the $F_{en}$ values and were not taken into account in Entergy's $F_{en}$ analyses. The list includes, in addition to the previously discussed strain rate, dissolved oxygen, surface finish, and coolant temperature below 150°C, eight other factors which Dr. Hopenfeld regarded as less significant: data scatter, size, flow rate, heat to heat variations, loading history, cyclic strain hardening, trace impurities in water, and sulfide morphology. Finding 77. Dr. Hopenfeld also includes potential feedwater base metal cracks in his list; that issue is discussed separately below.

Entergy testified that all but one of those eight factors were either directly or inherently included in its $CUF_{en}$ reanalysis and confirmatory analysis, as well as in the NUREG/CR-6909 guidance. Finding 78. The only factor that was not considered was trace impurities in the water. NUREG/CR-6909 states that it is very improbable that any kind of water impurity would be present during a transient event, thus such impurities need not be considered. Id.

**c. Cracks in the Feedwater Nozzle**

NEC raises the possibility that surface cracks may exist at the blend radius and the base metal of the feedwater nozzle. Finding 79. VY periodically inspects the feedwater nozzle for potential cracks in the base metal using ultrasonic testing ("UT") techniques. Id. The UT examinations, which are standard in the industry, are designed to detect the minimum size flaw that is postulated to occur in the base metal, on the order of 3/16 of an inch. Id.

Entergy has performed a fracture mechanics analysis that postulates the existence of a minimum size crack and estimates its growth with time. Finding 80. Based on that analysis, a nozzle inspection program is in place at VY, designed to inspect the nozzle at such frequency that cracks can be detected before they grow beyond allowable size. Id. One hundred percent UT inspections of all nozzles are performed every four refueling cycles (i.e., every six years). Id. In the last 20 years, the inspection program has not detected cracks in the feedwater nozzle base metal that are above the detection capability of the UT equipment. Finding 81. The most recent inspection was conducted during the 2007 refueling outage and showed no evidence of cracks in the base metal of the nozzle. Id.

The inspection program is conducted pursuant to ASME Section XI. The fatigue analysis, which is conducted under ASME Section III, inherently assumes there are no cracks in the nozzle. Finding 82. If a crack in the nozzle were detected, it would have to be repaired and otherwise dealt with under the existing ASME Section XI program. Id. Thus, its existence would not affect the Section III $CUF_{en}$ analysis. Id.

At the hearing, NEC witness Dr. Hopenfeld admitted that he has no evidence that the feedwater nozzle is cracked. Finding 83. He postulated the potential existence of cracks in the cladding that partially surrounds the base metal of the nozzle as points of stress initiation. Id. Again, he provided no evidence that such cracks exist and did not dispute that, if cracks were to develop in the base metal at those points, they would be detected as part of the plant Section XI program and repaired or otherwise addressed as necessary. Id.

### d. Heat Transfer Equations

NEC asserts that the heat transfer equations used by Entergy in its CUF calculations are wrong. In particular, Dr. Hopenfeld points out that the heat transfer coefficients and the resulting stresses on the component are dependent on the flow velocity. Finding 84. He describes the Entergy analyses as erroneously assuming a uniform velocity, both circumferentially and axially, along the surface of a component such as a nozzle. Id. Had Entergy computed the local velocity along and circumferentially around the surface of the nozzle, argues Dr. Hopenfeld, significantly higher stresses would have been obtained. Id.

In its reanalysis using the Green's Function methodology, Entergy applied bounding heat transfer coefficients at every region of the component where there was a change in flow velocity (e.g., a change in the diameter of a pipe). Finding 85. For each such location, the stress computation used the highest applicable flow rate to compute the heat transfer coefficient. Id. This procedure resulted in conservatively high heat transfer coefficients that maximized the estimated stresses on the component. Id. For Entergy's confirmatory analysis, in which each transient was modeled separately, it was possible to vary the temperature and flow rate throughout the transient, and this procedure was followed. Finding 86.

Neither model computes variations in heat transfer coefficient azimuthally around a pipe. Finding 87. However, no such variations are possible under the conditions that exist at VY, i.e., where fully developed, high flow rates do not allow for temperature variations around a pipe. Id.

Dr. Hopenfeld argued that the length of piping between an elbow and the entrance to the feedwater nozzle, 48 inches, is insufficient to allow for fully developed turbulent flow to exist which would justify the assumption of azimuthally uniform temperatures around the pipe. He opined that a minimum of 12 to 40 pipe diameter lengths of straight piping would be required in order for turbulent flow to be assured. Finding 88. Entergy responded that, for high flow lines, the effects of an entrance are minimized so that fully developed flow exists. Id. This is demonstrated by one of the exhibits to NEC's testimony, NEC Exh. NEC-JH_29 at 212, Fig. 8-9, which shows that for high flow velocities entrance effects are minimized. Id.

NEC raises other challenges to the heat transfer equations utilized in Entergy's CUF calculations. Finding 89. Entergy's testimony, however, argues persuasively that the claims are unsupported or technically unsound. Id. Since NEC witness Dr. Hopenfeld cannot tell whether the heat transfer coefficients that express the heat transfer from the coolant to the surface of a component yielded by the equations should be larger or smaller, those challenges can be disregarded as inconsequential. Id.

#### e. Number of Transients

To insure a realistic projection for the thermal transient cycles and events expected for 60 years of operation, Entergy used the Thermal Cycle Diagrams from a later vintage BWR as a starting point of its transient estimate. Finding 90. The VY Design Specification transients were mapped onto the BWR Transient Diagrams. Then, projections for 60 years were made based on the numbers for 40 years in the VY Design Specification, the numbers actually analyzed in the VY Design Stress Report, and the number of cycles experienced by VY in approximately 35 years of operation. Id. For example, 200 Startup / Shutdown cycles were included in the original VY Design Specification. However, 300 Startup / Shutdown cycles were conservatively used in the EAF analyses for 60 years of operation. Id. To date, only 92 Startup/Shutdown cycles have taken place. Id.

The 60 year number of operating events used in the VY EAF analyses are equal or greater in number to the numbers of cycles in the original VY Design Specification for 40 years. Actual plant cycle counts to date, projected out to 60 years of operation, will not exceed the number of cycles used in the Entergy analyses. Finding 91. In addition, the rates of temperature change during actual transients are less severe than those for the design transients analyzed, so that the actual fatigue usage that is experienced by the components is less than that calculated for a design transient. Id. Finally, bounding EPU conditions were used for all transient definitions and numbers of cycles, even though EPU operation did not apply to the first 35 years of plant operation. Therefore, there is significant margin incorporated in the number of transient cycles used in the EAF analysis, and their severity and the projected numbers of cycles used in the EAF analyses are conservative. Id.

Dr. Hopenfeld postulated, based on judgment, that the number of estimated transients used by Entergy in its $CUF_{en}$ calculations should be increased by 20% to account for the increased number of transients that will result from the 2006 power uprate. Finding 92. In reality, the numbers of transient cycles used in Entergy's CUF calculations are conservative projections of the numbers of cycles actually experienced by the plant over its operating history and the margin on the numbers of cycles analyzed over a linear extrapolation of those experienced significantly exceeds the factor of 1.2 suggested by NEC. Id.

At the Board's request, Entergy provided an up-to-date listing of those transients, having a potential impact on EAF, that have been actually experienced at VY since the beginning of plant operations. The list, which is organized by transient type, compares the actual number of transients thus far to the number assumed in Entergy's EAF analyses through the end of the license renewal period. Finding 93. The list shows that the number of actual transients experienced to date is one third or less than the number of transients used in the $CUF_{en}$ calculations. Accordingly, the number of transients used in Entergy's reanalysis and confirmatory analyses is conservatively high and unlikely to be exceeded during the license renewal period. Id.

In addition, Entergy has committed to monitor the transient count throughout the period of extended operation to verify that these assumptions remain valid, and will take appropriate corrective action if these assumptions do not remain valid. Finding 94. The NRC Staff has determined that the Fatigue Monitoring Program will ensure that the predicted number of transients is not exceeded. Id.

### f. Use of Green's Function Methodology

The use of the Green's Function methodology and associated simplifications in the reanalysis resulted in a 40% lower value of the CUF for the feedwater nozzle than the CUF for the nozzle computed in the confirmatory analysis. Finding 95. However, the actual difference is small and the difference in CUF due to changes in the Green's Function is also very small. Id. Therefore, use of the Green's Function methodology would not result in a substantial underestimation of the CUF. Also, the confirmatory analysis CUF for the safe end of the same nozzle was 60% lower than that calculated in the reanalysis. Id.

### g. Lack of Error Analysis

NEC criticizes Entergy for the failure to perform an error analysis to show the admissible range for each variable included in the analysis. Finding 96. In reality, performing an error analysis on the stress results is not traditional practice in analyses of this type, and is unnecessary given that input parameters (such as temperature, pressure, and heat transfer coefficients) were selected so as to maximize stresses and give a conservative, bounding estimate. Id. Any error analysis would give a lower stress estimate. Id.

### h. Analysis of Two Other Nozzles

NEC asserts that the confirmatory calculation for the feedwater nozzle does not bound the other two nozzles (the recirculation outlet nozzle and the core spray nozzle). Finding 97. However, as discussed earlier, the feedwater nozzle is the controlling nozzle because it experiences the most severe design transients and because it is the location where the relatively colder feedwater returns to the hot reactor vessel, thereby causing the most severe thermal stresses. Id. It is extremely improbable that either of the other two nozzles would experience $CUF_{en}$s in excess of that experienced by the feedwater nozzle. Id. This result was demonstrated in Entergy's reanalyses, where all three nozzles were evaluated on a consistent basis.

Performance of the confirmatory analysis of the other two nozzles will only serve to demonstrate, through obtaining similar results to those obtained for the feedwater nozzle, that the reanalysis results are conservative. Finding 98. Therefore, there is no technical basis requiring that these analyses be performed before the VY license renewal is approved. Id.

**i. Dr. Hopenfeld's $CUF_{en}$ Recalculation**

Dr. Hopenfeld performed a recalculation of the $CUF_{en}$s for the nine locations of interest at VY. Finding 99. Dr. Hopenfeld calculates irrelevant CUFs by using indefensible methodologies, including using $F_{en}$ values (17 for carbon steel and 12 for stainless steel) quoted in the abstract of NUREG/CR-6909 as potentially applicable "under certain environmental loading conditions" without determining whether they are applicable to VY conditions. Id. In fact, the conditions that would cause a multiplier of 17 to exist are primarily associated with high temperature and high dissolved oxygen content for carbon and low alloy steels; those conditions do not exist at VY because the plant is operating using hydrogen water chemistry. Id.

Dr. Hopenfeld also applies the CUF values from Table 4.3-3 of the Application, even though some of those are generic values and not VY-specific. Finding 100. As a result, the results of his recalculation are inapplicable to VY. Id.

Because Dr. Hopenfeld's recalculation is based on unrealistically conservative assumptions, it yields nonsensical results. Finding 101. For example, it predicts a 13.77 $CUF_{en}$ for one of the components, meaning that the component should have developed cracking after only four years in service, a result at odds with VY's over thirty years of operating experience without cracks. Id.

4. Conclusions on Contentions 2A and 2B

The evidence shows that Entergy has performed a more refined reanalysis, and a confirmatory EAF analyses at limiting piping and vessel locations for VY. These analyses, performed using conservative methodologies and bounding input parameters, have demonstrated that the $CUF_{en}$s are less than 1.0 for the sixty years of plant operation encompassed by the renewed VY operating license. These analyses collectively demonstrate that the critical VY components will not experience failure due to fatigue during the period of extended operation. For that reason, there is no support for the claims made in NEC Contentions 2A and 2B, which should be rejected.

## III. NEC CONTENTION 3

### A. Background

In its Petition to Intervene, NEC had alleged that Entergy's program for aging management of the steam dryer is not adequate to detect crack initiation and growth because it is not based on actual stress measurements, but instead relies on theoretical calculations of two computer models, the Computation Fluid Dynamic Model ("CFD") and the Acoustic Circuit Model ("ACM") – that have not been properly benchmarked and are subject to large uncertainties. NEC Petition to Intervene at 17; Declaration of Dr. Joram Hopenfeld (May 12, 2006) ("Hopenfeld First Declaration"), ¶ 19. In his affidavit in support of the Petition to Intervene, Dr. Hopenfeld also contended that "the status of the dryer cracks must be continuously monitored and assessed by a competent engineer." Hopenfeld's First Declaration, ¶ 18.

On April 19, 2007, Entergy filed a motion for summary disposition of NEC Contention 3. Entergy asserted, inter alia, that NEC's challenge to the CFD and ACM models was irrelevant because Entergy's aging monitoring program for the steam dryer "neither requires the use of computer models nor relies on the result of analyses using those models." Entergy's Motion for Summary Disposition of New England Coalition's Contention 3 (Steam Dryer) (April 19, 2008) at 6. On May 9, 2007, NEC filed an "Opposition to Entergy's Motion for Summary Disposition of NEC's Contention 3 (Steam Dryer), contending that the facts concerning Entergy's use of the ACM and CFD models and the validity of these models are still in genuine dispute and that if these models are not used, an aging management program consisting solely of visual inspection and parameter monitoring would not be sufficient to ensure the dryer's structural integrity.

In its "Memorandum and Order (Ruling on Motion for Summary Disposition of NEC Contention 3)" (September 11, 2007) ("September 11, 2007 Order"), the Board granted in part and denied in part Entergy's summary disposition Motion.[17] The Board granted Entergy's Motion as it related to NEC's claims of insufficient benchmarking of the CFD and ACM models and reliance upon them to assure safe operation of the steam dryer during the license renewal period. September 11, 2007 Order at 3. The Board so ruled because Entergy "flatly represented to the Board that CFD and ACM models will not be used or relied upon [to monitor potential steam dryer cracking] and that the steam dryer will be continuously monitored." Id. at 3. The Board denied Entergy's Motion as it related to NEC's assertion "that the status of the steam dryer must be continuously monitored and assessed by a competent engineer" because Entergy did not provide information regarding the qualifications of the personnel performing the monitoring. Id. at 11. The Board also denied Entergy's Motion as it related to asserted inadequacies in "Entergy's assessment of the monitoring data collected from the aging management program for the steam dryer" and failure to include some form of stress load analysis in its program. Id. 13-14.

As a result of the Board's rulings, the two remaining issues on NEC Contention 3 as to which a hearing needed to be held were: (1) whether Entergy has established sound evaluation and implementation procedures to assure that the integrity of the steam dryer is not jeopardized, including continuous monitoring and assessment by a competent engineer; and (2) whether a steam dryer aging management program that does not provide a means to estimate and predict stress loads on the dryer during operation for comparison to established fatigue limits is valid.

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[17] On June 19, 2007, NEC filed a motion requesting that Board withhold its decision on Entergy's motion for summary disposition to allow its expert witnesses to review Entergy's May 2007 steam dryer inspection report and, if necessary, to file a supplement to its answer opposing Entergy's motion. The Board granted this motion on July 13, 2007, and NEC filed a supplement to its opposition to Entergy's motion on July 19, 2007.

**B. Witnesses**

1. Entergy Witnesses

Entergy's testimony on NEC Contention 3 will be presented by a panel of two experts, each with extensive experience in monitoring the performance of the VY steam dryer and developing the plan for continued management of the steam dryer after renewal of the VY operating license. The first witness on the panel, Mr. John R. Hoffman, was until his retirement in September 2006, employed by Entergy as the Project Manager for the License Renewal Project at VY. Testimony of John R. Hoffman and Larry D. Lukens on NEC Contention 3 – Steam Dryer, attached to the Joint Declaration of John R. Hoffman and Larry D. Lukens on NEC Contention 3 – Steam Dryer, Entergy NEC 3 Dir. at A2. He has over 37 years of nuclear power engineering experience, and has been associated with VY since 1971. Id. at A2, A3. As Project Manager for the License Renewal Project at VY, he had the responsibility to ensure that all aspects of the license renewal application, including the steam dryer aging management program, were properly developed and were reviewed by the respective subject matter experts at VY. Id. at A5. Mr. Hoffman has a B.E. Degree in Mechanical Engineering from the Cooper Union for the Advancement of Science and Art in 1967, an M.S. Degree in Nuclear Engineering from the University of Lowell in 1977, and an M.S. Degree in Applied Management from Lesley College in 1985. Id. at A3.

The second witness in Entergy's panel is Mr. Larry D. Lukens. Prior to his retirement in July 2007, Mr. Lukens was employed by Entergy and had, among other responsibilities, that of Supervisor, Code Programs at VY. Entergy NEC 3 Dir. at A7. In that position, his responsibilities entailed ensuring that the activities required by industry codes, particularly those issued by ASME, that are applicable to VY and are the responsibility of Engineering were completed, evaluated, dispositioned, and documented. The required activities included, for example, those described by the ASME Operation and Maintenance Code for testing pumps and valves; the ASME Boiler & Pressure Vessel ("BPV") Code for inservice inspection ("ISI"), including containment inservice inspections; the primary containment integrity monitoring program described by 10 C.F.R.50, Appendix J; and the reactor vessel and internals management and monitoring program under the EPRI BWR Vessel & Internals Program ("BWRVIP"), an industry initiative implemented with the concurrence and participation of the NRC. Id. He was directly involved with the License Renewal audits and inspections of Code Programs activities including the inservice testing ("IST"), ISI, Containment ISI, Appendix J, and BWRVIP, and with the Fire Protection programs, and approved the VY License Renewal commitments relating to these programs. Id. With respect to the steam dryer, Mr. Lukens was responsible for ensuring the proper completion and evaluation of the steam dryer inspections conducted during the 2005 and 2007 refueling outages. He was also responsible for overseeing the license renewal aging management program as it applied to the steam dryer. Id. at A10. Mr. Lukens received a B.S. Degree in Nuclear Engineering from the University of Wisconsin, Madison, in 1978. He has over 38 years of nuclear power work experience, including being a qualified reactor operator in the U.S. Navy and an NRC licensed operator at the University of Wisconsin, and nearly 10 years of service as Program Manager for ASME Section XI inservice testing, inservice pressure testing, and containment leak rate testing at an operating nuclear power plant. Id. at A8.

Entergy witness Mr. Hoffman is qualified as an expert on Entergy's licensing commitments on the aging management of the VY steam dryer during the license renewal period, the program to be implemented at VY to manage steam dryer performance after license renewal, the mechanisms for crack formation in steam dryers, and the potential for steam dryer fatigue failure at VY during the period of extended operations.

Entergy witness Mr. Lukens is qualified as an expert on Entergy's licensing commitments on the aging management of the VY steam dryer during the license renewal period, the aging management of the VY steam dryer during the license renewal period, the steam dryer inspections performed to date at VY and their results, and the dryer inspection program to be implemented at VY after license renewal.

The testimony and opinions of the Entergy witnesses on NEC Contention 3 are based on both their technical expertise and experience and their first hand knowledge of the factual issues raised in the contention.

2. NRC Staff Witnesses

The NRC Staff provided testimony on NEC Contention 3 through a panel of three witnesses: Messrs. Kaihwa R. Hsu, Jonathan G. Rowley, and Thomas G. Scarbrough. Mr. Hsu is currently a senior mechanical engineer in the Division of Engineering in the Office of New Reactors ("NRO"). Statement of Professional Qualifications of Kaihwa R. Hsu (Staff NEC 3 Dir., following p. 14). Previously, he was a materials engineer in NRR's Division of License Renewal. Id. Mr. Hsu has 27 years of experience in the nuclear industry. Id. As part of his official duties, Mr. Hsu served as a technical lead for the license renewal safety audit at Vermont Yankee. Mr. Hsu is qualified as an expert witness on the subjects of the Staff's review of the aging management program for the VY steam dryer and the Staff's interpretation of the adequacy of the VY steam dryer inspection and monitoring program during VY's operation after license renewal.

Mr. Rowley has over fourteen years of experience in materials science and engineering. Statement of Professional Qualifications of Jonathan G. Rowley (Staff NEC 3 Dir., following p. 14). Mr. Rowley has been responsible for coordinating the Staff's review of Entergy's Application and the Staff's preparation of its "Safety Evaluation Report with Confirmatory Items Related to the License Renewal of Vermont Yankee Nuclear Power Station," dated March 2007 (ML070870378) and the FSER. Id. In addition to his involvement the Staff's review of Entergy's Application, Mr. Rowley was involved in the Staff's review of the license renewal applications for the D.C. Cook and R.E. Ginna plants. Mr. Rowley is qualified as an expert witness on the subjects of Entergy's Application and its review by the NRC Staff, NRC regulatory requirements and guidance pertaining to license renewal applications, and the bases for Staff approval of Entergy's Application as it relates to the aging management program for the steam dryer.

Mr. Scarbrough has 30 years of experience of technical experience in the field of nuclear engineering. Statement of Professional Qualifications of Thomas G. Scarbrough (Staff NEC 3 Dir., following p. 14). In the course of his career he served as Special Technical Advisor to the Atomic Safety and Licensing Appeal Panel ("ASLAP") for the restart of the Three Mile Island ("TMI") Unit 1 nuclear power plant and, later, was appointed as Technical Advisor to the ASLAP. Id. Following the failure of the steam dryer at Quad Cities Unit 2 in 2002, Mr. Scarbrough participated in the Staff's review of potential adverse flow effects on plant components during power uprate operation. Id. He worked on the power uprate license amendments for Vermont Yankee, Browns Ferry, Hope Creek, and Susquehanna. Id. Since February 2007, Mr. Scarbrough has worked in the Component Integrity, Performance, and Testing Branch II in NRO where he reviews component issues for proposed new reactors, and provides assistance to NRR on potential adverse flow effects for power uprates at operating nuclear power plants. Id. Mr. Scarbrough is qualified as an expert witness on the subjects of the operational performance of steam dryers in BWRs, the causes and consequences of steam dryer cracking, the impacts of steam dryer failure on plant safety, and the actions taken by Entergy at VY to modify its steam dryer prior to EPU implementation and the testing and monitoring of steam dryer performance during VY's power ascension to EPU operation.

3. NEC Witness

NEC's witness on this Contention, Dr. Joram Hopenfeld, has provided no indication that he has experience in the analysis or evaluation of BWR steam dryer performance or the issues associated with the structural integrity of steam dryers during normal plant operations. See "Curriculum Vitae for Dr. Joram (Joe) Hopenfeld," NEC Exhibit NEC-JH_02. Even though the need to estimate and measure stress loads on the dryer was the main thesis of Dr. Hopenfeld's testimony, he could not propose any method for performing either task nor could he specify for how long the loads on the dryer would need to be measured. Tr. at 1356-57, 1385 (Hopenfeld).

### C. Applicable Legal Standards

The applicable legal standards for the approval of VY's aging management program for the steam dryer is 10 C.F.R. §§ 54.21(a)(3) and 54.29(a), i.e. whether there is reasonable assurance that the aging on the steam dryer will be adequately managed so that the intended functions of the steam dryer will be maintained in accordance with the current licensing basis for the period of extended operation. 10 C.F.R. § 54.29(a). See also 60 Fed. Reg. at 22,479 (". . . the [license renewal] process is not intended to demonstrate absolute assurance that structures or components will not fail, but rather that there is reasonable assurance that they will perform such that the intended functions . . . are maintained consistent with the CLB").

As set forth in § 54.29(a), this reasonable assurance determination must be made against the current licensing basis of the plant, which is to be maintained during the license renewal period. As stated in 10 C.F.R. § 54.33(d):

> The licensing basis for the renewed license includes the CLB, as defined in § 54.3(a); the inclusion in the licensing basis of matters such as licensee commitments does not change the legal status of those matters unless specifically so ordered pursuant to paragraphs (b) or (c) of this section.

10 C.F.R. § 54.33(d). The licensing basis for a nuclear power plant during the renewal term will consist of the current licensing basis and new commitments to monitor, manage, and correct age-related degradation unique to license renewal, as appropriate. The current licensing basis includes all applicable NRC requirements and licensee commitments. Requirements for Renewal of Operating Licenses for Nuclear Power Plants, 56 Fed. Reg. 64,943, 64,946 (Dec. 13, 1991).

## D. Summary of the Evidence

### 1. Background

The steam dryer is a BWR stainless steel component, installed in the reactor vessel above the steam separator assembly, and supported by brackets welded to the inside of the vessel wall below the steam outlet nozzles, whose function is to remove moisture from the steam before it leaves the reactor. Finding 105. During plant operations, wet steam flows upward and outward through the dryer. Moisture is removed by impinging on the dryer vanes and flows down through drains to the reactor water in the downcomer annulus below the steam separators. Id.

The VY steam dryer is a non-safety-related, non-Seismic Category I component. Finding 106. Although the steam dryer is not a safety-related component, the assembly is designed to withstand design basis events without the generation of loose parts and the dryer is designed to maintain its structural integrity through all plant operating conditions. Id.

In 2002, steam dryer cracking and damage to components and supports for the main steam and feedwater lines were observed at the Quad Cities Unit 2 nuclear power plant, and it was discovered that loose parts shed by the dryer due to metal fatigue failure had damaged the supports. Finding 107. The cause of the dryer failure was determined to be high-cycle fatigue, which results from relatively low stresses over a large number of cycles, as opposed to high stresses over a low number of cycles. Id. Quad Cities had recently implemented a power uprate. Id.

Another steam dryer event occurred in Quad Cities Unit 1 in 2003. Finding 108. In both cases, the steam dryer failure was accompanied by a significant increase in measured moisture carryover in the steam leaving the reactor. Id. There was also a measurable change in the distribution of flow in the steam lines. Id.

The Quad Cities experience raised a concern that a loss of physical integrity of the dryer, resulting in loose dryer sections or parts being released to the reactor steam space could potentially migrate to other components and could have adverse impact on safety-related equipment.

Finding 110. While the formation of cracks on the surface of a steam dryer is not in itself cause for concern, the existence of those cracks needs to be identified and evaluated before the cracks progress to the point where they could cause a loss of physical integrity of the dryer, resulting in loose parts. Finding 111.

2. VY's response to industry experience

In response to the Quad Cities 2 event, Entergy substantially modified the steam dryer at VY during the Spring 2004 refueling outage to improve its capability to withstand the higher flow induced vibration loadings that could result from operation of the plant at EPU levels. Finding 112. The changes included replacing portions of the dryer with new ones made more robust. Id.

VY also instituted a program of dryer monitoring and inspections to provide assurance that the flow-induced loadings under operation at EPU levels did not result in the formation or propagation of cracks on the dryer. Finding 113. The program was reviewed and approved by the NRC and included as a license condition as part of the power uprate license amendment issued on March 2, 2006. Id.

As power was increased in 2006 from the original licensed power level to full extended power uprate conditions, there was continuous monitoring of plant parameters indicative of dryer performance, including measurement at least once per week of moisture carryover and periodic measurement of main steam line pressure. Finding 114. Following completion of EPU power ascension testing, moisture carryover measurements have continued to be made weekly, and other plant operational parameters that would be symptomatic of loss of steam dryer structural integrity (main steam line flow, reactor vessel water level, steam dome pressure) have continued to be monitored and their values trended. Id. This monitoring program will continue to be implemented during the period of extended operation after renewal of the VY license. Id.

In addition, the VY steam dryer was inspected during plant refueling outages in the Fall of 2005 (before completion of the EPU) and Spring of 2007 (after one year of operation at EPU power levels). Finding 115. The dryer is scheduled to be inspected again during the refueling outages in the Fall of 2008 and the Spring of 2010, with a partial inspection scheduled for the Fall of 2011. Id. Inspections will continue in the license renewal period starting with the first refueling after March 2012. Id.

### 3. VY's Steam Dryer Aging Management Program

VY's current licensing basis with respect to the steam dryer is set forth in Attachment 6 to Supplement 33 (dated September 14, 2005) of its EPU license amendment application. Finding 116. The commitments made in Supplement 33 to the EPU license amendment application have been made part of the current VY license. Finding 117. Those licensing commitments include the program of parameter monitoring and dryer inspections recommended in GE-SIL-644. Id.

The CLB for the steam dryer also includes some of the provisions of BWRVIP-139, "Steam Dryer Inspection and Flaw Evaluation Guidelines," which was issued by EPRI in 2005 and incorporated into VY's CLB as part of the requirements for dryer inspection. Finding 118.

In its License Renewal Application, Entergy commits VY to maintaining its CLB with respect to the steam dryer by continuing to implement the guidance in GE-SIL-644 until BWRVIP-139 is approved by the NRC Staff, at which point VY will either include the recommendations in the approved version of BWRVIP-139 in the VY BWR Vessel Internals Program or inform the Staff of VY's exceptions to that document. Finding 120.

If BWRVIP-139 is not approved by the Staff, VY has a license renewal commitment to "[c]ontinue inspections in accordance with the steam dryer monitoring plan, Revision 3 in the event that the BWRVIP-139 is not approved prior to the period of extended operation." Finding

121. These commitments bind Entergy to continue to implement the guidance in GE-SIL-644 through the license renewal period. Finding 122.

A contingency could develop when the NRC takes final action on whether and in what form to approve BWRVIP-139. Finding 123. If the Staff approves BWRVIP-139 with modifications, such approval would cause Entergy to evaluate the changes and determine whether VY can implement them. Id. If Entergy can implement the changes, the VY CLB would be modified by incorporating a revised version of BWRVIP-139; otherwise, the CLB as it exists today will remain unmodified through the license renewal period. Id.

Since a modification of the CLB is not currently being proposed, it is not part of the Application before the Board. The Board must proceed by evaluating the adequacy of the steam dryer management plan, which is based on continuing to implement the guidance in GE-SIL-644 during the license renewal period. Finding 124.

GE-SIL-644 recommends that BWR licensees institute a program for the long term monitoring and inspection of their steam dryers. It provides detailed inspection and monitoring guidelines (Exhibit E3-06, Appendices C and D). Finding 125.

The proposed VY steam dryer management program conforms to the guidance in the GALL Report (NUREG-1801), which was confirmed by the NRC Staff in its Final Safety Evaluation Report for the VY license renewal. Finding 126.

**a. Dryer Monitoring**

The monitoring component of the VY steam dryer management program consists of assessing the status of the steam dryer continuously by the plant operators and VY's technical staff through the monitoring of certain plant parameters. Finding 127.

Three parameters (main steam line flow, reactor vessel water level, and steam dome pressure) are measured continuously in the control room. Finding 128. In addition, weekly measurements of moisture carryover are performed. Id. The frequency of moisture carryover measurement has been increased after the uprate. Id.

If any changes in the other parameters reaching the limits set in VY Procedure ON-3178 are detected, an immediate measurement of moisture carryover is taken. Finding 129. Changes in moisture carryover are evaluated in accordance with the requirements of GE-SIL-644 to determine whether significant cracking has occurred. Id. Abnormal values of the monitored plant parameters would indicate that the steam leaving the reactor has a high moisture content, which in turn could indicate that steam is escaping through a crack in the dryer. Such escape would be symptomatic of a significant crack that might result in loss of physical integrity of the dryer. Finding 130.

Moisture carryover is measured by plant chemistry personnel using VY Procedure OP-0631. Finding 131. If moisture carryover is determined to be greater than the limit stated in the procedure, a Condition Report is to be written, the Shift Manager notified, and actions taken in accordance with Off-Normal Procedure ON-3178. Id. An engineering operability evaluation in accordance with EN-OP-104, "Operability Determinations," is also performed. Finding 132.

Experienced qualified engineering personnel will determine the significance of the abnormal moisture carryover measurement. Finding 133. The evaluation will be performed immediately upon determination that unexplained changes in the operating parameters had occurred. Id. If the engineering evaluation of plant data confirms that steam dryer damage may have occurred, a plant shutdown is initiated such that the plant is brought to a cold shutdown condition within 24 hours. Finding 134.

The personnel involved in determining the significance of the moisture carryover and other measured parameters are required to be qualified in the application of the operability determination procedure. Finding 135. A prerequisite for procedure qualification is the requirement that the individual(s) be enrolled in the Engineering Support Personnel ("ESP") training program and that their capability to perform independent engineering work be assessed by their supervisor. If an engineer or his supervisor feels the engineer needs additional training to maintain or enhance his level of expertise, that training is incorporated into the performance goals for the year. Id.

All engineering personnel at VY must be qualified through a prescribed Institute of Nuclear Power Operations ("INPO") ESP Training Program that prescribes the training methodology, the kind of training they need, and the experience they need before they can work independently. Finding 136. In addition to having that training, they need to have their supervisor certify that they have properly completed the training, and that they have performed the work under the guidance of someone else, and that the supervisor is satisfied with the level of the work that was performed. Id. The training program is audited by INPO, which conducts a thorough detailed assessment of the training program to ensure that it results in personnel being qualified to do perform their duties. Id.

The NRC Staff has reviewed the qualifications of the Entergy personnel who monitor and evaluate the plant parameter information, and has concluded that Entergy is capable of analyzing plant data related to steam dryer performance in an adequate manner. Finding 137. NEC has provided no testimony or exhibits challenging the qualifications of the Entergy personnel who monitor and evaluate plant parameters relating to the status of the dryer. Id.

The purpose of the measurements of plant parameters is not to enable Entergy to determine whether a dryer crack is about to form, but to provide early warning to the plant personnel that a crack may have developed so that appropriate, timely action may be taken before undesirable effects ensue as a result of the crack. Finding 138. This approach is consistent with the guidance in GE-SIL-644, which while noting that monitoring of steam moisture content and other parameters does not consistently predict imminent dryer failure, monitoring "does allow identification of a degraded dryer allowing appropriate action to be taken to minimize the damage to the dryer and the potential generation of loose parts." Id. All parties agreed that this is a correct statement as to the usefulness of a dryer monitoring program. Id.

A crack that developed sufficiently to lead to detectable amounts of steam bypassing of the dryer would be a through penetration crack, several inches in length, and wide enough for steam to leak out of the crack. Finding 139. The cracks that were identified at Quad Cities Unit 1, which were several inches wide and several feet in length, were accompanied by changes in moisture carryover and steam line flow distribution, whose detection led to the plant being quickly shut down. Id.

While there is no technology that will predict when a crack will initiate, the monitoring programs at VY ensure that any steam dryer cracks that develop are detected before they grow to a size that would be of concern. Finding 140. The monitoring program looks at operating parameters to identify potential indications of a breach in the steam dryer, regardless of the mechanism that causes the breach, and is therefore results-driven and not cause-driven. Id.

If a crack were to develop in the VY steam dryer and grew until was sufficiently large to be detected by the monitored plant parameters, it still would grow slowly enough to allow the plant to be shut down before it resulted in dryer failure and the generation of loose parts. Finding 141. The slow growth of the cracks found at VY is demonstrated by inspection data, and is consistent with the ductile nature of stainless steel, that inhibits fast-growing brittle fractures. Id.

Dr. Hopenfeld disagrees that the steam dryer monitoring program implemented at VY is sufficient to identify cracks on the dryer before they lead to structural failure of the dryer. Finding 142. However, he has cited no credible evidence that the VY steam dryer monitoring program would not be able to detect steam dryer cracks in time to allow for measures to be taken in response to their detection. Id.

### b. Dryer Inspections

Dryer inspections are performed during plant refueling outages. Finding 143. It is only feasible to inspect the steam dryer when the plant is shut down, because the inspection requires removing the reactor vessel head to take the steam dryer out. Id. The VY dryer inspections are to be performed in accordance with the VY BWRVIP Program Plan, VY-RPT-06-00006, which references GE-SIL-644, Revision 1 and BWRVIP-139. Finding 144.

The dryer examinations have consisted of VT-1 and (in earlier times) VT-3 examinations of accessible internal and external welds and plates in the steam dryer potentially susceptible to crack formation. Finding 145. VT-1 and VT-3 examinations are defined by ASME Boiler & Pressure Vessel ("BPV") Code Section XI, and the non-destructive examination technicians who perform and review these examinations are qualified in accordance with ASME BPV Section XI. Id. A VT-1 visual examination under BWRVIP standards (such as the steam dryer inspections) is one capable of achieving a resolution of 0.044 inch, slightly larger than the micro engraving on a dollar bill. Finding 146.

At VY, all locations in the steam dryer where fatigue cracks could develop are accessible for inspection, and are inspected. Finding 147. The inspections are performed by qualified non-destructive examination ("NDE") inspection personnel, using qualified NDE techniques appropriate for BWR steam dryer inspections. Finding 148. Because of the large number of individual examinations to be performed during a refueling outage, this work is typically contracted out to qualified reactor inspection specialists, including the reactor supplier (General Electric). Id. Entergy, however, specifies the scope of the inspections and the areas to be examined. Id.

The inspection data are reviewed by qualified Level III NDE personnel and are subject to final acceptance by Entergy Level III NDE personnel. Finding 150. The examinations also be reviewed by an Entergy Level III NDE technician, and Entergy Level III review and approval is required to be completed on 100% of the steam dryer examinations prior to the dryer's return to service. Id.

All detected indications (imperfections or unintentional discontinuities that may or may not be cracks) are evaluated by qualified structural engineers, who are experienced with BWR steam dryer crack evaluation. Finding 151. Typically, these indications are evaluated by engineers who are on the staff of the reactor vendor, and the evaluations and conclusions are reviewed and accepted by qualified Entergy structural engineers. Id.

An indication is classified as "recordable" or relevant if it is visible to the resolution of the examination technique. Finding 152. All recordable indications that are confirmed by the Level III NDE technician are evaluated by Engineering to determine whether or not they are "rejectable." Rejectable indications are those that must be repaired prior to restarting the plant. Repair of rejectable indications is an ASME BPV Code Section XI requirement. Id.

If the characteristics of a particular indication do not rule out fatigue, the indication is classified as a potential fatigue indication. Finding 153. The indication is tracked and examined in subsequent dryer inspections to determine whether it is growing. Id. All recordable indications are required to be reinspected at each refueling outage until at least two consecutive inspections show no growth. Id. However, as a matter of practice, Entergy is continuing to inspect all previously identified indications. Id.

During the Spring 2004 refueling outage, in preparation for EPU, the dryer received a baseline VT-1 inspection of all accessible areas deemed potentially susceptible to crack formation. These examinations comprised 287 weld and plate examinations. A total of 20 indications were identified, of which 2 were weld-repaired, and 18 were determined acceptable to use as-is. Finding 154.

The steam dryer inspections performed during the Fall 2005 outage examined all high-stress areas, as identified in GE-SIL-644. In addition, all areas that had been repaired or modified in the Spring 2004 outage were reinspected, as well as those indications that were found and evaluated to be acceptable for use as-is during the Spring 2004 outage. These examinations comprised 113 internal and external weld examinations. A total of 66 indications were identified, including previously identified indications and repaired areas from 2004, all of which were found acceptable for use as-is. Finding 155. None of the indications found in 2004 had grown. Id.

During the Spring 2007 outage, all accessible susceptible areas of the steam dryer were inspected. The previously repaired areas, the identified high stress areas, as well as those indications that were previously found and evaluated to be acceptable for use as-is were also examined. Finding 156. Approximately 448 individually identified steam dryer examinations were performed. A total of 66 indications were recorded, including 48 of those identified in 2005 and 27 new indications. Id. These previously unidentified indications were the result of the increased examination scope in 2007 compared to that in 2005, and the fact that all accessible susceptible areas of the steam dryer had been subjected to the improved resolution VT-1 examination. No growth was noted in the previously identified indications. All the indications identified in 2007 were accepted for use as-is. Id.

The steam dryer inspections conducted in the Spring of 2007 followed approximately one year of full power operation at the EPU condition. Finding 157. The examinations were conducted using enhanced examination resolution, which provides improved detection levels over those achievable by using the prescribed VT-1 examination process. Id. Each of the indications found in 2007 was evaluated by qualified structural engineers, experienced in evaluating indications in BWR steam dryers. Id. None of the cracks were determined to be associated with fatigue. Id.

Entergy presented the results of the 2007 steam dryer inspection to the NRC Staff and discussed them with the Staff. The Staff had no concerns about potential fatigue issues. Finding 158.

It is possible to determine by visual inspection whether a crack is due to intergranular stress corrosion cracking ("IGSCC") or is a fatigue crack. Finding 159. IGSCC typically occurs in the heat affected zone adjacent to a weld. Id. IGSCC develops between grain boundaries, giving the crack a jagged appearance. Id. By contrast, a fatigue crack is tends to develop as a straight line. Id. IGSCC develops in low stress areas, whereas fatigue occurs in highly stressed areas. Therefore, the location, shape, and stress level permit differentiating whether a crack is due to IGSCC or fatigue. Id.

All cracks identified in the VY steam dryer in the 2004, 2005 and 2007 inspections are the result of IGSCC, and are of a type that grows slowly if at all. Finding 160. At VY, IGSCC crack growth is arrested by the use of hydrogen injection and noble metal application on the feedwater. Id.

VY has not identified any steam dryer cracks that are consistent with fatigue, and this conclusion is supported by the fact that the identified indications have not grown during subsequent operating cycles. Finding 161.

While VY's monitoring and inspection procedures are not specifically cited in the aging management program for the steam dryer, they implement the program. Finding 162. The procedures cite the program and provide a direct link to it, so there is no possibility that their use may be inadvertently discontinued. Id.

4. Issues Raised in the NEC Testimony

Through Dr. Hopenfeld's steam dryer testimony, NEC raises two main claims regarding the VY steam dryer management program: (1) the monitoring of plant parameters indicative of potential dryer cracks is insufficient to prevent fatigue cracks from forming and propagating in the period between dryer inspections; and (2) a dryer management program must include estimating the stress loadings on the dryer and ensuring that they remain within the stress limits of the dryer material. In addition, Dr. Hopenfeld states in his rebuttal testimony that cracks occurring through IGSCC could lead to high-cycle fatigue, and that the steam dryer parameter monitoring and inspection program fails to comply with the General Design Criteria in Appendix A to

10 C.F.R. Part 50 "insofar as they require that protection must be provided against the dynamic effects of loss of coolant accidents ('LOCAs')."

**a. Adequacy of Plant Parameter Monitoring**

With respect to Dr. Hopenfeld statement that "[m]oisture monitoring only indicates that a failure has occurred; it does not prevent the failure from occurring," while monitoring of plant parameters will not predict the incipient formation of dryer cracks, it will identify the existence of a crack sufficiently large to adversely affect dryer performance and alert to the risk of a structural failure of the dryer, including the generation of loose parts. Finding 164. The steam dryer has completed over two years of EPU operations without the detection of fatigue cracks, and this operating performance provides a high degree of assurance that the steam dryer is not subject to high cycle fatigue that could cause rapid flaw growth. Id. The cracks that have been seen have been IGSCC cracks that have essentially not changed from one cycle to the next. Id. Therefore, if a crack were to start to develop in the dryer, it would develop slowly. Id. The monitoring program would be sufficient to provide an "early warning" of potential dryer failure so that action could be taken prior to the occurrence of such failure.

Dr. Hopenfeld asserts that most parameter monitoring may indicate the formation of only those steam dryer cracks that increase moisture carryover, but those cracks that do not lead to significant moisture carryover may continue to grow undetected. Finding 165. However, since all of the reactor steam flows through the steam dryer, it is very unlikely that any damage to the dryer would not also result in a decrease in efficiency of the steam dryer and an increase in moisture carry-over that would cause a change in one or more of the monitored parameters (steam flow rate, reactor vessel water level and/or steam dome pressure). Id.

**b. Adequacy of Dryer Inspection Program**

Dr. Hopenfeld further asserts that Entergy's program to date of visual inspection and moisture monitoring has been ineffective in identifying cracking at the time it occurs, when it occurs in between inspections. Finding 166. However, there is no need to identify a crack the moment it occurs, because the intent of VY's program is to monitor material conditions on a frequency sufficient to identify and mitigate any flaws before they can grow to a size that would be detrimental to the integrity of the component. Id. The overwhelming majority of visual indications at VY have not grown since they were first identified, and those few indications that were determined to need repair had not reached critical size (that is, they had not had a negative effect on steam dryer integrity) prior to repair. Id.

Dr. Hopenfeld states that, once fatigue cracks initiate, they propagate very fast when exposed to alternating stresses of sufficient magnitude and frequency, so that even if one does not find cracks during an inspection, there is absolutely no reason why such cracks would not start propagating once the plant is restarted. Finding 167. VY's operating experience after the EPU (exemplified by the data collected during the 2007 inspection and the continuous monitoring of plant operating parameters for over two years) demonstrates that the stresses experienced by the dryer are below the fatigue limit for the material and are insufficient to initiate and propagate high-cycle fatigue cracks. Id. Instead, the cracks that have been found in the dryer are due to IGSCC and are small cracks that grow very slowly, if at all. Id.

### c. Need for Stress Load Estimation and Measurement

Dr. Hopenfeld asserts that the aging management program for the VY steam dryer should include "some means of estimating and predicting stress loads on the dryer, establishing load fatigue margins, and establishing that stresses on the dryer will fall below ASME fatigue limits" Finding 168. At the hearing, however, Dr. Hopenfeld was unable to explain how he would go about estimating stress loads on the dryer. Id.

Stress load estimation and prediction are unnecessary because confirmation that stresses on the VY steam dryer remain within its fatigue limits is provided by the inspection results, which show that high cycle fatigue is not occurring. Finding 169.

Further confirmation that the stress loads on the steam dryer are below the fatigue limits is provided daily by the fact that the dryer has been able to withstand without damage the increased loads imparted on it during power ascension and for the over two years of operation since the EPU was implemented. Finding 170. If a dryer failure leading to the generation of loose parts was to occur at VY, it would have taken place prior to this time. Finding 171.

Dr. Hopenfeld could not cite any instances of steam dryer failures occurring beyond eighteen months after implementation of a power uprate. Finding 172. Mr. Scarbrough confirmed that most dryer failures occurred shortly after the implementation of a power uprate, and the longer occurring case, at Dresden Unit 3, was discovered no more than eighteen months after uprate implementation. Id.

It takes approximately 10 million cycles for a component to accumulate enough fatigue-inducing vibration cycles during normal operation to reach the fatigue limit. Finding 173. Such a limit is reached within a year or two of operations, so that operation of a steam dryer for a year or two is sufficient to accumulate enough fatigue cycles to cause significant cracking in susceptible areas of the dryer. Id. Conversely, good performance (such as exhibited by the VY steam dryer) during the first operating cycle after the uprate strongly suggests that the dryer will not experience a fatigue-induced failure. Id. The limit in number of vibration cycles that could lead to high-cycle fatigue has already been reached at VY without adverse consequences being observed; hence, the steam dryer at VY is not subject to failure through that mechanism. Id.

There will be no change in dryer loads or stresses during the license renewal period of operation; hence, there is no reason to expect that the dryer will be subjected to increased stresses in the future. Finding 174.

Dr. Hopenfeld also expresses the view that Entergy should have introduced additional analytical tools for predicting the loads on the dryer. Finding 175. However, the analytical tools that were used during the uprate proceeding to demonstrate that loads on the dryer will be below its endurance limits were utilized as part of the design validation process that demonstrated the adequacy of the design and established the current licensing basis. Id. Because the predicted loads on the dryer were shown to be below the endurance limit, the design analysis was not time limited and thus does not need to be revisited at the license renewal stage, where only time limited aging analyses need to be evaluated. Id. Further, the loadings on the dryer derive from plant geometries that have not changed since the uprate was implemented; so, there has been no change to the loadings on the dryer and the resulting stresses. Therefore, there is no reason for further analytical efforts. Id.

With respect to directly measuring dryer loads, Entergy testified that such measurements would not be feasible because they would require welding gauges to the dryer, which would introduce new high stress areas, and would also require running electric wires from the dryer to outside the vessel directly on the steam flow path. Finding 176. In addition to the difficulties of such installation, the gauges could become potential loose parts in the event of a dryer failure. Id. By contrast, all other instrumentation currently installed in the reactor is in place at the bottom of the vessel. Id.

In light of these difficulties, the only way to measure steam dryer loading is to install strain gauges on the steam line and extrapolate their reading to the dryer location, which is what Entergy did during the uprate. Finding 177. Dr. Hopenfeld questioned the accuracy of such measurements, but failed to describe an alternative way in which such measurements might be made. Id.

Dr. Hopenfeld was also unable to specify how long should his proposed dryer measurement program last. Finding 178. The Staff opined that the three inspections at consecutive outages after the uprate and the more graded approach recommended by GE-SIL-644 thereafter should be sufficient, if no fatigue cracking was observed, to manage the potential for steam dryer fatigue. Id.

Monitoring stresses on a component on an ongoing basis is not carried out for any other component at VY, nor anywhere in the industry. Finding 179.

### d. Effect of IGSCC Cracks

Dr. Hopenfeld states that "IGSCC can provide sites for corrosion attack which in turn accelerate crack growth under cyclic loading." Finding 180. The sole basis for this opinion is a statement, which he attributes to General Electric's report of its inspection of the VY dryer in 2007, to the effect that "[t]he dryer unit end plates are located in the dryer interior and are not subject to any direct main steam line acoustic loading. However, continued growth by fatigue cannot be ruled out." Id.

Fatigue cracks start at locations that have a high stress riser. Finding 181. Thus, for a crack due to IGSCC to become a site for fatigue failure, the IGSCC has to occur in a high stress area. Id. No IGSCC indications in high stress areas have been found to date in the VY steam dryer. Id. The IGSCC indications found at VY are in low stress areas, from which fatigue may not develop. Id.

The document quoted by Dr. Hopenfeld is an Entergy VY Engineering report used to clear one of the corrective actions associated with indications identified in the 2007 dryer inspection so that it would not be a bar to restarting the plant after the 2007 outage. Finding 182. The sentence quoted by Dr. Hopenfeld does not appear in the final, signed report but in an earlier draft of the report. Id. The phrase was deleted from the final version because it raises a possibility that does not in reality exist, since the indication is on the dryer interior and is not in an area of high stress. Id.

### e. Effects of Steam Dryer Failure on LOCA Response

At the hearing, Dr. Hopenfeld sought to elaborate on his concern that a failing steam dryer could release loose parts that interfered with VY's ability to respond safely to a loss of coolant accident ("LOCA"). Finding 183. However, he was unable to provide any specific scenarios under which this could occur. Id.

There is no regulatory requirement to consider a LOCA coincident with a steam dryer failure. Finding 184. Moreover, prior to implementing the EPU, Entergy analyzed the dryer response to various accident sequences and determined that there was no accident situation, including a LOCA, that would result in loads on the dryer in excess of its design allowables, which are set by the ASME Code endurance limit for the material. Id. The results of those analyses is consistent with Entergy's testimony that fatigue failure of the dryer is caused by long term exposure to normal operating loads, as opposed to short duration accident loads. Id.

The Staff did not review a potential failure of the steam dryer in conjunction with a design basis accident as part of the license renewal reviews because those analyses were performed as part of the uprate and remain valid as long as the steam dryer maintains its integrity, which is assured by the dryer monitoring program. Finding 185.

Mr. Scarbrough testified that he knows of no scenario where loose parts from a failing steam dryer could interfere with the ability to mitigate the effects of a LOCA. Finding 186. In Those instances in which a steam dryer has experienced a failure leading to loose part generation, the loose parts have not interfered with the operation of safety-related components. Id.

Dr. Hopenfeld postulated a scenario in which a dryer, already weakened by significant cracks, would fail upon being subjected to the loads imparted by a LOCA and would generate loose parts that would interfere with the safe shutdown of the plant. Finding 187. Occurrence of such a scenario would be unlikely because the plant parameter monitoring program would detect the existence of significant cracks and cause their evaluation and the potential shutdown of the plant. Id. Were such a situation to arise, however, the loose parts would probably escape with the steam released by the LOCA without adverse consequences. Id.

5. <u>Conclusions to be drawn from the evidence</u>

The evidence shows that Entergy has instituted a program, in effect as part of the plant's current licensing basis and to be continued after renewal of the VY license, to continuously monitor plant parameters indicative of potential cracking of the steam dryer and properly evaluate and respond to any significant departures of those parameters from their normal range. This program is conducted by highly qualified and trained individuals. Entergy also performs during each refueling outage thorough visual inspections, conducted in accordance with industry guidelines, of the areas of the steam dryer potentially susceptible to fatigue crack formation. The fact that the VY steam dryer has shown no evidence of fatigue induced cracks after over two years of EPU operation strongly indicates that routine inspection of the steam dryer during the period of extended operation will be sufficient to provide reasonable assurance of continued steam dryer integrity. In all, the steam dryer inspection and monitoring plan that Entergy will implement during the period of extended operation after license renewal will assure that the aging effects on the steam dryer will be adequately managed. For that reason, there is no support for the claims made in NEC Contention 3, which should be rejected.

NEC has argued that Entergy is still relying upon the analyses it performed in implanting the EPU. If that were true, there would be no need for any aging management program, because the determination through the analyses that the stresses will remain below the endurance limit would indicate that fatigue is not an aging effect for the dryer. In contrast, Entergy has committed to both inspections and monitoring of parameters on a continuing basis. It is the results of those inspections and monitoring that indicate that stresses are below the material's endurance limit.

## IV. NEC CONTENTION 4

### A. Background

As originally admitted into this proceeding, the claim raised by NEC Contention 4 was that "Entergy's License Renewal Application does not include an adequate plan to monitor and manage aging of plant piping due to flow-accelerated corrosion during the period of extended operation." LBP-06-20, 64 N.R.C at 192. The basis for that claim, however, was NEC's assertion that Entergy's plan at VY to monitor and manage the aging of plant piping and components due to flow-accelerated corrosion ("FAC") is inadequate in that its selection of the components that must be inspected for FAC relies on CHECWORKS, an "empirical code," which "must be continuously updated with plant-specific data," and which has not been benchmarked with sufficient data reflecting parameter changes associated with VY's EPU. NEC alleged that Entergy cannot rely on CHECWORKS because the recent power uprate has changed plant parameters, including coolant flow rates, and that the model cannot generate accurate recommendations because it has not been benchmarked with data reflecting these new parameters. For that reason, said NEC, "Entergy cannot assure the public that the minimum wall thickness of carbon steel piping and valve components will not be reduced by FAC to below . . . code limits during the period of extended operation." LBP-06-20, 64 N.R.C. at 192-94.

Entergy filed a motion for summary disposition of NEC Contention 4. Entergy's Motion for Summary Disposition of New England Coalition's Contention 4 (Flow Accelerated Corrosion) (June 5, 2007). In its "Memorandum and Order (Ruling on Motion for Summary Disposition of NEC 4)" (August 10, 2007) (unpublished) ("August 10, 2007 Order"), the Board denied Entergy's motion for summary disposition, concluding there were conflicting expert opinions between Entergy (whose experts asserted that extensive benchmarking of CHECWORKS is not necessary and the data from three refueling outages at EPU conditions is sufficient) and NEC's expert asserts, that 10-15 years worth of data is necessary. See August 10, 2007 Order at 7.

## B. Witnesses

### 1. Entergy Witnesses

Entergy's testimony on NEC Contention 4 was presented by a panel of two experts, each with extensive experience in the management of flow accelerated corrosion ("FAC") in boiling water reactor ("BWR") components. The first witness on the panel, Dr. Jeffrey S. Horowitz, has more than 36 years of experience in the field of nuclear energy and related disciplines and 22 years of experience specializing in FAC and nuclear safety analysis. Entergy NEC 4 Dir. at A3. Dr. Horowitz designed and implemented a computer program to assist utilities in determining the most likely places for FAC wear to occur, and thus the key locations to inspect for component wall thinning. He developed the computer programs CHEC (Chexal-Horowitz Erosion Corrosion) in 1987, CHECMATE (Chexal-Horowitz Methodology for Analyzing Two-Phase Environments) in 1989, and CHECWORKS (Chexal-Horowitz Engineering Corrosion Workstation) in 1993. Id. at A6. He has performed, by himself or with another engineer, audits of the FAC programs at over fifty nuclear units in the United States and Canada, including a FAC program audit at VY, in April 2007. Id. at A7. Dr. Horowitz played a significant role in drafting NSAC-202L, entitled "Recommendations for an Effective Flow-Accelerated Corrosion Program," and each of its three revisions, which has become the most important standard-setting document for the conduct of FAC control programs in the United States. Id. at A8. Dr. Horowitz has authored numerous articles and given numerous presentations regarding FAC. Id. at A9-A10.

The second witness on the panel, Mr. James C. Fitzpatrick, has thirty years of experience in design, construction, and modification of nuclear power plant structures, piping systems, pressure vessels, and other equipment. Id. at A13, A15 and A16. Mr. Fitzpatrick was the Cognizant Engineer for the VY FAC Program through June 2007, and was responsible for developing the scope of refueling outage inspections, providing on-site engineering support, screening and evaluating piping and components, determining if the sample of piping locations designated for inspection during a refueling outage needed to be expanded, coordinating piping and component repairs and replacements, updating the CHECWORKS models of plant piping systems, and maintaining the FAC Program Manual supporting documents. Id. at A16.

Entergy witness Dr. Horowitz is qualified as an expert in the analysis and prediction of the potential flow-accelerated corrosion of piping and other components in operating reactors, the effects of erosion and other mechanisms that may cause the degradation of piping and other components; industry standards and guidance relating to the analysis and prediction of FAC; the development and use of computer codes, particularly the CHECWORKS code, in the identification of piping and component locations susceptible to FAC; the ability of programs such as CHECWORKS to predict FAC susceptibility of piping and components in plants that have undergone a power uprate; and the operating experience in the United States and abroad with respect to FAC.

Entergy witness Mr. Fitzpatrick is qualified as an expert in the development and implementation of programs to control FAC; the selection of piping and component locations that need to be inspected for potential FAC effects; the history and status of the site-specific FAC program at VY; and the extent to which computer codes such as CHECWORKS are used to assist in selecting piping and component locations to be inspected for FAC.

The testimony and opinions of the Entergy witnesses on NEC Contention 4 are based on both their technical expertise and experience and their first hand knowledge of the factual issues raised in the contention.

2. NRC Staff Witnesses

The NRC Staff provided testimony on NEC Contention 4 through a panel of two witnesses, Messrs. Kaihwa R. Hsu and Jonathan G. Rowley. Mr. Hsu is currently a senior mechanical engineer in the Division of Engineering in the Office of New Reactors. Statement of Professional Qualifications of Kaihwa R. Hsu (Staff NEC 3 Dir., following p. 14). Previously, he was a materials engineer in NRR's Division of License Renewal. Id. Mr. Hsu has 27 years of experience in the nuclear industry, including significant experience with FAC-predictive codes through his work for the NRC and Westinghouse, where he was part of the team that developed a computer code, the Westinghouse Corrosion-Erosion Monitoring System, which, like CHECWORKS, predicts pipe thinning due to FAC. Id. As part of his official duties, Mr. Hsu served as a technical lead for the license renewal safety audit at VY, and reviewed Entergy's FAC program for VY. Id.

Mr. Hsu is qualified as an expert witness on the development and use of computer codes, particularly the CHECWORKS code, in the identification of piping and component locations susceptible to FAC; the ability of programs such as CHECWORKS to predict FAC susceptibility of piping and components in plants that have undergone a power uprate; the Staff's review of the FAC management program at VY; and the Staff's interpretation of the adequacy of the VY FAC inspection program during VY's operation after license renewal.

Mr. Rowley has over fourteen years of experience in materials science and engineering. Statement of Professional Qualifications of Jonathan G. Rowley (Staff NEC 3 Dir., following p. 14). Mr. Rowley has been responsible for coordinating the Staff's review of Entergy's Application and the Staff's preparation of its "Safety Evaluation Report with Confirmatory Items Related to the License Renewal of Vermont Yankee Nuclear Power Station," dated March 2007 (ML070870378) and the FSER. Id. In addition to his involvement the Staff's review of Entergy's Application, Mr. Rowley was involved in the Staff's review of the license renewal applications for the D.C. Cook and R.E. Ginna plants.

Mr. Rowley is qualified as an expert witness on the subjects of Entergy's Application and its review by the NRC Staff, NRC regulatory requirements and guidance pertaining to license renewal applications, and the bases for Staff approval of Entergy's Application as it relates to the FAC control program.

3. NEC Witness

The NEC testimony on Contention 4 was presented by a panel of three witnesses: Dr. Joram Hopenfeld, Dr. Rudolf Hausler, and Mr. Ulrich Witte. Dr. Hopenfeld lists as his relevant areas of expertise thermal/hydraulics, materials, environmental interaction, radioactivity transport, industrial instrumentation and environmental monitoring. Id. Dr. Hopenfeld's curriculum vitae (NEC Exhibit NEC-JH_02) does not state that he has any professional experience or training on matters relating to FAC management programs or the use of CHECWORKS or other computer codes to assist in identifying piping or component locations that may be susceptible to FAC.

Dr. Hopenfeld has little qualifications on corrosion and other degradation mechanisms in nuclear power plant piping and components.

Dr. Hausler has Bachelor's and Master's degrees in Chemical Process Technology and a Ph.D. in Chemical Engineering, all from the Swiss Federal Institute of Technology. His areas of expertise are "corrosion prevention, chemical inhibition, material selection, failure analysis, and trouble-shooting." Hausler Dir. at A2. He has served as a consultant to oil and engineering companies the selection, testing, and application of corrosion inhibitors. Id. His work in the nuclear industry appears to be limited to analyzing the safety of nuclear fuel storage casks. Id. Dr. Hausler's curriculum vitae does not reveal he has any experience on corrosion, erosion or other forms of degradation of nuclear power plants components, or any experience or expertise in the use of CHECWORKS or other computer codes to assist in identifying piping or component locations that may be susceptible to FAC.

Mr. Witte has a Bachelor's degree in Physics from the University of California, Berkeley (1983). He describes his professional experience as including "configuration management, engineering and design change controls, and licensing basis reconstitution" and work in "engineering, licensing, and regulatory compliance of commercial nuclear facilities." Witte Dir. at A2. His experience has "generally focused on assisting nuclear plant owners in reestablishing fidelity of the licensing bases with the current plan design configuration, and with actual plant operations." Id.

The Board had occasion to rule on Mr. Witte's qualifications to testify on NEC Contention 4. In granting in part Entergy's Motion in Limine to exclude Mr. Witte's testimony on NEC Contention 4 (Entergy's Motion in Limine, June 12, 2008, at 22-25), the Board found that Mr. Witte qualifies as expert on configuration management issues, but struck his testimony as to the predictive accuracy of the CHECWORKS model, the requirements necessary to benchmark it, and other technical aspects of predicting and modeling FAC." Order (Rulings on Motions to Strike and Motions in Limine) (July 16, 2008), slip op. at 7.[18]

### C. Applicable Legal Standards

The applicable legal standard for the Staff's approval of VY's aging management program with respect to FAC is whether there is reasonable assurance that the aging effects of FAC on reactor coolant pressure boundary piping and associated components will be adequately managed at VY so that the intended functions of the piping and associated components will be maintained consistent with the CLB for the period of extended operation. 10 C.F.R. §§ 54.21(a)(3), 54.29(a).

Thus, the legal issue to be addressed with respect to NEC Contention 4 is whether there is reasonable assurance that the proposed FAC aging management program adequately manages the aging effects of FAC on reactor coolant pressure boundary piping and associated components and is consistent with the CLB.

The FAC program at VY is an existing program currently in effect. Entergy NEC 4 Dir. at A19; Tr. at 1524 (Rowley). The Application states that "[t]he Flow-Accelerated Corrosion

[18] The Board declined to strike Mr. Witte's testimony on factual matters "as to events and activities that are primarily factual and otherwise historically verifiable in this proceeding." Id.

(FAC) Program at VYNPS is comparable to the program described in NUREG-1801, Section XI.M17, Flow-Accelerated Corrosion." Application, Appendix B, Section B.1-13 at B-47. The Board raised sua sponte the issue whether "a renewal application that contains a short written description of an aging management program that lacks content or details but instead states that it is 'comparable to' and 'based on' the relevant section of NUREG-1801 or EPRI NSAC-202L, 'demonstrate that the effects of aging will be adequately managed' as required by 10 C.F.R. §§ 54.21(a)(3) and 54.21(c)(1)(iii)." Order (Regarding the Briefing of Certain Legal Issues) (June 27, 2008) at 5.[19]

Under the NRC regulations, the Application may properly adopt the program descriptions in NUREG-1801 and NSAC-202L to establish an acceptable aging management program according to NRC rules. 10 C.F.R. § 54.17(e). Moreover, Section B.1.13 specifically states that "[t]he Flow-Accelerated Corrosion program at VYNPS is consistent with the program described in NUREG-1801, Section XI.M17, Flow-Accelerated Corrosion" and identifies the exceptions and enhancements to "NUREG-1801, Section XI.M17, Flow-Accelerated Corrosion" as "None." Application at B-47.[20] Therefore, the Application is not lacking in necessary specificity by incorporating by reference the guidance in NUREG-1801 and NSAC-202L.

With respect to the issue of whether compliance with the GALL Report satisfies the requirements of 10 C.F.R. § 54.21(a)(3), such compliance is not in itself incontrovertible evidence that the effects of aging will be adequately managed, but the GALL Report is entitled to particu-

[19] This issue was not raised by NEC in the text of NEC Contention 4 or any of the bases asserted in support of the Contention. Indeed, NEC's bases focused on whether CHECWORKS requires 10-15 years of data for "benchmarking." See LBP-06-20, 64 N.R.C. at 192-93. NEC Contention 4 did not challenge the Application's compliance with NUREG-1801 or EPRI NSAC-202L, or the incorporation by reference of those standards into the Application. Nor has NEC raised any objection to the specificity of the description of the VY FAC program in its contention.

[20] At the hearing, Entergy witness Mr. Fitzpatrick testified that any departures from the guidance in NUREG-1801 or EPRI NSAC-202L would be identified by Entergy and reviewed by the Staff. Tr. at 1493-94 (Fitzpatrick). No such departures have been identified in the record of this proceeding.

larly significant weight in addressing the issue of adequacy of aging management programs. It identifies aging management programs that have been determined by the NRC to be acceptable programs to manage the effects of aging on systems, structures and components within the scope of license renewal as required by 10 C.F.R. Part 54. The GALL Report is based on a systematic compilation of plant aging information and the evaluation of program attributes for managing the effects of aging on systems, structures and components for license renewal. GALL Report at 1-3.

The NRC Staff developed the GALL Report at the direction of the Commission to provide a basis for evaluating the adequacy of aging management programs for license renewal. GALL Report at 1, 4; Memorandum from A. Vietti-Cook to W. Travers, "Staff Requirements - SECY-99-148 - Credit for Existing Programs for License Renewal" (Aug. 27, 1999) (ADAMS Accession No. ML003751930). When the GALL Report was submitted to the Commission for approval in April 1991, the Staff stated:

> Applying the GALL report will reduce the need to review plant-specific aging management programs. .... <u>In addition, when applicants state that their aging management programs are bounded by the GALL programs, the staff's review will shift from reviewing each program in detail to verifying the applicant's assertion.</u> This will significantly reduce staff review resources and increase the efficiency of the review. The staff believes that the improved license renewal guidance documents will increase the stability and predictability of the license renewal review process because they describe the framework for a disciplined process that clearly articulates the evaluation criteria. <u>They also provide a clear and sound technical basis to support the staff's conclusion that (1) actions have been identified and have been or will be taken with respect to managing the effects of aging during the period of extended operation for structures, systems, and components within the scope of the license renewal rule, (2) and that actions have been identified and have been or will be taken with respect to time-limited aging analysis that are required to be reviewed in accordance with the license renewal rule</u>. These documents should also increase public confidence in the license renewal review process because the public was involved in developing them, and the public's comments

were considered and incorporated, and because the documents will make the staff's license renewal reviews more predictable.

SECY-01-0074, Memorandum from W. Travers to Commissioners, "Approval to Publish Generic License Renewal Guidance Documents" (Apr. 26, 2001) (ADAMS Accession No. ML010990201) at 4-5 (emphasis added).

The Commission approved the issuance of this guidance. The Commission commended the Staff for its outstanding efforts in developing these license renewal guidance documents, and stated: "These documents should serve to enhance the predictability, consistency, and efficiency of the NRC reviews of license renewal applications." Memorandum from A. Vietti-Cook to W. Travers, "Staff Requirements - SECY-01-0074 – Approval to Publish Generic License Renewal Guidance Documents (July 2, 2001) (ADAMS Accession No. ML011860168).

Thus, it is appropriate that compliance with guidance documents, such as the Gall Report, other NUREGs, or the Standard Review Plan, that are developed by the NRC to assist licensees or applicants to comply with applicable regulations, be afforded special weight. Such deference is particularly appropriate with respect to the GALL Report because it was developed at the Commission's direction with considerable public involvement and its issuance was approved by the Commission. Private Fuel Storage, L.L.C. (Independent Spent Fuel Storage Installation), CLI-01-22, 54 N.R.C. 255, 264 (2001) ("Where the NRC develops a guidance document to assist in compliance with applicable regulations, it is entitled to special weight.")[21]

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[21] In Florida Power & Light Co. (Turkey Point Nuclear Generating Plant), CLI-01-17, 54 N.R.C. 3 (2001) the Commission explained that the focus of license renewal adjudicatory hearings are the same as the scope of the Staff review:

> In sum, our license renewal safety review seeks to mitigate the "detrimental effects of aging resulting from operation beyond the initial license term." 60 Fed. Reg. at 22,463. To that effect, our rules "focus[ ] the renewal review on plant systems, structures, and components for which current [regulatory] activities and requirements may not be sufficient to manage the effects of aging in the period of extended operation." Id. at 22,469 (emphasis added). Adjudicatory hearings in individual license renewal proceedings will share the same scope of issues as our NRC Staff review, for our hearing process (like our Staff's review) necessarily examines only the questions our safety rules make pertinent.

54 N.R.C. at 10 (footnote omitted).

In summary, the Application is not lacking in content or specificity by incorporating by reference the guidance in NUREG-1801 and NSAC-202L. The legal issue on NEC Contention 4 is whether there is reasonable assurance that the current FAC management program, if continued during the license renewal period, will comply with the reference guidance documents and by doing so will adequately manage the aging effects of FAC on reactor coolant pressure boundary piping and associated components during the period of extended plant operation following renewal of the VY license.

## D. Summary of the Evidence

### 1. Summary of VY's FAC Management Program

#### a. Definition of FAC

Flow-accelerated corrosion is an age-related degradation mechanism that attacks carbon steel piping and components exposed to moving water or wet steam. Finding 189. FAC is caused by the protective oxide layer that builds on the surface of carbon steel piping and components being dissolved into the flow stream. Id.

If FAC is not detected, the piping or component walls will become progressively thinner, until the material in the affected area can no longer withstand internal pressure and other applied loads and a rupture eventually occurs. Finding 190. FAC wear causes a pipe rupture, whereas damage due to other mechanisms (e.g., erosion) causes leaks and does not impact the structural integrity of the piping. Id. FAC only attacks carbon steel components in the presence of purified flowing water or wet steam. It does not attack steels containing other fluids, such as oil. Id. Steels containing appreciable amounts of chromium have been found immune to FAC. Id.

FAC is only one of several mechanisms that can affect the physical integrity of piping and components. Finding 191. FAC is a chemical (corrosion) mechanism, not a mechanical damage mechanism (i.e., erosion). Erosive damage also occurs in nuclear piping, but such damage is normally confined to small leaks. Id.

The definition of FAC, as opposed to erosion-corrosion, was made more precise to avoid confusion that occurred when one used the term erosion-corrosion because the countermeasures that would be used to deal with a particular phenomenon differ depending upon whether the problem is erosion or flow-accelerated corrosion. Finding 192.

Cavitation is not an aging management issue, but is normally considered a design issue. Finding 193. Once a plant experiences cavitation the wear from cavitation is not trended, but instead the problem is fixed. Id.

Impingement damage normally occurs when high velocity streams of steam impinge on piping. Finding 194. As contrasted with FAC, impingement tends to be localized, causing little holes, and it often occurs under upset-type conditions, not normal operating conditions. Impingement damage is basically unpredictable. Id. Because it is a different mechanism from FAC, impingement is unaffected by replacing carbon steel with FAC-resistant material such as stainless steel or low-alloy steel. Id.

Although water flowing at a high enough velocity may cause erosion in addition to corrosion, that kind of erosive damage is not seen in light water reactors because the water velocities necessary to cause that damage in carbon steel piping are not seen in light water reactors. Finding 195.

Dr. Hopenfeld testified that he was not an expert on the characteristics of oxide layers in piping and could not separate erosion and corrosion, and deferred to Dr. Hausler's explanation of the concept of erosion-corrosion. Finding 196. Dr. Hausler testified that corrosion engineers have, in fact, used the term erosion like Dr. Horowitz and as Mr. Fitzpatrick indicated in areas of high turbulence, because where there is high turbulence you get somewhat localized corrosion. Id. Dr. Hausler agreed with Dr. Horowitz and Mr. Fitzpatrick that the mechanism being dealt with in FAC was a dissolution phenomenon. Id.

### b. Description of VY's Proposed FAC Program

Section B.1.13 of the Application states that the VY program for addressing FAC is consistent with the program described in the NRC guidance document "Generic Aging Lessons Learned (GALL) Report -- Tabulation of Results," NUREG-1801, Vol. 2, Rev. 1 (Sept. 2005), Section XI.M17, Flow Accelerated Corrosion. Finding 197. There are no exceptions in the Application to the guidance in NUREG-1801 with respect to FAC. Id.

The VY FAC Program currently in effect substantially follows the current version of NSAC-202L, NSAC-202L-R3 (Entergy Exh. E4-07). Finding 198. The VY FAC Program includes, as recommended in the GALL Report and the NSAC-202L guidelines, "procedures or administrative controls to assure that the structural integrity of all carbon steel lines containing high-energy fluids (two-phase as well as single-phase) is maintained." Id.

The VY FAC Program includes the following activities: (a) conducting an analysis to determine critical locations; (b) performing baseline inspections to determine the extent of thinning at these locations; and (c) performing follow-up inspections to confirm the predictions, or repairing or replacing components as necessary. Finding 199. NSAC-202L provides the general guidelines that are implemented in the FAC Program. Id.

The VY FAC Program during the license renewal period will be identical to the existing program and will conform to the EPRI guidelines contained in NSAC-202L. Finding 200. It will include "procedures or administrative controls to assure that the structural integrity of all carbon steel lines containing high-energy fluids (two-phase as well as single-phase) is maintained." Id. It will also provide detailed instructions on: (a) how to conduct the inspections; (b) how to evaluate the inspection data; (c) the acceptance criteria for inspected components; (d) the disposition of components failing to meet acceptance criteria; (e) the expansion of the sample to other components similar to those failing to meet acceptance criteria; and (f) the updating of CHECWORKS models to incorporate inspection data. Id.

VY has programs in place that deal with phenomena, other than FAC, that cause pipe wall thinning. Finding 201. The VY FAC Program, however, would detect pipe wall thinning in piping inspected under the program regardless of the cause of the thinning. Id. Other programs manage aging in piping systems not within the scope of the VY FAC Program. Id. Inspection locations to check for mechanical damage (non-FAC wear) are selected by operating experience. Id.

Phenomena that may cause wall thinning that are not FAC are not tracked in VY FAC Program because they are not aging management issues. Finding 202. Cavitation is a design issue, and is not subject to trending. Id. Impingement, although sometimes capable of being trended, is an unpredictable phenomenon that can be related to design issues. Id.

**c. Scope of the VY FAC Program**

All piping at VY that may experience FAC is included within the scope of the VY FAC Program. Finding 203. Compared to the majority of nuclear power plants in operation, VY is a relatively small and simple plant. Id. There are fewer FAC-susceptible systems and piping components than at a typical plant, and many of those were either originally constructed of FAC-resistant materials or have been replaced with FAC-resistant materials since their initial installation. Id.

The extraction steam system piping, which contains a significant portion of the two-phase piping in a power plant, is constructed from FAC-resistant materials. Finding 204. A number of other components and associated piping subject to two-phase flow have been replaced with FAC-resistant materials. Id. The original plant design and the component replacements have resulted in a significantly smaller amount of FAC-susceptible piping at VY as compared to the typical nuclear power plant of similar size. Id.

Since VY went into operation, carbon steel piping and equipment in a number of systems has been progressively replaced with FAC-resistant materials. These include: (a) all 10 of its feedwater heaters; (b) both low pressure turbine casings, including the attached extraction steam nozzles and piping; (c) all of the two phase flow piping in the moisture separator drains system; (d) the majority of the two phase flow piping in the heater drains system except at the lowest pressure feedwater heaters; (e) the majority of the turbine cross around piping; (f) small bore steam drain lines to the condenser coming from the high pressure cooling injection system, the reactor core isolation cooling system, and the advanced off-gas system; (g) small bore shell vent lines for all four of the high pressure feedwater heaters. Finding 205. Nearly all of the large bore piping at VY which is exposed to two-phase flow was either originally constructed with, or re-placed with, FAC-resistant material. Id.

#### d. Selecting Components for Inspection

The VY FAC Program conforms to the inspection recommendations contained in NSAC-202L. Finding 206. The FAC Program calls for piping and component inspections to be conducted at each refueling outage, with the items to be inspected being selected based on: (a) required re-inspections and recommendations from previous outages; (b) CHECWORKS susceptibility rankings; (c) industry/utility/station experience including items identified through work orders and condition reports; (d) susceptible non-modeled large bore and small bore program piping; (e) engineering judgment. Id. Entergy's FAC program also takes risk significance and component susceptibility to failure into account. Id.

In a typical inspection, approximately one-third of the piping and components will be selected based on CHECWORKS results, approximately one-third will be chosen based on previous inspection data, and the remainder will be selected based on operating experience. Finding 207. Although CHECWORKS is a useful analytical tool, the VY FAC Program is not dependent on the CHECWORKS results to select the piping to inspect, and such selection could be managed without resorting to CHECWORKS. Id.

The trending of wear on piping is not based on the use of CHECWORKS but on actual inspection data. Finding 208. The actual trend wear rates from inspection data are also used to select components for inspection. Id. The selection of components for inspection by the FAC engineer is subject to peer review by another engineer. Id.

At VY, the initial scoping and inspection selection of small bore piping was performed in 1993 and 1995. Finding 209. The scope and criteria for determining the inspection locations is documented in FAC Program documents. Id. The small bore inspections were initiated prior to the inclusion of small bore guidance provided in NSAC-202L. Id.

**e. Inspecting Components**

When components are selected for inspection, Entergy follows an Engineering Standard, "Flow Accelerated Corrosion Component Scanning and Gridding Standard," to perform the inspections. Finding 210. This standard defines the methodology for gridding the components that are to be inspected and the size of the grids. Id.

Historically, grid size is related to the physical size of the component being inspected. There are two aspects to grid size: (1) where piping degradation is found, the grid size is normally made smaller to more accurately define the wear area; and (2) the larger the pipe, the larger amount of material that may be lost before the component fails, allowing for a "larger" grid (i.e., the defect size that would cause failure varies directly with the size of the pipe). Finding 211. Both of these approaches are consistent with NSAC-202L. Id.

Under the VY FAC Program, the size of the grid varies with the outside diameter of the pipe. Finding 212. The grid sizes are based on the recommendations in NSAC-202L. Id. Rather than recording the thickness reading at particular grid points, however, the VY FAC Program takes an additional step in performing the inspections: the components inspected at VY are scanned in their entirety. Id. This ensures that the thinnest thickness readings in the component are found. Id.

The extent to which the thickness of a component is measured depends on the component. Finding 213. If, for example, the component is an elbow, the entire elbow is scanned. Id. The pipe downstream will be inspected axially for two diameters in length, as provided in NSAC-202L. Id. If there is another component within two diameters of the component being measured, the components are modeled in CHECWORKS and are inspected consistent with the inspection program. Id.

With respect to how far downstream of a component such as an elbow should piping be inspected, Dr. Hopenfeld testified that a minimum distance of 25 to 45 diameters should be inspected, but could give no basis for that range other than his opinion that it is a customary number that has been around for many years. Finding 214. However, VY has done axial inspections on four different lines beyond a distance of two diameters from a component and has not found any excessive wear more than two diameters past the component. Id.

**f. Role of Water Chemistry in the VY FAC Program**

In 1980, an oxygen injection system was added to VY to improve the water chemistry towards minimizing FAC. Finding 217. Oxygen is injected into the condensate and feedwater trains just downstream of the condensate pumps in order to mitigate the effects of FAC on piping exposed to single phase flow. Id. This treatment results in about 40 parts per billion ("ppb") dissolved oxygen in the condensate and feedwater trains. Id. This level of dissolved oxygen serves to reduce the rate of FAC because, by maintaining this concentration of dissolved oxygen in the condensate and feedwater lines, the stability of the oxide film is enhanced, the rate of dissolution is reduced, and the potential for corrosion is decreased. The change to hydrogen water chemistry in 2003 did not change, nor was it expected to change, the oxygen concentrations in the feedwater system, as demonstrated by measured plant data. Id.

### 2. Use of CHECWORKS

#### a. Description of CHECWORKS

In December 1986, an elbow in the condensate system at the Surry Unit 2 nuclear plant failed catastrophically. Finding 219. This failure caused steam and hot water to be released into the turbine building, resulting in the deaths of four workers and severe injuries to others. Id. Post-accident investigations revealed that FAC was the cause of the degradation to the elbow. Id.

In response to the Surry accident, EPRI became committed to developing a computer program that would assist utilities in determining the most likely places for FAC wear to occur, and thus the key locations to inspect for pipe wall thinning. Finding 220. Dr. Horowitz developed the computer program CHEC (Chexal-Horowitz Erosion Corrosion) and demonstrated and released it to U.S. utilities in 1987. Id. CHEC was replaced by CHECMATE (Chexal-Horowitz Methodology for Analyzing Two-Phase Environments) in 1989. Id. CHECMATE expanded on the capabilities of CHEC by adding algorithms to calculate FAC under two-phase conditions. Id.

CHECMATE was later re-placed by the current program, CHECWORKS (Chexal-Horowitz Engineering Corrosion Workstation), in 1993. Finding 221. Dr. Horowitz remained the technical lead person in the development of the new and revised versions. Id.

CHECWORKS is a multi-purpose computer program designed to assist FAC engineers in identifying potential locations of FAC vulnerability. Finding 222. CHECWORKS is designed to be used by plant engineers as a tool in identifying piping locations susceptible to FAC, predicting FAC wear rates, planning inspections, evaluating inspection data, and managing inspection data. Id. It predicts FAC wear rates based on a number of variables that define: (1) the water chemistry; (2) the flow rate; (3) the geometry of the components; (4) the material properties of the components; (5) temperature; and (6) steam quality. Id.

CHECWORKS creates a calculation of FAC wear rate that is composed of seven factors: FAC Rate = F1 * F2 * F3 * F4 * F5 * F6 * F7. The seven factors are: (1) F1 = Temperature factor; (2) F2 = Mass transfer factor; (3) F3 = Geometry factor; (4) F4 = pH factor; (5) F5 = Oxygen factor; (6) F6 = Alloy factor; and (7) F7 = Void fraction factor. Finding 223. The CHECWORKS model is based on an established method for calculating FAC wear and is based upon laboratory data and plant data. Id.

The correlations built into CHECWORKS are based on laboratory experiments on modeled geometries, published correlations, and operating data from many nuclear units. Finding 224. These data support the nearly linear relationship between flow velocity and FAC wear rate. Id.

Dr. Hopenfeld testified that he believed that the geometry factors used in CHECWORKS were inaccurate because they used an average value to calculate "pressure drops" at a fitting, when he believed that the local flow velocity value, not the average value should be used and that correction would result in an order of magnitude of difference in results. Finding 225. However, nuclear plants operate under the same conditions for long periods of time. So, so it is appropriate for CHECWORKS to use the average velocity corresponding to the power plants' run time. Id. With respect to changes in local velocity due to geometric discontinuities such as elbows and fittings, CHECWORKS uses the average velocity in the cross-section, and the geometry factors to correct for the different flow patterns that occur at those locations. Id.

Dr. Hopenfeld testified that he did not believe that the relationship between corrosion and velocity used in CHECWORKS was accurate because it was based on copper dissolution in hydrochloric acid, and hydrochloric acid is not used to cool reactors and most material in reactors is not copper. Finding 226. However, the copper tests were not used to establish wear rates, but serve only as a fast way of doing tests of various geometries. Id. All the geometry factors used by CHECWORKS are based on actual plant data. Id.

CHECWORKS uses a nearly linear velocity relationship with mass transfer. Finding 227. The linear velocity relationship is based on Dr. Horowitz's review of experimental data and plant experience. Id. Unlike erosion mechanisms, FAC damage is linear with time (i.e., there is a constant corrosion rate). Id. This linear relationship has been demonstrated in numerous laboratory tests and by the fact that field measurements match predictions using a linear model. Id. Studies from nuclear power plants that have undergone power uprates show increases in FAC wear rates proportional to velocity. Id.

If CHECWORKS' model of the relationship between velocity and mass transfer were inaccurate, a user would immediately see discrepancies on the predicted wear of piping upstream and downstream of components such as reducers – e.g., the wear rate predictions of an eight inch diameter pipe would be over-predicted, and that of a six inch diameter pipe would be under-predicted. Finding 228. No plant using CHECWORKS has reported such erroneous results. Id.

The rate of FAC is constant as long as conditions remain constant and, under constant conditions, FAC rate can be determined by two data points. Finding 229. Although Dr. Hausler testified that surface finish could affect the rate of FAC wear, the variation in wear rate with roughness is very small; in large pipes, the extent of roughness does not have much of an effect on flow once the surface of a pipe has become rough. Id.

The single most important variable in FAC wear rates is the chromium content of the piping. Finding 230. CHECWORKS conservatively assumes that steel components contain the lowest amount of alloying elements allowed by the specification (typically, zero). Id. Such an assumption disregards the beneficial effects of chromium in retarding the onset of FAC. Id.

CHECWORKS produces for every component in each analysis line a predicted wear rate, and predicted total wear for that component. Finding 231. For components with measured data, it also compares the predicted wear with the measured wear at the time of that inspection and presents the time to reach a user-defined critical thickness in tabular fashion. Id.

The CHECWORKS User Group has met twice a year since 1989 and has been the major source of feedback on the adequacy of the program. Finding 232. In no instance has a pipe failure been determined by the NRC to be a result of inaccurate predictions by CHECWORKS. Id.

The CHECWORKS model is periodically checked against laboratory and plant operating data as it becomes available to examine how well the CHECWORKS correlation performs. Finding 233. For example, inspection results at VY show that CHECWORKS over predicts FAC wear rates for the feedwater line. Id. This is common to all BWR feedwater lines. Id. When a new issue occurs or when users report that something is not working as well as CHECWORKS would like, EPRI will conduct a separate study to look at that individual parameter. Id.

Dr. Hopenfeld testified that an alternative to the use of CHECWORKS would be for each plant to have an expert that would be working on, and completely dedicated to, FAC evaluations. Finding 234. VY has a dedicated FAC engineer whose job is solely to maintain the FAC program. Id.

**b. Modeling of a Plant in CHECWORKS**

CHECWORKS modeling of a nuclear unit starts with the development of a plant heat balance diagram ("HBD"), which is a schematic representation of the major lines and connectivity of the power producing portion of the nuclear plant. Finding 235. The HBD model constructed in CHECWORKS is then populated with the thermodynamic conditions representative of each power level at which the plant has operated at, or is contemplated to operate. Id. The user then inputs the oxygen concentration conditions that have been used or are anticipated. Id. These inputs define the operational history of the plant in terms of what power levels have been used with what water chemistry for how long. Id.

The user enters as input to CHECWORKS information concerning the piping systems to be analyzed. Finding 236. Most of this information is at the component level and deals with geometry, wall thickness, operating conditions, and pipe material. Id. CHECWORKS includes over fifty geometry models to represent various component geometries. Id. In cases where the component geometry does not match any of the models, the CHECWORKS user is instructed to either use a conservative model or schedule the component for inspection. Id.

The VY FAC Program uses a thickness measuring process that scans the components in their entirety by moving an ultrasonic transducer over the entire surface within a grid "square." Finding 237. The data logger automatically records the minimum reading anywhere within the grid square and the qualified inspector verifies that reading. Id.

Dr. Hopenfeld testified that he believed that a one inch grid for component inspection would take care of all uncertainties about flow turbulence and the effect of geometric discontinuities. Finding 238. However, under the methodology used at VY the size of the grid is immaterial, since the entire component is scanned. Id.

In the CHECWORKS analysis, the plant is divided into a number of analysis lines, which do not have to be physically connected, but represent components which have the same water chemistry and generally the same temperature. Finding 239. These components are expected to behave in the same way. Id. Depending on the complexity of the reactor itself and the amount of inspection data available, there is typically between 20 and 50 or more of these lines. Id.

In CHECWORKS, analysis lines are used in determining the FAC wear rate of components because the FAC wear rate for any individual component in that line is relatable to any other component in that line because components in a line will have the same dissolved oxygen, the same pH, the same temperature, the same flow rate. Finding 240. Use of analysis lines allows the user to compare wear rate inspection data from different components in the same analysis line to determine if there is a good comparison. Id. By taking data on an analysis line and comparing inspections, the random scatter in the data is minimized and potential inconsistencies can be identified and investigated. Id.

### c. CHECWORKS Analysis Process

Based on the user inputs, an initial "Pass 1 Analysis" is conducted to report predicted wear rates. Finding 241. The results of the Pass 1 Analysis, together with other information including operating experience at similar units, are normally used by the FAC engineer to generate a list of components for inspection. Id. Once this information is specified in the plant database, the plant engineers are able to conduct wear rate analyses of any or all of the piping defined in the database. Id.

Although inspection data are not required for a Pass 1 Analysis, inspection data may also be used as inputs into CHECWORKS. Finding 242. Inspection data may be input in the form of a matrix of thickness readings covering the component. Id. Typically, these data sets are from ultrasonic measurements of the wall thickness at local points (i.e., grid points) or from scanning the component and recording the minimum thickness at grid points. Id.

When inspection data are available, a "Pass 2 Analysis" can be run. Finding 243. A Pass 2 Analysis compares the measured inspection results to the calculated wear rates and adjusts the FAC rate calculations to account for the inspection results. Id. The program does this by comparing the predicted amount of degradation with the measured degradation for each of the inspected components. Id. Pass 2 Analyses provide the analyst with the opportunity to evaluate the goodness of fit of the model to actual results, the location of any outliers, and the possibility of modeling improvements. Id. However, none of the algorithms in CHECWORKS are modified by the incorporation of plant-specific data. Id.

Using statistical methods, a correction factor is determined, which is applied to all components in a given pipe line – whether or not they were inspected. Finding 244. A line correction factor is calculated by CHECWORKS for each analysis line in its entirety. Id. The line correction factor is used to fine-tune the results of the CHECWORKS analysis so to improve its predictive ability. Id.

### d. Modeling Changes in Plant Conditions

The use of CHECWORKS does not change as a result of a power uprate or any other change in operating parameters. Finding 245. CHECWORKS was designed to handle changes in plant operating conditions. Id. CHECWORKS can be used to forecast what impact a proposed change in operating conditions will have on FAC wear rates. Id.

When a power uprate is implemented, the user updates the relevant input parameters (e.g., thermodynamic conditions, temperature, oxygen concentration, etc.), and lets the program calculate the predicted FAC wear under the new conditions. Finding 246. With the implementation of the power uprate at VY the only CHECWORKS inputs which affect wear rates that changed were the flow rate and the operating temperature. Id.

The Pass 2 Analysis can be used as a planning tool by performing it in advance of the uprate to determine if, under uprate conditions, systems and sub-systems would experience significantly greater FAC rates than those predicted before the uprate. Finding 247. CHECWORKS was specifically designed to accommodate power uprates and is routinely used throughout the U.S. nuclear industry for this purpose. Id.

Dr. Hopenfeld testified that CHECWORKS would need ten to fifteen years of VY uprate data to calibrate CHECWORKS to reflect uprate operating conditions; Dr. Hausler likewise testified that 12-15 years would be a reasonable estimate of time to calibrate CHECWORKS. Finding 249. However, for purposes of CHECWORKS use, power uprates are no different from other operational changes. Id. The differences in rates experienced in a power uprate are generally smaller than those experienced by units when their water chemistry changes. Id. It has never been necessary to "re-calibrate," "re-baseline" or "benchmark" CHECWORKS when plants have changed their water chemistry, power output has been increased, or other operational changes have taken place. Id.

Dr. Hopenfeld is of the view that, in order to establish the rate of FAC, many years of inspection data would be needed, including inspection of every potentially susceptible run of piping three times over five inspection periods. Finding 250. However, the development of CHECWORKS and the EPRI guidance have eliminated the need for such an approach. Findings 249 and 250. It is not necessary to have ten to fifteen years of inspection data collected after a power uprate for an effective FAC Program. Finding 250. The new values for flow rate and temperature are simply used as inputs into CHECWORKS and CHECWORKS provides FAC rate calculations for the modeled components under the uprated conditions. Id. Locations that CHECWORKS shows as having the highest wear rates are typically those with the most tortuous geometry, such as around control valves, across reducers, and downstream of valves. The effect of such geometric discontinuities does not change with a power uprate. Id. Because only the flow rate and temperature are changed at VY due to the power uprate, any FAC rates established after the uprate will be constant and the effect of the uprate on FAC will, therefore, be apparent with the first inspection after the uprate. Id.

Plant operational data used for the correlations in CHECWORKS involves approximately thirty different units. Finding 251. VY, under power uprate conditions, is a fairly small plant in terms of power level compared to the other units. Id. CHECWORKS covers the range of conditions at operating light water reactors, including VY. The change at VY with the power uprate is primarily one of velocity, and the maximum velocity is comparable to that at any number of other nuclear power plants. Id. The algorithms used to predict the FAC wear rate are based on extensive laboratory and plant data, including data on FAC wear rates where the flow rate and the temperature exceed those present at VY after the uprate. Id. This assures that the FAC wear rates predicted by CHECWORKS are accurate. Id.

**e. CHECWORKS and Quality Assurance**

CHECWORKS is not used for nuclear design but only to provide information to FAC engineers. Finding 251. The information used by CHECWORKS is not directly used for functions covered by "nuclear level" quality assurance. Id.

### 3. Use of NSAC-202L and CHECWORKS in the VY FAC Program

#### a. NSAC-202L

Dr. Horowitz played a key role in drafting the original version of NSAC-202L and in each of the three subsequent revisions to NSAC-202L. Finding 252. NSAC-202L has become the most important standard-setting document for the conduct of FAC control programs in the United States, and has also been accepted as a valuable guidance tool by INPO and the NRC. Id.

The original VY FAC Program was instituted prior to the issuance of EPRI's guidance document, NSAC-202L. Finding 253. However, the FAC Program's documents have been revised as necessary over time to conform to the recommendations in the various revisions to NSAC-202L. Id. The VY FAC Program currently in effect substantially follows the current version of NSAC-202L. Id.

#### b. Use of CHECWORKS in the VY FAC Program

VY uses five criteria for selecting which components and locations will be inspected for potential FAC effects during a plant refueling outage. Finding 254. Those factors, which are consistent with the guidance in NSAC-202L, are: (1) pipe wall thickness measurements from past outages; (2) predictive evaluations performed using the CHECWORKS computer code; (3) industry experience related to FAC; (4) results from other plant inspection programs; and (5) engineering judgment. Id.

Currently, the FAC Program at VY primarily uses CHECWORKS as a tool in planning inspections, evaluating inspection data, and managing the ultrasonic thickness ("UT") data compiled over the past thirteen refueling outages at Vermont Yankee. Finding 256.

#### c. Updating VY Plant Data for Use in CHECWORKS

NSAC-202L, Rev. 2 does not specify a specific interval for model updates. It merely states: "It is recommended that whenever possible, the Predictive Plant Model utilize the results of wall thickness inspections to enhance the FAC predictions. In CHECWORKS this is called Pass 2 analysis." Finding 257.

All applicable inspection data were updated for VY during the Summer and Fall of 2000. Finding 258. Additional updates were performed for the feedwater system in 2003. Id. In addition, inspections performed in 2001, 2002, 2004, and 2005 showed that the wear rates predicted by the CHECWORKS model were consistently conservative. Id.

Inspection data were not entered into CHECWORKS immediately following inspection outages in 2004 and 2005. Finding 259. Mr. Fitzpatrick wrote condition reports regarding the failure to update the CHECWORKS model with inspection data. Id. These condition reports were intended to identify to management the need for additional human resources. Id. VY now has a dedicated FAC engineer whose job is to keep the FAC program current. Id.

The inspection planning and component selections made during the outages where inspection data had not been entered into the CHECWORKS model were based in part on the conservatively high wear rates previously predicted by CHECWORKS. Finding 260. The CHECWORKS update performed in 2006 confirmed again that the previously predicted wear rates were conservative. Id. Runs of the updated model did not identify any instance where recommended inspections were not performed. Id.

Comparison of the CHECWORKS predictions with subsequent inspection data have uniformly shown that the CHECWORKS predictions are conservative (i.e., they predict higher wear rates than those observed during the inspection). Finding 261. Thus, even if the most recent inspection data had not been entered into the CHECWORKS program, the result would have been over-estimation of FAC wear. Id. The condition report written by Mr. Fitzpatrick concludes, therefore, that not updating the CHECWORKS database with the most recent inspection data was not necessary in order to determine the appropriate scope of the RFO 25 inspection. Id.

VY updated the version of CHECWORKS it used from CHECWORKS FAC 1.0D to CHECWORKS FAC 1.0F in 2000. Finding 262. Version 1.0F was used for the 2003 and 2006 model updates. CHECWORKS FAC 1.0G was installed in 2006. Id. There were no differences in versions 1.0D, 1.0F, and 1.0G with respect to water chemistry and wear rate predictions for BWRs. Id.

#### d. FAC Inspections Since VY Uprate

The scoping process for the FAC inspection in the 2007 refueling outage ("RFO 26") started before RFO 25 was complete. Finding 263. The RFO 26 scoping was performed using the same criteria as contained in Section 5.3 of ENN-DC-315, Rev.1. Id.

As an added measure of conservatism, Entergy is increasing the FAC inspection scope by at least 50% for the first three outages following the EPU. Finding 264. In 2005, in RFO 25, the last refueling outage prior to the EPU, a total of 35 FAC inspections were performed, including 27 large bore inspections. Id. In RFO 26, the first outage since the EPU, the inspection scope was increased by more than 50%, as there were a total of 63 inspections performed, including 49 large bore inspections. Id. These additional inspections provide further confirmatory data points for the use of the FAC Program. Id.

The results of the 2007 FAC inspection demonstrate that data from repeat inspections (before and after the uprate) of large bore components in the feedwater system show that essentially no wear has occurred since the commencement of the EPU in March 2006. Finding 265. Because no significant increase in wear was shown in the first post-uprate inspection, there is a high level of confidence that no significant change in the rate of wear will be found in the next two inspections. Id.

#### e. Use of CHECWORKS after License Renewal

After license renewal, Entergy will continue to use the CHECWORKS program to assist in identifying the locations where piping inspections should be performed. Finding 266. Data collected at VY since 1989 and in the three sets of inspections that will be conducted during refueling outages between the implementation of the EPU and the expiration of the current license will be sufficient to use CHECWORKS effectively. Id. Those inspections will yield data for four and a half years of operation at the EPU levels. Id.

### f. VY FAC Program Quality Assurance Issues Raised by NEC

VY Quality Assurance Audit No. QA-8-2004-VY-1 of the VY FAC Program (Entergy Exh. E4-26) resulted in the issuance of two condition reports ("CRs") against the program: for (1) not getting inspection data into the data management system on time; and (2) not finalizing the draft outage inspection report. Finding 267. Despite these CRs, the Audit report states that "[n]one of the findings or areas for improvement, individually or in the aggregate, were indicative of significant programmatic weaknesses which would impact the overall effectiveness of the Engineering Programs assessed." Id.

With respect to the CR on inspection data not being put into the data management system on time, the data were in fireproof cabinets, but not into the record management system (i.e., microfilmed). Finding 268. The data were in the programs that the VY FAC Program used to trend wear. Id.

The second CR was written because a draft outage report had not been timely finalized. Finding 269. The inspection scoping for the next outage was based on the draft report. Id. The CHECWORKS model at that time was not being updated based on the conservative wear rates and the inspection data were showing no wear. Id. Even though the inspection data were not input into the CHECWORKS model, inspections were conducted, data were evaluated, and component wear rate was trended (based on inspection data), all in accordance with the FAC Program. Id.

Mr. Witte testified that in 2004 at least four VYNPS components, including the condensate system and the extraction steam systems, were determined to have "negative time to Tmin," meaning that wall thinning was being predicted as beyond operability limits and should be considered unsafe with potential rupture at any time. Finding 270. VY Scoping Worksheets developed in preparation for the 2004 refueling outage included four components that had "negative times to Tmin." This is a theoretical conclusion based on the results of CHECWORKS, and is not based on actual inspection data. Id. As such, there would be no need to write condition reports with respect to those results. Id. Condition reports are written when inspection data indicate there is an actual problem, and additional inspections are then performed as corrective actions. Id. In any event, of the four items identified in the 2004 Scoping Worksheets as having negative times to Tmin, three were made of FAC-resistant material. The only FAC susceptible component in the list (CD30TE02DS) was inspected and determined to meet design requirements with significant margin. Id.

Mr. Witte testified that the 2006 cornerstone report shows a number of indicators as yellow, with lists of open CR corrective actions, and a new CR written in August 30, 2006. The report lists six corrective actions and four CRs that were written as early as 2003 that remain open. Finding 271. The referenced report, "Cornerstone Rollup," shows the overall FAC Program status as Green. Id. The report rates twenty-seven different areas. Of these, only two were rated as "Yellow," signifying that an action item is more than one year old. Id. Six LO-VTYLO action items are listed. These are not condition reports, nor are they corrective actions from condition reports. They are commitments without safety significance; the items listed are for completion of program administrative tasks. Id.

4. Conclusions to be drawn from the evidence

The evidence shows that Entergy has instituted a program, in effect as part of the plant's current licensing basis and to be continued after renewal of the VY license, to continuously monitor FAC wear. This program is conducted under industry and NRC accepted standards and run by a dedicated FAC engineer. Inspection locations are selected in accordance with industry standards based on several factors, including past inspection data, plant operating experience and predicted FAC wear rates from CHECWORKS. CHECWORKS is an industry-standard analytical tool that is based on a large amount of plant operational data and laboratory data and has been demonstrated to be an effective planning tool throughout 20 years of plant operating experience. CHECWORKS' correlations are based on data from over 30 plants, many of which have considerably higher power levels than those found at VY after its EPU.

Entergy performs FAC inspections during each refueling outage thorough ultrasonic measurement of the entire component and attached piping that has been selected for inspection. As a measure of conservatism, Entergy has increased by at least 50% the number of components selected for inspection in the three refueling outages following the EPU at VY. The inspection data from the first refueling outage show no increase in FAC wear rates due to the power uprate. In all, the FAC management program that Entergy will implement during the period of extended operation after license renewal will assure that the aging effects of FAC will be adequately managed. For that reason, there is no support for the claims made in NEC Contention 4, which should be rejected.

## V. FINDINGS OF FACT

### A. NEC CONTENTIONS 2A AND 2B

#### 1. <u>Summary of VY's EAF Management Program</u>

1. Metal fatigue is an age-related degradation mechanism caused by cyclic mechanical and thermal stresses at a location on a metallic component. The results of fatigue can be observed in the cracking of components subjected to cyclic stresses of sufficient magnitude and duration. Entergy NEC 2 Dir. at A5.
2. The design specifications for a given safety-related component specify the number of mechanical and thermal cycles that the component is expected to experience during its de-

sign life, and define the safety limits and applicable codes that must be satisfied. For components exposed to the primary reactor coolant pressure boundary, the specified requirements for evaluation of cyclic loading and thermal conditions are contained in Section III of the ASME Code for Class 1 components. Id. at A6.

3. For a Class 1 component, stress cycles from the loadings specified in the governing design specification will produce total stresses of several different magnitudes. The number of times these stress magnitudes occur also varies. The allowable number of cycles for a given alternating stress range is determined from the ASME Code design fatigue curve for the material being evaluated. The fatigue usage for that stress cycle is the ratio of the number of applied stress cycles (n) to the allowable number of stress cycles (N) from the ASME Code design fatigue curve. The CUF for the component is the sum of the individual usage factors for all of the various stress magnitudes. Id. at A7. At any point in time, the CUF for a component represents the fraction of the allowable fatigue cycles that the component has experienced up to that time. Id.

4. ASME Code Section III requires that the CUF for a Class 1 component not exceed unity; that is, the total number of applied stress cycles is not to exceed the allowable number of stress cycles. Id. at A8.

5. For components (equipment and piping) exposed to reactor coolant water, the fatigue life, as measured by the allowable number of stress cycles, may be reduced compared to the components' fatigue life when exposed to an air environment. The ASME Code design fatigue curves were developed based on laboratory testing of specimens in an air environment, with safety factors incorporated into the curves to account for several factors including atmosphere, surface finish, size effects, etc. Laboratory testing of specimens in water under reactor operating conditions indicate that, under certain situations additional environmental factors may need to be included in the calculated CUF to fully accommo-

date reactor coolant environmental conditions. Accounting for the effects of operating in a reactor coolant environment in the fatigue analysis is called environmentally assisted fatigue (EAF) analysis. Id. at A9.

6. To quantify the effect of the reactor coolant environment on component fatigue, the CUF for a component exposed to reactor coolant may multiplied by an adjustment factor or "EAF multiplier", when appropriate environmental conditions exist. This results in an environmentally adjusted CUF, or $CUF_{en}$. The resulting $CUF_{en}$ must still not exceed unity. Id. at A12.

7. Section 4.3.3 of the Application presents Entergy's initial assessment of the effects of the reactor coolant environment on fatigue life for nine plant-specific locations of six reactor components at VY selected in accordance NUREG/CR-6260 (Entergy Exh. E2-04) and the GALL Report (Entergy Exh. E2-05). Joint Stipulation, para. 1.

8. The component locations identified in NUREG/CR-6260 and endorsed by the GALL Report 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. Due to the inclusion of both piping and nozzles, as well as the different materials for the nozzle forgings and nozzle safe ends, a total of nine locations for the six components identified in the NUREG/CR-6260 list above were evaluated for EAF at VY. NRC Exh. 6 at 5-102; Entergy NEC 2 Dir. at A19.

9. The initial $CUF_{en}$s computed by Entergy for VY are presented in Table 4.3.3 of the Application. As that Table shows, seven of the nine locations specified in NUREG/CR-6260 had $CUF_{en}$s greater than unity, and therefore greater than the specified criterion of the ASME Code. Joint Stipulation, para. 2.

10. To address these results, the Application states (Application, Section 4.3.3 at 4.3-7) that, prior to entering the period of extended operation, for each location that may exceed a $CUF_{en}$ of 1.0, VY will implement one of three possible courses of action, including "further refinement of the fatigue analyses to lower the predicted CUFs to less than 1.0." Joint Stipulation, para. 3.

11. This commitment was modified in Amendment 35 to the Application, which states in part as follows:

> At least 2 years prior to entering the period of extended operation, for the locations identified in NUREG/CR-6260 for BWRs of the VY vintage, VY will refine our current fatigue analyses to include the effects of reactor water environment and verify that the cumulative usage factors (CUFs) are less than 1. This includes applying the appropriate $F_{en}$ factors to valid CUFs determined in accordance with one of the following:
>
> > 1. For locations, including NUREG/CR-6260 locations, with existing fatigue analysis valid for the period of extended operation, use the existing CUF to determine the environmentally adjusted CUF.
> >
> > 2. More limiting VY-specific locations with a valid CUF may be added in addition to the NUREG/CR-6260 locations.
> >
> > 3. Representative CUF values from other plants, adjusted to or enveloping the VY plant specific external loads may be used if demonstrated applicable to VY.
> >
> > 4. An analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case) may be performed to determine a valid CUF.

Entergy Exh. E2-09, Attachment 3, Commitment 27; FSER, Appendix A at A8 – A10.

12. Entergy engaged SIA in 2007 to implement Option 1, that is, perform reanalyses to calculate the CUFs, $F_{en}$s and $CUF_{en}$s for all nine locations of interest in accordance with the approach described in the GALL Report. Joint Stipulation, para. 4.

13. Final versions of fifteen calculations comprising the 2007 reanalysis were issued in August and December 2007. Joint Stipulation, para. 5; Entergy Exh. E2-10 through E2-24.

14. The results of the 2007 reanalysis show that the $CUF_{en}$s for the nine limiting piping and vessel locations for the sixty years through VY's extended license period are in all cases less than unity, signifying that component failure due to fatigue will not be a concern at VY during the period of extended operation. Entergy NEC 2 Dir. at A26.

15. Since performance of the 2007 reanalysis has demonstrated that environmentally assisted fatigue will not be a concern during the period of extended operation, it is Entergy's position that no further actions regarding metal fatigue are currently necessary. Id. at A27. Nonetheless, the condition of piping and components will continue to be monitored under the plant's in-service inspection program through the period of extended plant operation. Id. In addition, the VY Fatigue Monitoring Program will continue to track plant cycles and transients to ensure that the numbers of transient cycles experienced by the plant remain within the analyzed numbers of cycles for all transients. Id.

16. If, at some future time, the results of continued monitoring suggest that the evaluations no longer encompass 60 years of plant operation, further refined analyses, submittal of an inspection program for NRC review or replacement of the component in question may become necessary or desirable. Id.

2. Initial EAF Assessment

17. The initial assessment of environmentally assisted fatigue contained in the VY Application consisted of the evaluation of EAF effects for all nine locations in accordance with the provisions of Section X.M1 of the GALL Report by performing $CUF_{en}$ calculations

for the nine locations and demonstrating that the total $CUF_{en}$s for 60 years of plant operation remains less than unity. Entergy NEC 2 Dir. at A19.

18. The initial $CUF_{en}$s computed by Entergy for VY are tabulated in Table 4.3.3 of the Application. As that table shows, seven of the nine locations had $CUF_{en}$s greater than unity, and therefore greater than the specified criterion of the ASME Code. Joint Stipulation, para. 2.

19. In the Application, Entergy used CUFs for the reactor components derived from either the original design reports for the plant or updated analyses prepared in 2003 by General Electric, the plant's vendor, to account for the effects of the proposed power uprate on the CUFs. Tr. at 887-88 (Fitzpatrick).

20. The piping components of interest had been designed using the ANSI B31.1 Power Piping Code, which did not require performing fatigue analyses. Tr. at 888 (Fitzpatrick). For that piping, CUFs were reported in the Application utilizing generic values provided in NUREG/CR-6260 because the VY design basis for the piping under the ANSI B31.1 Code does not require explicit fatigue analyses. Entergy NEC 2 Dir. at A21.

3. <u>Reanalysis and Confirmatory Analysis</u>

21. Entergy engaged SIA in 2007 to perform a reanalysis of the reactor and piping component locations specified by NUREG/CR-6260. The objectives of SIA's reanalysis were twofold: to perform more refined analyses of the locations that showed $CUF_{en}$s greater than unity, and to perform plant-specific analyses for piping, for which the Application analysis had used generic CUF values. Tr. at 892-93 (Stevens).

22. The 2007 reanalysis and the subsequent confirmatory analysis sought to demonstrate the acceptability of the $CUF_{en}$s for the locations at issue, not to calculate the available margin. Tr. at 893 (Stevens). Accordingly, once the analysis had shown the $CUF_{en}$ is less

than one, acceptability had been demonstrated and there would be no need to proceed to calculate what the actual margin is. Id.

23. The 2007 reanalysis and the confirmatory analysis both compute the $CUF_{en}$ for each location in the same three steps used in the initial analysis: (1) the CUF for a component location is calculated, (2) the environmental multiplier, $F_{en}$, is calculated, and (3) the $CUF_{en}$ is calculated as the product of the CUF for the component and the corresponding $F_{en}$. Entergy NEC 2 Dir. at A20.

24. The CUF for a component location starts by constructing a finite element "mesh" model of the component, which was then used to determine the stresses to which each element in the model is subjected. Tr. at 807 (Stevens). The model is developed using ANSYS, a reliable, benchmarked computer code widely used in the nuclear industry. Tr. at 813 (Stevens). The locations at which the plant-specific VY analyses are performed do not have discontinuities that could lead to higher localized stresses were a finer mesh to be selected for the modeling. Tr. at 814 (Stevens).

25. The next step in the analysis is to identify the pressure and temperature transients to which each component will be subjected during VY's operation. The magnitude and frequency of each transient are given by the design specifications, and have been proved to be conservative on the high side (i.e., overestimate the severity of the temperature and pressure fluctuations and the frequency of occurrence of the transients). Tr. at 820-21 (Stevens).

26. For each type of transient, the analysis develops a time history of the temperature and pressure stresses imparted on the component location during the transient. The magnitudes of those stresses are added linearly so as to develop a stress time history for that type of transient. Tr. at 807-08 (Stevens).

27. Since the fatigue of a component is due to stress fluctuations and the order of occurrence of the transients is not known a priori, the ASME Code conservatively requires that transients be "paired" so as to create the largest possible stress fluctuations, and thus a transient resulting in the highest stresses is paired with (followed by) a transient with the lowest stresses. Tr. at 810 (Stevens). This is done until all the available transient pairs have been exhausted, taking into account the relative frequency of occurrence of each transient. Tr. at 819-20, 825-29 (Stevens). That process results in the most conservative estimate of the stress fluctuations to which a component will be subjected by effectively assuming the worst possible ordering of the transients. Tr. at 817-18 (Stevens).

28. Each set of stress fluctuations results in a number of fatigue cycles being imparted on the component at a given location. This number of applied cycles is used to compute a fraction of the component's fatigue life using the number of cycles that are allowed by the ASME Code for the material at each stress level. The number of allowable cycles at each stress level is determined using an ASME Code "S-N" (fatigue) curve. Tr. at 834 (Stevens). The addition of all fractions for all stress levels experienced by the component yields the cumulative usage factor, or CUF, for the component at the location of interest. Tr. at 834-36 (Stevens).

29. The overall methodology is standard in the industry and has been in place for over 30 years, so there is no disagreement as to its appropriateness. Tr. at 833 (Hopenfeld).

30. The process leading to the computation and addition of stresses is very labor intensive, requiring approximately twelve to fourteen man-weeks for each reanalysis calculation. Tr. at 916 (Stevens). The confirmatory analysis took less time, on the order of nine weeks, because it could utilize some of the inputs generated in the reanalysis. Id.

31. The ASME Code requires that the CUF be less than unity. Tr. at 837 (Stevens). This is a criterion for acceptability established by the Code, but exceeding the criterion does not

mean the component will fail, given the number of factors of conservatism included in the analytical process. Tr. at 837, 895 (Stevens). A CUF of unity means that there is a 1 to 5 percent probability that a small crack, 3 millimeters deep, may have formed on the component. Tr. at 901-02 (Fair).

32. The S-N curves provided in the ASME Code are established based on strain testing of various material specimens in air at room temperature. Tr. at 896 (Stevens). To translate the results obtained from those curves to the fatigue limits for the materials in a reactor water environment, it is necessary to apply correction factors known as $F_{en}$s to compute the environmentally adjusted fatigue cumulative usage factors when appropriate. Tr. at 838-39 (Stevens).

33. In addition to the conservatisms embedded in the methodology for computing CUFs, the VY EAF calculations in the 2007 reanalysis incorporated a number of VY-specific conservatisms, including:

a. The numbers of transient cycles for 60 years used in the calculations is conservative relative to the numbers of transients expected to occur through 60 years of operation. Entergy NEC 2 Dir. at A30; Tr. at 850 (Stevens).

b. The calculations use conservative, design basis transient pressure and temperature fluctuation definitions, as opposed to the less abrupt changes experienced during actual plant transients. Entergy NEC 2 Dir. at A30; Tr. at 851-54 (Stevens).

c. The calculations use bounding values for pressure, temperature and flow rate at EPU conditions for the entire 60-year period of plant operation, instead of using the actual conditions that existed during the first thirty-four years of plant operation before the uprate. Entergy NEC 2 Dir. at A30; Tr. at 855 (Stevens). This choice of parameters results in having the transients that occurred before EPU implementation being treated as more severe than they in reality were. Tr. at 855-56 (Stevens).

d. The calculations obtain bounding $F_{en}$ multipliers through the use of temperature, strain rate and sulfur content values selected to maximize the $F_{en}$ multipliers. Entergy NEC 2 Dir. at A30.

34. SIA Report No. SIR-07-132-NPS, "Summary Report of Plant-Specific Environmental Fatigue Analyses for the Vermont Yankee Nuclear Power Station" (Revision 1, dated December 2007) provides summaries of the methodology used in the 2007 reanalysis and the results of the calculations comprising the reanalysis. Id. at A32; Entergy Exh. E2-24. The results of the analyses, as summarized in Table 3-10 of that report demonstrate that the environmentally adjusted fatigue usage factors for all locations and components analyzed remain within the allowable value of 1.0 through 60 years of VY operation. Entergy NEC 2 Dir. at A33; Entergy Exh. E2-24 at 3-18, Table 3-10.

35. Upon review of Entergy's 2007 reanalysis, the NRC Staff determined that the overall approach proposed by Entergy was acceptable and found that the performance of the reanalysis calculations would be in conformance with the recommendations in both the Standard Review Plan for License Renewal and in GALL Report Section X.M1. FSER, Section 4.3.3.2 at 4-32 through 4-38.

36. The Staff had no concerns with the reanalysis calculations prepared by Entergy for the six piping and component locations that did not involve nozzle corners. Id.

37. For those six locations, Entergy's reanalysis had consisted of performing conservative calculations based on ASME Code methodology. For example, the piping locations were evaluated with ASME Code Section NB-3600 methodology for piping, which accounts for stresses on the piping in a conservative fashion. Tr. at 926 (Stevens). Those locations could be shown to have acceptable $CUF_{en}$s without using many analytical refinements. Tr. at 928, 947 (Stevens).

38. At the hearing, NEC witness Dr. Hopenfeld acknowledged he was not familiar with the computer code used for the $CUF_{en}$ calculations for these six locations. Tr. at 1101-02 (Hopenfeld).

39. The Staff asked Entergy to explain how the stress intensity for thermal transients (including shear stresses) was calculated for the analyzed components and locations in the reanalysis calculations. Entergy explained that, in most cases, shear stresses are negligible for thermal transients for cylindrical components like those used in reactor pressure vessels and piping. The Staff, however, took the position that shear stresses cannot always be neglected in the calculation of stress intensities used to determine CUFs of all locations, and that while it is appropriate to do so for locations where non-symmetric loadings are not significant, neglecting shear stresses for locations with significant geometric discontinuity, such as nozzle corners, could lead to non-conservative results. These Staff concerns were applicable to locations at the blend radius (nozzle corner) regions of three reactor pressure vessel components evaluated in the refined calculations: the feedwater nozzle, the recirculation outlet nozzle, and the core spray nozzle. FSER, Section 4.3.3.2 at 4-38 through 4-40.

40. At the three nozzle corner locations, the reanalysis had used a single stress difference component as output from the Green's Function analysis to generate stress difference histories for all transients. Tr. at 926, 928-29 (Stevens). The Staff felt that this simplification might introduce potential non-conservatisms in the computation of CUFs, specifically in the nozzle corner regions, compared to the use of all six stress components of the stress tensor. FSER, Section 4.3.3.2 at 4-38.

41. To resolve the Staff's concerns associated with the use of a simplified single stress difference, Entergy proposed, and the NRC Staff accepted, that Entergy perform a confirma-

tory $CUF_{en}$ analysis of the feedwater nozzle corner using methods that would be acceptable to the NRC. Joint Stipulation, para. 6.

42. The feedwater nozzle was selected for analysis because (1) it is the limiting nozzle (i.e., has the highest $CUF_{en}$) in the VY refined evaluations among the three nozzles regarding which the Staff had questions, (2) it is subjected to more transients and cycles than the other two nozzles, and (3) the transients it experiences are more severe than the transients experienced by the other two nozzles. Entergy NEC 2 Dir. at A38.

43. The Staff agreed to this suggestion and to the choice of the feedwater nozzle corner as the limiting location of interest. FSER Section 4.3.3.2 at 4-40 – 4-41.

44. The confirmatory analysis used the same finite element model, thermal transient definitions, numbers of transient cycles, and water chemistry inputs as the 2007 reanalysis performed by SIA for Entergy, but differed from the reanalysis in several respects. Entergy NEC 2 Dir. at A38. In particular: (1) while both methods use a detailed finite element model of the feedwater nozzle, when the thermal transient stress histories were determined, the confirmatory analysis computed 6-component stress histories for each transient, whereas the refined analysis used a simplified single stress component difference to obtain the stress time history for all of the transients; (2) in the confirmatory analysis, six stress components are paired and combined to obtain maximum stress intensity ranges for all evaluated transients. In the reanalysis, on the other hand, only the maximum stress component difference, which is essentially equal to the stress intensity computed from the finite element program, is used with Green's Function techniques to estimate stress intensity histories for all transients. (3) In the confirmatory calculation, a maximum $F_{en}$ is computed for each incremental stress load-pair, which constitute the paired transient stress state points into which the applied loading history is separated for calculating the CUF. Each $F_{en}$ value is based on the maximum transient temperature unique to each load

pair, and the contributions of all load pairs are added to produce a composite $CUF_{en}$. In the reanalysis, on the other hand, a single, maximum $F_{en}$ is applied to the total CUF resulting from all load pairs, and is based on the maximum transient temperature for all load pairs. This $F_{en}$ selection technique is more conservative than the approach used in the confirmatory calculation, as it results in a single bounding $F_{en}$ value. Entergy NEC 2 Dir. at A39.

45. The bounding $F_{en}$ value used for the feedwater nozzle corner in the reanalysis was 10.05. Id. at 40.

46. The methodology and results of the confirmatory analysis are presented in three calculations (Entergy Exh. E2-25 through E2-27). The feedwater nozzle EAF evaluation was performed for the two controlling locations on the nozzle, the inside surface of the nozzle blend radius (nozzle corner) and at the inside surface of the nozzle safe end. The confirmatory calculation yielded a CUF (before application of $F_{en}$ factors) of 0.089 at the nozzle corner, versus a CUF of 0.064 at the same location using the refined analysis methodology (Entergy Exh. E2-27, Section 4.0 and Entergy Exh. E2-24, Table 3-10). The corresponding results for the safe end location of the nozzle showed a reduction in the computed CUF when using the confirmatory analysis methodology versus the results of the reanalysis for that location. Entergy NEC 2 Dir. at A40.

47. The forty percent (0.064 versus 0.089) difference between the CUFs for the feedwater nozzle corner computed using the reanalysis methodology versus that used in the confirmatory analysis could be due to a number of factors besides the simplifications inherent in the Green's Function approach. Tr. at 936-37 (Stevens).

48. The confirmatory calculation yielded an environmentally adjusted $CUF_{en}$ of 0.353 for the nozzle corner, significantly below the acceptable ASME Code limit. Entergy Exh. E2-27 at 7, Table 1; Tr. at 945 (Stevens). The reanalysis calculation, on the other hand, yielded

an environmentally adjusted CUF of 0.639 for the nozzle corner, which was higher than the confirmatory analysis result but still significantly below the acceptable limit of 1.0. Entergy Exhibit E2-24, Table 3-10.

49. The Staff requested that Entergy compute the $CUF_{en}$s using the confirmatory analysis methodology and the single, limiting value of $F_{en}$ correction factor used in the reanalysis (instead of using individual, applicable $F_{en}$s for each transient pair). Using the bounding $F_{en}$ value of 10.05, the confirmatory analysis results increased to 0.893, still below unity. FSER Section 4.3.3.2 at 4-42; Tr. at 938 (Fair); Entergy NEC 2 Dir. at A41.

50. After review of the confirmatory analysis results for the feedwater nozzle locations, the Staff found that Entergy correctly applied the ANSYS finite element software; used appropriate input parameters; added the stresses correctly; and applied proper $F_{en}$ factors for each transient, so that the results of the confirmatory analysis were appropriate and acceptable. Entergy NEC 2 Dir. at A41; FSER Section 4.3.3.2 at 4 – 41 through 4 – 43. NEC witness Dr. Hopenfeld agreed that the confirmatory analysis methodology had resolved the issues arising from the use of the Green's Function methodology. Tr. at 935 (Hopenfeld).

51. The Staff imposed a license condition requiring similar confirmatory analyses for two other nozzles, the recirculation outlet nozzle and the core spray nozzle. FSER Section 4.3.3.2 at 4 – 41 through 4 – 43. Entergy is to submit these analyses to the Staff no later than two years prior to the start of the period of extended operation, in March 2012. Joint Stipulation, para. 7.

52. The Board finds that it is acceptable to perform these two confirmatory analyses after the VY license renewal decision is reached for several reasons. First, there is no requirement to address the effects of coolant environment on component fatigue life prior to license renewal. Entergy NEC 2 Dir. at A43; Entergy Exh. E2-03 at 1. Second, the methods for

performing the calculations have been defined, and the required performance of the calculations two years prior to the license renewal period ensures that the environmental effects are addressed prior to entering the period of extended operation. Entergy NEC 2 Dir. at A43. Third, because the $CUF_{en}$s at those two nozzle corner locations from Entergy's reanalysis are very small (0.084 for the recirculation outlet nozzle and 0.167 for the core spray nozzle), and based on comparisons to the feedwater nozzle, which is the limiting component, it is extremely improbable that the results of the confirmatory analyses of these two nozzles would yield $CUF_{en}$s greater than unity. Id.; Entergy Exh. E2-09, Amendment 35 to Application, Attachment 1.

53. There is no practical difference between the approaches in the reanalysis and confirmatory analyses because they both yield conservatively calculated $CUF_{en}$s for all nine limiting piping and vessel locations that are well within the acceptable limit. Thus, regardless of what method one chooses to apply, the conclusion is the same – the critical reactor components will not experience failure due to fatigue during the period of extended operation. Entergy NEC 2 Dir. at A44.

4. Issues raised by NEC

54. NEC asserts the following "errors" in Entergy's 2007 reanalysis and confirmatory analysis: (1) Entergy used "outdated" statistical equations to calculate the $F_{en}$ parameters, and should have used instead the results in the 2007 guidance document NUREG/CR-6909 (NEC Exh. at 10-12); (2) Entergy failed to account for factors that affect the values of the $F_{en}$ parameters (Hopenfeld Reb. at A5); (3) Entergy has not provided proof that the base metal of the feedwater nozzles is not cracked (NEC-JH_03 at 15-16); (4) Entergy used inappropriate heat transfer equations to calculate the thermal stress for each transient (id. at 12-15); (5) the number of plant transients estimated to occur during the operating life of VY is not sufficiently conservative (id. at 16); (6) Entergy's calculation of the $F_{en}$ pa-

rameters does not appropriately account for oxygen concentrations and resulting changes in water chemistry (id. at 16-17); and (7) Entergy failed to perform an error analysis on its calculations (id. at 18). In addition, NEC criticizes the 2007 reanalysis because it uses a simplified Green's Function methodology, which allegedly results in "the underestimation of CUF values by approximately 40%" (id. at 18-19).

**a. Use of NUREG/CR-6909 Methodology**

55. Entergy performed its reanalysis and confirmatory analysis following the criteria and methodology for performing EAF analyses specified in Section X.M1 of the GALL Report (Entergy Exh. E2-05). The methodology is comprised of three steps: (1) the CUF for a component is calculated in accordance with ASME Code guidance; (2) the environmental multiplier, $F_{en}$, is calculated in accordance with the guidance in NUREG/CR-6583 (Entergy Exh. E2-06) for carbon and low alloy steels, and in NUREG/CR-5704 (Entergy Exh. E2-07) for stainless steels; and (3) the $CUF_{en}$ is calculated as the product of the CUF for the component and the corresponding $F_{en}$. Entergy NEC 2 Dir. at A20.

56. NUREG/CR-6909, issued in February 2007, incorporates additional information from an expanded database with respect to the fatigue behavior of stainless steels, and presents revised fatigue curves for calculating CUFs in air that are somewhat different from those contained in the ASME Code. Tr. at 787-88, 791-92 (Fair); NUREG-1.207 (Staff Exh. 13) at 2. The NRC Staff has approved the use of NUREG/CR-6909 in licensing of new reactors, but has not required its use in fatigue analyses for operating plants or plants undergoing license renewal. Tr. at 794-95 (Fair); Staff Exh. 13 at 6.

57. One reason the Staff has not required utilization of the NUREG/CR-6909 methodology for operating plants or plants seeking renewal of their licenses is that application of the NUREG/CR-6909 methodology would produce less conservative results (i.e., lower $CUF_{en}$ estimates) than use of existing ASME Code fatigue curves coupled with the meth-

odology in NUREG/CR-6583 and NUREG/CR-5704. Tr. at 795-96 (Fair); Entergy Exh. E2-31 at 96-97.

58. A set of calculations performed by Entergy prior to the hearing confirmed the Staff's assessment that use of the NUREG/CR-6909 methodology would produce less conservative EAF estimates than those obtained using the ASME Code fatigue curves and the methodology in NUREG/CR-6583 and NUREG/CR-5704, as employed in Entergy's reanalysis and confirmatory analysis. Tr. at 797-805 (Stevens).

59. Entergy's $CUF_{en}$ recomputation started with the stress results for the 2007 reanalysis, replaced the fatigue curves and the $F_{en}$s used in the reanalyses with ones computed following NUREG/CR-6909, and obtained new $CUF_{en}$s that incorporated all the NUREG/CR-6909 methodology. Tr. at 802 (Stevens).

60. Entergy recomputed the $CUF_{en}$s for all nine locations specified in NUREG/CR-6260 by applying both the revised air curves set forth in NUREG/CR-6909 and the methodology for computing $F_{en}$ factors for carbon, low alloy and stainless steel provided in that NUREG. Tr. at 799-800, 1174 (Stevens). The values of $CUF_{en}$ computed following NUREG/CR-6909 guidance are in every case (i.e., at all nine locations) less than unity and lower than the corresponding values obtained in Entergy's reanalysis using the ASME Code air curves and the methodology in NUREG/CR-6583 and NUREG/CR-5704. Tr. at 800-01, 1175 (Stevens); Entergy Exh. E2-24 at 3-18, Table 3-10.

61. The computation of the $CUF_{en}$s that Entergy performed prior to the hearing using NUREG/CR-6909 methodology was completed in eight man-hours because it involved only taking the computed stresses from the reanalysis and using the NUREG/CR-6909 fatigue curves to recompute the CUFs and $CUF_{en}$s. Tr. at 932-33 (Stevens).

### b. Factors Affecting $F_{en}$ Parameter Values

62. While a number of environmental factors may affect the potential fatigue of components subjected to a reactor coolant environment, the most significant ones as reflected in laboratory data are the strain rate, the presence of dissolved oxygen in the coolant, the fluid temperature, and the sulfur content in the material. Tr. at 952 (Stevens). All of those factors were taken into account in the Entergy reanalysis and confirmatory analysis. Entergy NEC 2 Dir. at A51. The Staff confirmed that Entergy properly considered dissolved oxygen, strain rate, temperature and sulfur content in calculating Fens. FSER, Section 4.3.3.2 at 4-41 to 4-42. The Staff also verified that values of strain rate, temperature, and sulfur content used in the calculation of $F_{en}$s would remain valid for the period of extended operations. FSER at 4-42.

63. NEC, on the other hand, contends that as many as thirteen factors need to be considered, and Entergy has failed to properly (or at all) account for them. Hopenfeld Reb. at 4-6, Table 1.

(1) Strain Rate

64. When a component is subjected to cyclic strains, the oxide layer that protects the base material may be cracked, thereby exposing the base material to the environment. If the rate of strain application is sufficiently high, the underlying material may not be affected. However, if the rate of strain application is low, the material is exposed to the environment for a longer period of time. Once above a strain threshold value, the environmental effect no longer depends on the magnitude of the strain but rather on how long the base material is exposed to the environment. Tr. at 1036-37 (Fair). Entergy used a value of strain rate in its $F_{en}$ computations that maximized the $F_{en}$ values. Tr. at 1034-35 (Stevens). NEC did not disagree with Entergy's treatment of the strain rate. Tr. at 1037-38 (Hopenfeld).

(2) Dissolved Oxygen in Feedwater

65. Increased dissolved oxygen in the feedwater is detrimental to carbon and low alloy steels in terms of inducing fatigue. Lower dissolved oxygen levels are more detrimental for stainless steels. Tr. at 954 (Stevens).

66. Entergy investigated the variability in dissolved oxygen in the feedwater from plant data, including water chemistry excursions, over a thirteen year period, and obtained a mean plus one sigma value for dissolved oxygen to which each component of interest is exposed. Attachment 2 to Amendment 35 to the Application (Entergy Exh. E2-09); Entergy NEC 2 Dir. at A56; Tr. at 972 (Fitzpatrick). Entergy then provided those bounding values to SIA for use in the $CUF_{en}$ analyses. Tr. at 953 (Stevens). These values range from 40 to 128 parts per billion ("ppb"). Entergy Exh. E2-12, Table 1, p. 14; Tr. at 995 (Fitzpatrick).

67. NEC claims that, in calculating the $F_{en}$ parameters, Entergy did not properly account for unanticipated increases in oxygen levels during transients. NEC Exh. NEC-JH_03 at 16-17; Tr. at 970 (Hopenfeld). Dr. Hopenfeld's opinion that dissolved oxygen levels increase during transients is based in part on a figure in an EPRI Report (NEC Exh. NEC-JH_64 at p. 4-18, Fig. 4-6) that he interprets as showing that $F_{en}$s can rise to levels of 80 or higher due to high oxygen levels in the feedwater during plant transients. Tr. at 985 (Hopenfeld). However, Mr. Stevens (the author of the EPRI Report introduced into evidence as NEC Exh. NEC-JH_64) explained that the portion of the curve whose values Dr. Hopenfeld cites was intended to show the range of possible variations in $F_{en}$ and the higher values of $F_{en}$ apply to conditions of high oxygen levels and low strain rates that do not exist at VY. Tr. at 986-87 (Stevens). Moreover, the report makes the recommendation that bulk oxygen levels should be time-averaged before they are used as inputs to the fatigue analysis, which is the approach that Entergy took in its EAF analyses. Tr. at 1003 (Stevens).

68. Plant data indicate that the oxygen concentration does not vary significantly during transients. Tr. at 973-74, 990 (Fitzpatrick). In addition, the transients where increased oxygen concentration could be observed (startup and shutdown) are very small contributors to the total CUF. Tr. at 989 (Fitzpatrick); Tr. at 1005 (Stevens). Dr. Chang agreed with this assessment, observing that "no significant thermal transients occur during [the startup and shutdown] period so that practically no fatigue usage factor is accrued during this period." NRC Staff Exh. 2 at 14.

69. Another factor cited by Dr. Hopenfeld in support of his conclusion that high levels of dissolved oxygen occur during transients is "plain physics," by which he means that when the temperature of the coolant goes down, the solubility of gases like oxygen increases. Tr. at 991 (Hopenfeld). However, for the transients of interest, the feedwater is not in a saturated condition. When the temperature decreases, the vessel is still pressurized such that the solubility does not decrease. Tr. at 1034 (Fitzpatrick).

70. Also, if the transient is very rapid, there is a concurrent increase in strain rate that may cancel the effect of the increase in oxygen level. Tr. at 1002, 1034 (Fitzpatrick).

71. Finally, Dr. Hopenfeld cites the recommendation in NUREG/CR-6909 (Entergy Exh. E2-30) at A5 that "A value of 0.4 ppm [400 ppb] for carbon and low-alloy steels and 0.05 ppm [50 ppb] for austenitic stainless steels can be used for the [dissolved oxygen] content to perform a conservative evaluation." Tr. at 996-97 (Hopenfeld). Mr. Fair, however, clarified that the 400 ppb value was used in NUREG/CR-6909 as a default value if no plant-specific data are available. Tr. at 997-98 (Fair). There is no need to apply this default value at VY, since site-specific oxygen data exist for the plant.

72. NEC further claims (NEC Exh. NEC-JH_03 at 17) that VY uses the EPRI-BWVRIA computer code to assess oxygen concentrations and has not described how that code was benchmarked against plant data. However, the BWRVIA model was benchmarked by fit-

ting it to chemistry data for one plant and then comparing the model with chemistry data from six other plants. Entergy NEC 2 Dir. at A57.

73. NEC also claims that fatigue is sensitive to electrochemical potential, but its witness Dr. Hopenfeld acknowledges that electrochemical potential cannot be measured at a plant in all applicable locations and one must resort to dissolved oxygen measurement as an alternative. Tr. at 963-64 (Hopenfeld). The $F_{en}$ expressions used by Entergy, and documented in NUREG/CR-6583 and NUREG/CR-5704, are not dependent on electrochemical potential. Entergy NEC 2 Dir. at A56.

(3) Surface Finish

74. In the development of the ASME design curves, an adjustment factor of 4 was applied to account for the difference between the specimens tested in the laboratory, which were mirror polished specimens, and the components actually installed in the plant. Tr. at 1088-89 (Stevens); Entergy Exh. E2-30 at 76, Table 12. In developing the revised fatigue air curve in NUREG/CR-6909, this adjustment factor was scaled down to a range between 2.0 and 3.5. Tr. at 1079 (Fair). In its $CUF_{en}$ calculations, Entergy used the more conservative 4.0 adjustment factor, since the ASME Code fatigue curves were used to calculate CUF values. Tr. at 1079 (Stevens).

75. The surface finish adjustment is included in the ASME Code design curve, so it does not need to be duplicated when computing the $F_{en}$. Tr. at 1089 (Stevens).

(4) Coolant Temperature Below 150°C

76. When the temperature of the coolant drops below 150°C, the value of $F_{en}$ drops to a constant value, becoming a number close to two for carbon steel and just over two for stainless steel. Entergy Exh. E2-30 at A.1 – A.2; Tr. at 1028 (Fair).

(5) Other Environmental Factors

77. NEC provided a list of environmental factors that it alleges affect the $F_{en}$ values and were not taken into account in Entergy's $F_{en}$ analyses. The list includes, in addition to the previously discussed strain rate, dissolved oxygen, surface finish, and coolant temperature below 150°C, eight other factors which Dr. Hopenfeld regarded as less significant: data scatter, size, flow rate, heat to heat variations, loading history, cyclic strain hardening, trace impurities in water, and sulfide morphology. Tr. at 1013 (Hopenfeld); NEC Exhibit NEC-JH_63 at 4-6, Table 1. (Dr. Hopenfeld also includes potential feedwater base metal cracks in his list; that issue is discussed separately below.)

78. Entergy testified that all but one of those eight factors were either directly or inherently included in its $CUF_{en}$ reanalysis and confirmatory analysis, as well as in the NUREG/CR-6909 guidance. Tr. at 1094 (Stevens); see also Entergy NEC 2 Dir. at A51. The only factor that was not considered was trace impurities in the water. NUREG/CR-6909 states that it is very improbable that any kind of water impurity would be present during a transient event, thus such impurities need not be considered. Tr. at 1094 (Stevens).

**c. Cracks in the Feedwater Nozzle**

79. NEC raises the possibility that surface cracks may exist at the blend radius and the base metal of the feedwater nozzle (NEC Exhibit NEC-JH_03 at 11-12, 15-16). VY periodically inspects the feedwater nozzle for potential cracks in the base metal using ultrasonic testing ("UT") techniques. Tr. at 1040-41 (Fitzpatrick). The UT examinations, which are standard in the industry, are designed to detect the minimum size flaw that is postulated to occur in the base metal, on the order of 3/16 of an inch. Tr. at 1047 (Fitzpatrick).

80. Entergy has performed a fracture mechanics analysis that postulates the existence of a minimum size crack and estimates its growth with time. Tr. at 1063 (Stevens). Based on that analysis, a nozzle inspection program is in place at VY, designed to inspect the nozzle at such frequency that cracks can be detected before they grow to appreciable size.

Tr. at 1048 (Fitzpatrick). One hundred percent UT inspections of all nozzles are performed every four refueling cycles (i.e., every six years). Tr. at 1061 (Fitzpatrick).

81. In the last 20 years, the inspection program has not detected cracks in the feedwater nozzle base metal that are above the detection capability of the UT equipment. Tr. at 1051 (Fitzpatrick). The most recent inspection was conducted during the 2007 refueling outage and showed no evidence of cracks in the base metal of the nozzle. Entergy NEC 2 Dir. at A53; Entergy Exh. E2-33.

82. The inspection program is conducted pursuant to ASME Code Section XI. The fatigue analysis, which is conducted under ASME Code Section III, inherently assumes there are no cracks present in the nozzle. Tr. at 1051 (Fitzpatrick). If a crack in the nozzle were detected, it would have to be repaired or otherwise addressed under the existing ASME Code Section XI program. Thus, its existence would not affect the Section III $CUF_{en}$ analysis. Tr. at 1059-60 (Stevens).

83. At the hearing, NEC witness Dr. Hopenfeld admitted that he has no evidence that the feedwater nozzles are cracked. Tr. at 1063 (Hopenfeld). He postulated the potential existence of cracks in the cladding that partially surrounds the base metal of the nozzle as points of stress initiation. Tr. at 1067 (Hopenfeld). Again, he provided no evidence that such cracks exist and did not dispute that, if cracks were to develop in the base metal at those points, they would be detected as part of the plant Section XI program and repaired or otherwise addressed as necessary. Tr. at 1059 (Stevens).

**d. Heat Transfer Equations**

84. NEC asserts that the heat transfer equations used by Entergy in its CUF calculations are wrong. In particular, Dr. Hopenfeld points out that the heat transfer coefficients and the resulting stresses on the component are dependent on the flow velocity. He describes the Entergy analyses as erroneously assuming a uniform velocity, both circumferentially and

axially, along the surface of a component such as a nozzle. Tr. at 1108-09 (Hopenfeld). Had Entergy computed the local velocity along and circumferentially around the surface of the nozzle, argues Dr. Hopenfeld, significantly higher stresses would have been obtained. Id. at 1109-10.

85. In its reanalysis using the Green's Function methodology, Entergy applied bounding heat transfer coefficients at every region of the component where there was a change in flow velocity (e.g., a change in the diameter of a pipe). Tr. at 1112-13 (Stevens). For each such location, the stress computation used the highest applicable flow rate to compute the heat transfer coefficient. Id. at 1113. This procedure resulted in conservatively high heat transfer coefficients that maximized the estimated stresses on the component. Id.

86. For Entergy's confirmatory analysis, in which each transient was modeled separately, it was possible to vary the temperature and flow rate throughout the transient, and this procedure was followed. Id. at 1115.

87. Neither analysis computes variations in heat transfer coefficient azimuthally around a pipe. However, no such variations are possible under the conditions that exist at VY, i.e., where fully developed, high flow rates do not allow for temperature variations around a pipe. Id. at 1118.

88. Dr. Hopenfeld argued that the length of piping between an elbow and the entrance to the feedwater nozzle, 48 inches, is insufficient to allow for fully developed turbulent flow to exist which would justify the assumption of azimuthally uniform temperatures around the pipe. He opined that a minimum of 12 to 40 pipe diameter lengths of straight piping would be required in order for turbulent flow to be assured. Tr. at 1119-22, 1126 (Hopenfeld). Entergy responded that, for high flow lines, the effects of an entrance are minimized so that fully developed flow exists. Tr. at 1123-25 (Stevens). This is demon-

strated by one of the exhibits to NEC's testimony, NEC Exh. NEC-JH_29 at 212, Fig. 8-9, which shows that, for high flow velocities, entrance effects are minimized.

89. NEC raises other challenges to the heat transfer equations utilized in Entergy's CUF calculations. NEC Exh. NEC-JH_03 at 12-15. Entergy's testimony, however, argues persuasively that the claims are unsupported or technically unsound. Entergy NEC 2 Dir. at A54. Since NEC witness Dr. Hopenfeld cannot tell whether the heat transfer coefficients that express the heat transfer from the coolant to the surface of a component yielded by the equations should be larger or smaller, those challenges can be disregarded as inconsequential. Tr. at 1095 (Hopenfeld).

**e. Number of Transients**

90. To insure a realistic projection for the thermal transient cycles and events expected for 60 years of operation, Entergy used the Thermal Cycle Diagrams from a later vintage BWR as a starting point of its transient estimate. Entergy NEC 2 Reb. at A17. The VY Design Specification transients were mapped onto the BWR Transient Diagrams. Then, projections for 60 years were made based on the numbers for 40 years in the VY Design Specification, the numbers actually analyzed in the VY Design Stress Report, and the number of cycles experienced by VY in approximately 35 years of operation. Id. For example, 200 Startup / Shutdown cycles were included in the original VY Design Specification. However, 300 Startup / Shutdown cycles were conservatively used in the EAF analyses for 60 years of operation. Id. To date, only 92 Startup/Shutdown cycles have taken place. Entergy Exh. E2-39.

91. The 60 year quantity of operating events used in the VY EAF analyses are equal or greater in number to the numbers of cycles in the original VY Design Specification for 40 years. Actual plant cycle counts to date, projected out to 60 years of operation, will not exceed the number of cycles used in the Entergy analyses. Id. In addition, the rates of

temperature change during actual transients are less severe than those for the design transients analyzed, so that the actual fatigue usage that is experienced by the components is less than that calculated for a design transient. Id. Finally, bounding EPU conditions were used for all transient definitions and numbers of cycles, even though EPU operation did not apply to the first 35 years of plant operation. Therefore, there is significant margin incorporated in the number of transient cycles used in the EAF analysis and their severity, and the projected numbers of cycles used in the EAF analyses are conservative. Id.

92. Dr. Hopenfeld postulated that the number of estimated transients used by Entergy in its $CUF_{en}$ calculations should be increased by 20% to account for the increased number of transients that will result from the 2006 power uprate. Tr. at 1141 (Hopenfeld). The number was arrived based on Dr. Hopenfeld's judgment, because he "needed a hanger to hang [his] hat on." Tr. at 1141 (Hopenfeld). In reality, only three transients have been experienced at VY since the uprate was implemented in March 2006. Tr. at 1170-72 (Fitzpatrick). The numbers of transient cycles used in Entergy's CUF calculations are conservative projections of the numbers of cycles actually experienced by the plant over its operating history. The margin on the numbers of cycles analyzed over a linear extrapolation of those experienced significantly exceeds the factor of 1.2 suggested by NEC. Entergy NEC 2 Dir. at A55.

93. At the Board's request (Tr. at 1161-62, 1165), Entergy provided an up-to-date listing of those transients, having a potential impact on EAF, that have been actually experienced at VY since the beginning of plant operations. The list, which is organized by transient type, compares the actual number of transients thus far to the number assumed in Entergy's EAF analyses through the end of the license renewal period. Entergy Exh. E2-39; see also Entergy Exh. E2-11 at 18, Table 5 and Tr. at 1166-68 (Stevens). Entergy

Exh. E2-39 shows that the number of actual transients experienced to date is one third or less than the number of transients used in the $CUF_{en}$ calculations. Accordingly, the number of transients used in Entergy's reanalysis and confirmatory analyses is conservatively high and unlikely to be exceeded during the license renewal period.

94. In addition, Entergy has committed to monitor the transient count throughout the period of extended operation to verify that these assumptions remain valid, and will take appropriate corrective action if these assumptions do not remain valid. This Fatigue Monitoring Program is described in Section A.2.11 of the Application. Entergy NEC 2 Dir. at A55. The NRC Staff has determined that the Fatigue Monitoring Program will ensure that the predicted number of transients is not exceeded. FSER Section 3.0.3.2.10.

**f. Use of Green's Function Methodology**

95. The use of the Green's Function methodology and associated simplifications in the reanalysis resulted in a 40% lower value of the CUF for the feedwater nozzle than the CUF for the nozzle computed in the confirmatory analysis. Tr. at 930 (Stevens). However, the actual difference is small and the difference in CUF due to changes in the Green's Function is also very small. Entergy Exh. E2-35 (Amendment 33 to Application, Attachment 1 at 4); Entergy NEC 2 Dir. at A58. Therefore, use of the Green's Function methodology would not result in a substantial underestimation of the CUF. Also, the confirmatory analysis CUF for the safe end of the same nozzle was 60% lower than that calculated in the reanalysis. Entergy NEC 2 Dir. at A58.

**g. Lack of Error Analysis**

96. NEC criticizes Entergy for the failure to perform an error analysis to show the admissible range for each variable included in the analysis (NEC Exh. NEC-JH_03 at 18). In reality, performing an error analysis on the stress results is not traditional practice in analyses of this type, and is unnecessary given that input parameters (such as temperature, pressure,

and heat transfer coefficients) were selected so as to maximize stresses and give a conservative, bounding estimate. Any error analysis would give a lower stress estimate. Entergy NEC 2 Dir. at A59; Tr. at 910-12 (Stevens).

### h. Analysis of Two Other Nozzles

97. NEC Exh. NEC-JH_03 (at 18) asserts that the confirmatory calculation for the feedwater nozzle does not bound on the other two nozzles (the recirculation outlet nozzle and the core spray nozzle). However, the feedwater nozzle is the controlling nozzle because it experiences the most severe design transients and because it is the location where the relatively colder feedwater returns to the hot reactor vessel, thereby causing the most severe thermal stresses. It is extremely improbable that either of the other two nozzles would experience $CUF_{en}$s in excess of that experienced by the feedwater nozzle. Entergy NEC 2 Dir. at A61; Tr. at 922-23 (Stevens); Tr. at 923, 951 (Fair). This result was demonstrated in Entergy's reanalyses, where all three nozzles were evaluated on a consistent basis.

98. Performance of the confirmatory analysis of the other two nozzles will only serve to demonstrate, through obtaining results similar to those obtained for the feedwater nozzle, that the reanalysis results are conservative. Tr. at 921-22 (Fitzpatrick). Therefore, there is no technical basis requiring that these analyses be performed before the VY license renewal is approved.

### i. Dr. Hopenfeld's $CUF_{en}$ Recalculation

99. NEC witness Dr. Hopenfeld performed a recalculation of the $CUF_{en}$s for the nine locations of interest at VY (NEC Exh. NEC-JH_03 at 19-20). Dr. Hopenfeld calculates irrelevant CUFs by using indefensible methodologies, including using $F_{en}$ values (17 for carbon steel and 12 for stainless steel) quoted in the abstract of NUREG/CR-6909 as potentially applicable "under certain environmental loading conditions" (Entergy Exh. E2-

30, Abstract at iii), without determining whether they are applicable to VY conditions. In fact, the conditions that would cause a multiplier of 17 to exist are primarily associated with high temperature and high dissolved oxygen content for carbon and low alloy steels; those conditions do not exist at VY because the plant is operating using hydrogen water chemistry. Entergy NEC 2 Dir. at A50.

100. Dr. Hopenfeld also applies the CUF values from Table 4.3-3 of the Application, even though some f those are generic values and not VY-specific. NEC 2 Dir. at A62. As a result, the results of his recalculation are inapplicable to VY.

101. Because Dr. Hopenfeld's recalculation is based on unrealistically conservative assumptions, it yields nonsensical results. For example, it predicts a 13.77 $CUF_{en}$ for one of the components, meaning that the component should have developed cracking after only four years in service, a result at odds with VY's over thirty years of operating experience without cracks. Tr. at 1128-32 (Hopenfeld).

#### j. Other Issues

102. Dr. Hopenfeld asserts that Entergy has not supplied to NEC information necessary to establish the validity of Entergy's $CUF_{en}$ reanalyses, i.e.: "adequate layout drawings of the plant piping" and "a complete description of the methods or models used to determine velocities and temperatures during transients." NEC Exh. NEC-JH_03 at 8. However, Entergy supplied to NEC 36 drawings, which showed nozzles and connecting headers for the components in question. In addition, Entergy supplied to NEC a copy of the Design Information Record ("DIR") that lists all the drawings and other inputs used in the reanalysis calculations. If NEC or its consultants had required additional drawings they could have identified them through the DIR and requested them in discovery, as they requested other materials. No such a request was ever made. Entergy NEC 2 Dir. at A48; Entergy Exh. E2-29.

103. With respect to the alleged lack of "a complete description of the methods or models used to determine velocities and temperatures during transients," NEC asked for such a description through counsel and it was provided to NEC on April 14, 2008. Again, if additional specifics on the calculations were needed, NEC could have asked for them. Entergy NEC 2 Dir. at A48.

5. Conclusions to be drawn from the evidence

104. The evidence shows that Entergy has performed a more refined reanalysis and a confirmatory EAF analysis at limiting piping and vessel locations for VY. These analyses, performed using conservative methodologies and bounding input parameters, have demonstrated that the $CUF_{en}$s are less than 1.0 for the sixty years of plant operation encompassed by the renewed VY operating license. These analyses collectively demonstrate that the critical VY components will not experience failure due to fatigue during the period of extended operation. For that reason, there is no support for the claims made in NEC Contentions 2A and 2B, which should be rejected.

**B. NEC CONTENTION 3**

1. Industry Experience in Steam Dryer Operation

105. The steam dryer is a BWR stainless steel component, installed in the reactor vessel above the steam separator assembly, and supported by brackets welded to the inside of the vessel wall below the steam outlet nozzles, whose function is to remove moisture from the steam before it leaves the reactor. Entergy NEC 3 Dir. at A11. During plant operations, wet steam flows upward and outward through the dryer. Moisture is removed by impinging on the dryer vanes and flows down through drains to the reactor water in the downcomer annulus below the steam separators. Id.

106. The VY steam dryer is a non-safety-related, non-Seismic Category I component. Although the steam dryer is not a safety-related component, the assembly is designed to withstand design basis events without the generation of loose parts and the dryer is designed to maintain its structural integrity through all plant operating conditions. Id.

107. In 2002, steam dryer cracking and damage to components and supports for the main steam and feedwater lines were observed at the Quad Cities Unit 2 nuclear power plant, and it was discovered that loose parts shed by the dryer due to metal fatigue failure had damaged the supports. Id. at A12; Tr. at 1259- 61 (Hoffman); Entergy Exh. E3-06, Appendix A. The cause of the dryer failure was determined to be high-cycle fatigue, which results from relatively low stresses over a large number of cycles, as opposed to high stresses over a low number of cycles. Tr. at 1263 (Hoffman). The fatigue was the result of high frequency resonant acoustic waves traveling back from the steam line towards the dryer and imparting loads on the face of the dryer that exceeded the dryer's fatigue limit. Tr. at 1329-32 (Scarbrough). Quad Cities had implemented a power uprate. Entergy NEC 3 Dir. at A16.

108. Another steam dryer event occurred in Quad Cities Unit 1 in 2003. Tr. at 1265-66 (Hoffman); Entergy Exh. E3-06, Appendix B. In both cases the steam dryer failure was accompanied by a significant increase in measured moisture carryover in the steam leaving the reactor. Tr. at 1266, 1268 (Hoffman); Tr. at 1340 (Scarbrough). There was also a measurable change in the distribution of flow in the steam lines. Tr. at 1336-38 (Scarbrough); Tr. at 1339-40 (Hoffman).

109. Another instance steam dryer cracking occurred at a BWR-5 dryer (VY has a BWR-3) in 2003. Unlike other failures, this occurred after approximately nine years of operation at a 5% uprate level. Tr. at 1279 (Hoffman); Entergy Exh. E3-06 at 3. It was concluded that the uprate was not a factor in the failure, and high-cycle fatigue was not the failure

mechanism. Tr. at 1279 (Hoffman). The cracks were discovered during a refueling outage inspection of the dryer. Id.

110. The Quad Cities experience raised a concern that a loss of physical integrity of the dryer, resulting in loose dryer sections or parts being released to the reactor steam space (that is, the space in the reactor where steam is confined above the water) could potentially migrate to other components and could have adverse impact on safety-related equipment. Entergy NEC 3 Dir. at A13-A14.

111. While the formation of cracks on the surface of a steam dryer is not in itself cause for concern, the existence of those cracks needs to be identified and evaluated before the cracks progress to the point where they could cause a loss of physical integrity of the dryer, resulting in loose parts. Id. at A15.

2. VY's response to industry experience

112. In response to the Quad Cities 2 event, Entergy substantially modified the steam dryer at VY during the Spring 2004 refueling outage to improve its capability to withstand the higher flow induced vibration loadings that could result from operation of the plant at EPU levels. Id. at A16. The changes included replacing the vertical section of the hood, the reinforcing gussets in that section, the end plates and the tie bars. Tr. at 1415 (Lukens); Exhibit E3-04. All of the replaced components were made more robust. For example, the thickness of the vertical hood material was increased from half an inch to five-eights of an inch, and partial length gussets were replaced with full length ones. Tr. at 1415-17 (Lukens).

113. VY also instituted a program of dryer monitoring and inspections to provide assurance that the flow-induced loadings under operation at EPU levels did not result in the formation or propagation of cracks on the dryer. Entergy NEC 3 Dir. at A16; Exhibit E3-05. The program was reviewed and approved by the NRC and included as a license condition

as part of the power uprate license amendment issued on March 2, 2006. Entergy NEC 3 Dir. at A16.

114. As power was increased in 2006 from the original licensed power level to full extended power uprate conditions, there was continuous monitoring of plant parameters indicative of dryer performance, including measurement at least once per week of moisture carry-over and periodic measurement of main steam line pressure. Id. at A17. Following completion of EPU power ascension testing, moisture carryover measurements have continued to be made weekly, and other plant operational parameters that would be symptomatic of loss of steam dryer structural integrity (main steam line flow, reactor vessel water level, steam dome pressure) have continued to be monitored and their values trended. Id. This monitoring program will continue to be implemented during the period of extended operation after renewal of the VY license. Id.

115. In addition, the VY steam dryer was inspected during plant refueling outages in the Fall of 2005 (before completion of the EPU) and Spring of 2007 (after one year of operation at EPU power levels). Id. at A17. The dryer is scheduled to be inspected again during the refueling outages in the Fall of 2008 and the Spring of 2010, with a partial inspection scheduled for the Fall of 2011. Id. Inspections will continue in the license renewal period starting with the first refueling after March 2012. Id.

3. VY's Steam Dryer Aging Management Program

116. VY's current licensing basis with respect to the steam dryer is set forth in Attachment 6 to Supplement 33 (dated September 14, 2005) of its EPU license amendment application, Entergy Exh. E3-05. It includes the following "long term" actions to be accomplished after VY completed its power ascension to EPU levels:

> Long Term Actions
>
> The VYNPS steam dryer will be inspected during the refueling outages scheduled for the Fall 2005, Spring 2007 Fall 2008 and

> Spring 2010. The inspections conducted after power uprate implementation will be comparable to the inspection conducted during the Spring 2004 refueling outage and will meet the recommendations of SIL 644, Rev. 1.
>
> Following completion of power ascension testing, moisture carryover measurements will continue to be made periodically, and other plant operational parameters that may be affected by steam dryer structural integrity will continue to be monitored, in accordance with GE SIL 644 and plant procedures.

Entergy Exh. E3-05 at 7.

117. These CLB commitments have been made part of the current VY license, as follows:

> 5. During each of the three scheduled refueling outages (beginning with the Spring 2007 refueling outage), a visual inspection shall be conducted of all accessible, susceptible locations of the steam dryer, including flaws left "as-is" and modifications.
>
> 6. The results of the visual inspections of the steam dryer conducted during the three scheduled refueling outages (beginning with the Spring 2007 refueling outage) shall be reported to the NRC staff within 60 days following startup from the respective refueling outage. The results of the SDMP shall be submitted to the NRC staff in a report within 60 days following the completion of all EPU power ascension testing.
>
> ****
>
> 8. This license condition shall expire upon satisfaction of the requirements in paragraphs 5, 6, and 7 provided that a visual inspection of the steam dryer does not reveal any new unacceptable flaw or unacceptable flaw growth that is due to fatigue.

Staff Exh. 14 at 5.

118. The CLB for the steam dryer also includes the inspection guidelines in BWRVIP-139, “Steam Dryer Inspection and Flaw Evaluation Guidelines,” which was issued by EPRI in 2005 and incorporated into VY’s CLB as part of the requirements for dryer inspection. Tr. at 1213-14, 1220 (Lukens); Entergy Exh. E3-12 at 40.

119. In its License Renewal Application, Entergy addresses aging management of the VY steam dryer as follows:

> Cracking due to flow-induced vibration in the stainless steel steam dryers is managed by the BWR Vessel Internals Program. The BWR Vessel Internals Program currently incorporates the guidance of GE-SIL-644, Revision 1. VYNPS will evaluate BWRVIP-139 once it is approved by the staff and either include its recommendations in the VYNPS BWR Vessel Internals Program or inform the staff of VYNPS's exceptions to that document.

Application, § 3.1.2.2.11 "Cracking due to Flow-Induced Vibration."

120. The Application commits VY to maintaining its CLB with respect to the steam dryer by continuing to implement the guidance in GE-SIL-644 until BWRVIP-139 is approved by the NRC Staff, at which point VY will either include the recommendations in the approved version of BWRVIP-139 in the VYNPS BWR Vessel Internals Program or inform the Staff of VYNPS's exceptions to that document. Tr. at 1195, 1198 (Scarbrough); Tr. at 1203 (Rowley).

121. If BWRVIP-139 is not approved by the Staff, VY has a license renewal commitment to "[c]ontinue inspections in accordance with the steam dryer monitoring plan, Revision 3 in the event that the BWRVIP-139 is not approved prior to the period of extended operation." FSER, Commitment 37, Appendix A at A-12; Tr. at 1192-93 (Rowley).

122. The NRC Staff has approved these Application commitments. Staff NEC 3 Dir. at A4(c); FSER § 2.1.2.1.6 at 3.174-3.175.[22] These commitments bind Entergy to continue to im-

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[22] FSER Section 3.1.2.1.6 states:

Cracking Due to Flow-Induced Vibration

In the discussion column of Application Table 3.1.1, item 3.1.1-29, the applicant stated that the BWR Vessel Internals Program will manage cracking in the stainless steel steam dryers. During the audit and review, the staff asked the applicant for additional information on the AMP. VYNPS technical personnel stated that a steam dryer monitoring plan had been submitted as part of the power uprate application and approved by the staff. In addition, BWRVIP-139, "Steam Dryer Inspection and Flaw Evaluation Guidelines," has been submitted to the NRC for review and approval. It is expected that this BWRVIP will be approved by the NRC prior to the period of extended operation and as such will become a part of the BWR Vessel Internals Pro-

Footnote continued on next page

plement the guidance in GE-SIL-644 through the license renewal period. Tr. at 1206 (Lukens); Tr. at 1210-12 (Rowley).

123. A contingency could develop when the NRC takes final action on whether and in what form to approve BWRVIP-139. Such action could take place by the end of 2008. Tr. at 1215 (Scarbrough). If the Staff approves BWRVIP-139 with modifications, such approval would cause Entergy to evaluate the changes and determine whether VY can implement them. Tr. at 1218-19 (Lukens). If Entergy can implement the changes, the VY CLB would be modified by incorporating a revised version of BWRVIP-139; otherwise, the CLB as it exists today will remain unmodified through the license renewal period. Tr. at 1221, 1226-28 (Lukens); Tr. at 1229 (Rowley).

124. Since a modification of the CLB is not currently being proposed, it is not part of the Application before the Board. The Board must proceed by evaluating the adequacy of the steam dryer management plan, which is based on continuing to implement the guidance in GE-SIL-644 during the license renewal period. Application, § 3.1.2.2.11; Tr. at 1223 (Lukens).

125. GE-SIL-644 recommends that BWR licensees institute a program for the long term monitoring and inspection of their steam dryers. It provides detailed inspection and monitoring guidelines (Exhibit E3-06, Appendices C and D); Tr. at 1206-09 (Lukens).

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> gram. VYNPS will manage cracking of the steam dryers per the BWR Vessel Internals Program during the period of extended operation if BWRVIP-139 is approved. Exceptions, if any, will be subject to review and approval by the staff.
>
> The staff finds that since the applicant committed to implement BWRVIP-139, if approved by the staff prior to the period of extended operation, this aging effect or mechanism will be adequately managed as recommended by the GALL Report. If the staff does not issue an SER approving the use of BWRVIP-139, steam dryer inspections will continue in accordance with the steam dryer monitoring plan, Revision 3. The steam dryer monitoring plan would also assure that this aging effect or mechanism will be adequately managed.

FSER at 3.174-3.175.

126. The proposed VY steam dryer management program conforms to the guidance in the NRC GALL Report (NUREG-1801), which was confirmed by the NRC Staff in its Final Safety Evaluation Report for the VY license renewal. Entergy NEC 3 Dir. at A23; FSER at 3-175.

**a. Dryer Monitoring**

127. The monitoring component of the VY steam dryer management program consists of assessing the status of the steam dryer continuously by the plant operators and VY's technical staff through the monitoring of certain plant parameters. Entergy NEC 3 Dir. at A24; VY Off-Normal Procedure ON-3178, Exhibit E3-07 at 2.

128. VY Off-Normal Procedure ON-3178 alerts the operators that any of the following events could be indicative of significant dryer damage: (a) sudden drop in main steam line flow >5%; (b) >3 inch difference in reactor vessel water level instruments; and (c) sudden drop in steam dome pressure >2 psig. Id. These parameters are measured continuously in the control room. Entergy NEC 3 Dir. at A29; Tr. at 1306 (Hoffman). In addition, weekly measurements of moisture carryover are performed. Id. at 1304, 1305 (Hoffman). The frequency of moisture carryover measurement has been increased after the uprate. Tr. at 1305 (Lukens).

129. If any changes in the other parameters reaching the limits set in Procedure ON-3178 are detected, an immediate measurement of moisture carryover is taken. Entergy NEC 3 Dir. at A30; Exhibit E3-07 at 2; Tr. at 1307, 1341 (Hoffman). Changes in moisture carryover are evaluated in accordance with the requirements of GE-SIL-644 to determine whether significant cracking has occurred. Entergy NEC 3 Dir. at A24.

130. Abnormal values of the monitored plant parameters would indicate that the steam leaving the reactor has a high moisture content, which in turn could indicate that steam is escaping through a crack in the dryer. Such escape would be symptomatic of a significant

crack that might result in loss of physical integrity of the dryer. Entergy NEC 3 Dir. at A25; Exhibit E3-06, Appendix D.

131. Moisture carryover is measured by plant chemistry personnel using procedure OP-0631, Appendix F (Exhibit E3-10). If moisture carryover is determined to be greater than the limit stated in the procedure (currently 0.19%), the procedure requires that a Condition Report ("CR") be written, the Shift Manager notified, and actions taken in accordance with Off-Normal Procedure ON-3178. Entergy NEC 3 Dir. at A27; Tr. at 1307 (Hoffman).

132. If the moisture carryover is in the range of 0.19% to 0.35%, Off-Normal Procedure ON-3178 requires that plant management and engineering be informed. Additionally, the Operational Decision Making Initiative process would be initiated, to provide an especially methodical, systematic, conservative decision making process affecting station operation. An engineering operability evaluation in accordance with EN-OP-104 "Operability Determinations" would also be performed. Entergy NEC 3 Dir. at A28; Entergy Exh. E3-11; Tr. at 1308, 1343-44 (Hoffman).

133. Experienced qualified engineering personnel will determine the significance of the abnormal moisture carryover measurement. Entergy NEC 3 Dir. at A28. The evaluation would be performed immediately upon determination that unexplained changes in the operating parameters had occurred. Tr. at 1269-70 (Hoffman).

134. According to Procedure ON-3178, if the engineering evaluation of plant data confirms that steam dryer damage may have occurred, a plant shutdown is initiated such that the plant is brought to a cold shutdown condition within 24 hours. Entergy NEC 3 Dir. at A32; Tr. at 1268-69, 1342 (Hoffman). Assurance that plant would be shut down immediately upon such a situation is provided by the learning from industry experience, such as the Quad City failures. 1269-70 (Hoffman).

135. The personnel involved in determining the significance of the moisture carryover and other measured parameters are required to be qualified in the application of the operability determination procedure EN-OP-0104 (Exhibit E3-11). Entergy NEC 3 Dir. at A31. A prerequisite for procedure qualification is the requirement that the individual(s) be enrolled in the Engineering Support Personnel ("ESP") training program and that their capability to perform independent engineering work be assessed by their supervisor. If an engineer or his supervisor feels the engineer needs additional training to maintain or enhance his level of expertise, that training is incorporated into the performance goals for the year. Id.

136. All engineering personnel at VY must be qualified through a prescribed Institute of Nuclear Power Operations ("INPO") ESP Training Program that prescribes the training methodology, the kind of training they need, and the experience they need before they can work independently. In addition to having that training, they need to have their supervisor certify that they have properly completed the training, and that they have performed the work under the guidance of someone else, and that the supervisor is satisfied with the level of the work that was performed. Tr. at 1393 (Hoffman). The training program is audited by INPO, which conducts a thorough detailed assessment of the training program to ensure that it results in personnel being qualified to do perform their duties. Id. at 1394.

137. The NRC Staff has reviewed the qualifications of the Entergy personnel who monitor and evaluate the plant parameter information, and has concluded that Entergy is capable of analyzing plant data related to steam dryer performance in an adequate manner. Staff NEC 3 Dir. at A11; Tr. at 1395-97 (Hsu, Rowley). NEC has provided no testimony or exhibits challenging the qualifications of the Entergy personnel who monitor and evaluate plant parameters relating to the status of the dryer.

138. The purpose of the measurements of moisture carryover, main steam line flow, reactor vessel water level, and steam dome pressure is not to enable Entergy to determine whether a dryer crack is about to form, but to provide early warning to the plant personnel that a crack may have developed so that appropriate, timely action may be taken before undesirable effects ensue as a result of the crack. Entergy NEC 3 Dir. at A33; Tr. at 1296-97 (Hoffman). This approach is consistent with the guidance in GE-SIL-644, which while noting that monitoring of steam moisture content and other parameters does not consistently predict imminent dryer failure, monitoring "does allow identification of a degraded dryer allowing appropriate action to be taken to minimize the damage to the dryer and the potential generation of loose parts." Entergy Exh. E3-06 at 6. All parties agreed that this is a correct statement as to the usefulness of a dryer monitoring program. Tr. at 1356-57 (Hoffman, Lukens, Scarbrough, Hopenfeld).

139. A crack that developed sufficiently to lead to detectable amounts of steam bypassing of the dryer would be a through penetration crack, several inches in length, and wide enough for steam to leak out of the crack. Tr. at Tr. at 1296-97 (Hoffman). The cracks that were identified at Quad Cities Unit 1, which were several inches wide and several feet in length, were accompanied by changes in moisture carryover and steam line flow distribution, whose detection led to the plant being quickly shut down. Tr. at 1336-37, 1340 (Scarbrough).

140. While there is no technology that will predict when a crack will initiate, the monitoring programs at VY ensure that any steam dryer cracks that develop are detected before they grow to a size that would be of concern. Entergy NEC 3 Dir. at A34. The monitoring program looks at operating parameters to identify potential indications of a breach in the steam dryer, regardless of the mechanism that causes the breach, and is therefore results-driven and not cause-driven. Tr. at 1301 (Hoffman).

141. If a crack were to develop in the VY steam dryer and grew until it was sufficiently large to be detected by the monitored plant parameters, it still would grow slowly enough to allow the plant to be shut down before it resulted in dryer failure and the generation of loose parts. Tr. at 1301, 1313, 1334 (Hoffman). The slow growth of the cracks found at VY is demonstrated by inspection data, and is consistent with the ductile nature of stainless steel, that inhibits fast-growing brittle fractures. Id. at 1302-03 (Hoffman).

142. Dr. Hopenfeld disagrees that the steam dryer monitoring program implemented at VY is sufficient to identify cracks on the dryer before they lead to structural failure of the dryer. He cites the following statement in a Pacific Northwest Laboratory report: "Unlike the previously discussed mechanisms (corrosion) vibration fatigue does not lend itself to periodic in-service examinations (volumetric, surface, etc) as means of managing this degradation mechanism. . . . Once a crack initiates failure quickly follows." Hopenfeld Reb. at A32, quoting NEC Exh. NEC-JH_69 at 62. However, the cited statement refers to high-cycle fatigue effects on small bore piping and socket welded vent and drain connections in immediate proximity to vibration sources, and has nothing to do with steam dryer fatigue prospects. Tr. at 1391-92 (Lukens). As such, it does not support the proposition that steam dryer monitoring program at VY would not be able to detect steam dryer cracks in time to allow for measures to be taken in response to their detection.

**b. Dryer Inspections**

143. Dryer inspections are performed during plant refueling outages. Entergy NEC 3 Dir. at A37. It is only feasible to inspect the steam dryer when the plant is shut down, because the inspection requires removing the reactor vessel head to take the steam dryer out. Tr. at 1417-18 (Lukens).

144. The details of the program to be implemented during the dryer inspections are set forth in the section of GE-SIL-644 devoted to BWR-3 steam dryers (Entergy Exh. E3-06, Appen-

dix C at 15-16). The dryer inspections are to be performed in accordance with the VY BWRVIP Program Plan, VY-RPT-06-00006, which references GE-SIL-644, Revision 1 and BWRVIP-139. Entergy NEC 3 Dir. at A35; Entergy Exh. E3-12.

145. The dryer examinations have consisted of VT-1 and (in earlier times) VT-3 examinations of accessible internal and external welds and plates in the steam dryer potentially susceptible to crack formation. Tr. at 1367 (Lukens). VT-1 and VT-3 examinations are defined by ASME Boiler & Pressure Vessel ("BPV") Code Section XI, and the non-destructive examination technicians who perform and review these examinations are qualified in accordance with ASME BPV Section XI. Entergy NEC 3 Dir. at A35-A36.

146. A VT-1 visual examination under BWRVIP standards (such as the steam dryer inspections) is one capable of achieving a resolution to discern a 0.044 inch (slightly over 1/32 inch) high lower case character with no ascender or descender strokes (e.g., an a, c, e, or o) on an 18% neutral gray card. Entergy NEC 3 Dir. at A36; Tr. at 1363 (Lukens). This resolution level is slightly larger than the micro engraving on a dollar bill. Id.

147. GE-SIL-644 instructs that all accessible areas of the dryer that are susceptible to fatigue are to be inspected. Tr. at 1371 (Lukens). At VY, all locations in the steam dryer where fatigue cracks could develop are accessible for inspection, and are inspected. These areas include the outer vertical sections of the dryer, the sides, and the cover plates. Tr. at 1367 (Lukens).

148. The inspections are performed by qualified non-destructive examination ("NDE") inspection personnel, using qualified NDE techniques appropriate for BWR steam dryer inspections. Because of the large number of individual examinations to be performed during a refueling outage, this work is typically contracted out to qualified reactor inspection specialists, including the reactor supplier (General Electric). Entergy NEC 3 Dir. at A38.

Entergy, however, specifies the scope of the inspections and the areas to be examined. Tr. at 1373-74 (Lukens).

149. Steam dryer examinations at VY are performed using high resolution color cameras and are recorded directly to digital video disks ("DVDs") for review and evaluation. A "resolution demonstration" is performed by aiming the camera at a "Sensitivity Resolution Contrast Standard" and verifying that the lighting and the equipment setup at actual test conditions meets the 0.044 inch resolution requirement for the examination being performed. The resolution demonstration is also recorded on DVDs for future review. Therefore, the technicians and structural engineers reviewing the inspection results see exactly the same surface conditions that the inspecting technician saw during the examination. Entergy NEC 3 Dir. at A36.

150. The inspection data are reviewed by qualified Level III NDE personnel and are subject to final acceptance by Entergy Level III NDE personnel. VY typically contracts both the Level II and Level III services for reactor vessel and internals examinations, including the steam dryer examinations. As an additional quality step, VY requires that these examinations also be reviewed by an Entergy Level III NDE technician. The Entergy Level III review and approval is required to be completed on 100% of the steam dryer examinations prior to the dryer's return to service. Id. at A40.

151. All detected indications (imperfections or unintentional discontinuities that may or may not be cracks) are evaluated by qualified structural engineers, who are experienced with BWR steam dryer crack evaluation. Typically, these indications are evaluated by engineers who are on the staff of the reactor vendor, and the evaluations and conclusions are reviewed and accepted by qualified Entergy structural engineers. Id. at A42.

152. An indication is classified as "recordable" or relevant if it is visible to the resolution of the examination technique. All recordable indications found by the Level II NDE techni-

cian who performs the examination are identified and documented in the corrective action program. All recordable indications that are confirmed by the Level III NDE technician are evaluated by Engineering to determine whether or not they are "rejectable." Rejectable indications are those that must be repaired prior to restarting the plant. Repair of rejectable indications is an ASME BPV Code Section XI requirement. Id. at A43.

153. If the characteristics of a particular indication do not rule out fatigue, the indication is classified as a potential fatigue indication. The indication is tracked and examined in subsequent dryer inspections to determine whether it is growing. Tr. at 1369 (Lukens). A fatigue crack would be expected to grow; a crack that does not grow would not show the characteristic behavior of fatigue and would not be of concern. NEC 3 Dir. at A44. At VY, all recordable indications are required to be reinspected at each refueling outage until at least two consecutive inspections show no growth. Id. However, as a matter of practice, Entergy is continuing to inspect all previously identified indications. Tr. at 1369 (Lukens).

154. During the Spring 2004 refueling outage, in preparation for EPU, the dryer received a baseline VT-1 inspection of all accessible areas deemed potentially susceptible to crack formation. These examinations comprised 287 weld and plate examinations. A total of 20 indications were identified, of which 2 were weld-repaired, and 18 were determined acceptable to use as-is. Tr. at 1358 (Lukens); Entergy NEC 3 Dir. at A46; Entergy Exh. E3-13.

155. The steam dryer inspections performed during the Fall 2005 outage examined all high-stress areas, as identified in GE-SIL-644. In addition, all areas that had been repaired or modified in the Spring 2004 outage were reinspected, as well as those indications that were found and evaluated to be acceptable for use as-is during the Spring 2004 outage. These examinations comprised 113 internal and external weld examinations. A total of 66

indications were identified, including previously identified indications and repaired areas from 2004, all of which were found acceptable for use as-is. The increase in identified VT-1 indications was due to increased resolution of the VT-1 examinations that were conducted in 2005 versus those performed previously. Tr. at 1362-63 (Lukens); Entergy NEC 3 Dir. at A46; Entergy Exh. E3-14. None of the indications found in 2004 had grown. Tr. at 1365 (Lukens).

156. During the Spring 2007 outage, all accessible susceptible areas of the steam dryer were inspected, consistent with the guidance in GE-SIL-644, Revision 1. The previously repaired areas, the identified high stress areas, as well as those indications that were previously found and evaluated to be acceptable for use as-is were also examined. Approximately 448 individually identified steam dryer examinations were performed. A total of 66 indications were recorded, including 48 of those identified in 2005 and 27 new indications (nine previously identified indications were determined to be non-relevant). These previously unidentified indications were the result of the increased examination scope in 2007 compared to that in 2005, and the fact that all accessible susceptible areas of the steam dryer had been subjected to the improved resolution VT-1 examination. No growth was noted in the previously identified indications. All the indications identified in 2007 were accepted for use as-is. Tr. at 1365 (Lukens); Entergy NEC 3 Dir. at A46; Entergy Exh. E3-15.

157. The steam dryer inspections conducted in the Spring of 2007 followed approximately one year of full power operation at the EPU condition. The examinations were conducted using enhanced examination resolution, which provides improved detection levels over those achievable by using the prescribed VT-1 examination process. Each of the indications found in 2007 was evaluated by qualified structural engineers, experienced in evalu-

ating indications in BWR steam dryers. None of the cracks were determined to be associated with fatigue. Entergy NEC 3 Dir. at A49.

158. Entergy presented the results of the 2007 steam dryer inspection to the NRC Staff and discussed them with the Staff. The Staff had no concerns about potential fatigue issues. Tr. at 1386-87 (Scarbrough).

159. It is possible to determine by visual inspection whether a crack is due to intergranular stress corrosion cracking ("IGSCC") or is a fatigue crack. IGSCC typically occurs in the heat affected zone adjacent to a weld. The heat affected zone in a stainless steel component is the typical location for IGSCC to occur. Tr. at 1360 (Lukens). IGSCC develops between grain boundaries, giving the crack a jagged appearance. Id. By contrast, a fatigue crack is tends to develop as a straight line. Id. IGSCC develops in low stress areas, whereas fatigue occurs in highly stressed areas. Therefore, the location, shape, and stress level permit differentiating whether a crack is due to IGSCC or fatigue. Id.

160. All cracks identified in the VY steam dryer in the 2004, 2005 and 2007 inspections are the result of IGSCC, and are of a type that grows slowly if at all. Tr. at 1299 (Hoffman); Tr. at 1360 (Lukens). The industry-accepted growth rate of IGSCC is approximately half an inch per year. Tr. at 1359 (Lukens). At VY, IGSCC crack growth is arrested by the use of hydrogen injection and noble metal application on the feedwater. Entergy NEC 3 Dir. at A45.

161. VY has not identified any steam dryer cracks that are consistent with fatigue, and this conclusion is supported by the fact that the identified indications have not grown during subsequent operating cycles. Id.

162. While VY's monitoring and inspection procedures are not specifically cited in the aging management program for the steam dryer, they implement the program. Tr. at 1273,

1277-78 (Hoffman). The procedures cite the program and provide a direct link to it, so there is no possibility that their use may be inadvertently discontinued. Id. at 1275.

4. Issues Raised in the NEC Testimony

163. Through Dr. Hopenfeld's steam dryer testimony, NEC raises two main claims regarding the VY steam dryer management program: (1) the monitoring of plant parameters indicative of potential dryer cracks is insufficient to prevent fatigue cracks from forming and propagating in the period between dryer inspections; and (2) a dryer management program must include estimating the stress loadings on the dryer and ensuring that they remain within the stress limits of the dryer material. Assessment of Proposed Program to Manage Aging of the Vermont Yankee Steam Dryer Due to Flow-Induced Vibrations (April 25, 2008), NEC Exhibit NEC-JH_54. In addition, Dr. Hopenfeld states in his rebuttal testimony that cracks occurring through IGSCC could lead to high-cycle fatigue, Hopenfeld Reb. at A29-31, and that the steam dryer parameter monitoring and inspection program fails to comply with the General Design Criteria in Appendix A to 10 C.F.R. Part 50 "insofar as they require that protection must be provided against the dynamic effects of loss of coolant accidents ('LOCAs')." Id at A28.

**a. Adequacy of Plant Parameter Monitoring**

164. With respect to Dr. Hopenfeld statement that "[m]oisture monitoring only indicates that a failure has occurred; it does not prevent the failure from occurring" (NEC Exh. NEC-JH_54 at 5), while monitoring of plant parameters will not predict the incipient formation of dryer cracks, it will identify the existence of a crack sufficiently large to adversely affect dryer performance and alert to the risk of a structural failure of the dryer, including the generation of loose parts. Entergy NEC 3 Dir. at A53; Tr. at 1295-97, 1400-03 (Hoffman). The steam dryer has completed over two years of EPU operations without the detection of fatigue cracks, and this operating performance provides a high degree of assur-

ance that the steam dryer is not subject to high cycle fatigue that could cause rapid flaw growth. Tr. at 1298-1300 (Hoffman); Tr. at 1349-50 (Scarbrough). The cracks that have been seen have been IGSCC cracks that have essentially not changed from one cycle to the next. Tr. at 1300 (Hoffman). Therefore, if a crack were to start to develop in the dryer, it would develop slowly. Tr. at 1298-1300 (Hoffman). The monitoring program would be sufficient to provide an "early warning" of potential dryer failure so that action could be taken prior to the occurrence of such failure. Entergy NEC 3 Dir. at A53.

165. Dr. Hopenfeld asserts that most parameter monitoring may indicate the formation of only those steam dryer cracks that increase moisture carryover, but those cracks that do not lead to significant moisture carryover may continue to grow undetected (NEC Exh. NEC-JH_54 at 4). However, since all of the reactor steam flows through the steam dryer, it is very unlikely that any damage to the dryer would not also result in a decrease in efficiency of the steam dryer and an increase in moisture carry-over that would cause a change in one or more of the monitored parameters (steam flow rate, reactor vessel water level and/or steam dome pressure). Entergy NEC 3 Dir. at A54.

**b. Adequacy of Dryer Inspection Program**

166. Dr. Hopenfeld further asserts (NEC Exh. NEC-JH_54 at 1-2) that Entergy's program to date of visual inspection and moisture monitoring has been ineffective in identifying cracking at the time it occurs, when it occurs in between inspections. However, there is no need to identify a crack the moment it occurs, because the intent of VY's program is to monitor material conditions on a frequency sufficient to identify and mitigate any flaws before they can grow to a size that would be detrimental to the integrity of the component. The overwhelming majority of visual indications at VY have not grown since they were first identified, and those few indications that were determined to need repair had

not reached critical size (that is, they had not had a negative effect on steam dryer integrity) prior to repair. Entergy NEC 3 Dir. at A55.

167. Dr. Hopenfeld states (NEC Exh. NEC-JH_54 at 3) that, once fatigue cracks initiate, they propagate very fast when exposed to alternating stresses of sufficient magnitude and frequency, so that even if one does not find cracks during an inspection, there is absolutely no reason why such cracks would not start propagating once the plant is restarted. However, that postulation assumes that there will be stresses of sufficient magnitude and frequency to cause cracks to propagate rapidly. VY's operating experience after the EPU (exemplified by the data collected during the 2007 inspection and the continuous monitoring of plant operating parameters for over two years) demonstrates that the stresses experienced by the dryer are below the fatigue limit for the material and are insufficient to initiate and propagate high-cycle fatigue cracks. Entergy NEC 3 Dir. at A56; Tr. at 1350 (Scarbrough). Instead, the cracks that have been found in the dryer are due to IGSCC and are small cracks that grow very slowly, if at all. Tr. at 1299 (Hoffman).

**c. Need for Stress Load Estimation and Measurement**

168. Dr. Hopenfeld asserts that the aging management program for the VY steam dryer should include "some means of estimating and predicting stress loads on the dryer, establishing load fatigue margins, and establishing that stresses on the dryer will fall below ASME fatigue limits" (Hopenfeld Dir. at A16); Tr. at 1315-16, 1350-51 (Hopenfeld). At the hearing, however, Dr. Hopenfeld was unable to explain how he would go about estimating stress loads on the dryer. Tr. at 1355-56 (Hopenfeld).

169. Stress load estimation and prediction are unnecessary because confirmation that stresses on the VY steam dryer remain within its fatigue limits is provided by the inspection results, which show that high cycle fatigue is not occurring. Entergy NEC 3 Dir. at A61.

170. Further confirmation that the stress loads on the steam dryer are below the fatigue limits is provided daily by the fact that the dryer has been able to withstand without damage the increased loads imparted on it during power ascension and for the over two years of operation since the EPU was implemented. Tr. at 1294-95 (Hoffman).

171. If a dryer failure leading to the generation of loose parts was to occur at VY, it would have taken place prior to this time. Id.

172. Dr. Hopenfeld could not cite any instances of steam dryer failures occurring beyond eighteen months after implementation of a power uprate. Tr. at 1325-27 (Hopenfeld). Mr. Scarbrough confirmed that most dryer failures occurred shortly after the implementation of a power uprate, and the longer occurring case, at Dresden Unit 3, was discovered no more than eighteen months after uprate implementation. Tr. at 1327-29, 1347-48 (Scarbrough).

173. It takes approximately 10 million cycles for a component to accumulate enough fatigue-inducing vibration cycles during normal operation to reach the fatigue limit. Tr. at 1263-64 (Hoffman). Such a limit is reached within a year or two of operations, so that operation of a steam dryer for a year or two is sufficient to accumulate enough fatigue cycles to cause significant cracking in susceptible areas of the dryer. Id. Conversely, good performance (such as exhibited by the VY steam dryer) during the first operating cycle after the uprate strongly suggests that the dryer will not experience a fatigue-induced failure. Id.; Entergy NEC 3 Dir. at A62. The limit in number of vibration cycles that could lead to high-cycle fatigue has already been reached at VY without adverse consequences being observed; hence the steam dryer at VY is not subject to failure through that mechanism. Tr. at 1299 (Hoffman).

174. There will be no change in dryer loads or stresses during the license renewal period of operation; hence, there is no reason to expect that the dryer will be subjected to increased stresses in the future. Entergy NEC 3 Dir. at A61.

175. Dr. Hopenfeld also expresses the view (NEC Exh. NEC-JH_54 at 6) that Entergy should have introduced additional analytical tools for predicting the loads on the dryer. However, the analytical tools that were used during the uprate proceeding to demonstrate that loads on the dryer will be below its endurance limits were utilized as part of the design validation process that demonstrated the adequacy of the design and established the current licensing basis. Because the predicted loads on the dryer were shown to be below the endurance limit, the design analysis was not time limited and thus does not need to be revisited at the license renewal stage, where only time limited aging analyses need to be evaluated. Entergy NEC 3 Dir. at A63. Further, the loadings on the dryer derive from plant geometries (pipe lengths, diameters, flows, pipe connections, etc.) that have not changed since the uprate was implemented, so there has been no change to the loadings on the dryer and the resulting stresses. Therefore, there is no reason for further analytical efforts. Id.

176. With respect to directly measuring dryer loads, Entergy testified that such measurements would not be feasible because they would require welding gauges to the dryer, which would introduce new high stress areas, and would also require running electric wires from the dryer to outside the vessel directly on the steam flow path. Tr. at 1379 (Lukens). In addition to the difficulties of such installation, the gauges could become potential loose parts in the event of a dryer failure. Id. at 1378. By contrast, all other instrumentation currently installed in the reactor is in place at the bottom of the vessel. Id. at 1379.

177. In light of these difficulties, the only way to measure steam dryer loading is to install strain gauges on the steam line and extrapolate their reading to the dryer location, which

is what Entergy did during the uprate. Tr. at 1380-81 (Lukens). Dr. Hopenfeld questioned the accuracy of such measurements, but failed to describe an alternative way in which such measurements might be made.

178. Dr. Hopenfeld was also unable to specify how long should his proposed dryer measurement program last. Tr. at 1383 (Hopenfeld). The Staff opined that the three inspections at consecutive outages after the uprate and the more graded approach recommended by GE-SIL-644 thereafter should be sufficient, if no fatigue cracking was observed, to manage the potential for steam dryer fatigue. Tr. at 1384-85(Scarbrough).

179. Monitoring stresses on a component on an ongoing basis is not carried out for any other component at VY, nor anywhere in the industry. Tr. at 1285(Hoffman, Lukens); Entergy NEC 3 Dir. at A64.

#### d. Effect of IGSCC Cracks

180. Dr. Hopenfeld states that "IGSCC can provide sites for corrosion attack which in turn accelerate crack growth under cyclic loading." Hopenfeld Reb. at A31. The sole basis for this opinion is a statement, which he attributes to General Electric's report of its inspection of the VY dryer in 2007, to the effect that "[t]he dryer unit end plates are located in the dryer interior and are not subject to any direct main steam line acoustic loading. However, **continued growth by fatigue cannot be ruled out.**" Hopenfeld Reb. at A29, quoting NEC Exh. NEC-JH_68, emphasis by Dr. Hopenfeld.

181. Fatigue cracks start at locations that have a high stress riser. Tr. at 1347-48 (Scarbrough). Thus, for a crack due to IGSCC to become a site for fatigue failure, the IGSCC has to occur in a high stress area. No IGSCC indications in high stress areas have been found to date in the VY steam dryer. Tr. at 1374, 1418-19 (Lukens). The IGSCC indications found at VY are in low stress areas, from which fatigue may not develop. Id.

182. The document introduced as NEC Exh. NEC-JH_68 is an Entergy VY Engineering report used to clear one of the corrective actions associated with indications identified in the 2007 dryer inspection so that it would not be a bar to restarting the plant after the 2007 outage. Tr. at 1376 (Lukens). The sentence quoted by Dr. Hopenfeld does not appear in the final, signed report but in an earlier draft of the report. The phrase was deleted from the final version because it raises a possibility that does not in reality exist, since the indication is on the dryer interior and is not in an area of high stress. Tr. at 1377(Lukens).

### e. Effects of Steam Dryer Failure on LOCA Response

183. At the hearing, Dr. Hopenfeld sought to elaborate on his concern that a failing steam dryer could release loose parts that interfered with VY's ability to respond safely to a loss of coolant accident ("LOCA"). Tr. at 1242-49, 1253-54, 1256-57 (Hopenfeld). However, he was unable to provide any specific scenarios under which this could occur.

184. NRC Staff witness Scarbrough testified that there is no regulatory requirement to consider a LOCA coincident with a steam dryer failure. Tr. at 1250 (Scarbrough). Moreover, prior to implementing the EPU, Entergy analyzed the dryer response to various accident sequences and determined that there was no accident situation, including a LOCA, that would result in loads on the dryer in excess of its design allowables, which are set by the ASME Code endurance limit for the material. Id. at 1250, 1316-18 (Scarbrough); Tr. at 1271-73 (Hoffman). The results of those analyses are consistent with Entergy's testimony that fatigue failure of the dryer is caused by long term exposure to normal operating loads, as opposed to short duration accident loads. Tr. at 1263 (Hoffman).

185. The Staff did not review a potential failure of the steam dryer in conjunction with a design basis accident as part of the license renewal reviews because those analyses were performed as part of the uprate and remain valid as long as the steam dryer maintains its

integrity, which is assured by the dryer monitoring program. Tr. at 1397 (Scarbrough); Tr. at 1398 (Hsu).

186. Mr. Scarbrough testified that he knows of no scenario where loose parts from a failing steam dryer could interfere with the ability to mitigate the effects of a LOCA. Tr. at 1252, 1255, 1316-18 (Scarbrough). In those instances in which a steam dryer has experienced a failure leading to loose part generation, the loose parts have not interfered with the operation of safety-related components. Id. at 1254-55.

187. Dr. Hopenfeld postulated a scenario in which a dryer, already weakened by significant cracks, would fail upon being subjected to the loads imparted by a LOCA and would generate loose parts that would interfere with the safe shutdown of the plant. Tr. at 1315-16 (Hopenfeld). Occurrence of such a scenario would be unlikely because the plant parameter monitoring program would detect the existence of significant cracks and cause their evaluation and the potential shutdown of the plant. Tr. at 1320-22 (Hoffman); Tr. at 1319-20 (Hsu). Were such a situation to arise, however, the loose parts would probably escape with the steam released by the LOCA without adverse consequences. Tr. at 1317-18 (Scarbrough).

5. Conclusions to be drawn from the evidence

188. The evidence shows that Entergy has instituted a program, in effect as part of the plant's current licensing basis and to be continued after renewal of the VY license, to continuously monitor plant parameters indicative of potential cracking of the steam dryer and properly evaluate and respond to any significant departures of those parameters from their normal range. This program is conducted by highly qualified and trained individuals. Entergy also performs during each refueling outage thorough visual inspections, conducted in accordance with industry guidelines, of the areas of the steam dryer potentially susceptible to fatigue crack formation. The fact that the VY steam dryer has shown no

evidence of fatigue induced cracks after over two years of EPU operation strongly indicates that routine inspection of the steam dryer during the period of extended operation will be sufficient to provide reasonable assurance of continued steam dryer integrity. In all, the steam dryer inspection and monitoring plan that Entergy will implement during the period of extended operation after license renewal will assure that the aging effects on the steam dryer will be adequately managed. For that reason, there is no support for the claims made in NEC Contention 3, which should be rejected.

## C. NEC CONTENTION 4

### 1. Summary of VY's FAC Management Program

#### a. Definition of FAC

189. Flow-accelerated corrosion is an age-related degradation mechanism that attacks carbon steel piping and components exposed to moving water or wet steam. Entergy NEC 4 Dir. at A5. FAC is caused by the protective oxide layer that builds on the surface of carbon steel piping and components being dissolved into the flow stream. Id. This attack occurs under specific water chemistry conditions.

190. If FAC is not detected, the piping or component walls will become progressively thinner, normally globally (i.e., over a broad area of the component), until the material in the affected area can no longer withstand internal pressure and other applied loads and a rupture (rather than a leak) eventually occurs. Id. It is the global nature of FAC wear that causes a pipe rupture, whereas localized damage due to other mechanisms (e.g., erosion) causes leaks and does not impact the structural integrity of the piping. Id. FAC only attacks carbon steel components in the presence of purified flowing water or wet steam. It does not attack steels containing other fluids, such as oil. Id. Steels containing appreciable amounts of chromium have been found immune to FAC. Id.; Tr. at 1635-36 (Horowitz).

191. FAC is only one of several mechanisms that can affect the physical integrity of piping and components. The term "FAC" was coined to avoid the ambiguities present in the previously used term – "erosion-corrosion." Specifically, FAC is a chemical (corrosion) mechanism, not a mechanical damage mechanism (i.e., erosion). Erosive damage also occurs in nuclear piping, but such damage is normally confined to small leaks. Entergy NEC 4 Dir. at A5; Tr. at 1700-02 (Horowitz).

192. The definition of FAC, as opposed to erosion-corrosion, was made more precise to avoid confusion that occurred when one used the term erosion-corrosion because the counter-measures that would be used to deal with a particular phenomenon differ depending upon whether the problem is erosion or flow-accelerated corrosion. Tr. at 1470 (Horowitz).

193. Cavitation is not an aging management issue but is normally considered a design issue. Once a plant experiences cavitation the wear from cavitation is not trended, but instead the problem is fixed. Tr. at 1471 (Horowitz).

194. Impingement damage normally occurs when high velocity streams of steam impinge on piping. As contrasted with FAC, impingement tends to be localized, causing little holes, and it often occurs under upset-type conditions, not normal operating conditions. Impingement damage is basically unpredictable. Because it is a different mechanism from FAC, impingement is unaffected by replacing carbon steel with FAC-resistant material such as stainless steel or low-alloy steel. Tr. at 1471-72 (Horowitz).

195. Although water flowing at a high enough velocity may cause erosion in addition to corrosion, that kind of erosive damage is not seen in light water reactors because the water velocities necessary to cause that damage in carbon steel piping are not seen in light water reactors. Tr. at 1473-74 (Horowitz); Tr. 1479 (Hausler).

196. Dr. Hopenfeld testified that he was not an expert on the characteristics of oxide layers in piping and could not separate erosion and corrosion, and deferred to Dr. Hausler's expla-

nation of the concept of erosion-corrosion. Tr. at 1476-78 (Hopenfeld). Dr. Hausler testified that corrosion engineers have, in fact, used the term erosion like Dr. Horowitz and as Mr. Fitzpatrick indicated in areas of high turbulence, because where there is high turbulence you get somewhat localized corrosion. Tr. at 1478 (Hausler). Dr. Hausler testified that it is extremely difficult to generate sheer forces high enough for fluids to cause erosion. Tr. at 1479 (Hausler). He agreed with Dr. Horowitz and Mr. Fitzpatrick that the mechanism being dealt with in FAC was a dissolution phenomenon. Tr. at 1479 (Hausler).

**b. Description of VY's Proposed FAC Program**

197. Section B.1.13 of the Application (Entergy Exh. E4-04), states that the VY program for addressing FAC is consistent with the program described in the NRC guidance document "Generic Aging Lessons Learned (GALL) Report -- Tabulation of Results," NUREG-1801, Vol. 2, Rev. 1 (Sep. 2005) ("NUREG-1801" or "GALL Report"), Section XI.M17, Flow Accelerated Corrosion (Entergy Exh. E4-05). Entergy Exh. E4-04 at B-47. There are no exceptions in the Application to the guidance in NUREG-1801 with respect to FAC. Entergy NEC 4 Dir. at A17.

198. The VY FAC Program currently in effect (set forth in Entergy Procedure EN-DC-315, Rev. 0, Entergy Exh. E4-06) substantially follows the current version of NSAC-202L, NSAC-202L-R3 (Entergy Exh. E4-07). Entergy NEC 4 Dir. at A17. The VY FAC Program includes, as recommended in the GALL Report and the NSAC-202L guidelines, "procedures or administrative controls to assure that the structural integrity of all carbon steel lines containing high-energy fluids (two-phase as well as single-phase) is maintained." Id.; Entergy Exh. E4-05 at XI.M-61.

199. The VY FAC Program includes the following activities: (a) conducting an analysis to determine critical locations; (b) performing baseline inspections to determine the extent of

thinning at these locations; and (c) performing follow-up inspections to confirm the predictions, or repairing or replacing components as necessary. Entergy NEC 4 Dir. at A18. NSAC-202L (Entergy Exh. E4-07) provides the general guidelines that are implemented in the FAC Program. Id.

200. The VY FAC Program during the license renewal period will be identical to the existing program and will conform to the EPRI guidelines contained in NSAC-202L. Entergy NEC 4 Dir. at A19. It will include "procedures or administrative controls to assure that the structural integrity of all carbon steel lines containing high-energy fluids (two-phase as well as single-phase) is maintained." It will also provide detailed instructions on: (a) how to conduct the inspections; (b) how to evaluate the inspection data; (c) the acceptance criteria for inspected components; (d) the disposition of components failing to meet acceptance criteria; (e) the expansion of the sample to other components similar to those failing to meet acceptance criteria; and (f) the updating of CHECWORKS models to incorporate inspection data. Id.; Entergy Exh. E4-06, Section 5.0.

201. VY has programs in place that deal with phenomena, other than FAC, that cause pipe wall thinning. Tr. at 1469 (Fitzpatrick). The VY FAC Program, however, would detect pipe wall thinning in piping inspected under the program regardless of the cause of the thinning. Tr. at 1468 (Fitzpatrick). Wall thinning due to mechanical erosion would be detected in the inspections under the VY FAC Program. Tr. at 1491 (Fitzpatrick). Other programs manage aging in piping systems not within the scope of the VY FAC Program. Tr. at 1492 (Fitzpatrick). Inspection locations to check for mechanical damage (non-FAC wear) are selected by operating experience. Tr. at 1508 (Horowitz).

202. Phenomena that may cause wall thinning that are not FAC are not tracked in VY FAC Program because they are not aging management issues. Cavitation is a design issue (Tr. at 1469 (Fitzpatrick); Tr. at 1471 (Horowitz)) and is not subject to trending. Tr. at 1471

(Horowitz). Impingement, although sometimes capable of being trended, is an unpredictable phenomenon that can be related to design issues. Tr. at 1472 (Horowitz).

**c. Scope of the VY FAC Program**

203. All piping at VY that may experience FAC is included within the scope of the VY FAC Program. Tr. at 1492 (Fitzpatrick). Compared to the majority of nuclear power plants in operation, VY is a relatively small and simple plant. Entergy NEC 4 Dir. at A22. There are fewer FAC-susceptible systems and piping components than at a typical plant, and many of those were either originally constructed of FAC-resistant materials or have been replaced with FAC-resistant materials since their initial installation. Id. VY has vane-type moisture separators with no reheat steam system. This eliminates a large amount of FAC-susceptible piping and a number of components known to be susceptible to FAC found in a typical nuclear power plant. Id.

204. The extraction steam system piping, which contains a significant portion of the two-phase piping in a power plant, is constructed from FAC-resistant materials. Id.; Tr. at 1657 (Fitzpatrick). A number of other components and associated piping subject to two-phase flow (wet steam) have been replaced with FAC-resistant materials. Entergy NEC 4 Dir. at A22; Tr. at 1671 (Fitzpatrick). The original plant design and the component replacements have resulted in a significantly smaller amount of FAC-susceptible piping at VY as compared to the typical nuclear power plant of similar size. Entergy NEC 4 Dir. at A22.

205. Since VY went into operation, carbon steel piping and equipment in a number of systems has been progressively replaced with FAC-resistant materials. These include: (a) all 10 of its feedwater heaters; (b) both low pressure turbine casings, including the attached extraction steam nozzles and piping; (c) all of the two phase flow piping in the moisture separator drains system; (d) the majority of the two phase flow piping in the heater drains system except at the lowest pressure feedwater heaters; (e) the majority of the turbine cross

around piping; (f) small bore steam drain lines to the condenser coming from the high pressure cooling injection system, the reactor core isolation cooling system, and the advanced off-gas system; (g) small bore shell vent lines for all four of the high pressure feedwater heaters. Nearly all of the large bore piping at VY which is exposed to two-phase flow was either originally constructed with, or re-placed with, FAC-resistant material. Entergy NEC 4 Dir. at A23.

**d. Selecting Components for Inspection**

206. The VY FAC Program conforms to the inspection recommendations contained in NSAC-202L. Entergy NEC 4 Dir. at A20; Exhibits E4-06, Section 5.0, and E4-07. The FAC Program calls for piping and component inspections to be conducted at each refueling outage, with the items to be inspected being selected based on: (a) required re-inspections and recommendations from previous outages (Tr. at 1528 (Fitzpatrick)); (b) CHECWORKS susceptibility rankings; (c) industry/ utility/ station experience including items identified through work orders and condition reports (Tr. at 1528 (Fitzpatrick)); (d) susceptible non-modeled large bore and small bore program piping (Tr. at 1527, 1535, 1712-13 (Fitzpatrick)); (e) engineering judgment. Entergy NEC 4 Dir. at A20; Entergy Exh. E4-06, Section 5.3. Entergy's FAC program also takes risk significance and component susceptibility to failure into account. Entergy NEC 4 Dir. at A56; Entergy Exh. E4-06, Sections 5.2 and 5.3.

207. In a typical inspection, approximately one-third of the piping and components will be selected based on CHECWORKS results, approximately one-third will be chosen based on previous inspection data, and the remainder will be selected based on operating experience. Tr. at 1673-74 (Fitzpatrick). Although CHECWORKS is a useful analytical tool, the VY FAC Program is not dependent on the CHECWORKS results to select the piping

to inspect, and such selection could be managed without resorting to CHECWORKS. Tr. at 1674 (Fitzpatrick).

208. The trending of wear on piping is not based on the use of CHECWORKS but on actual inspection data. Tr. at 1546, 1681 (Fitzpatrick). The actual trend wear rates from inspection data are also used to select components for inspection. Tr. at 1645 (Fitzpatrick). The selection of components for inspection by the FAC engineer is subject to peer review by another engineer. Tr. at 1645 (Fitzpatrick).

209. At VY, the initial scoping and inspection selection of small bore piping was performed in 1993 and 1995. The scope and criteria for determining the inspection locations is documented in FAC Program documents (Entergy Exh. E4-41 and E4-42). The small bore inspections were initiated prior to the inclusion of small bore guidance provided in NSAC-202L. Entergy NEC 4 Dir. at A68.

**e. Inspecting Components**

210. When components are selected for inspection, Entergy follows an Engineering Standard, "Flow Accelerated Corrosion Component Scanning and Gridding Standard," (Entergy Exh. E4-25) to perform the inspections. This standard defines the methodology for gridding the components that are to be inspected, including the size of the grids. Id.

211. Historically, grid size is related to the physical size of the component being inspected. There are two aspects to grid size: (1) where piping degradation is found, the grid size is normally made smaller to more accurately define the wear area; and (2) the larger the pipe, the larger amount of material that may be lost before the component fails, allowing for a "larger" grid (i.e., the defect size that would cause failure varies directly with the size of the pipe). Both of these approaches are consistent with NSAC-202L, Rev. 2, Section 4.5.3. Entergy NEC 4 Dir. at A56.

212. Under the VY FAC Program, the size of the grid varies with the outside diameter of the pipe. With one to six inch diameter pipe, a one inch grid spacing is used. With eight to ten inch diameter pipe, a two inch grid spacing is used. With twelve to fourteen inch diameter pipe, a three inch grid spacing is used. With sixteen to eighteen inch diameter pipe, a four inch grid spacing is used. Tr. at 1658 (Fitzpatrick). These grid sizes are based on the recommendations in NSAC-202L. Tr. at 1659 (Fitzpatrick). Rather than recording the thickness reading at particular grid points, however, the VY FAC Program takes an additional step in performing the inspections: the components inspected at VY are scanned in their entirety. This is done by moving an ultrasonic transducer over the entire surface within a grid "square." The data logger automatically records the minimum reading anywhere within the grid square and a qualified inspector verifies that reading. This procedure ensures that the thinnest thickness readings in the component are found. Entergy NEC 4 Dir. at A56, A57.

213. The extent of the axial measurements of the thickness of a component depends on the component. If, for example, the component is an elbow, the entire elbow is scanned. The pipe downstream will be inspected axially for two diameters in length. Tr. at 1660-61 (Fitzpatrick). Two diameters axially is the standard provided in NSAC-202L. Id. If there is another component within two diameters of the component being measured, the components are modeled in CHECWORKS and are inspected consistent with the inspection program. Tr. at 1663-64 (Fitzpatrick).

214. With respect to how far downstream of a component such as an elbow should piping be inspected, Dr. Hopenfeld testified that a minimum distance of 25 to 45 diameters should be inspected, but could give no basis for that range other than his opinion that it is a customary number that has been around for many years. Tr. at 1664 (Hopenfeld). However, VY has done axial inspections on four different lines beyond a distance of two diameters

from a component and has not found any excessive wear more than two diameters past the component. Tr. at 1665 (Fitzpatrick).

215. Dr. Hausler testified that the important consideration is where the point of reattachment occurs, such as if you have a flow through an orifice, there will be an eddy nearby the orifice and the point of reattachment will be further away. Dr. Hausler, however, could not give a specific distance axially where he believed reattachment would occur. Tr. at 1666 (Hausler). At VY, a Venturi pipe is used for flow measurement in orifices in the main process steam feedwater and condensate systems to smooth the transition. Inspections have been conducted two, three, and four diameters downstream of the orifices and a number of fittings downstream have been inspected, and the inspections have not found any wear. Tr. at 166-67 (Fitzpatrick). The two diameter inspection standard is based on British data that have determined that the point of maximum wear downstream of an orifice is about one and a quarter diameters away, so measuring pipe wear up to two diameters beyond the orifice would capture the wear effects from the orifice. Tr. at 1667 (Horowitz).

216. Dr. Hausler testified that he believed that not just erosion or corrosion could occur, but that particle erosion (abrasion) could be caused prior to an orifice where the velocity is high and the flow carries sand. Tr. at 1706-09 (Hausler). However, particle erosion it is not relevant to operating conditions in BWRs. Tr. at 1709-10 (Horowitz).

### f. Role of Water Chemistry in the VY FAC Program

217. In 1980, an oxygen injection system was added to VY to improve the water chemistry towards minimizing FAC. Entergy NEC 4 Dir. at A24. Oxygen is injected into the condensate and feedwater trains just downstream of the condensate pumps in order to mitigate the effects of FAC on piping exposed to single phase flow. This treatment results in about 40 parts per billion ("ppb") dissolved oxygen in the condensate and feedwater

trains. Id. This level of dissolved oxygen serves to reduce the rate of FAC because, by maintaining this concentration of dissolved oxygen in the condensate and feedwater lines, the stability of the oxide film is enhanced, the rate of dissolution is reduced, and the potential for corrosion is decreased. Id. at A25. The reduction of rates is shown in Section 5.3.2.1, Table 5-2 of Entergy Exh. E4-07. Id. The change to hydrogen water chemistry in 2003 did not change, nor was it expected to change, the oxygen concentrations in the feedwater system, as demonstrated by measured plant data. Entergy NEC 4 Dir. at A44; Entergy Exh. E4-18.

218. Hydrogen water chemistry is not affected by a power uprate. The dissolved oxygen concentration is still about 40 parts per billion. Tr. at 1667-68 (Horowitz). Dr. Hausler concurred that a power uprate does not change the water chemistry. Tr. at 1668 (Hausler). Although more oxygen may be consumed from the water due to increased mass transfer (Tr. at 1668 (Hausler)), this effect would be small. Tr. at 1669 (Horowitz).

### 2. Use of CHECWORKS

#### a. Description of CHECWORKS

219. In December 1986, an elbow in the condensate system at the Surry Unit 2 nuclear plant failed catastrophically. This failure caused steam and hot water to be released into the turbine building, resulting in the deaths of four workers and severe injuries to others. Post-accident investigations revealed that FAC was the cause of the degradation to the elbow. At that time, the U.S. nuclear fleet did not have programs in place to deal with single-phase (i.e., water only) piping degradation caused by FAC. Some programs were in place to deal with two-phase (i.e., water and steam) piping degradation, but in general, these programs were very limited in their scope. Entergy NEC 4 Dir. at A6; Tr. at 1439-40 (Horowitz).

220. In response to the Surry accident, EPRI became committed to developing a computer program that would assist utilities in determining the most likely places for FAC wear to occur, and thus the key locations to inspect for pipe wall thinning. Id. Dr. Horowitz developed the computer program CHEC (Chexal-Horowitz Erosion Corrosion) and demonstrated and released it to U.S. utilities in 1987. CHEC was replaced by CHECMATE (Chexal-Horowitz Methodology for Analyzing Two-Phase Environments) in 1989. CHECMATE expanded on the capabilities of CHEC by adding algorithms to calculate FAC under two-phase conditions. CHECMATE was the first program to accurately predict two-phase FAC. Entergy NEC 4 Dir. at A6.

221. CHECMATE was later re-placed by the current program, CHECWORKS (Chexal-Horowitz Engineering Corrosion Workstation), in 1993. Each new version built on the success of the previous program and incorporated user feedback, improvements in software technology, and available laboratory and plant data into the modeling used in the programs. Dr. Horowitz remained the technical lead person in the development of the new and revised versions. Entergy NEC 4 Dir. at A6.

222. CHECWORKS is a multi-purpose computer program designed to assist FAC engineers in identifying potential locations of FAC vulnerability. CHECWORKS is designed to be used by plant engineers as a tool in identifying piping locations susceptible to FAC, predicting FAC wear rates, planning inspections, evaluating inspection data, and managing inspection data. Id. at A26. It predicts FAC wear rates based on a number of variables that define: (1) the water chemistry; (2) the flow rate; (3) the geometry of the components; (4) the material properties of the components; (5) temperature; and (6) steam quality. Id. at A27; Entergy Exh. E4-07, Section 1.1.

223. CHECWORKS uses an approach for calculating FAC wear that follows the Keller and Kastner correlations and the Berge model, to create a calculation of FAC wear rate that is

composed of seven factors: FAC Rate = F1 * F2 * F3 * F4 * F5 * F6 * F7. The seven factors are: (1) F1 = Temperature factor; (2) F2 = Mass transfer factor; (3) F3 = Geometry factor; (4) F4 = pH factor; (5) F5 = Oxygen factor; (6) F6 = Alloy factor; and (7) F7 = Void fraction factor. Entergy Exh. E4-43; Tr. at 1442-43 (Horowitz). The CHECWORKS model is based on an established method for calculating FAC wear and is based upon laboratory data and plant data. Tr. at 1441-42, 1655 (Horowitz).

224. The correlations in one of the predecessor programs to CHECWORKS, CHEC, were initially based on FAC laboratory testing data from France and the United Kingdom and a combination of laboratory and plant operational data from Germany. When CHECMATE was written, and again when CHECWORKS was revised in the mid-1990s, a large amount of plant inspection data were used to refine the accuracy of the program's predictions. These data sources included assembled data from a variety of U.S. nuclear units as well as available laboratory data from England, France and Germany. Entergy NEC 4 Dir. at A30; Entergy Exh. E4-08 at 7-20 – 7-33. The correlations built into CHECWORKS are based on laboratory experiments on modeled geometries, published correlations, and operating data from many nuclear units. Entergy NEC 4 Dir. at A45, A50. These data support the nearly linear relationship between flow velocity and FAC wear rate. Entergy NEC 4 Dir. at A49.

225. Dr. Hopenfeld testified that he believed the geometry factors used in CHECWORKS were inaccurate because they used an average value to calculate "pressure drops" of a fitting, when he believed that the local flow velocity value, not the average value, should be used and that correction would result in an order of magnitude of difference in results. Tr. at 1616-17 (Hopenfeld). However, nuclear plants operate under the same conditions for long periods of time. So, it is appropriate for CHECWORKS to use the average velocity corresponding to the power plants' run time. With respect to changes in local velocity

due to geometric discontinuities such as elbows and fittings, CHECWORKS uses the average velocity in the cross-section, and the geometry factors to correct for the different flow patterns that occur at those locations. Since CHECWORKS is designed to use average velocities, the user does not have to be concerned about providing as inputs the variations in velocity, for example, downstream of orifices or elbows. Tr. at 1647-48 (Horowitz); Entergy NEC 4 Dir. at A48.

226. Dr. Hopenfeld testified that he did not believe that the relationship between corrosion and velocity used in CHECWORKS was accurate because it was based on copper dissolution in hydrochloric acid, and hydrochloric acid is not used to cool reactors and most material in reactors is not copper. Tr. at 1615 (Hopenfeld). However, the copper tests were not used to establish wear rates, but serve only as a fast way of doing tests of various geometries. All the geometry factors used by CHECWORKS are based on actual plant data. Tr. at 1623-24 (Horowitz).

227. CHECWORKS uses a nearly linear velocity relationship with mass transfer. Tr. at 1648 (Horowitz). The linear velocity relationship is based on Dr. Horowitz's review of experimental data (Tr. at 1621-22 (Horowitz)) and plant experience. Tr. at 1648-49 (Horowitz). Unlike erosion mechanisms, FAC damage is linear with time (i.e., there is a constant corrosion rate). This linear relationship has been demonstrated in numerous laboratory tests and by the fact that field measurements match predictions using a linear model. For that reason, laboratory tests designed to measure the impact of water chemistry on FAC rates are often run for a period of just hours – enough time to establish a trend. In fact, when operating conditions are changed, the rate of FAC responds almost immediately to the new conditions. Entergy NEC 4 Dir. at A47; Entergy Exh. E4-19; Entergy Exh. E4-08 at 7-6 and Figures 3-6 and 3-7. An extensive body of laboratory results supports this linear dependence. Entergy NEC 4 Dir. at 56; Exhibits E4-22 at Figure 8

and E4-23 at Figures 5, 6, 8 and 9. Studies from nuclear power plants that have undergone power uprates show increases in FAC wear rates proportional to velocity. Tr. at 1693 (Fitzpatrick).

228. If CHECWORKS' model of the relationship between velocity and mass transfer were inaccurate, a user would immediately see discrepancies on the predicted wear of piping upstream and downstream of components such as reducers – e.g., the wear rate predictions of an eight inch diameter pipe would be over-predicted, and that of a six inch diameter pipe would be under-predicted. No plant using CHECWORKS has reported such erroneous results. Tr. at 1621-22, 1648-49 (Horowitz).

229. The rate of FAC is constant as long as conditions remain constant and under constant conditions FAC rate can be determined by two data points. Tr. at 1686 (Horowitz); NEC Exh. NEC-JH_72. Although Dr. Hausler testified that surface finish could affect the rate of FAC wear if all other conditions remained constant (Tr. at 1687-89 (Hausler)), the variation in wear rate with roughness is very small; in large pipes, the extent of roughness does not have much of an effect on flow once the surface of a pipe has become rough. Tr. at 1689-91 (Horowitz).

230. The single most important variable in FAC wear rates is the chromium content of the piping. Tr. at 1635-37 (Horowitz). CHECWORKS conservatively assumes that steel components contain the lowest amount of alloying elements allowed by the specification (typically, zero). Such an assumption disregards the beneficial effects of chromium in retarding the onset of FAC. Entergy NEC 4 Dir. at A28; Tr. at 1445-46 (Horowitz).

231. CHECWORKS produces for every component in each analysis line a predicted wear rate, and predicted total wear for that component. For components with measured data, it also compares the predicted wear with the measured wear at the time of that inspection and

presents the time to reach a user-defined critical thickness in tabular fashion. Tr. at 1642-43 (Horowitz).

232. The CHECWORKS User Group ("CHUG"), has met twice a year since 1989 and has been the major source of feedback on the adequacy of the program. Entergy NEC 4 Dir. at A30. In no instance has a pipe failure been determined by the NRC to be a result of inaccurate predictions by CHECWORKS. Entergy NEC 4 Dir. at A42, A52; Tr. at 1694-96 (Horowitz).

233. The CHECWORKS model is periodically checked against laboratory and plant operating data as it becomes available to examine how well the CHECWORKS correlation performs. Tr. at 1444 (Horowitz). For example, inspection results at VY show that CHECWORKS over predicts FAC wear rates for the feedwater line. Entergy Exh. E4-30 at 57-58. This is common to all BWR feedwater lines. Tr. at 1627-28 (Horowitz). When a new issue occurs or when users report that something is not working as well as CHECWORKS would like, EPRI will conduct a separate study to look at that individual parameter. Tr. at 1444 (Horowitz).

234. Dr. Hopenfeld testified that an alternative to the use of CHECWORKS would be for each plant to have an expert that would be working on, and completely dedicated to, FAC evaluations. Tr. at 1606 (Hopenfeld). VY has a dedicated FAC engineer whose job is solely to maintain the FAC program. Tr. at 1574 (Fitzpatrick).

**b. Modeling of a Plant in CHECWORKS**

235. CHECWORKS modeling of a nuclear unit starts with the development of a plant heat balance diagram ("HBD"), which is a schematic representation of the major lines and connectivity of the power producing portion of the nuclear plant. The HBD model constructed in CHECWORKS is then populated with the thermodynamic conditions representative of each power level at which the plant has operated at, or is contemplated to op-

erate. The user then inputs the oxygen concentration conditions that have been used or are anticipated. These inputs define the operational history of the plant in terms of what power levels have been used with what water chemistry for how long. Entergy NEC 4 Dir. at A28; Tr. at 1445 (Horowitz).

236. The user enters as input to CHECWORKS information concerning the piping systems to be analyzed. Most of this information is at the component level and deals with geometry, wall thickness, operating conditions, and pipe material. CHECWORKS includes over fifty geometry models to represent various component geometries. In cases where the component geometry does not match any of the models, the CHECWORKS user is instructed to either use a conservative model or schedule the component for inspection. Entergy NEC 4 Dir. at A28.

237. The VY FAC Program uses a thickness measuring process that scans the components in their entirety by moving an ultrasonic transducer over the entire surface within a grid "square." The data logger automatically records the minimum reading anywhere within the grid square and the qualified inspector verifies that reading. Entergy NEC 4 Dir. at A47.

238. Dr. Hopenfeld testified that he believed that a one inch grid for component inspection would take care of all uncertainties about flow turbulence and the effect of geometric discontinuities. Tr. at 1609-10, 1684 (Hopenfeld). However, under the methodology used at VY the size of the grid is immaterial, since the entire component is scanned. Entergy NEC 4 Dir. at A56.

239. In the CHECWORKS analysis, the plant is divided into a number of analysis lines, which do not have to be physically connected, but represent components which have the same water chemistry and generally the same temperature. These components are expected to behave in the same way. Depending on the complexity of the reactor itself and the

amount of inspection data available, there are typically between 20 and 50 or more of these lines. Tr. at 1446-47 (Horowitz).

240. In CHECWORKS, analysis lines are used in determining the FAC wear rate of components because the FAC wear rate for any individual component in that line is relatable to any other component in that line because components in a line will have the same dissolved oxygen, the same pH, the same temperature, and the same flow rate. They will differ in the geometry factor which is specific to a type of fitting. Use of analysis lines allows the user to compare wear rate inspection data from different components in the same analysis line to determine if there is a good comparison. By taking data on an analysis line and comparing inspections, the random scatter in the data is minimized and potential inconsistencies can be identified and investigated. Tr. at 1548-50 (Horowitz).

### c. CHECWORKS Analysis Process

241. Based on the user inputs, an initial "Pass 1 Analysis" is conducted to report predicted wear rates. The results of the Pass 1 Analysis, together with other information including operating experience at similar units, are normally used by the FAC engineer to generate a list of components for inspection. Once this information is specified in the plant database, the plant engineers are able to conduct wear rate analyses of any or all of the piping defined in the database. Entergy NEC 4 Dir. at A28. In using CHECWORKS, the engineer breaks the operating time of the plant into a number of periods with a nominally constant power level and reasonably constant water chemistry. For each of these periods, the program calculates a corrosion rate for each component considered in the analysis. The product of the corrosion rate and operating time (i.e., the predicted degree of corrosion) is added up for all the operating periods (Entergy NEC 4 Dir. at A29) to obtain the total predicted wear. Tr. at 1447 (Horowitz).

242. Although inspection data are not required for a Pass 1 Analysis, inspection data may also be used as inputs into CHECWORKS. Entergy NEC 4 Dir. at A28. Inspection data may be input in the form of a matrix of thickness readings covering the component. Typically, these data sets are from ultrasonic measurements of the wall thickness at local points (i.e., grid points) or from scanning the component and recording the minimum thickness at grid points. Id.

243. When inspection data are available, a "Pass 2 Analysis" can be run. A Pass 2 Analysis compares the measured inspection results to the calculated wear rates and adjusts the FAC rate calculations to account for the inspection results. The program does this by comparing the predicted amount of degradation with the measured degradation for each of the inspected components. Id. Pass 2 Analyses provide the analyst with the opportunity to evaluate the goodness of fit of the model to actual results, the location of any outliers, and the possibility of modeling improvements. Entergy NEC 4 Dir. at A28. However, none of the algorithms in CHECWORKS are modified by the incorporation of plant-specific data. Entergy NEC 4 Dir. at A31; Entergy Exh. E4-09 at ¶ 23.

244. Using statistical methods, a correction factor is determined, which is applied to all components in a given pipe line – whether or not they were inspected. Entergy NEC 4 Dir. at A31. A line correction factor is calculated by CHECWORKS for each analysis line in its entirety. Tr. at 1698-99 (Horowitz). The line correction factor is used to fine-tune the results of the CHECWORKS analysis so to improve its predictive ability. Tr. at 1651 (Horowitz).

**d. Modeling Changes in Plant Conditions**

245. The use of CHECWORKS does not change as a result of a power uprate or any other change in operating parameters. CHECWORKS was designed to handle changes in plant operating conditions. Tr. at 1447-49 (Horowitz); Entergy NEC 4 Dir. at A51.

CHECWORKS can be used to forecast what impact a proposed change in operating conditions will have on FAC wear rates. Tr. at 1692 (Horowitz).

246. When a power uprate is implemented, the user updates the relevant input parameters (e.g., thermodynamic conditions, temperature, oxygen concentration, etc.), and lets the program calculate the predicted FAC wear under the new conditions. Entergy NEC 4 Dir. at A33. With the implementation of the power uprate at VY the only CHECWORKS inputs which affect wear rates that changed were the flow rate and the operating temperature. Id.

247. The Pass 2 Analysis can be used as a planning tool by performing it in advance of the uprate to determine if, under uprate conditions, systems and sub-systems would experience significantly greater FAC rates than those predicted before the uprate. Id. CHECWORKS was specifically designed to accommodate power uprates and is routinely used throughout the U.S. nuclear industry for this purpose. Id.; Entergy Exh. E4-09 at ¶¶ 19, 20.

248. Dr. Hopenfeld testified that CHECWORKS would need ten to fifteen years of VY uprate data to calibrate CHECWORKS to reflect uprate operating conditions (Hopenfeld Dir. at A22); Dr. Hausler likewise testified that 12-15 years would be a reasonable estimate of time to calibrate CHECWORKS. Hausler Dir. at A6. However, for purposes of CHECWORKS use, power uprates are no different from other operational changes. The differences in rates experienced in a power uprate are generally smaller than those experienced by units when their water chemistry changes. It has never been necessary to "re-calibrate," "re-baseline" or "benchmark" CHECWORKS when plants have changed their water chemistry, power output has been increased, or other operational changes have taken place. Entergy NEC 4 Dir. at A34.

249. Dr. Hopenfeld is of the view that, in order to establish the rate of FAC that many years of inspection data would be needed, including inspection of every potentially susceptible run of piping three times over five inspection periods. NEC Exh. NEC-JH_36 at 15-16. However, the development of CHECWORKS and the EPRI guidance have eliminated the need for such an approach. It is not necessary to have ten to fifteen years of inspection data collected after a power uprate for an effective FAC Program. Entergy NEC 4 Dir. at A40, A42. There is no need to "recalibrate" CHECWORKS when operating conditions change due to an uprate. The new values for flow rate and temperature are simply used as inputs into CHECWORKS and CHECWORKS provides FAC rate calculations for the modeled components under the uprated conditions. Entergy NEC 4 Dir. at A41. Locations that CHECWORKS shows as having the highest wear rates are typically those with the most tortuous geometry, such as around control valves, across reducers, and downstream of valves. The effect of such geometric discontinuities does not change with a power uprate. Tr. at 1670 (Fitzpatrick). Because only the flow rate and temperature are changed at VY due to the power uprate, any FAC rates established after the uprate will be constant and the effect of the uprate on FAC will, therefore, be apparent with the first inspection after the uprate. Entergy NEC 4 Dir. at A41.

250. Plant operational data used for the correlations in CHECWORKS involves approximately thirty different units. Tr. at 1654 (Horowitz). VY, under power uprate conditions, is a fairly small plant in terms of power level compared to the other units. Tr. at 1655 (Horowitz). CHECWORKS covers the range of conditions at operating light water reactors, including VY. The change at VY with the power uprate is primarily one of velocity, and the maximum velocity is just under 25 feet a second in the feedwater system, which is comparable to any number of other nuclear power plants. Tr. at 1653 (Horowitz). The algorithms used to predict the FAC wear rate are based on extensive laboratory and plant

data, including data on FAC wear rates where the flow rate and the temperature exceed those present at VY after the uprate. This assures that the FAC wear rates predicted by CHECWORKS are accurate. Entergy NEC 4 Dir. at A45.

#### e. CHECWORKS and Quality Assurance

251. CHECWORKS is not used for nuclear design but solely to provide information to FAC engineers. The information used by CHECWORKS is not directly used for functions covered by "nuclear level" quality assurance. Tr. at 1596 (Horowitz).

### 3. Use of NSAC-202L and CHECWORKS in the VY FAC Program

#### a. NSAC-202L

252. Dr. Horowitz, recognizing the need for a guidance document that would help utilities improve and standardize their FAC programs, played a key role in drafting the original version of NSAC-202L and resolving numerous utility and NRC comments on it. Dr. Horowitz has played a significant role in each of the three subsequent revisions to NSAC-202L. Entergy NEC 4 Dir. at A8. NSAC-202L has become the most important standard-setting document for the conduct of FAC control programs in the United States, and has also been accepted as a valuable guidance tool by INPO and the NRC. Id.

253. The original VY FAC Program was instituted prior to the issuance of EPRI's guidance document NSAC-202L. However, the FAC Program's documents have been revised as necessary over time to conform to the recommendations in the various revisions to NSAC-202L. The VY FAC Program currently in effect (set forth in Entergy Procedure EN-DC-315, Rev. 0, Entergy Exh. E4-06) substantially follows the current version of NSAC-202L, NSAC-202L-R3 (Entergy Exh. E4-07). Entergy NEC 4 Dir. at A17.

#### b. Use of CHECWORKS in the VY FAC Program

254. VY uses five criteria for selecting which components and locations will be inspected for potential FAC effects during a plant refueling outage. Those factors, which are consistent

with the guidance in NSAC-202L, are: (1) pipe wall thickness measurements from past outages; (2) predictive evaluations performed using the CHECWORKS computer code; (3) industry experience related to FAC; (4) results from other plant inspection programs; and (5) engineering judgment. Entergy NEC 4 Dir. at A40.

255. In implementing the FAC Program at VY, the original CHECMATE and later CHECWORKS Pass 1 Analyses (performed with the initial use of CHECMATE and CHECWORKS) calculated FAC wear rates and predicted time to minimum wall thickness based on plant-specific variables. Under the VY FAC Program, inspections were then performed for the components with the highest wear rates and lowest time to minimum wall thickness. As inspection data were obtained and incorporated into the models, Pass 2 Analyses were performed and the predicted wear rates were correlated to the measured data. In all cases, the inclusion of the inspection data reduced the predicted wear rates and increased the times to minimum wall thickness. Entergy NEC 4 Dir. at A62.

256. Currently, the FAC Program at VY primarily uses CHECWORKS as a tool in planning inspections, evaluating inspection data, and managing the ultrasonic thickness ("UT") data compiled over the past thirteen refueling outages at Vermont Yankee. Entergy NEC 4 Dir. at A21.

### c. Updating VY Plant Data for Use in CHECWORKS

257. Section 4 of NSAC-202L, Rev. 2, does not specify a specific interval for model updates. It merely states: "It is recommended that whenever possible, the Predictive Plant Model utilize the results of wall thickness inspections to enhance the FAC predictions. In CHECWORKS this is called Pass 2 analysis." Entergy NEC 4 Dir. at A63.

258. All applicable inspection data were updated for VY during the Summer and Fall of 2000. Additional updates were performed for the feedwater system in 2003. In addition, inspec-

tions performed in 2001, 2002, 2004, and 2005 showed that the wear rates predicted by the CHECWORKS model were consistently conservative. Entergy NEC 4 Dir. at A62.

259. Inspection data were not entered into CHECWORKS immediately following inspection outages in 2004 and 2005. Tr. at 1716 (Fitzpatrick). Mr. Fitzpatrick wrote condition reports regarding the failure to update the CHECWORKS model with inspection data. Tr. at 1584-86, 1715, 1716 (Fitzpatrick). These condition reports were intended to identify to management the need for additional human resources. Tr. at 1715 (Fitzpatrick). VY now has a dedicated FAC engineer whose job is to keep the FAC program current. Tr. at 1574 (Fitzpatrick).

260. The inspection planning and component selections made during the outages where inspection data had not been entered into the CHECWORKS model were based in part on the conservatively high wear rates previously predicted by CHECWORKS. The CHECWORKS update performed in 2006 confirmed again that the previously predicted wear rates were conservative. Entergy NEC 4 Dir. at A62; Exhibits NEC-UW_10 and E4-31. Runs of the updated model did not identify any instance where recommended inspections were not performed. Entergy Exh. E4-31.

261. Comparison of the CHECWORKS predictions with subsequent inspection data have uniformly shown that the CHECWORKS predictions are conservative (i.e., they predict higher wear rates than those observed during the inspection). Entergy NEC 4 Dir. at A63. Thus, even if the most recent inspection data had not been entered into the CHECWORKS program, the result would have been over-estimation of FAC wear. Id. The condition report written by Mr. Fitzpatrick concludes, therefore, that not updating the CHECWORKS database with the most recent inspection data was not necessary in order to determine the appropriate scope of the RFO 25 inspection. Id.

262. VY updated the version of CHECWORKS it used from CHECWORKS FAC 1.0D to CHECWORKS FAC 1.0F in 2000. Entergy NEC 4 Dir. at A65; Entergy Exh. E4-28. Version 1.0F was used for the 2003 and 2006 model updates. CHECWORKS FAC 1.0G was installed in 2006. Entergy NEC 4 Dir. at A65. There were no differences in versions 1.0D, 1.0F, and 1.0G with respect to water chemistry and wear rate predictions for BWRs. Id.; Entergy Exh. E4-39.

#### d. FAC Inspections Since VY Uprate

263. The scoping process for the FAC inspection in the 2007 refueling outage ("RFO 26") started before RFO 25 was complete. The RFO 26 scoping was performed using the same criteria as contained in Section 5.3 of ENN-DC-315, Rev.1. The scoping criteria in ENN-DC-315, Rev. 1, is the same as under the superseded VY procedure PP 7028 (Entergy Exh. E4-34, Appendix E, Section E.2). Entergy NEC 4 Dir. at 71.

264. As an added measure of conservatism, Entergy is increasing the FAC inspection scope by at least 50% for the first three outages following the EPU. In 2005, in RFO 25, the last refueling outage prior to the EPU, a total of 35 FAC inspections were performed, including 27 large bore inspections. Entergy NEC 4 Dir. at A41; Entergy Exh. E4-38. In RFO 26, the first outage since the EPU, the inspection scope was increased by more than 50%, as there were a total of 63 inspections performed, including 49 large bore inspections. Entergy NEC 4 Dir. at A41; Entergy Exh. E4-10. These additional inspections provide further confirmatory data points for the use of the FAC Program. Entergy NEC 4 Dir. at A41.

265. The results of the 2007 FAC inspection demonstrate that data from repeat inspections (before and after the uprate) of large bore components in the feedwater system show that essentially no wear has occurred since the commencement of the EPU in March 2006. Entergy NEC 4 Dir. at A41; Entergy Exh. E4-10, Section 8. Because no significant increase

in wear was shown in the first post-uprate inspection, there is a high level of confidence that no significant change in the rate of wear will be found in the next two inspections. Tr. at 1672 (Fitzpatrick).

#### e. Use of CHECWORKS after License Renewal

266. After license renewal, Entergy will continue to use the CHECWORKS program to assist in identifying the locations where piping inspections should be performed. Entergy NEC 4 Dir. at A40. Data collected at VY since 1989 and in the three sets of inspections that will be conducted during refueling outages between the implementation of the EPU and the expiration of the current license will be sufficient to use CHECWORKS effectively. Id. Those inspections will yield data for four and a half years of operation at the EPU levels. Id.

#### f. VY FAC Program Quality Assurance Issues Raised by NEC

267. VY Quality Assurance Audit No. QA-8-2004-VY-1 of the VY FAC Program (Entergy Exh. E4-26) resulted in the issuance of two condition reports ("CRs") against the program for: (1) not getting inspection data into the data management system on time; and (2) not having finalized the draft outage inspection report. Tr. at 1581-83 (Fitzpatrick). Despite these CRs, the Audit report states that "[n]one of the findings or areas for improvement, individually or in the aggregate, were indicative of significant programmatic weaknesses which would impact the overall effectiveness of the Engineering Programs assessed." Entergy Exh. E4-26 at 2.

268. With respect to the CR on inspection data not being put into the data management system on time, the data were in fireproof cabinets, but not into the record management system (i.e., microfilmed). Tr. at 1581-82 (Fitzpatrick). The data were in the programs that the VY FAC Program used to trend wear. Tr. at 1582 (Fitzpatrick).

269. The second CR was written because a draft outage report had not been timely finalized. Tr. at 1583 (Fitzpatrick). The inspection scoping for the next outage was based on the draft report. Tr. at 1584 (Fitzpatrick). The CHECWORKS model at that time was not being updated based on the conservative wear rates and the inspection data were showing no wear. Id. Even though the inspection data were not input into the CHECWORKS model, inspections were conducted, data were evaluated, and component wear rate was trended (based on inspection data), all in accordance with the FAC Program. Entergy NEC 4 Dir. at A65.

270. Mr. Witte testified that "[i]n 2004, at least four VYNPS components, including the condensate system and the extraction steam systems, were determined to have 'negative time to Tmin,' meaning that wall thinning was being predicted as beyond operability limits and should be considered unsafe with potential rupture at anytime." NEC Exh. NEC-UW_03 at 17. VY Scoping Worksheets (Entergy Exh. E4-40 at 5) developed in preparation for the 2004 refueling outage included four components that had "negative times to Tmin" is a theoretical conclusion based on the results of CHECWORKS, and is not based on actual inspection data. As such, there would be no need to write condition reports with respect to those results. Condition reports are written when inspection data indicate there is an actual problem, and additional inspections are then performed as corrective actions. If a planning tool, like CHECWORKS, indicates an area of potential concern, inspections of that area are scheduled. In any event, of the four items identified in the 2004 Scoping Worksheets as having negative times to Tmin, three were made of FAC-resistant material. The only FAC susceptible component in the list (CD30TE02DS) was inspected and determined to meet design requirements with significant margin. Entergy NEC 4 Dir. at A67; Entergy Exh. E4-37 at 12.

271.Mr. Witte testified: "The 2006 cornerstone report shows a number of indicators as yellow, with lists of open CR corrective actions, and a new CR written in August 30, 2006. The report lists six corrective actions and four CRs that were written as early as 2003 that remain open." NEC Exh. NEC-UW_03 at 19. The referenced report, FAC Program Health Report, "Cornerstone Rollup," shows the overall FAC Program status as Green. NEC Exh. NEC-UW_07 at 1. The report rates twenty-seven different areas. Of these, two were rated as "Yellow": (1) Owner Availability and (2) Open Actions Items. Id. at 4, 6. A yellow indicator for Open Action Items is triggered if any action item, regardless of its importance, is more than one year old. Six LO-VTYLO action items are listed. These are not condition reports, nor are they corrective actions from condition reports. They are commitments. The Corrective Action Program is used to track all commitments. There is no safety significance to these commitments. The items listed are for completion of program administrative tasks. Entergy NEC 4 Dir. at A67.

4. Conclusions to be drawn from the evidence

The evidence shows that Entergy has instituted a program, in effect as part of the plant's current licensing basis and to be continued after renewal of the VY license, to continuously monitor FAC wear. This program is conducted under industry and NRC accepted standards and run by a dedicated FAC engineer. Inspection locations are selected in accordance with industry standards based on several factors, including past inspection data, plant operating experience and predicted FAC wear rates from CHECWORKS. CHECWORKS is an industry-standard analytical tool that is based on a large amount of plant operational data and laboratory data and has been demonstrated to be an effective planning tool throughout 20 years of plant operating experience. CHECWORKS' correlations are based on data from over 30 plants, many of which have considerably higher power levels than those found at VY after its EPU.

Entergy performs FAC inspections during each refueling outage through ultrasonic measurement of the entire component and attached piping that has been selected for inspection. As a measure of conservatism, Entergy has increased by at least 50% the number of components selected for inspection in the three refueling outages following the EPU at VY. The inspection data from the first refueling outage show no increase in FAC wear rates due to the power uprate. In all, the FAC management program that Entergy will implement during the period of extended operation after license renewal will assure that the aging effects of FAC will be adequately managed. For that reason, there is no support for the claims made in NEC Contention 4, which should be rejected.

## VI. CONCLUSIONS OF LAW

The applicable legal standards for the approval of VY's aging management program the license renewal period, as set forth in 10 C.F.R. §§ 54.21(a)(3) and 54.29(a), are whether there is reasonable assurance that the aging of plant components will be adequately managed so that their intended functions will be maintained in accordance with the current licensing basis for the period of extended operation. It is the burden of the applicant, Entergy, to show that its aging management program meets this criterion.

NEC Contention 2A challenges Entergy's 2007 environmentally assisted fatigue reanalysis of critical reactor components, and Contention 2B challenges the confirmatory EAF analysis Entergy performed in early 2008. Therefore, the legal issue with respect to those contentions is whether the 2007 EAF reanalysis and the 2008 confirmatory analysis provide reasonable assurance that that the effects of aging of the critical reactor components will be adequately managed for the period of extended operation.

The Board concludes, based on the evidence presented in this proceeding, that the analyses performed by Entergy demonstrate that the critical VY components will not experience failure due to fatigue during the period of extended operation. Thus, NEC Contentions 2A and 2B are resolved in favor of Entergy. NEC Contention 2, which has been held in abeyance pending the adjudication of Contentions 2A and 2B is dismissed.

The other contentions admitted for adjudication deal with compliance with the aging management requirements with respect to specified components: the steam dryer (Contention NEC-3), and piping and components potentially subject to flow-accelerated corrosion (Contention NEC-4). Entergy has the burden of showing that there is reasonable assurance that the current aging management program for the steam dryer and the current FAC management program, if continued during the license renewal period, will adequately manage the effects of aging of the steam dryer and the aging effects of FAC on reactor coolant pressure boundary piping and associated components during the period of extended plant operation following renewal of the VY license.

The Board is persuaded by the evidence presented that there is reasonable assurance that the current aging management program for the steam dryer and the current FAC management program, if continued during the license renewal period, will adequately manage the effects of aging of the steam dryer and the aging effects of FAC on reactor coolant pressure boundary piping and associated components during the period of extended plant operation following renewal of the VY license. Thus, NEC Contentions 3 and 4 are also resolved in favor of Entergy.

The NRC Staff and the Advisory Committee on Reactor Safeguards have considered the matters raised in the NEC contentions and concluded that they raise no impediment to the renewal of the VY operating license.

## VII. PROPOSED ORDER

For the foregoing reasons it is hereby ordered that NEC Contentions 2, 2A, 2B, 3 and 4 are resolved in favor of the applicant, Entergy. This initial decision shall constitute the final decision of the Commission forty (40) days from the date of its issuance, unless, within fifteen (15) days of its service, a petition for review is filed in accordance with 10 C.F.R. §§ 2.1212 and

2.341(b). Filing of a petition for review is mandatory for a party to exhaust its administrative remedies before seeking judicial review. 10 C.F.R. § 2.341(b)(1).

It is so ORDERED.

Respectfully Submitted,

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Dated: August 25, 2008

**UNITED STATES OF AMERICA**
**NUCLEAR REGULATORY COMMISSION**

Before the Atomic Safety and Licensing Board

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| 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) | ) | |

## CERTIFICATE OF SERVICE

I hereby certify that copies of the foregoing "Entergy's Proposed Findings of Fact and Conclusions of Law on New England Coalition Contentions" were served on the persons listed below by deposit in the U.S. Mail, first class, postage prepaid, and where indicated by an asterisk by electronic mail, this 25th day of August, 2008.

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