UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
ENTERGY NUCLEAR VERMONT YANKEE, LLC, and ENTERGY NUCLEAR OPERATIONS, INC.	)	Docket No. 50-271-LR
	)	ASLBP No. 06-849-03-LR
	)	
(Vermont Yankee Nuclear Power Station)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of the "NRC STAFF REBUTTAL TESTIMONY CONCERNING NEC CONTENTION 4" and "NOTICE OF APPERANCE" of Susan L. Uttal in the above-captioned proceeding have been served on the following by electronic mail with copies by deposit in the NRC's internal mail system or, as indicated by an asterisk, by electronic mail, with copies by U.S. mail, first class, this 2nd day of June, 2008.

Alex S. Karlin, Chair
Administrative Judge
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
E-mail: ask2@nrc.gov

William H. Reed*
Administrative Judge
Atomic Safety and Licensing Board
1819 Edgewood Lane
Charlottesville, VA 22902
E-mail: whrcville@embarqmail.com

Richard E. Wardwell
Administrative Judge
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
E-mail: rew@nrc.gov

Office of Commission Appellate
Adjudication
Mail Stop: O-16G4
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
E-mail: OCAAmail@nrc.gov

Office of the Secretary
Attn: Rulemakings and Adjudications Staff
Mail Stop: O-16G4
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
E-mail: hearingdocket@nrc.gov

Marcia Carpentier, Law Clerk
Atomic Safety and Licensing Board Panel
Mail Stop: T-3F23
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
E-mail: mxc7@nrc.gov

Lauren Bregman, Law Clerk
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Mail Stop: T-3 F23
Washington, D.C. 20555-0001
E-mail: lauren.bregman@nrc.gov

Peter C.L. Roth, Esq*
Office of the Attorney General
33 Capitol Street
Concord, NH 3301
E-mail: peter.roth@doj.nh.gov

Ronald A. Shems, Esq.*
Karen Tyler, Esq.
Shems Dunkiel Kassel & Saunders, PLLC
91 College Street
Burlington, VT 05401
E-mail: rshems@sdkslaw.com
Ktyler@sdkslaw.com

Anthony Z. Roisman, Esq.*
National Legal Scholars Law Firm
84 East Thetford Rd.
Lyme, NH 03768
E-mail: aroisman@nationallegalscholars.com

David R. Lewis, Esq.*
Matias F. Travieso-Diaz, Esq
Elina Teplinsky, Esq
Blake J. Nelson, Esq
Pillsbury Winthrop Shaw Pittman LLP
2300 N Street, NW
Washington, DC 20037-1128
E-mail: david.lewis@pillsburylaw.com
matias.travieso-diaz@pillsburylaw.com
elina.teplinsky@pillsburylaw.com
blake.nelson@pillsburylaw.com

Sarah Hofmann, Esq.*
Director of Public Advocacy
Department of Public Service
112 State Street - Drawer 20
Montpelier, VT 05620-2601
E-mail: sarah.hofmann@state.vt.us

Diane Curran*
Harmon, Curran, Spielberg, & Eisenberg, LLP
1726 M Street N.W., Suite 600
Washington, D.C. 20036
E-mail: dcurran@harmoncurran.com

James R. Milkey*
Assistant Attorney General, Chief
Environmental Protection Division
Office of the Attorney General
One Ashburton Place, 18th Floor
Boston, MA 02108
E-mail: jim.milkey@state.ma.us

/RA/

Lloyd B. Subin
Counsel for NRC Staff

June 2, 2008

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NRC STAFF REBUTTAL TESTIMONY CONCERNING NEC CONTENTION 4

INTRODUCTION

Pursuant to 10 C.F.R. § 2.1207(a)(2) and the "Initial Scheduling Order" (Nov. 17, 2006) (unpublished), the staff of the U.S. Nuclear Regulatory Commission ("Staff") hereby files rebuttal testimony of Kaihwa R. Hsu, a supporting affidavit and exhibits in response to New England Coalition, Inc.'s ("NEC") initial statement of position and testimony.[1] For the reasons set forth in the rebuttal testimony, the Staff again submits that NEC's challenge to the Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc.'s ("Entergy") application for renewal of the Vermont Yankee operating license cannot be sustained.

DISCUSSION

The issue in this rebuttal testimony addresses NEC contention 4, which is "a challenge to Entergy's plans for aging management of plant components subject to [flow-accelerated corrosion] FAC." LBP-06-20, 64 NRC 131, 194 (2006). The Staff maintains its position that

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[1] New England Coalition, Inc. Initial Statement of Position, Direct Testimony and Exhibits (Apr. 28, 2008).

Entergy's program for monitoring FAC is adequate. *See* NRC Staff Initial Statement of Position on NEC Contentions 2A, 2B, 3, and 4 (May 13, 2008), at 21.

Respectfully submitted,

/RA/

Lloyd B. Subin
Counsel for NRC Staff

Dated at Rockville, Maryland
this 2nd day of June, 2008

June 2, 2008

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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NRC STAFF REBUTTAL TESTIMONY OF KAIHWA R. HSU CONCERNING NEC CONTENTION 4

Q1. Please state your name, occupation, and by whom you are employed.

A1. My name is Kaihwa R. Hsu ("Hsu"). I am employed by the US Nuclear Regulatory Commission ("NRC") as a senior mechanical engineer in the Engineering Division in the Office of New Reactors. Previously I was employed as a materials engineer in the office of Nuclear Reactor Regulation Division of License Renewal. A statement of my professional qualifications was attached to the staff's "Affidavit of Kaihwa R. Hsu, Jonathan G. Rowley, and Thomas G. Scarbrough Concerning NEC Contention 3 (Steam Dryer)," filed May 13, 2008.

Q2. What is the purpose of this testimony?

A2. The purpose of this rebuttal testimony is to address the pre-filed written testimony and exhibits of Dr. Hausler, Mr. Witte, and Dr. Hopenfeld regarding New England Coalition, Inc.'s ("NEC") Contention 4, which were submitted on behalf of NEC on April 28, 2008.

Q3. In his "Discussion of the Empirical Modeling of Flow-Induced Localized Corrosion of Steel under High Shear Stress" ("NEC-RH_03"), Dr. Hausler expressed his concern for uncertainties in the methodology of ultrasonic thickness ("UT") measurements due to: 1) "[t]he inherent variability of the instrument with which the measurements are being made"; and 2) "[t]he inherent difficulty of placing the handheld UT probe at exactly the same location for repeat measurements one-and-a-half to two years apart." Exhibit NEC-RH_03 at Appx. A. Do you agree that these are areas of concern?

A3. No, I do not agree that these are areas of concern for UT measurements. First, inherent variability of the instrument is not a concern because in the nuclear industry, standards require that UT instruments be properly calibrated and UT technicians be trained to perform UT measurements. In addition, information regarding the accuracy of handheld UT probes is provided to the user by the manufacturer. Recent UT wall thickness technology has demonstrated that UT measurements are capable of attaining measurement accuracy for a high frequency UT transducer of +/- 0.01mm, which is significantly lower than +/- 1% to 2% of wall thickness claimed by Dr. Hausler, *see id.*

Second, any inherent difficulty in placing the probe in the same location for temporally separate repeat measurements has been eliminated because the plant has painted a permanent grid on the outside surface of the pipes. This permanent grid provides assurance that the probe will be placed in the same location for repeat measurements.

Q4. In NEC-RH_03, Dr. Hausler concluded "that the absolute minimum number of thickness measurements required for reasonably accurate prediction of failure is three, if an assessment of the confidence limits of the resulting trend is to be made." *Id.* at Appx. A. Do you agree with this statement?

A4. The Staff agrees that three measurements are required. In order for the aging management program to be consistent with GALL Report's recommendation, "limited baseline inspections to determine the extent of thinning at these locations" can be accomplished by performing two measurements, and "follow-up inspections to confirm the predictions" can be accomplished by performing a third measurement. NRC Staff Initial Statement of Position on

NEC contentions 2A, 2B, 3, and 4 (May 13, 2008), Exhibit 7 at XI M-61.

Although Dr. Hausler stated "that at least two measurements are needed to determine the rate of deterioration," he concludes that a minimum of three measurements are required for a reasonably accurate prediction of failure, if an assessment of the confidence limits is required. Exhibit NEC-RH_03 at Appx. A. This notion is based on his concern of uncertainty in the methodology of measurement. As described above in A3, the uncertainty has been essentially eliminated due to the accuracy of the measuring equipment. Therefore, the uncertainty concerns are not valid.

Q5. Mr. Witte, in his report regarding proposed aging management programs for flow-accelerated corrosion ("FAC"), stated that a concern "regarding deficiencies in implementation of the program brings into question the results of FAC inspection during RFO 25 and RFO 26 . . . ." Exhibit NEC-UW_03 at 2. Do you agree with this statement?

A5. No, I do not agree with this statement. There is no basis to question the results of the FAC inspection during RFO 25 and RFO 26. The results of these FAC inspections are actual UT wall thickness measurements that are independent of the CHECWORKS software or model update. CHECWORKS is used to manage and evaluate the actual wall thickness data to help trend/predict pipe failure due to FAC.

Q6. In his report regarding proposed aging management programs for FAC, Mr. Witte stated that

> [w]ith the exception of VY's [Vermont Yankee's] strength in reactively replacing piping or components with FAC-resistant material during repairs or maintenance, the program itself was not effective as a predictive modeling tool. Simply stated, once something ruptured or was found to be outside of its design margin, it was replaced in a reactive management approach. Proactive management of the program to *predict failures* has been inadequate in the FAC Program . . . .

Exhibit NEC-UW_03 at 7 (emphasis in original). Do you agree with this statement?

A6. No, I do not agree with this statement. The Staff's position is that a FAC program that performs inspections successfully, identifies critical FAC-susceptible components, and allows for replacement of piping and components with FAC-resistant material to prevent failure due to FAC, is effective.

Q7. In "Review of License Renewal Application for Vermont Yankee Nuclear Power Station: Program for Management of Flow-Accelerated Corrosion," Dr. Hopenfeld stated that

> [a]ccording to NUREG/CR-6936, *Probabilities of Failure and Uncertainty Estimate Information for Passive Components – a Literature Review* (May 2007) at Table 5.15, there were 250 through-wall pipe failures from FAC in BWRs and PWRs between 1988 and 2005, compared to 183 failures that occurred between 1976 and 1987. On a yearly basis, this represents a reduction of 2 failures per year during 1988-2005 period compared to the previous period, disregarding the number of reactors and their age. Since the CCC codes were introduced in 1987, one could attribute the 10% reduction to the CCC codes.

Exhibit NEC-JH_36 at 9. Do you agree with this statement?

A7. No, I do not agree with this statement. Dr. Hopenfeld's conclusions and use of the data is incorrect. NUREG/CR-6936 does not support his testimony.

Dr. Hopenfeld relies on data reported in NUREG/CR-6936, which was extracted from Appendix D of NUREG-1829 (Exhibit A, *Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process*, (Apr. 2008)). However, the service data extracted from NUREG-1829, as recorded in PIPExp. (a proprietary database for pipe failure experience), included applicable **worldwide** service experience, not just US nuclear industry data.

NUREG-1829 states that

> *D.3.2 2 FW Piping Service Experience* - Figures D.9 and D.10 summarize the service experience with FW piping. With respect to plant designed by General Electric, the Code Class I portion of BWR carbon steel feedwater piping has performed well in the field. There are no reported leaks in medium-or large diameter RCPB piping. Foreign plants have experienced (and in some cases, continue to experience) thermal fatigue damage due to thermal mixing and stratification. In fact, 80% of the degradation of the RCPB portions of FW piping has occurred in foreign plants with a piping system design that differs from that of U.S. BWR plants.

> The U.S. service experience includes a few instances of non-through wall cracking of FW nozzle-to-safe-end (bimetallic) welds. The root cause of the cracking is attributed to weld defects from original construction. As documented in Information Notice 92-35 [D. 19], Susquehanna Unit I has experienced flow-accelerated corrosion damage about 250 mm (10 inches) from a weld connecting NPS12 piping to a 20-inch by 12-inch reducing tee. **There have been no reported flaws in any U.S. plant beyond T = 15 years of operation.**

Id. at D-22 (emphasis added). Contrary to Dr. Hopenfeld's statement, this information indicates that the frequency of FAC related events at US nuclear plants has declined significantly and there have been no FAC-related injuries at US nuclear plants since improved FAC programs (e.g., EPRI guidelines, CHECWORKS) have been used in the US nuclear industry.

Q8. Mr. Witte stated in his report regarding proposed aging management programs for FAC that "VY is the first plant modified to achieve Constant Pressure Power Up-rate to 120% power and only one other plant out of the fleet of 104 was licensed to 120% increase in power in one step. Given the uniqueness of the design of VY's power up-rate, CHECWORKS has little industry benchmarking data, and is of marginal use." Exhibit NEC-UW_03 at 8. Is this a correct statement?

A8. No, this is not a correct statement. There is enough industry data regarding BWR extended uprates to demonstrate CHECWORKS benchmarking. For example, Dresden Units 2 & 3 extended 17% of their power from 2527 MWt to 2957 MWt; Quad Cities Units 1 & 2 extended 17.8% of their power from 2511 MWt to 2957 MWt; and Clinton extended 20% of its power from 2894 MWt to 3473 MWt. Exhibit B, *Approved Application for Power Uprates,* http://www.nrc.gov/reactors/operating/licensing/power-uprates/approved-applications.html (last visited June 2, 2008). In comparison, VY extended 20% of its power from 1593 MWt to 1912 MWt. *Id.* The original power levels of Dresden, Quad Cities, and Clinton are much greater than VY's extended power level. The Staff's position is that the above plants are comparable to VY, and therefore, there is enough industry data to demonstrate benchmarking for extended power uprates.

CHECWORKS was designed for FAC prediction at the power levels at which the plant is being operated. The program does not recognize whether the power levels have been uprated or remain at lower levels, the use of the program remains unchanged. Data from plants that have a power level much higher than VY's extended power level have already been considered in the CHECWORKS development. Therefore, it is not accurate to say CHECWORKS only has a marginal use.

Q9. In "Review of License Renewal Application for Vermont Yankee Nuclear Power Station," Dr. Hopenfeld asserts that "[t]o account for local turbulence . . . , the grid should be kept to below 1" x 1" inch." Exhibit NEC-JH_36 at 15. Do you agree with this statement?

A9. No, I do not agree with this statement. If the grid is kept to below 1 inch by 1 inch, then inspection of more than 6,000 points for a 30 inch diameter, long radius elbow would be required. This is not necessary or feasible. The FAC failure cases have demonstrated that a large bore pipe failure occurs over much more than a 1 inch x 1 inch area.

Q10. Dr. Hopenfeld has previously stated that "[i]t is important to realize that wall thinning rate from FAC is not necessarily constant with time, and therefore a considerable number of cycles are needed to establish the FAC rate on a given component at a particular plant." Exhibit C, Petition for Leave to Intervene, Request for Hearing and Contentions (May 26, 2006) at Exhibit 7 ¶ 24 (ADAMS ML061640032). In contrast, NEC's pre-filed exhibit, NEC-JH_37, indicates that there is a linear relationship between FAC degradation and time. Do you agree with Dr. Hopenfeld's previous statement or with the information submitted in NEC Exhibit NEC-JH_37?

A10. I agree with the information in NEC-JH_37. The laboratory data and plant data shown in Figures 4 and 5 of NEC-JH_37 clearly demonstrate that the FAC degradation is linear with time. This is the basis for trending/predicting FAC failure date. This algorithm has been adopted for all plants, including those plants not using CHECWORKS.

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AFFIDAVIT OF KAIHWA R. HSU

I, Kaihwa R. Hsu, do hereby declare under penalty of perjury that my statements in the foregoing testimony are true and correct to the best of my knowledge and belief.

Kaihwa R. Hsu

KAIHWA R. HSU

Executed at Rockville, MD
this 2nd day of June, 2008

EXHIBIT A

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

NUREG-1829
Vol. 1

Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process

Main Report

Office of Nuclear Regulatory Research

U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

NUREG-1829
Vol. 1

Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process

Main Report

Manuscript Completed: March 2008
Date Published: April 2008

Prepared by
R. Tregoning (NRC), L. Abramson (NRC)
P. Scott (Battelle-Columbus)

A. Csontos, NRC Project Manager

Office of Nuclear Regulatory Research

U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

NUREG-1829
Vol. 2

Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process

Appendices A through M

Office of Nuclear Regulatory Research

U.S.NRC
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NUREG-1829
Vol. 2

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Appendices A through M

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A. Csontos, NRC Project Manager

Office of Nuclear Regulatory Research

APPENDIX D

PIPING BASE CASE RESULTS OF BENGT LYDELL

An Application of the Parametric Attribute-Influence Methodology to Determine Loss of Coolant Accident (LOCA) Frequency Distributions

Report No. 2 to the NRC Expert Panel on LOCA Frequency Distributions

Prepared for

U.S. Nuclear Regulatory Commission
Washington (DC)

June 2004

D.3.2.2 FW Piping Service Experience - Figures D.9 and D.10 summarize the service experience with FW piping. With respect to plant designed by General Electric, the Code Class 1 portion of BWR carbon steel feedwater piping has performed well in the field. There are no reported leaks in medium-or large-diameter RCPB piping. Foreign plants have experienced (and in some cases, continue to experience) thermal fatigue damage due to thermal mixing and stratification. In fact, 80% of the degradation of the RCPB portions of FW piping has occurred in foreign plants with a piping system design that differs from that of U.S. BWR plants.

The U.S. service experience includes a few instances of non-through wall cracking of FW nozzle-to-safe-end (bimetallic) welds. The root cause of the cracking is attributed to weld defects from original construction. As documented in Information Notice 92-35 [D.19], Susquehanna Unit 1 has experienced flow-accelerated corrosion damage about 250 mm (10 inches) from a weld connecting NPS12 piping to a 20-inch by 12-inch reducing tee. There have been no reported flaws in any U.S. plant beyond T = 15 years of operation.

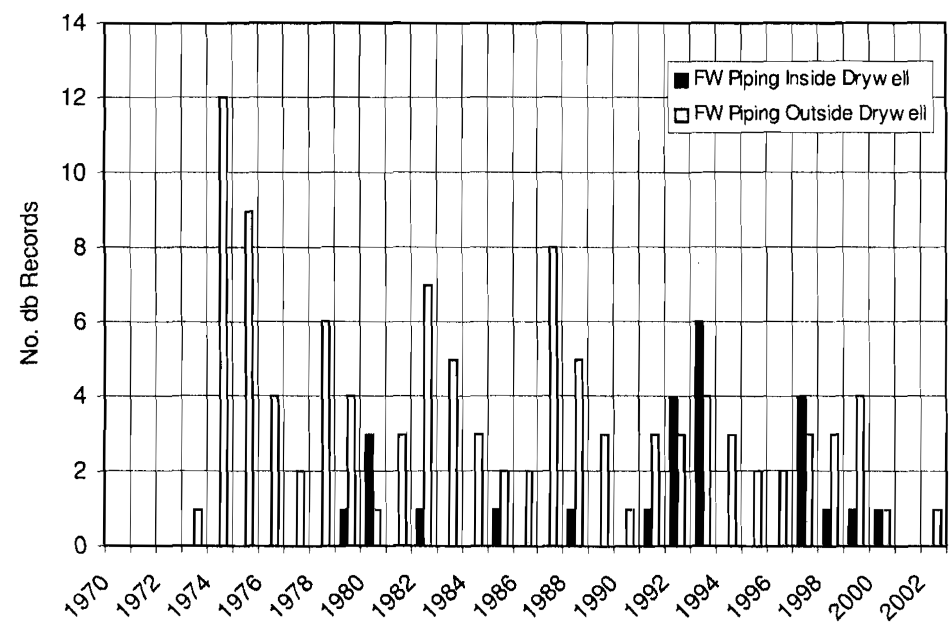

Figure D.9 Service Experience with FW Piping (i)

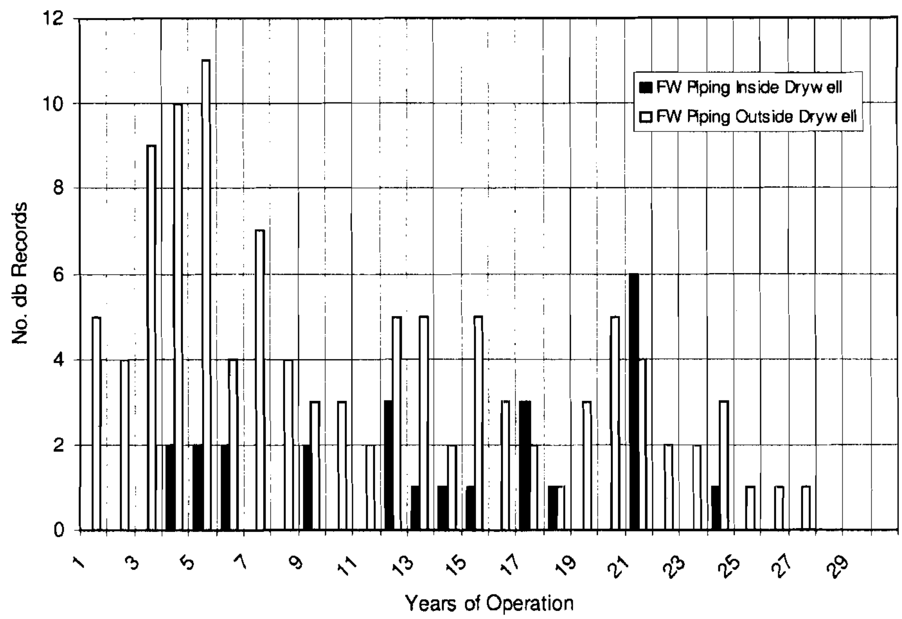

Figure D.10 Service Experience with FW Piping (ii)

D.3.3 Review of PWR-Specific Piping Service Experience

Limited to the PWR Base Case systems, this section summarizes the service experience with Code Class 1 piping. The results of this review are input to the pipe failure rate estimation.

D.3.3.1 RC & HPI/NMU Piping Service Experience - There have only been a limited number of events involving through-wall cracks in the large-diameter RC piping and the Class 1 portion of SI/CV piping. Evidence of axial primary water stress corrosion cracking (PWSCC) in the bimetallic safe-end to RPV nozzle welds of the RC-HL piping has been reported at Ringhals [D.20] and V.C. Summer [D.21].

During an eight-year period, the now decommissioned Trojan nuclear power plant experienced pressurizer surge line movement, which was attributed to thermal stratification [D.22]. In response, the NRC issued Bulletin 88-11 in December of 1988 [D.23] requesting that licensees perform visual inspections of the pressurizer surge line at the first available cold shutdown. Purpose of the inspections was to determine presence of any "gross discernible distress or structural damage in the entire pressurizer surge line, including piping, pipe supports, pipe whip restraints, and anchor bolts."

The current version (June 2004) of the PIPExp database includes four records associated with degradation of pressurizer surge lines:

- Record # 19849; during the Three Mile Island-1 2003 Refueling Outage (18-Oct-2003 to 3-Dec-2003), a UT examination found an axial flaw about 13 mm (0.5-inch) deep in the surge line nozzle-to-safe end interface in dissimilar metal weld No. SR0010BM. This weld connects a 10-inch Schedule 140, carbon steel nozzle to stainless steel safe end.
- Record # 19736; in November 2002 during UT examination of RC piping in the Belgian plant Tihange-2 (a 900 MWe series plant designed by Framatome), code rejectable indications were

EXHIBIT B

Approved Applications for Power Uprates

The following power uprates have been reviewed and accepted by the NRC. The licenses for the following plants have been amended to reflect the increase in power level shown in the table.

(TYPE -- S = Stretch; E = Extended; MU = Measurement Uncertainty Recapture)

The following links on this page are to documents in our Agencywide Documents Access and Management System (ADAMS). ADAMS documents are provided in either Adobe Portable Document Format (PDF) or Tagged Image File Format (TIFF). To obtain free viewers for displaying these formats, see our Plugins, Viewers, and Other Tools. If you have problems with viewing or printing documents from ADAMS, please contact the Public Document Room staff.

NO.	PLANT	% UPRATE	MWt	DATE APPROVED	TYPE	ACCESSION #
1	Calvert Cliffs 1	5.5	140	09/26/77	S	
2	Calvert Cliffs 2	5.5	140	11/08/77	S	
3	Millstone 2	5	140	06/25/79	S	7907240100*
4	H. B. Robinson	4.5	100	06/29/79	S	7907180064*
5	Fort Calhoun	5.6	80	08/15/80	S	8008280223*
6	St. Lucie 1	5.5	140	11/23/81	S	ML013530273
7	St. Lucie 2	5.5	140	03/01/85	S	ML013600080
8	Duane Arnold	4.1	65	03/27/85	S	ML021890435
9	Salem 1	2	73	02/06/86	S	ML011660249
10	North Anna 1	4.2	118	08/25/86	S	ML013460131
11	North Anna 2	4.2	118	08/25/86	S	ML013460131
12	Callaway	4.5	154	03/30/88	S	ML021650524
13	TMI-1	1.3	33	07/26/88	S	ML003779786
14	Fermi 2	4	137	09/09/92	S	ML020720520
15	Vogtle 1	4.5	154	03/22/93	S	ML012330056
16	Vogtle 2	4.5	154	03/22/93	S	ML012330056
17	Wolf Creek	4.5	154	11/10/93	S	ML022030519
18	Susquehanna 2	4.5	148	04/11/94	S	ML010170334
19	Peach Bottom 2	5	165	10/18/94	S	ML011490143
20	Limerick 2	5	165	02/16/95	S	ML011560773
21	Susquehanna 1	4.5	148	02/22/95	S	9503070354*
22	Nine Mile Point 2	4.3	144	04/28/95	S	9505090259*

23	WNP-2	4.9	163	05/02/95	S	ML022120154
24	Peach Bottom 3	5	165	07/18/95	S	ML021580312
25	Surry 1	4.3	105	08/03/95	S	ML012710328
26	Surry 2	4.3	105	08/03/95	S	ML012710328
27	Hatch 1	5	122	08/31/95	S	ML013020073
28	Hatch 2	5	122	08/31/95	S	ML013020073
29	Limerick 1	5	165	01/24/96	S	ML011560244
30	V. C. Summer	4.5	125	04/12/96	S	ML012320013
31	Palo Verde 1	2	76	05/23/96	S	ML021710572
32	Palo Verde 2	2	76	05/23/96	S	ML021710572
33	Palo Verde 3	2	76	05/23/96	S	ML021710572
34	Turkey Point 3	4.5	100	09/26/96	S	ML013390234
35	Turkey Point 4	4.5	100	09/26/96	S	ML013390234
36	Brunswick 1	5	122	11/01/96	S	9611070136*
37	Brunswick 2	5	122	11/01/96	S	9611070136*
38	Fitzpatrick	4	100	12/06/96	S	9612180303*
39	Farley 1	5	138	04/29/98	S	ML012140259
40	Farley 2	5	138	04/29/98	S	ML012140259
41	Browns Ferry 2	5	164	09/08/98	S	ML042670045
42	Browns Ferry 3	5	164	09/08/98	S	ML042670045
43	Monticello	6.3	105	09/16/98	E	ML020920138
44	Hatch 1	8	205	10/22/98	E	ML013030084
45	Hatch 2	8	205	10/22/98	E	ML013030084
46	Comanche Peak 2	1	34	09/30/99	MU	ML021820306
47	LaSalle 1	5	166	05/09/00	S	ML003716743
48	LaSalle 2	5	166	05/09/00	S	ML003716743
49	Perry	5	178	06/01/00	S	ML003724441
50	River Bend	5	145	10/06/00	S	ML003762072
51	Diablo Canyon 1	2	73	10/26/00	S	ML003764792
52	Watts Bar	1.4	48	01/19/01	MU	ML010260074
53	Byron 1	5	170	05/04/01	S	ML011420274
54	Byron 2	5	170	05/04/01	S	ML011420274
55	Braidwood 1	5	170	05/04/01	S	ML011420274
56	Braidwood 2	5	170	05/04/01	S	ML011420274
57	Salem 1	1.4	48	05/25/01	MU	ML011350051
58	Salem 2	1.4	48	05/25/01	MU	ML011350051
59	San Onofre 2	1.4	48	070/6/01	MU	ML012180231
60	San Onofre 3	1.4	48	07/06/01	MU	ML012180231
61	Susquehanna 1	1.4	48	07/06/01	MU	ML011760551
62	Susquehanna 2	1.4	48	070/6/01	MU	ML011760551

63	Hope Creek	1.4	46	07/30/01	MU	ML011910345
64	Beaver Valley 1	1.4	37	09/24/01	MU	ML012490569
65	Beaver Valley 2	1.4	37	09/24/01	MU	ML012490569
66	Shearon Harris	4.5	138	10/12/01	S	ML012830516
67	Comanche Peak 1	1.4	47	10/12/01	MU	ML012550246
68	Comanche Peak 2	0.4	13	10/12/01	MU	ML012550246
69	Duane Arnold	15.3	248	11/06/01	E	ML013050342
70	Dresden 2	17	430	12/21/01	E	ML013540187
71	Dresden 3	17	430	12/21/01	E	ML013540187
72	Quad Cities 1	17.8	446	12/21/01	E	ML013540222
73	Quad Cities 2	17.8	446	12/21/01	E	ML013540222
74	Waterford 3	1.5	51	03/29/02	MU	ML020910734
75	Clinton	20	579	04/05/02	E	ML021650543
76	South Texas 1	1.4	53	04/12/02	MU	ML020800263
77	South Texas 2	1.4	53	04/12/02	MU	ML020800263
78	ANO-2	7.5	211	04/24/02	E	ML021130826
79	Sequoyah 1	1.3	44	04/30/02	MU	ML021220060
80	Sequoyah 2	1.3	44	04/30/02	MU	ML021220060
81	Brunswick 1	15	365	05/31/02	E	ML021440346
82	Brunswick 2	15	365	05/31/02	E	ML021440346
83	Grand Gulf	1.7	65	10/10/02	MU	ML022630304
84	H. B. Robinson	1.7	39	11/05/02	MU	ML023100365
85	Peach Bottom 2	1.62	56	11/22/02	MU	ML031000317
86	Peach Bottom 3	1.62	56	11/22/02	MU	ML031000317
87	Indian Point 3	1.4	42.4	11/26/02	MU	ML023290636
88	Point Beach 1	1.4	21.5	11/29/02	MU	ML023370133
89	Point Beach 2	1.4	21.5	11/29/02	MU	ML023370133
90	Crystal River 3	0.9	24	12/04/02	S	ML023380800
91	D.C. Cook 1	1.66	54	12/20/02	MU	ML023470126
92	River Bend	1.7	52	01/31/03	MU	ML030340294
93	D.C. Cook 2	1.66	57	05/02/03	MU	ML030990129
94	Pilgrim	1.5	30	05/09/03	MU	ML031220007
95	Indian Point 2	1.4	43	05/22/03	MU	ML031420375
96	Kewaunee	1.4	23	07/08/03	MU	ML031530734
97	Hatch 1	1.5	41	09/23/03	MU	ML032590944
98	Hatch 2	1.5	41	09/23/03	MU	ML032590944
99	Palo Verde 2	2.9	114	09/29/03	S	ML032720538
100	Kewaunee	6	99	02/27/04	S	ML040430633
101	Palisades	1.4	35.4	06/23/04	MU	ML040970622
102	Indian Point 2	3.26	101.6	10/28/04	S	ML042960007

103	Seabrook	5.2	176	02/28/05	S	ML050140453
104	Indian Point 3	4.85	148.6	03/24/05	S	ML050600380
105	Waterford	8.0	275	04/15/05	E	ML051030068
106	Palo Verde 1	2.9	114	11/16/05	S	ML053130286
107	Palo Verde 3	2.9	114	11/16/05	S	ML053130286
108	Vermont Yankee	20	319	03/02/06	E	ML060050024
109	Seabrook	1.7	61	05/22/06	MU	ML061430044
110	Ginna	16.8	255	07/11/06	E	ML061380133
111	Beaver Valley 1	8	211	07/19/06	E	ML061720274
112	Beaver Valley 2	8	211	07/19/06	E	ML061720274
113	Browns Ferry 1	5	165	03/06/07	S	ML070680307
114	Crystal River 3	1.6	41	12/26/07	MU	ML073610197
115	Susquehanna 1	13	463	01/30/08	E	
116	Susquehanna 2	13	463	01/30/08	E	
117	Vogtle 1	1.7	60.6	02/27/08	MU	ML080350345
118	Vogtle 2	1.7	60.6	02/27/08	MU	ML080350345
	Total MWt		15788.2			
	Total MWe		5263			

*Documents can be requested from the Public Document Room

Capacity Recapture Power Uprates for Provisional Operating License Plants are not included in this table. These are Haddam Neck uprate of 24% in 1969, Oyster Creek uprate of 14% in 1971, Palisades uprate of 15% in 1977, Ginna uprate of 17% in 1984, Maine Yankee uprate of 10% in 1989, and Indian Point 2 Uprate of 11% in 1990.

NOTE:The NRC staff approved an MUR power uprate for Fort Calhoun on January 16, 2004, which authorized an increase in the licensed thermal power limit to 1,524 megawatts-thermal. The Omaha Public Power District was subsequently informed by Westinghouse that the potential instrument inaccuracies in the Advanced Measurement and Analysis Group (AMAG) ultrasonic flow meter would not allow implementation of the MUR power uprate at Fort Calhoun. As a result, on May 7, 2004, prior to implementation of the MUR power uprate, the Omaha Public Power District submitted an exigent license amendment request to return Fort Calhoun's licensed thermal power limit to 1,500 megawatts-thermal, the pre-MUR level. On May 14, 2004, the NRC staff approved this license amendment.

EXHIBIT C

RAS 11759

SHEMS DUNKIEL KASSEL & SAUNDERS PLLC

RONALD A. SHEMS
BRIAN S. DUNKIEL*
JOHN B. KASSEL
MARK A. SAUNDERS

GEOFFREY H. HAND
KAREN L. TYLER
ASSOCIATE ATTORNEYS

ANDREW N. RAUBVOGEL
EILEEN I. ELLIOTT
OF COUNSEL

DOCKETED
USNRC

May 30, 2006 (3:30pm)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

May 26, 2006

Office of the Secretary
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001
Attn: Rulemaking and Adjudications Staff

Re: *In the matter of* ENTERGY NUCLEAR VERMONT YANKEE, LLC and ENTERGY NUCLEAR OPERATIONS, INC., Vermont Yankee Nuclear Power Station License Renewal Application, Docket No. 50-271

Dear Sir or Madam:

Please find enclosed for filing in the above stated matter New England Coalition's Petition for Leave to Intervene, Request for Hearing, and Contentions; and the Notice of Appearance on behalf of New England Coalition of Shems Dunkiel Kassel & Saunders PLLC by attorneys Ronald A. Shems, and Karen Tyler.

Thank you for your attention to this matter.

Sincerely,

Karen Tyler

Karen Tyler

Enclosures

cc: see attached Certificate of Service

91 COLLEGE STREET BURLINGTON, VERMONT 05401
TEL 802 / 860 1003 FAX 802 / 860 1208 - www.sdkslaw .com

Template = SECY-037

SECY 02

*Also admitted in the District of Columbi

EXHIBIT 7

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the matter of		
ENTERGY NUCLEAR VERMONT YANKEE, LLC	)	
and ENTERGY NUCLEAR OPERATIONS, INC.	)	NO. 50-271
Vermont Yankee Nuclear Power Station	)	
License Renewal Application	)	

DECLARATION OF DR. JORAM HOPENFELD

1. My name is Dr. Joram Hopenfeld. The New England Coalition (NEC) has retained me as an expert witness in proceedings concerning the application of Entergy Nuclear Operations, Inc. ("Entergy") to renew its operating license for Vermont Yankee Nuclear Power Station ("Vermont Yankee") for twenty years beyond the current expiration date of March 21, 2012.

2. I am a mechanical engineer and hold a doctorate in engineering. I have 45 years of professional experience in the fields of instrumentation, design, project management, and nuclear safety, including 18 years in the employ of the U.S. Nuclear Regulatory Commission. My curriculum vitae is attached to this declaration as Attachment A.

3. I have reviewed Entergy's License Renewal Application, and such publicly available documents as are relevant to the subjects of my declaration.

CONTENTION TWO

4. Paragraphs 4 - 14 of this declaration concern NEC's "Contention Two." I refer to the following documents:

22. The Entergy program to manage the effects of Flow-Accelerated Corrosion (FAC) is based on NUREG 1801 § XI.M17 and EPRI Report NSAC-202L-R2. License Renewal Application Table 3.4.1 ¶ 3.4.1-29, and Appendix B § B.1.13. These guidance documents recommend use of a computer code, CHECWORKS, to recommend the scope and frequency of in-service inspections. It can be reasonably deduced that Entergy proposes to use the CHECWORKS code to manage FAC during the new license term.

23. Because Entergy has recently increased the operating power level of its plant by 20%, CHECWORKS would require additional inputs before it can be used at the VY plant as an adequate FAC management tool. Consequently, the proposed program, as presented in the Entergy Application, will not be valid throughout the entire period of the extended plant operation.

24. The theoretical basis of FAC is not completely understood; however, it is well established that turbulence intensity, steam quality, material compositions, oxygen content and coolant pH are the main variables that affect FAC. The CHECWORKS computer code is not a mechanistic code; it is an empirical code that must be updated continuously with plant-specific data. Inspection results are routinely used as inputs to the code. The code can be used to predict pipe wall thinning as long as plant parameters (velocity, coolant chemistry, etc.) do not change drastically and the data has been collected for a long period of time. It is important to realize that wall thinning rate from FAC is not necessarily constant with time, and therefore a considerable number of cycles are needed to establish the FAC rate on a given component at a particular plant. Since Vermont Yankee has recently increased the coolant flow rate by 20%, which also

significantly accelerates local wall thinning, it would take at least 10-15 years before CHECWORKS can be benchmarked with the Vermont Yankee inspection data.

25. The inability to reliably predict wall thinning from FAC has been very costly. In 1986, a feed water pipe elbow ruptured at the Surry Nuclear plant. There were several fatalities and the reactor was down for several months. The accident resulted from severe pipe walls thinning due to FAC (References a & b). In 1991 and in 1993, the feed ring and the J tubes at San Onofre's steam generators (References c and d) failed from FAC. In 1997, extraction steam piping ruptured at the Fort Calhoun Station (Reference e). In July 2004, several workers were killed at the Mihama nuclear power plant due to FAC in the secondary loop (Reference f).

26. The above is only a partial list of the failures that occurred from FAC in nuclear plants. This list alone, however, is sufficient to demonstrate that CHECWORKS (developed in 1987) has not been successful in averting major catastrophes and costly outages. The prediction of FAC is an art not a science and must be obtained empirically and with expert engineering judgment. The above plant experience indicates that a lack of proper FAC controls can lead to very serious consequences.

27. The key to a valid FAC program is the ability to adequately specify the frequency of inspections. CHECWORKS cannot be used to provide adequate guidance regarding inspection frequencies. Therefore, 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 the ASME code limits during the period of extended operation.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
ENTERGY NUCLEAR VERMONT	)	Docket Nos. 50-271-LR
YANKEE, LLC, and ENTERGY	)	
NUCLEAR OPERATIONS, INC	)	ASLBP No. 06-849-03-LR
	)	
(Vermont Yankee Nuclear Power Station)	)	

NOTICE OF APPEARANCE

Notice is hereby given that the undersigned attorney herewith enters an appearance in the captioned matter in accordance with 10 C.F.R. § 2.314(b).

Name:	Susan L. Uttal
Address:	U.S. Nuclear Regulatory Commission Office of the General Counsel Mail Stop: O-15-D21 Washington, D.C. 20555-0001
Telephone Number:	(301) 415-1582
Fax Number:	(301) 415-3725
E-Mail Address:	susan.uttal@nrc.gov
Admissions:	New Jersey; Pennsylvania; U.S. Court of Appeals 3rd Circuit
Name of Party:	NRC Staff

Respectfully submitted,

/RA/

Susan L. Uttal
Counsel for NRC Staff

Dated at Rockville, Maryland
this 2nd day of June, 2008