RAS M-394

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December 18, 2009 (8:00am)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	December 17, 2008
ENTERGY NUCLEAR VERMONT YANKEE, L.L.C., and ENTERGY NUCLEAR) OPERATIONS, INC.	Docket No. 50-271-LR
	ASLBP-08-25
(Vermont Yankee Nuclear Power Station)	

NEW ENGLAND COALITION'S MOTION FOR RECONSIDERATION OF THE LICENSING BOARD'S PARTIAL INITIAL DECISION

I. INTRODUCTION

New England Coalition, Inc ("NEC ") through its pro se representative, Raymond Shadis, hereby moves for reconsideration of the Atomic Safety and Licensing Board's ("Board") Partial Initial decision of November 24,2008.

Pursuant to 10 C.F.R. 2.345[1] and 2.345 (2)(b), NEC submits that reconsideration is warranted because the Board's Partial Initial Decision (Ruling on Contentions 2A, 2B, 3

[1] **§ 2.345 Petition for Reconsideration.**
(a)(1) Any petition for reconsideration of a final decision must be filed by a party within ten (10) days after the date of the decision.
(2) Petitions for reconsideration of Commission decisions are subject to the requirements in § 2.341(d).
(b) A petition for reconsideration must demonstrate a compelling circumstance, such as the existence of a clear and material error in a decision, which could not have been reasonably anticipated, which renders the decision invalid. The petition must state the relief sought.
Within ten (10) days after a petition for reconsideration has been served, any other party may file an answer in opposition to or in support of the petition.

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and 4) rests on clearly erroneous findings of fact that could not be anticipated and conclusions based on the erroneous findings of fact that therefore also could not be anticipated.

The Board's ruling also made a number of clearly erroneous findings regarding:

(A) the nature and relative importance of the many physical phenomena that must be known and understood, and

(B) the informed engineering considerations that must be brought to bear in order to provide adequate assurance of public health and safety, in aging management of reactor components and high energy piping systems.

The Board's ruling also made a number of clearly erroneous findings that are unsupportable in light of the record viewed in its entirety.

The Board ruling contained findings and conclusions that unfairly favored, as more credible, the verbal opinions of less qualified witnesses unsupported by any documents or data, over the document and data supported written and oral testimony of much more highly qualified witnesses. In fact the Board itself provided testimony regarding metal fatigue factors that it later reiterated as a basis for a finding of fact even though it was based on an assumption that was disputed by both Entergy and NEC experts.

(c) Neither the filing nor the granting of the petition stays the decision unless the Commission orders otherwise.

The Board permitted the licensee to introduce new testimony to the evidentiary hearing in the form of a slide show-illustrated tutorial by licensee witness and vendor but refused to permit NEC to make a countervailing presentation. In fact, the Board refused to permit NEC to show, for discussion purposes, an enlarged version of an exhibit graph that in its original size had already been introduced into evidence. The Board permitted the licensee to produce and distribute to the Board a table purporting to list a history of plant transients, albeit without authentication or the opportunity for the intervenors to review beforehand. The Board permitted the last minute (eve of hearing) of written testimony in the form of calculations by Entergy witness Fitzpatrick even though intervenors had no opportunity to review the material, The Board permitted the introduction of the testimony of NRC's witness, Dr. Kenneth Chang, even though Dr. Chang was not present to be examined on it.

Thus the Board repeatedly deprived the intervenor of a reasonable opportunity to review and controvert opposing witness testimony-both written and oral.

In a Subpart L Hearing, where intervenors may not cross-examine, the Board's careful, probing, examination is all the more important to fact finding and to building a sound record.

The Board's Ruling is made under the influence of false and misleading testimony. Under 10 C.F.R. §2.337 (f)(2)[2] NEC will controvert that testimony with relevant documents.

The Board's seemingly uncritical acceptance in the evidentiary hearing of the unsupported opinions or recollections of licensee and NRC Staff witnesses as, "evidence", on a par with prefiled written testimony and in preference to the document and data supported testimony of NEC witnesses calls into question if this was a fair hearing meeting the NRC's requisite standards of "fair" or not.

The ASLB's first duty is to assure the adequate protection of public health and safety,

> An application to renew the operating license of a commercial nuclear power plant may be granted only if the Commission finds that the continued operation of the facility "will be in accord with the common defense and security and will provide adequate protection to the health and safety of the public."
> 42 U.S.C. § 2232(a). PID 8

A finding of "reasonable assurance that there will be adequate protection to the health and safety of the public" is based on judgment Like "adequate protection," the phrase "reasonable assurance" is a determination that the NRC bases upon full consideration of all relevant information.[3]

Therefore, NEC respectfully requests that the Board suspend, reverse or modify

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[2] **10 C.F.R. §2.337 ((2)** If a decision is stated to rest in whole or in part on official notice of a fact which the parties have not had a prior opportunity to controvert, a party may controvert the fact by filing an appeal from an initial decision or a petition for reconsideration of a final decision.
The appeal must clearly and concisely set forth the information relied upon to controvert the fact.

[3] **PID- 11 - Footnote 26** A finding of "reasonable assurance that there will be adequate protection to the health and safety of the public" is based on judgment, not on the application of a mechanical verbal formula, a set of objective standards, or specific confidence interval. See Union of Concerned Scientists v. NRC, 880 F.2d 552, 558 (D.C. Cir. 1989)

decision with respect to NEC Contentions 2, 2A and 2B, consider anew the evidence in the light of the discussion in this Motion to Reconsider, and if need be reopen the record to take new evidence.[4] In the alternative, NEC respectfully requests that the Board submit its findings and the evidentiary record to review by a panel of competent, knowledgeable experts in the disciplines required to ascertain within the highest professional standards that the Board's Partial Initial Decision provides adequate assurance of public health and safety.

Further, NEC respectfully requests, in consideration of the information herein presented by Dr. Hausler, an amended decision and order requiring the Checworks program at Vermont Yankee be precisely benchmarked through a campaign of detailed measurement, while taking into consideration extended power uprate parameters, of all piping points known to be in FAC or FILC susceptible locations; entering them into entering them into the Vermont Yankee Checworks database and then applying a rigorous regimen to the program maintenance.

DISCUSSION

In support of its Motion for Reconsideration, New England Coalition presents:

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[4] [T]he mechanism of post-hearing resolution must not be employed to obviate the basic findings prerequisite to an operating license - **including a reasonable assurance that the facility can be operated without endangering the health and safety of the public**. In short, the 'post-hearing' approach should be employed sparingly and only in clear cases. **In doubtful cases, the matter should be resolved in the adversary framework prior to issuance of license, reopening the record if necessary**. Indian Point, CLI-74-23, 7 AEC at 951-52. ***Emphasis added***

A. **The Declaration of Dr. Joram Hopenfeld.** Dr. Hopenfeld holds a PhD in nuclear engineering and has absorbed a lifetime of learning and experience in the various physical, chemical, and mechanical phenomena associated with the metal fatigue and pipe thinning issues addressed in NEC Contentions 2, 2a, 2b, 3, and 4. In his declaration Dr. Hopenfeld provides clarification of phenomena and engineering considerations which were reflected in the findings of fact, conclusions, and order in the PID. Dr. Hopenfeld further offers criticism of the conduct of examination of witnesses in the evidentiary hearing; not from a legal standpoint, but from the standpoint of scientific inquiry. Dr. Hopenfeld attests to critical examples of instances where not only were the findings of fact incorrectly drawn, but the very examination questions (and responses) that led to the findings, connote technically incorrect bases and assumptions. Dr. Hopenfeld's Declaration (Exhibit JH MR 1) is attached together with a supporting document (JH MR 2 Industry Comments to NRC ...).

B. The **Declaration of Dr. Rudolf Hausler.** Dr. Hausler is a corrosion specialist and a principal of Corro-Consulta of Kaufman, Texas. Dr. Hausler holds a PhD in chemical engineering. Dr. Hausler takes issue with the Board's dismissal of the issues surrounding proper implementation of the Checworks program for flow accelerated corrosion (FAC). Dr. Hausler explains the importance of accurately locating multiple piping inspection points susceptible to flow induced localized corrosion (FILC. Dr. Hausler provides a short treatise on the subject and explains why it is important for both operators and regulators to have a clear understanding of the phenomena involved; something that was not developed in the proceedings; therefore leading to erroneous findings and a defective PID. Dr. Hausler further explains both the necessity to an Aging Management Program and the shortcomings in the application of Checworks as it is applied a Vermont Yankee. Dr. Hausler recommends re-base lining or more precisely benchmarking the Checworks program taking into consideration extended power uprate parameters and then applying a rigorous maintenance regimen to the program application. Dr. Hausler recommends an amended decision and order to that effect. Attached Exhibits: RH MR 1 Declaration, and RH MR 2 Signed Non-Disclosure

C. **The Declaration of Mr. Ulrich Witte.** Mr. Witte holds a Degree in Physics and had been employed in the nuclear industry for over seventeen years; specializing in systems configuration management. Mr. Witte has review the PID and the testimony , as well as voluminous Vermont Yankee FAC management documents and records. Mr. Witte explains why Vermont Yankee's FAC program as it is proposed for the extended period of operation will not provide adequate assurance of public health and safety

Exhibits: UW MR 1 Affidavit and Declaration, Exhibits NEC Motion for Reconsideration 1-4

D. New Evidence, is sponsored by NEC's pro se representative, that clearly illustrates that the Board was mislead by Entergy witness, James Fitzpatrick, on an issue of some consequence regarding both metal fatigue and flow accelerated corrosion, that is, the presence of water impurities in the circulating steam and feedwater systems at Vermont Yankee. NEC's particular concern is that because the issue of impurities in the steam/water system was raised by NEC, the Board's acceptance of Entergy's dismissal of the issue results in an impeachment of NEC's concerns and a tainting of NEC testimony overall. On July 22nd, Judge Reed engaged Mr. Fitzpatrick in the following exchange:

TR. 1172
JUDGE REED: We're really confused, still,
Tr.1172
about this issue of trace elements and impurities.
And, first, I want to clarify that we are talking
about trace elements in the fluid itself, not in the
metal. Is that correct? In your earlier testimony
about trace elements --
MR. STEVENS: Yes.
JUDGE REED: -- we were speaking about
impurities within the coolant.
MR. STEVENS: Correct.
JUDGE REED: Okay. And so I believe your
testimony was that they were not considered because
you felt it was unlikely that they would be present
during a transient.
MR. STEVENS: Correct....
JUDGE REED: Now, it has been brought to
our attention that there was an incident in which
there was a leakage of service water through the
condenser. Was it -- is it possible that impurities
were injected into the system as a result of that
incident?
MR. STEVENS: I can't speak to that.
MR. FITZPATRICK: What date is the
incident?
JUDGE REED: I'm assuming it was probably
this incident in 2004, but I'm not certain.
1173
MR. FITZPATRICK: Service water?--
JUDGE REED: Pardon me?
MR. FITZPATRICK: Service water does not connect
[to] the condenser that -- under normal operations.
JUDGE REED: All right. So that answers
our question. Thank you.

Joint Proposed transcript Changes August 13, 2008
(changes entered above and underlined)

7/22/08 1172/4 Change "mud" to "metal"
7/22/08 1172/8 Change "cooling" to "coolant"
7/22/08 1173/1 Change "Some sort of-" to "Service water?"
7/22/08 1173/3 Change "Some sort of impact to" to "Service water does not connect"

Mr. Fitzpatrick misled the Judge one more than one count. The service water lows into the cooling tower basin and over the steam condenser, as does mineral and halogen laden river water. Although he testified that the coolant was essential pure, as a former plant supervisor, Mr. Fitzpatrick knows otherwise. Please see attached, NEC MR COPPER EXHIBIT A,

BVY 08-013, Letter to USNRC, March 31, 2008,
Subject: Vermont Yankee Nuclear Power Station License No. DPR-28 (Docket No. 50-271) Deviation from BWRVIP-130

> In accordance with BWRVIP-94, Entergy hereby informs the NRC of a specific deviation from the BWRVIP-130 Action Level 1 for total Feedwater System copper. The attachment to this letter provides the history and technical basis for the deviation.

Also attached two Vermont Yankee Power Reports , EXHIBITS B and C, which detail condenser tube leaks AND chlorination of the cooling tower basin, the circulating waters of which flood the condenser. Finally, attached is a Vermont news article, EXHIBIT D, quoting an Entergy spokesman expressing concern about the potential for chloride damage due to leaking condenser tubes.

New England Coalition submits that if a witness cannot be relied to give accurate and truthful information in small matters, the trier of fact should certainly be on notice to give probing skeptical examination on matters that are large. It should be said that the witness at the time of the hearing was no longer employed at Vermont Yankee and hence could plead ignorance of this particular condenser leak event. However, seated in the courtroom were current Entergy Vermont Yankee operations personnel and management. If the record is , for whatever reason, reopened, NEC respectfully requests, in light of the above, that it be moved to a Subpart G proceeding allowing for cross examination and discovery rights for intervenors.

NEC respect fully submits that forgoing evidence and testimony abundantly shows the quality of information flowing to the Board and the quality of examination and post-hearing analysis has not provided a sufficient basis on which to register a sound decision.

MOTION

Wherefore, NEC now respectfully moves the Board to reconsider its Partial Initial decision. Further, NEC respectfully requests that the Board suspend, reverse or modify decision with respect to NEC Contentions 2, 2A and 2B, consider anew the evidence in the light of the discussion in this Motion to Reconsider, and if need be reopen the record to take new evidence.[5] In the alternative, NEC respectfully requests that the Board submit its findings and the evidentiary record to review by a panel of independent, competent, knowledgeable experts in the disciplines required to ascertain within the highest professional standards that the Board's Partial Initial Decision provides adequate assurance of public health and safety.

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[5] [T]he mechanism of post-hearing resolution must not be employed to obviate the basic findings prerequisite to an operating license - **including a reasonable assurance that the facility can be operated without endangering the health and safety of the public.** In short, the 'post-hearing' approach should be employed sparingly and only in clear cases. **In doubtful cases, the matter should be resolved in the adversary framework prior to issuance of license, reopening the record if necessary.** Indian Point, CLI-74-23, 7 AEC at 951-52. ***Emphasis added***

Further, NEC respectfully requests, in consideration of the information herein presented by Dr. Hausler, an amended decision and order requiring the Checworks program at Vermont Yankee be precisely benchmarked through a campaign of detailed measurement, while taking into consideration extended power uprate parameters, of all piping points known to be in FAC or FILC susceptible locations; entering them into entering them into the Vermont Yankee Checworks database and then applying a rigorous regimen to the program maintenance.

for New England Coalition, Inc.

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Entergy

Entergy Nuclear Operations, Inc.
Vermont Yankee
P.O. Box 0250
320 Governor Hunt Road
Vernon, VT 05354
Tel 802 257 7711

March 31, 2008

BVY 08-013

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

References: (a) BWRVIP-94, Revision 1, "BWR Vessel and Internals Project Program Implementation Guide"
(b) BWRVIP-130, "BWR Vessel and Internals Project BWR Water Chemistry Guidelines - 2004 Revision"

Subject: **Vermont Yankee Nuclear Power Station**
License No. DPR-28 (Docket No. 50-271)
Deviation from BWRVIP-130

Dear Sir or Madam:

In accordance with BWRVIP-94, Entergy hereby informs the NRC of a specific deviation from the BWRVIP-130 Action Level 1 for total Feedwater System copper. The attachment to this letter provides the history and technical basis for the deviation.

This notification is for information only and no action on the part of the NRC is requested.

There are no new regulatory commitments being made in this submittal.

If you have any questions concerning this submittal, please contact Mr. David J. Mannai at (802) 451-3304.

Sincerely,

Ted A. Sullivan
Site Vice President
Vermont Yankee Nuclear Power Station

Attachment: Vermont Yankee Nuclear Power Station, Technical Justification for Deviation from BWRVIP-130

cc: (next page)

A001
NRR

cc: Mr. Samuel J. Collins, Region 1 Administrator
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406-1415

Mr. James S. Kim, Project Manager
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

USNRC, BWRVIP Project Manager
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

USNRC Resident Inspector
Vermont Yankee Nuclear Power Station
320 Governor Hunt Road
P.O. Box 157
Vernon, VT 05354

Mr. David O'Brien, Commissioner (w/o attachment)
VT Department of Public Service
112 State Street, Drawer 20
Montpelier, VT 05620-2601

Attachment 1

Vermont Yankee Nuclear Power Station

Technical Justification for Deviation from BWRVIP-130

Vermont Yankee Nuclear Power Station
Technical Justification for Deviation from BWRVIP-130

Introduction

In accordance with BWRVIP-94 "BWRVIP Program Implementation Guide", Revision 1, a Deviation Disposition is required when utility procedures, inspections, methodology, or guidelines are inconsistent with the intent of the supporting BWRVIP guidelines. BWRVIP-94 Appendix A provides guidance on the document structure for a technical justification for a deviation.

BWRVIP Requirement

BWRVIP-130, "BWR Water Chemistry Guidelines", 2004 Revision, Section 6-14 in Table 6-6 identified that the Action Level 1 value for feedwater copper is 0.2 ppb. In Section 6-5 the following statement is made concerning an Action Level 1 condition for feedwater copper: "if not restored within 96 hours, perform a review to assess the impact of long-term system reliability. Identify and evaluate corrective actions. Develop and obtain management approval of a written plan and schedule to implement appropriate corrective actions."

The basis for the BWRVIP documents is that the presence of copper in the reactor coolant "can cause delamination of nodular oxide on zircaloy cladding or deposit in a tenacious crud, potentially leading to cladding damage". Crud Induced Localized Corrosion (CILC) type failures have been associated with elevated levels of copper in BWR feedwater. Plants with copper alloy condensers such as Vermont Yankee (VY) should carefully evaluate fuel concerns and take preventative measures, which should include the following:

- Use only fuel cladding with high resistance to nodular corrosion.
- Consult with the fuel vendor if planning to add zinc in the feedwater.
- Perform an engineering risk assessment including the potential effect on fuel integrity due to redistribution and deposition on the fuel of Fe, Cu, and Zn prior to making significant chemistry changes.

The justification below addresses these recommendations and risk assessment.

VY Deviation

Feedwater copper is not controlled to <0.2 ppb, BWRVIP-130 Action Level 1, under all operating conditions. The Cycle 25 average feedwater copper concentration was 0.47 ppb.

Background

Elevated feedwater copper levels for plants with admiralty brass condensers and filter demineralizers have been a noted industry problem for several years. The admiralty brass condenser tubes contain approximately 78% copper and approximately 20% zinc. Filter demineralizers are approximately 90% efficient for removal of soluble species due to the very short residence time on the thin ion exchange resin layer on the precoat.

The basis for this deviation is as follows:

1. VY cannot meet the 0.2 ppb feedwater copper limit under all operating conditions.

2. There have been no fuel failures at VY within the past 15 years where the root cause was copper when the cycle average copper concentration was ~ 0.5 ppb or greater. Actual fuel failures were attributed to fretting, manufacturing defects, FME, and accelerated corrosion.

3. Currently, all the fuel cladding in VY's core is process 8, which is more resistant to accelerated corrosion.

4. Feedwater copper concentrations > 0.2 ppb do not impact the effectiveness of hydrogen water chemistry in a plant (VY) that has injected noble metals.

5. VY procedures contain Fuel Warranty Limits and has a Continuous Limit of 1.0 ppb for feedwater copper. The Fuel Contract has a continuous limit of 1.0 ppb for feedwater copper.

6. Fuel inspections performed during RFO-26 showed no fuel failures after one year operation under Extended Power Uprate (EPU) conditions and testing zinc addition at the end of the cycle. However, higher than expected "lift off", within the GE experience base, was observed on second cycle fuel. This could mean more tenacious crud. Fuel inspections will be performed again in RFO-27 and the thrice burned fuel will not be put back in the core.

The following table shows cycle averages for total feedwater copper:

Cycle Average Total Feedwater Copper	
Cycle	Total Copper (ppb)
19	0.50
20	0.43
21	0.41
22	0.45
23	0.23
24	0.25
25	0.47

The data shows that VY feedwater total copper on a cycle average basis has been consistently greater than 0.2 ppb (Action Level 1 of BWRVIP-130). The cycle averages for cycles 23 and 24 show an improvement in total copper concentration. This is due to improvements in the resin mix of the condensate demineralizers. The optimized copper removal continued into cycle 25 until March 2006. For the EPU condensate flow was increased by 20% and a bypass line was added to the condensate demineralizers to balance flow at 100% power with a demineralizer out of service. These factors have resulted in an increase in feedwater copper concentration for the rest of cycle 25 and cycle 26. Since EPU, feedwater copper concentration has typically been 0.5-0.6 ppb. Higher copper concentrations are seen during the summer months and periods of operation with four condensate demineralizers and the bypass line.

VY is modifying the condensate demineralizer system to optimize copper removal. The modification of the system involves complete replacement of the internal components of the condensate demineralizer vessels and adding an integrated flow distributor to each vessel. This will present more filter area and resin to condensate flow and maximize copper removal.

The concern with copper is for copper oxide on the fuel surface. This caused the industry CILC fuel failures of the 1980s. Fuel vendors have developed corrosion resistant fuel cladding to prevent CILC failures, such as the P-8 cladding used at VY. The current fuel concern involves copper oxide precipitating onto a tenacious layer of zinc ferrite on the fuel cladding, causing an increased temperature gradient and affecting heat transfer, which could lead to fuel failures. A concentration of 0.2 ppb in the feedwater was chosen as Action Level 1 for BWRVIP-130 because there was copper with large concentrations of iron and zinc in the River Bend fuel failures.

Review of Operating Experience

The River Bend Station (RBS) fuel failure incident of 1999 was thoroughly evaluated and discussed at several EPRI meetings. RBS experienced fuel failures in 7 fuel assemblies that appeared to be related to fuel crud (copper + zinc + iron). Although there was an elevated amount of copper in the fuel crud, the failure mechanism was more a result of heavy deposition of iron oxide-based tenacious crud. Two conductivity excursions resulting from a chemical decontamination of an Residual Hear Removal (RHR) system heat exchanger during the October 1997 refueling outage and the subsequent startup are the suspected causes for a large influx of corrosion products early in the operating cycle. Their feedwater iron levels were around 3.7 ppb. This did not account for all iron deposits on the fuel inside the core and it was not clear where this extra iron came from. At VY, feedwater iron is maintained below the EPRI Guideline value of 5 ppb and is infrequently above 2 ppb. Therefore, the RBS event does not apply to VY because VY has much less iron and zinc, and iron and zinc were the major contributors to RBS's fuel failures. As a result, the type of fuel failures seen at RBS are not expected at VY, even with a feedwater copper concentration >0.2 ppb. Based on a review of the EPRI Guidelines, the RBS incident was used as the basis for the guideline value for feedwater copper being reduced from 0.5 ppb to 0.2 ppb.

The General Electric (GE) BF2/VY Root Cause Investigation Report dated 03/17/2003 did not determine a root cause for the 5 fuel failures identified during Cycle 22. The report indicated that the high levels of copper likely contributed to accelerating the corrosion process along with some unknown initiating event. However, high levels of copper crud did not affect the performance of the 3rd cycle fuel that was discharged during RFO-22 (these assemblies were exposed to less flux). Copper concentrations were very low on the GE BF2 failed fuel. Fuel examinations at VY indicated relatively high copper deposits on Cycle 19, 20 and 21 fuel. The 5 fuel rod failures were from the same tubing lot that failed in VY reload number 20. The data indicate that other reloads residing in the core are not exhibiting the accelerated corrosion. It was noted in later fuel inspections that Reload 22 had significant accelerated corrosion and probably could not be used for another cycle. The root cause evaluation did not provide any recommendations for copper control. Following the VY fuel failures, the Reactor Engineering department contacted Aquarius Services Corporation and requested an evaluation of the data associated with the fuel failures. This included GE evaluations and material, two cycles of Chemistry Data and plant operating history. Fuel manufacturing data was also reviewed. Some conclusions and notes from the report are as follows:

- Nodular corrosion should not occur on an in-process heat treated cladding. Of the two causes, corrosion by high copper chemistry water is unlikely, since GE work in the past showed that this does not occur either in or outside the reactor. High copper chemistry with noble metals might induce nodular corrosion by the change in redox conditions at the cladding surface. The previously proposed poor in-process heat treating control could be a second cause.

- The continued evaluation of the fuel examination tapes confirm previous conclusions that there is a correlation between the level of corrosion observed, some of the cladding lot numbers and some of the local peaking factor histories of the rods.

- The author concurs with GNF's conclusion that three cladding lot numbers behaved poorly.

- A cursory comparison of fuel rod local peaking factor histories of rods from the same cladding lot indicates a reasonable correlation of power with corrosion control.

- Based on GE information, there does not appear to be a correlation between copper content and liftoff measurements, and there does not appear to be a correlation between linear power generation and liftoff either. This indicates a lack of correlation between copper content and corrosion.

- The maximum concentration of copper at a discreet axial location was 1885 $\mu g/cm^2$ that occurred at the 31" elevation of Rod D* Bundle YJF493. This fuel rod was without a fuel defect.

Duration of Technical Justification

This deviation will remain in effect until such time that the admiralty brass condenser is replaced with one that does not contain copper alloys. This is currently scheduled for 2011. The current revision of the Deviation Disposition will be reviewed after the condensate demineralizer modifications are completed. This is currently scheduled for May 2009.

OPERATING DATA REPORT

DOCKET:	271	PREPARER NAME:	Greg Wallin
UNIT_NME:	Vermont Yankee Unit 1	PREPARER TELEPHONE:	802-451-3309
RPT_PERIOD:	200804		

1. Design Electrical Rating: 617
2. Maximum Dependable Capacity (MWe-Net) 605

	This Month	Yr-to-Date	Cumulative
3. Number of Hours the Reactor was Critical	720.00	2,903.00	267,533.42
4. Number of Hours Generator On-line	720.00	2,903.00	263,722.42
5. Reserve Shutdown Hours	0.00	0.00	0.00
6. Net Electrical energy Generated (MWHrs)	421,705.00	1,781,169.00	130,182,432.00

UNIT SHUTDOWNS

No.	Date	Type F: Forced S: Scheduled	Duration (Hours)	Reason 1	Method of Shutting Down 2	Cause - Corrective Action Comments

SUMMARY:	Date	Activity	Losses in MWe hours	Type of Loss (S) or (F)
	4/01-4/04/08	Power reduction for a condenser tube leak	27255.0	F
	4/04/08	Power reduction for a rod pattern exchange	31.0	S
	4/05/08	Power recuction for a rod pattern exchange	73.0	S
	4/13/08	Recirc gate adjustment for trash rack backwash	18.0	S
	4/22/08	Recirc gate adjustment for trash rack backwash	2.0	S
	Total Losses for the month were:		124.0	S
			27255.0	F
			27379.0	

OPERATING DATA REPORT

DOCKET:	271	PREPARER NAME:	Greg Wallin
UNIT_NME:	Vermont Yankee Unit 1	PREPARER TELEPHONE:	802-451-3309
RPT_PERIOD:	200806		

1. Design Electrical Rating: 617
2. Maximum Dependable Capacity (MWe-Net) 605

	This Month	Yr-to-Date	Cumulative
3. *Number of Hours the Reactor was Critical*	720.00	4,367.00	268,997.42
4. Number of Hours Generator On-line	720.00	4,367.00	265,186.42
5. Reserve Shutdown Hours	0.00	0.00	0.00
6. Net Electrical energy Generated (MWHrs)	426,842.00	2,662,892.00	131,064,155.00

UNIT SHUTDOWNS

No.	Date	Type F: Forced S: Scheduled	Duration (Hours)	Reason 1	Method of Shutting Down 2	Cause - Corrective Action Comments

SUMMARY: Date	Activity	Losses in MW hours	Type of Losses (S) or (F)
06/07/08	Power reduction to maintain condenser backpressure <5 inches due to closed cycle operations for chlorination	270.0 MWe	S
06/11/08	Power reduction for a rod pattern exchange, turbine quarterly and stop valve testing, plus MSIV testing	5485.0 MWe	S
06/12-06/30/08	Power production losses due to condenser cleanliness	2135.0 MWe	S
06/13/08	Power reduction for a rod pattern adjustment	80.0 MWe	S
06/14/08	Power reduction for a rod pattern adjustment	306.0 MWe	S
06/19/08	Power reduction for chlorination	174.0 MWe	S
06/22/08	Power reduction for chlorination	105.0 MWe	S
06/25/08	A CWBP valve binding. Power reduction to maintain Cond B/P <5 inches	777.0 MWe	F
06/27/08	Power reduction for a rod pattern adjustment	17.0 MWe	S
06/29/08	Power reduction for chlorination	124.0 MWe	S
Total Losses for the month were:		8696.0 MWe	S
		777.0 MWe	F
		9473.0 MWe	

Yankee set to return to full power

April 4, 2008

By Susan Smallheer Herald Staff

BRATTLEBORO — Entergy Nuclear has given up trying to find the leak in its condenser at the Vermont Yankee nuclear plant, and is returning the reactor to full power.

Entergy said Thursday that it would closely monitor the leak or leaks in the condenser, and it said it would hold open the option of reducing power once again to try and find the leak. The plant reduced power Monday night because of the problem.

Robert Williams, spokesman for Entergy Nuclear, said that in any case, the leak will be fixed next fall, during the plant's next regularly scheduled refueling and maintenance outage.

"We are returning to full power. We were not able to pinpoint the location of the leak," he said, noting the leak was "very small," since it was only leaking a quarter a minute, or 16 gallons an hour, of Connecticut River water into the reactor's coolant system.

He said the leak was small when compared to the total amount of water circulating in the condenser, 360,000 gallons per minute.

The condenser is not directly related to the nuclear side of the power plant, but it is important because it cools the water that cools the reactor.

Despite the leak, the condenser is designed so that the radioactive water in the reactor will not leak out into the Connecticut River water.

"We're going to continue to monitor it closely," he said. "We may try again with other methods, but it's prudent to come back to a steady state."

According to the Nuclear Regulatory Commission Web site, the plant was at 41 percent power on Wednesday. Williams would not say what the power level was Thursday afternoon.

Williams said the leak was traced to one of the quarter sections of the condenser, called "water boxes," comprised of 5,500 tubes. There are 22,000 tubes in all in the condenser, which acts much like a car radiator.

He said not all of the 5,500 tubes were checked in recent days for the leaks.

"We didn't check all of them. Some are inaccessible. It may be that the leak sealed itself because the temperature change," he said.

According to Neil Sheehan, spokesman for the regional NRC headquarters, Vermont Yankee has had condenser leaks before, the last being five years ago.

"They decided there was a point of diminishing returns," Sheehan said. "There are thousands of tubes, and it really is hunt and peck."

He said the leak, which amounts to 16 gallons an hour, "doesn't challenge the plant's water chemistry."

He said the biggest problem is chlorides that exist in the Connecticut River would interact with the nuclear fuel, but he said the chloride level is in the 2 to 3 parts per billion state, while if it gets in the parts per million, Entergy Nuclear staff will have to act.

Sheehan said that while condenser leaks are not unheard of in the nuclear industry, there were no other plants in this northeast region with leaks in its condenser besides Vermont Yankee.

Contact Susan Smallheer at susan.smallheer@rutlandherald.com.

 RUTLAND HERALD

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Search Results

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Returned: 1 displays of 1 matches

Yankee set to return to full power

Author: SUSAN SMALLHEER Herald Staff
Date: April 4, 2008
Publication: Rutland Herald (VT)

BRATTLEBORO - Entergy Nuclear has given up trying to find the leak in its condenser at the Vermont Yankee nuclear plant, and is returning the reactor to full power. Entergy said Thursday that it would closely monitor the leak or leaks in the condenser, and it said it would hold open the option of reducing power once again to try and find the leak. The plant reduced power Monday night because of the problem.

Robert Williams, spokesman for Entergy Nuclear, said that in any case, the leak will...
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EXHIBIT
JH HR 1
ASLB-06-849-03 LR

UNITED STATES
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the matter of ENTERGY NUCLEAR VERMONT YANKEE, LLC and ENTERY NUCLEAR OPERATIONS, INC. Vermont Yankee Nuclear Power Station License Renewal Application	Docket No. 50-271-LR ASLB No.06-849-03-LR

DECLARATION OF DR. JORAM HOPENFELD
IN SUPPORT OF
NEW ENGLAND COALITION'S MOTION FOR RECONSIDERATION

1. My name is Dr. Joram Hopenfeld. New England Coalition, Inc. ("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 2 1, 2012.

2. I am a mechanical engineer and I hold a doctorate in mechanical engineering. My curriculum vitae was attached to my first declaration in support of NEC's Petition to Intervene, filed May 26, 2006.

3. I submit the following comments in support of New England Coalition's Motion For Reconsideration.

4. With one exception, the Atomic Safety and Licensing Board Panel ("ASLBP, "Board" or "Panel") decided to dismiss wholesale the technical issues that were raised by the New England Coalition during lengthy proceedings regarding Entergy's application for a twenty-year life extension of the VY nuclear power plant.

5. In rejecting the NEC's issues the Board ignored or dismissed, by and large, the technical data that was presented by NEC without the Board providing a valid technical rationale for doing so.

6. Conversely, the Board cited and relied on Entergy's statements which were not supported by data. For example, I have provided data showing that it takes 25-40 diameters for turbulent flow to become fully developed when entering a pipe. Without providing a supporting technical argument or data the ASLB accepted Entergy's unsupported position that four diameters is sufficient to attain a fully developed flow.(PID 46-47, Rebttal Decl,,Post Tr.779 at 13, Tr. 1124-1126 (Hopenfeld)

> Dr. Hopenfeld's concern that it was inappropriate to assume that the flow at the feedwater nozzles is fully developed has not been substantiated and instead has been fairly rebutted by the evidence presented by Mr. Stevens and Mr. Fitzpatrick. Nor is there fair indication that Dr. Hopenfeld's other concerns are warranted { ii Findings, PID 49)

The Board's position disregards well-established hydraulic principles. (Tr.1125-1128) (PID 123-124}

7. The ASLBP allowed very little time to NEC's witnesses to speak or explain in comparison to the time allowed to Entergy and NRC Staff witnesses at the evidentiary hearings.

8. Ironically, one explanation provided by the ASLB for favoring Entergy's testimony was that the NEC witness did not provide sufficient information to the Board; while the Board chose to ignore information that I did provide.

In the PID, the Board points to reliance on a Table in a standard text pointedly describing it as an "excerpt from a textbook" as if that settled the weight the data should be accorded.

> Dr. Hopenfeld testified that it is unlikely that the flow in the VYNPS feedwater nozzle is fully developed because the upstream pipe has a straight section only 48 inches in length and a diameter of 9.7 inches, and this, according to an excerpt from a textbook,[69] Dr. Hopenfeld says, is not sufficient for fully developed flow. Id.; Hopenfeld Rebuttal Decl. Post Tr. 779, at 13; Tr. at1120-21 (Hopenfeld).
>
> [69] NEC Exh. NEC-JH_29, E.R.G. ECKHERT, HEAT AND MASS TRANSFER at 212, Fig. 8-9 (2d Ed. 1959).
>
> PID, 46

9. The ASLB questioned witnesses for Entergy, the NRC Staff, and NEC in a panel format; seating the witnesses in the jury box together. However, the conversation was largely limited to the ASLB and Entergy and NRC Staff witnesses.

10. Entergy and NRC Staff witnesses were permitted to interject comment, speaking out and offering "clarification" in the form of lengthy testimony on technical issues under discussion. (E.g., Tr. 986, 990, 1025, 1025, 1049, 1058, 1065, 1088, 1107 – Testimony of Stevens, Fitzpatrick, and Fair)

NEC witnesses were not often permitted to offer countervailing views. In one instance when a critical technical issue was under discussion, rather than to interject I raised my hand just to shoulder height. I had observed Entergy's witness raising his hand from time to time and getting a positive response from the Board. Therefore, I thought it was a polite and acceptable signal that I would like to offer testimony on the subject which might "clarify" at least one point in NEC's view. Judge Richard E.Wardwell sharply told me, "From now on if I don't have a question for you, I'd like for you not to raise your hand. We're not in school here. Okay?" Tr. 1636-1638

> DR. HOROWITZ: A little more complicated, but essentially.
>
> JUDGE WARDWELL: I've got to call you one more time. Dr. Hopenfeld. From now on, if I don't
>
> 1637
>
> have a question for you, I'd like for you not to raise your hand. We're not in school here. Okay? I don't have a question, but go ahead.
>
> DR. HOPENFELD: Well, I just wanted to make a comment on the line -- I wasn't telling you anything different. Dr. Hausler has information, or would like to comment about the completeness of my -
>
> JUDGE WARDWELL: And I understand that, and if I have a question for Dr. Hausler, I will ask it. The reason I say that, Dr. Horowitz, is because - - I mean Dr. Hopenfeld
>
> DR. HOPENFELD: I understand.
>
> JUDGE-WARDWELL: -- is that we have the pre-filed testimony. Some of the testimony is clearer to understand than others. And it's not to say that the amount of questioning is any relationship to the weight of the testimony. It's all weighted equally, and then evaluated in regards to its credibility. But it may be just that his testimony is clearer, so I personally don't have questions.
>
> DR. HOPENFELD: I apologize.
>
> JUDGE WARDWELL: Well, I'll get it to when I come down. I have a list of questions in regards to velocity. Rather than trying to find it, I'd rather go through mine in the order of things. It will take
>
> 1638
>
> more time for me to find it than when we eventually get to it.

He did not inquire further as to NEC's evaluation of the issue.

12. When, as NEC's witness, I was trying during the oral hearings to explain and provide technical background to the Board, I was continuously interrupted; often not being permitted to finish a simple sentence. I was certainly not prepared for the issues to be treated in this manner.

Open skepticism, conflicting expressions of opinion and rational data-reliant argument are nutritional necessities for solid science. I had presumed that fact finding in an evidentiary hearing would pursue a similar course and one that was even-handed. That was not to be and I believe that as a result the Panel left the hearing with less than a whole picture of the issues.

13. The ASLB Panel was chaired by a legal expert, who shared the duties of inquiry with two members having a scientific and/or technical background. However, the issues in this case involve very specific and not broadly understood materials, mechanics, energy, and plant operations phenomena beyond the depth of most generalists. It appears now that the ASLB Panel in this case lacked the on board expertise to competently weigh conflicting testimony on all of the topics presented.

The transcript of the evidentiary hearings that were held in Vermont this past July and the Board decision, which relies heavily on the testimony presented in those hearings, clearly demonstrate that the Board lacks a fundamental understanding of the principles of safety risk assessment, material fatigue, material corrosion and nuclear plant instrumentation

The ASLB's lack of rudimentary knowledge of these subjects is illustrated by several examples.

A. Cumulative Usage Factor, CUF. The Board concluded that my "CUFen recalculations are unsound" because , the Board explains, '…the recalculations predict that the regulatory requirement would have been exceeded within 4.63 years after VYNPS commenced operations, and it is obvious that this did not occur. Tr. At 1129-30" (Initial Decision, pages 56 and 57) Actually, a reading of the transcript finds that the discussion the Board refers to spans pages 1128 -1136. In the finding the Board says that it obvious that the regulatory requirement was not exceeded. But in the transcript it is clear that Dr. Reed is of the opinion that a calculated CUF that is larger than unity is an indication that the component must fail and what is obvious to him is that there have been no failures.

> JUDGE REED: Given that the plant has not failed, that none of these nozzles has failed, how can you justify proposing that the CUFen numbers could possibly be as large as what you propose?
>
> DR. HOPENFELD: How can I justify? All this says, all these numbers say, and I think that's what the ASME code, to the best of my understanding, and what the guidance are, to say if you have -- and I believe that Mr. Stevens talked about that too -- it doesn't mean everything falls apart once that number is about one. All it says, when you reach about one you have got to do something. I cannot buy your supposition --
>
> JUDGE REED: Even if I accept your point, that it doesn't fall apart, just major cracking occurs, we have not seen major cracking in any of these components in 30-something years of operation. And yet your CUFens predict that they fail in periods of time that would be substantially shorter than that. Hence I have to infer that your calculations are extremely excessively conservative.
>
> Tr.1130-31

Based on the above apparent misunderstanding of how the CUF is determined and what exceeding unity means, the Board concluded on page 56 of the PID, that:

> "As was elicited in testimony during the hearing, Dr.Hopenfeld's recalculations predict that the regulatory requirement (i.e., unity) would have been exceeded within 4.63 years after the VYNPS commenced operations, and it is obvious to the Board that this did not occur. Tr. at 1129-30. "

In my opinion the Board erred in deciding that my calculations lead to the conclusion that the regulatory requirements would have been exceeded within 4.63 years of the time in which VY commenced operations. No such number as 4.63 was elicited from me or any other expert witness. The numbers that I provided do not lend themselves to the Board's conclusion.

The ASME code requires that calculated CUFs not exceed one. The two sets of CUF numbers I have provided were obtained by two different methods indicating only that the CUF would exceed some time during the 20 year life extension. In one method I have used the exact same CUF values provided by Entergy and corrected them only by using more appropriate oxygen concentrations.

Dr. Reed erroneously concluded that the NEC calculations are not valid because none of the NEC components with CUF larger than unity has failed. In fact the CUF is a convenient design criterion that incorporates safety factors because it is based on a formation of small crack (known as an engineering crack) and it is not based on running tests to failure. In fact, the license has reported the presence of such cracks in the lining of the nozzles in question. Both I and Mr. Stevens are in agreement that the fact that the calculated CUF exceed one does not mean that the components would fail. It only means that that it could potentially fail, Dr. Reed apparently does not agree with this concept. The Board chose to ignore experts on both sides to come to its finding. From Dr. Reed's assertions regarding CUF as an indicator of failure, Dr. Ward's off-point questions during the hearings, and the findings in the Board decision it is apparent that the Board misunderstood what the CUF means and how it was defined.

I have explained to the Board that I do not know how to relate my CUF predictions (Pages 1128-1129) to the 4.63 years calculated by the Board. Since the ASLB dismissed my calculations on that basis it is necessary to establish the technical validity of how relating my results to the 4.63 years calculated by the Board. The Board should be required to provide a technical explanation how the above decision was reached given the definition of the CUF, the experimental data used to determine the CUF and how one measures whether the regulatory requirements have been exceeded. (The Board did not specify what those requirements were, yet at the hearing Dr. Reed was referring to component failures).

In summary, the Board replaced my expert opinion on this subject, and that of the licensee's witness, Mr. Stevens, with their own assumptions, calculations, and conclusions which I believe are grossly in error reflecting almost a complete lack of understanding of fatigue technology.

It would greatly improve public confidence in NRC if the Board would provide a detailed technical rationale for substituting the own view of technical issues for that of technical experts, and to avoid couching their findings in technical vagaries.

C. CUFen and Metal Fatigue – The fatigue analysis of nuclear power plant components is discussed in **Comments on Proposed NRC Generic Communication Regulatory Issue Summary (RIS) 2008-XX "Fatigue analysis of Nuclear Power Plant Components" May 1, 2008** , (attached as NEC Exhibit , Motion for Reconsideration JH-2).[1] The document is authored by Structural Integrity Associates, Inc. and incorporates, with their own comments, the comments of four nuclear utilities. Comment 3 states their understanding of the relative importance of one aspect which is associated with the application of Green's Function, i.e. the use of one stress component vs. the use of six stress components, in the context of ASME requirements:

[1] This document was provided by Entergy in Disclosures on July 1, 2008; too late for inclusion in NEC's Prefiled Direct Testimony

ASME Code, Subsection NB, Subarticle NB-3200 methodology is not prescriptive. As a result all analyses performed using this methodology rely on the judgment of the analyst, including judgment on items such stress components, transient definitions, heat transfer coefficients, material properties, and other input parameters to ensure that the analysis results are appropriate and bounding for the intended application. In fact, the confirmatory analysis performed for one boiling water reactor feedwater nozzle referenced in the RIS uses many of the same judgments – judgments that have routinely been applied in CLB analyses for Class1 components throughout the industry.

Given the lack of specific requirements related to environmental fatigue assessment, any methodology may be nonconservative if not correctly applied. Why is the single-stress analysis method singled out in the RIS. Has NRC reviewed all approaches used to assess environmental effects and determined that all other methods are always conservative?

I share Structural Integrity Associates, Inc.'s concern: why did the NRC ignore the uncertainties in the heat transfer, material and other inputs and focused attention only on the uncertainties in the stress analysis? Based on my observations during the two years prior to, and during the hearing the answer became apparent to me: the NRC reviewers were experts in stress analysis but did not have the required knowledge in other but equally important areas. Unlike NEC, neither the NRC Staff nor the Licensee presented any witnesses with credentials or experience qualifying them to give expert testimony on electrochemistry, mass transfer, heat transfer, and hydraulics. The NRC did not present experts in these areas for the ACRS review of the Vermont Yankee License Renewal Safety Evaluation Report, as well.

From the testimonies provided at the hearing it is clear that the uncertainty in using one stress component vs. six component pales in importance in comparison to the uncertainties in heat transfer and oxygen inputs. In my opinion the uncertainty in oxygen input is at least 10 times more important than the number of components used in the Green's function as opposed to the number of component used in the conventional ASME analysis.

The Board chose, following on NRC Staff's example to focus only on one aspect which is associated with the application of Green's Function, i.e. the use of one stress component vs. the use of six stress components. Even though NEC provided data on heat transfer and oxygen which contradicted Entergy's and NRC mere unsupported statements the Board chose to believe the NRC and Entergy as discussed below.

It appears that the ASLB copied and reinforced NRC's request for essentially disallowing the use of the Green's function without weighing the uncertainties of other parameters. Since the issue whether Green's function should or should not be used was hardly debated at the hearing the Board has done disservice to both the Industry for imposing perhaps unnecessary financial burden and to the public for completely ignoring the essential and real technical issues associated with metal fatigue.

C. Heat Transfer. Entergy assumed that the heat transfer input to certain nozzles is uniform because the flow at the entrance to the nozzle is fully developed. This is a major non conservative oversimplification of the problem

It is a fundamental engineering fact, known for at least for 100 years, that it takes about 25-40 diameters for the flow at the entrance to a pipe to become fully developed. NEC provided data showing how the heat transfer would vary in the feedwater nozzle. Entergy stated that it takes only 4 diameters to establish a fully developed flow without any supporting data. Yet, the ASLB accepted Entergy's position on the basis that their explanation, without providing any supporting data, was more "credible' than NEC's presentation. The Board did not provide a technical rational to support their decision.

D. Oxygen Effect The equations for calculating fatigue factors were formulated in a laboratory under conditions where the parameters effecting fatigue were known. In the reactor environment during transients these conditions, oxygen levels for example, are not known. This fact is commonly accepted by those researches that developed the fatigue equations. Only by using available data and known laws of physics one can asses the effects of these parameters. Entergy ignored the specifications provided by the developer of the fatigue equations, and consequently calculated low CUF values.

The ASLB, dismissing all experimental data and the fact that oxygen solubility increases as the temperature decreases, agreed with Entergy, claiming without any proof that the oxygen during the transients is known at the surface of a given components. The ASLB apparently mistakenly believed, (PID page 37) that the fact that VT performed daily measurements of oxygen for 13 years during steady state operations represents oxygen levels at the components surface during the transients where the temperature varies.

Even though the specifications for calculating the FEN requires that the oxygen be used at 0.4PPM the Board accepted Mr. Fairs testimony (Page 37) that this was meant to be a default value. While this is a crucially important point, the Board did not question Mr. Fair as to where it is specified as a default value in the relevant NUREGS 6909 and 6587 reports or from where Mr. Fair obtained his information. Nor did the Board inquire how one decides when default values should or should not be used. Nor did the Board ask Mr. Fair to relate the steady oxygen measurements to the values that exceed them during the transients. The fact that the recommendation to use 0.4 PPM is couched in permissive language (can) rather than prescriptive language (must) is not cause to dismiss. Since the guidance is not regulation, but rather a guide toward meeting regulation a more proper interpretation of "can" is that if industry uses 0.04 PPM, they need provide reviewers with no further justification than to invoke the guidance. If they pick some other concentration, then they must show analysis to support that choice will not result in a non-conservative outcome. The Board overrode the written prescription of NUREGS 6909 and 6587 on how to calculate oxygen by a mere unsupported verbal statement of an NRC witness, Mr. Fair.

The Board conclusion is fundamentally incorrect when it finds that

> **"Entergy used actual DO data and otherwise demonstrated that its approach to this phenomenon is sound.** "(PID 39)

As already stated above and in my testimony, plants do not perform actual DO measurements during the transients at the location of the component in question. There is no practical method of performing such measurements.

E. CDF NEC claimed that the safety consequences of pipe failure from corrosion or formation of loose parts plant from the dryer must be studied in terms of Core Damage Frequency, CDF. Even though the concept of CDF is commonly used in all NRC safety studies and Commission papers, the ASLB members displayed no apparent familiarity with this concept. (Tr. 1613-1621). This point is important because Entergy's witness repeatedly referred to how safe certain components are without quantifying their statements. When the decision makers are not aware that safety can be to some degree described in terms of the CDF it raises into question the technical quality of the entire hearing process.

F. Effect of Velocity on Corrosion. NEC provided experimentally derived data showing the sensitivity of flow accelerated corrosion (FAC) to velocity. The ASLB dismissed the data without providing rationale for doing so. By stating (Initial Decision, Page 146) that Bench marking is not required the Board accepted Entergy's witness testimony that FAC is **not** very sensitive to flow velocity. The Board did not discuss why the data that was provided by NEC, showing a marked effect of velocity on FAC of steel in water was rejected. Nor did the Board discuss why, instead they relied on data of corrosion of copper in an acid which was cited by Entergy's witness. Apparently the ASLB has the mistaken impression that local velocities in a plant are known parameters.

G. In General - Were the assumptions of Entergy and the ALSB resulting Findings of Fact to be reviewed by a competent technical panel, it is in my profession opinion that they would not survive, without censure, a first reading.

It is my opinion that if the ASLBP, and the NRC want to retain credibility in the field of nuclear review, the ASLBP should now re-open the record; contract independent expert consultants and with their assistance review the submissions and testimony on the record for completeness, credibility, and veracity within the acceptance criteria of the various relevant scientific and technical disciplines.

Advice

It is my advice that, in order to recover scientific and technical integrity in these proceedings, NEC should, at the least, request that the ASLB or the Commission appoint an independent panel of competent technical experts to review the Board's technical rationale for rejecting the NEC contentions.

I, Dr. Joram Hopenfeld, declare under penalty of perjury that the foregoing

DECLARATION OF DR. JORAM HOPENFELD IN SUPPORT OF
NEW ENGLAND COALITION'S MOTION FOR RECONSIDERATION

is true and correct.

Signed in the original, Joram Hopenfeld

Dr Joram Hopenfeld

Executed this day of , 2008 at Rockville, Maryland.

I, Dr. Joram Hopenfeld, declare under penalty of perjury that the foregoing

DECLARATION OF DR. JORAM HOPENFELD IN SUPPORT OF
NEW ENGLAND COALITION'S MOTION FOR RECONSIDERATION

is true and correct.

Joram Hopenfeld

Dr, Joram Hopenfeld

Executed this fifteenth day of December, 2008 at Rockville, Maryland.

Structural Integrity Associates, Inc.

EXHIBIT
JH HR 2
ASLB 06-849-03 LR
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June 16, 2008
GLS-08-013

Chief, Rulemaking, Directives and Editing Branch
Division of Administrative Services
Office of Administration
U.S. Nuclear Regulatory Commission
Mail Stop T6-D59,
Washington, DC 20555–0001

Subject: Comments on Proposed Generic Communication, "Fatigue Analysis of Nuclear Power Plant Components"

Reference: U.S. Federal Register, Vol. 73, No. 85, Thursday, May 1, 2008, Notices, p. 24094.

To Whom It May Concern:

Attached, please find comments on the subject Proposed Generic Communication. These comments reflect compiled input from Structural Integrity Associates, Inc. and four U.S. nuclear utilities.

If you have any questions on the enclosed comments, please do not hesitate to contact me.

Very truly yours,

Gary

Gary L. Stevens, P. E.
Senior Associate

cc (via e-mail): J. Fair (NRC)
K. Chang (NRC)
M. Case (NRC)

Comments on Proposed NRC Generic Communication Regulatory Issue Summary (RIS) 2008-XX "Fatigue Analysis of Nuclear Power Plant Components," May 1, 2008[1]

Each comment includes a quotation from the proposed RIS text being addressed by the comment. The quoted text is indented and italicized to separately identify it from the comment.

Comment 1:

> ***INTENT***
>
> *The U.S. Nuclear Regulatory Commission (NRC) is issuing this regulatory issue summary (RIS) to inform licensees of an analysis methodology used to demonstrate compliance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) fatigue acceptance criteria that could be nonconservative if not correctly applied.*

The Intent section of the RIS indicates that nonconservative results could be obtained if the methodology is not correctly applied. However, the final results of the example boiling-water reactor feedwater nozzle confirmatory analysis cited in the RIS do not support this statement. For the sample boiling-water reactor plant citied in the RIS, the cumulative usage factor (CUF), including environmental effects, at the feedwater nozzle corner was calculated to be 0.63 in the original (refined) analysis. This value is conservative compared to the CUF value (including environmental effects) of 0.35 calculated at the feedwater nozzle corner in the follow-on confirmatory analysis. Whereas the CUF value, prior to adjustment for environmental effects, was higher for the confirmatory analysis than for the refined analysis, the higher value of CUF in the confirmatory analysis was the result of the different implicit conservatisms present in each analysis. When these conservatisms are all collectively considered, the refined analysis methodology is observed to be conservative, as demonstrated by the final CUF results. Similar reductions in CUF (including environmental effects) were also reported for a second boiling-water reactor confirmatory analysis reported since the publication of the draft RIS.

Please clarify the intent of the RIS.

---

1. U.S. Federal Register, Vol. 73, No. 85, Thursday, May 1, 2008, Notices, p. 24094.

Comments on Proposed NRC Generic Communication Regulatory Issue Summary (RIS) 2008-XX "Fatigue Analysis of Nuclear Power Plant Components," May 1, 2008[1]

Comment 2:

INTENT

The U.S. Nuclear Regulatory Commission (NRC) is issuing this regulatory issue summary (RIS) to inform licensees of an analysis methodology used to demonstrate compliance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) fatigue acceptance criteria that could be nonconservative if not correctly applied.

BACKGROUND INFORMATION

Title 10 of the Code of Federal Regulations (10 CFR) Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," requires that applicants for license renewal perform an evaluation of time-limited aging analyses relevant to structures, systems, and components within the scope of license renewal. The fatigue analysis of the reactor coolant pressure boundary components is an issue that involves time-limited assumptions. In addition, the staff has provided guidance in NUREG-1800, Rev. 1, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," issued September 2005. NUREG-1800, Rev. 1, specifies that the effects of the reactor water environment on fatigue life be evaluated for a sample of components to provide assurance that cracking because of fatigue will not occur during the period of extended operation. Since the reactor water environment has a significant impact on the fatigue life of components, many license renewal applicants have performed supplemental detailed analyses to demonstrate acceptable fatigue life for these components.

To our knowledge, the ASME Code fatigue analysis methodology never has been explicitly required for environmental fatigue calculations. The NRC has not defined the specifics of the underlying fatigue analysis requirements to address environmental fatigue effects for license renewal. As a result, there are no clear rules for performing such fatigue evaluation, beyond the environmental fatigue (F_{en}) methodology referenced in the GALL Report (NUREG-1801, Revision 1) and specified in associated documents NUREG/CR-6583 and NUREG/CR-5704. Since the evaluation of environmental effects is not associated with the current licensing basis (CLB), but rather for license renewal purposes, it seems that any approach that can be defended technically as conservative with respect to fatigue can be used to establish a fatigue usage factor upon which to apply environmental factors. For example, the use of strain rates for CLB transients may not be bounding for use in an environmental fatigue assessment, since F_{en} values are increased for lower strain rates that are typical of actual plant operation. An additional example is those plants that have a piping design basis of ANSI B31.1 where no explicit fatigue evaluation exists. In these cases, most plants choose to perform fatigue calculations using ASME Code Section III methodology to provide a fatigue basis to evaluate the effects of environmental fatigue, but there does not seem to be any requirement that the ASME Code methodology be used in these circumstances. Is it the intent of the RIS to establish the ASME Code fatigue analysis methodology as the only NRC-approved method for environmental fatigue evaluations?

Comments on
Proposed NRC Generic Communication
Regulatory Issue Summary (RIS) 2008-XX
"Fatigue Analysis of Nuclear Power Plant Components," May 1, 2008[1]

Comment 3:

> *The detailed stress analysis requires consideration of six stress components, as discussed in ASME Code, Section III, Subsection NB, Subarticle NB-3200. Simplification of the analysis to consider only one value of the stress may provide acceptable results for some applications; however, it also requires a great deal of judgment by the analyst to ensure that the simplification still provides a conservative result.*

ASME Code, Subsection NB, Subarticle NB-3200 methodology is not prescriptive. As a result, all analyses performed using this methodology rely on the judgment of the analyst, including judgment on items such as stress components, transient definitions, heat transfer coefficients, material properties, and other input parameters to ensure that the analysis results are appropriate and bounding for the intended application. In fact, the confirmatory analysis performed for the one boiling-water reactor feedwater nozzle component referenced in the RIS uses many of the same judgments – judgments that have routinely been applied in CLB analyses for Class 1 components throughout the industry.

Given the lack of specific requirements related to environmental fatigue assessment, any methodology may be nonconservative if not correctly applied. Why is the single-stress analysis method singled out in the RIS? Has the NRC reviewed all approaches used to assess environmental effects and determined that all other methods are always conservative?

Comment 4:

The detailed stress analysis requires consideration of six stress components, as discussed in ASME Code, Section III, Subsection NB, Subarticle NB-3200. Simplification of the analysis to consider only one value of the stress may provide acceptable results for some applications; however, it also requires a great deal of judgment by the analyst to ensure that the simplification still provides a conservative result.

The staff has requested that recent license renewal applicants that have used this simplified Green's function methodology perform confirmatory analyses to demonstrate that the simplified Green's function analyses provide acceptable results. The confirmatory analyses retain all six stress components. To date, the confirmatory analysis of one component, a boiling-water reactor feedwater nozzle, indicated that the simplified input for the Green's function did not produce conservative results in the nozzle bore area when compared to the detailed analysis. However, the confirmatory analysis still demonstrated that the nozzle had acceptable fatigue usage.

Whereas the ASME Code methodology is intended to use six stress components in fatigue evaluation, allowance is made to simplify the analysis when the situation warrants. Specifically, ASME Code, Paragraph NB-3215(d) states:

> "In many pressure component calculations, the t, l, and r directions may be so chosen that the shear stress components are zero and σ_1, σ_2, and σ_3 are identical to σ_t, σ_l, and σ_r."

The above is true for cylindrical component geometries such as those prevalent throughout the nuclear industry (e.g., reactor vessels and piping). In fact, CLB fatigue analyses have traditionally used only component (σ_x, σ_y, σ_z or σ_t, σ_l, σ_r) stresses. This practice assumes shear stresses are negligibly small such that the component stresses essentially equal the principal stresses, and simplifies the evaluation by negating the need to solve a cubic equation to resolve a six-component stress tensor into three principal stresses. This simplified approach has been widely adopted over many years of industry use for a variety of component analyses, including nozzle corner locations. In fact, responses to additional information (RAIs) associated with the one boiling-water reactor feedwater nozzle confirmatory analysis cited in the RIS demonstrated that shear stresses were negligible, and Advisory Committee on Reactor Safeguards (ACRS) testimony earlier this year indicated that the nonconservatism in those results was the result of "twenty differences... of conservatisms" and approximations between the refined and confirmatory analyses.

In view of all of the foregoing discussion, it is unclear why the RIS requires the use of all six stress components, why it is acceptable for CLB analyses to not do so and why the RIS is limited to those select few environmental fatigue evaluations that have used a simplified Green's Function methodology associated with license renewal. Please clarify.

Comment 5:

> *The staff identified a concern regarding the methodology used by some license renewal applicants to demonstrate the ability of nuclear power plant components to withstand the cyclic loads associated with plant transient operations for the period of extended operation. This particular analysis methodology involves the use of the Green's function to calculate the fatigue usage during plant transient operations such as startups and shutdowns.*
>
> *The Green's function approach involves performing a detailed stress analysis of a component to calculate its response to a step change in temperature. This detailed analysis is used to establish an influence function, which is subsequently used to calculate the stresses caused by the actual plant temperature transients. This methodology has been used to perform fatigue calculations and as input for on-line fatigue monitoring programs. The Green's function methodology is not in question. The concern involves a simplified input for applying the Green's function in which only one value of stress is used for the evaluation of the actual plant transients.*

The RIS is misleading in that the Green's Function methodology does not have anything to do with the potential non-conservatism. Rather, it is the single stress calculation methodology used after the Green's Function analysis that is the area of concern. Therefore, all references to Green's Function methodology should be removed from the RIS to avoid misinterpretation.

Comment 6:

> *The Green's function approach involves performing a detailed stress analysis of a component to calculate its response to a step change in temperature. This detailed analysis is used to establish an influence function, which is subsequently used to calculate the stresses caused by the actual plant temperature transients. This methodology has been used to perform fatigue calculations and as input for on-line fatigue monitoring programs. The Green's function methodology is not in question. The concern involves a simplified input for applying the Green's function in which only one value of stress is used for the evaluation of the actual plant transients.*

It is not clear based on the reference to fatigue monitoring programs whether those applications are also being questioned. If not, reference to "fatigue monitoring systems" should be removed from the RIS to avoid misinterpretation. If so, please clarify what aspects of those applications are in question, what actions are necessary, and identify whether the NRC is familiar with the fatigue monitoring literature that has been published over the past 20 years that documents the technology used by these applications and its acceptability for ASME Code evaluation.

Comments on Proposed NRC Generic Communication Regulatory Issue Summary (RIS) 2008-XX "Fatigue Analysis of Nuclear Power Plant Components," May 1, 2008[1]

Comment 7:

> *Licensees may have also used the simplified Green's function methodology in operating plant fatigue evaluations for the current license term. For plants with renewed licenses, the staff is considering additional regulatory actions if the simplified Green's function methodology was used.*

If this RIS is intended for license renewal only, the first sentence of this paragraph should be stricken, as any statements concerning the current license term are extraneous.

Comment 8:

The staff identified a concern regarding the methodology used by some license renewal applicants to demonstrate the ability of nuclear power plant components to withstand the cyclic loads associated with plant transient operations for the period of extended operation. This particular analysis methodology involves the use of the Green's function to calculate the fatigue usage during plant transient operations such as startups and shutdowns.

The Green's function approach involves performing a detailed stress analysis of a component to calculate its response to a step change in temperature. This detailed analysis is used to establish an influence function, which is subsequently used to calculate the stresses caused by the actual plant temperature transients. This methodology has been used to perform fatigue calculations and as input for on-line fatigue monitoring programs. The Green's function methodology is not in question. The concern involves a simplified input for applying the Green's function in which only one value of stress is used for the evaluation of the actual plant transients. The detailed stress analysis requires consideration of six stress components, as discussed in ASME Code, Section III, Subsection NB, Subarticle NB-3200. Simplification of the analysis to consider only one value of the stress may provide acceptable results for some applications; however, it also requires a great deal of judgment by the analyst to ensure that the simplification still provides a conservative result.

The staff has requested that recent license renewal applicants that have used this simplified Green's function methodology perform confirmatory analyses to demonstrate that the simplified Green's function analyses provide acceptable results. The confirmatory analyses retain all six stress components. To date, the confirmatory analysis of one component, a boiling-water reactor feedwater nozzle, indicated that the simplified input for the Green's function did not produce conservative results in the nozzle bore area when compared to the detailed analysis. However, the confirmatory analysis still demonstrated that the nozzle had acceptable fatigue usage.

The text of the RIS seems to suggest that the following four conditions are relevant:

1. Fatigue analyses are being performed to support operation during the period of extended operation.
2. These fatigue analyses are being performed in accordance with ASME Code, Subarticle NB-3200 methodology.
3. Green's Functions are being used.
4. An abbreviated stress tensor that ignores some of the non-zero terms is used.

Is it intended that confirmatory analyses are required only for situations where all four of the above conditions are satisfied? If the answer to this question is "yes", why is this issue limited to license renewal evaluations and not the other legacy work where the four conditions above are satisfied? If the answer to this question is "no", please clarify under which conditions that confirmatory analyses are required.

Comment 9:

> *The staff has requested that recent license renewal applicants that have used this simplified Green's function methodology perform confirmatory analyses to demonstrate that the simplified Green's function analyses provide acceptable results. The confirmatory analyses retain all six stress components. To date, the confirmatory analysis of one component, a boiling-water reactor feedwater nozzle, indicated that the simplified input for the Green's function did not produce conservative results in the nozzle bore area when compared to the detailed analysis. However, the confirmatory analysis still demonstrated that the nozzle had acceptable fatigue usage.*

It is not clear from the language in the RIS whether utilities must perform confirmatory analyses and submit notice of such work to the NRC, or whether utilities are being informed of the issue and that no actions are necessary unless specifically requested by the NRC. Please clarify.

Also, there have been several other confirmatory analyses performed to-date, in addition to the one boiling-water reactor feedwater nozzle analysis identified in the RIS, all of which demonstrate acceptable fatigue usage factors with environmental fatigue effects incorporated. Don't these results collectively suggest that the RIS is unnecessary?

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MEMORANDUM

December 3, 2008

Discussion of the ASLB Decision with regards to

Contention 4

The Distinction between Flow Assisted Corrosion and Erosion Corrosion

I. Introduction

On November 24, 2008 the Atomic Safety and Licensing Board (ASLB or the Board) reached a Partial Initial Decision with regards Contention 4[1)]. The Board found that "Entergy had demonstrated that its proposed aging management program (AMP) for the flow accelerated corrosion (FAC) of plant piping will adequately manage the aging effects during the 20-year license renewal period". However, I think that this decision was reached on the basis of a misunderstanding of fundamental facts, misapplication of the major AMP tool (Checworks), and in part misrepresentation of the AMP by Entergy.

II. Flow Assisted Corrosion – Erosion-Corrosion – Erosion

On page 104 of the decision the Board makes a clear distinction between **chemical corrosion** and physical erosion. The definition of corrosion is "the chemical reaction *between a material, usually a metal, and its environment that produces a deterioration of the material and it properties*"[2)]. (This definition implicitly says that all corrosion is

[1)] Atomic Safety and Licensing Board, Partial Initial Decision in the Matter of Entergy Nuclear Vermont Yankee, L.L.C., and Entergy Nuclear Operations, Inc., Docket No 50-271-LR, ASLB No. 06-849-03-LR, November, 24, 2008
[2)] Corrosion Tests and Standards, Robert Baboian Ed. ASTM International, Jan. 2005, pg 8

chemical corrosion). Erosion is defined as the "*progressive loss of material from a solid surface due to mechanical interaction between that surface and a fluid, a multi-component fluid, or solid particles carried in that fluid.* (Erosion therefore is not considered to be chemical in nature). Erosion-corrosion is defined as the "*con-joint action involving corrosion and erosion in the presence of a moving corrosive fluid, leading to the accelerated loss of material*[2)]". These definitions in essence agree with the definitions the Board has used in the Decision, which in turn had been extracted from testimony by Entergy experts, (Horowitz and Fitzpatrick). However, at no point have the conditions under which these phenomena occur been quantitatively circumscribed. Neither has the nature of "mechanical" been clearly defined. This lack of quantitative specificity has led to the misunderstandings of the true nature of erosion-corrosion as will be explained below.

The important point missing from these definitions is this: It is well known that moving fluids accelerate corrosion (see below), and it is also well known that extremely fast moving fluids can damage metal mechanically. The Checworks program is based on the fact that increased velocity increases the corrosion of the base metal, hence the term "Flow Assisted Corrosion" (FAC). Flow in this context acts to accelerate mass transfer as is well recognized by many workers in the field (see for instance NEC RH-03 and NEC JH-36, JH-37.)

On the other hand, the use of high velocity water jets for the precise cutting of almost any material has become a widely used industrial tool. As it turns out the **physical manifestation** of locally flow accelerated corrosion is nearly identical to the action of a random high velocity jet which is why historically both have been identified as "erosional effects", while in essence from a fundamental mechanistic point of view they are entirely different. It is the objective of this discussion to delineate the precise meaning of these terms, highlight where the Board may have misunderstood the usage of the terms and put in perspective which phenomena are the basis for Checworks.

Since there is clearly a transition from "fluid flow accelerating the chemical reactions" to "flow actually destroying metal" the question becomes where such transition will occur and whether velocities which effect such transitions could indeed occur in nuclear piping installation.

The answer is no, not in general but perhaps in rare occasions as mentioned by Dr. Horowitz [3)]. Such velocities do not generally occur in nuclear facility piping. If they were occurring then the damage would be much more rapid, days rather than years to failure as are actually found.

In summary: The confusion between erosion-corrosion and erosion arises from the fact that the physical manifestation of either effects are very similar (localized metal loss) while mechanistically and kinetically the causes for either are entirely different.

III. Corrosion – Flow Assisted Corrosion (FAC) – Flow Induced Localized Corrosion

These phenomena are being discussed by means of the example of corrosion of carbon steel in high energy fluids in power generation. The details of the mechanism can be found in NEC RH-005 (June 2, 2008). It is first and foremost important to recognize that under the prevailing conditions the steel is covered by a layer of iron oxide, or magnetite (see Fig. 1 loc. cit.). Magnetite at high temperature and pure water has a certain low solubility. As a consequence an equilibrium iron ion concentration establishes itself on the surface of the oxide. Iron ions will be removed from the surface by diffusion first through the boundary layer and then by convection into the adjacent turbulent fluid. As iron ions diffuse away from the oxide surface, they are being replaced from below by corrosion. The rate determining step is the diffusion through the laminar boundary layer, hence the corrosion rate is mass transfer controlled. The mass transfer rate is controlled by the thickness of the laminar boundary layer and the thickness of the laminar boundary layer is controlled by the local flow rate.

[3)] Horowitz Tr part 4 pg 22 at 24

Therefore: **The corrosion rate is controlled by the flow rate,** not in terms of "eroding the metal surface" but rather by accelerating the diffusion of iron ions away from the magnetite surface. If and when locally high flow rates occur because of turbulence generated by changes in the geometry of the flow channel (orifices, elbows, flanges, valves etc.) then locally higher corrosion rates occur. This is precisely what Checworks was designed to handle and what had been discussed in the various papers which had been submitted by Entergy and Dr. Horowitz in support of the model[4]. The concern in Checworks is not with straight runs of process piping, which would lead to flow accelerated general corrosion, but with all the features which can distort normal flow patterns[5]. Since these distortion are local, and since they lead to turbulence and therefore higher local flow rates (or shear stresses), hence higher local corrosion rates, we have called these phenomena flow induced localized corrosion, which describes phenomena identical to FAC and identical to the ones embedded in Checworks, but in a more descriptive manner. The lesson learned from this analysis is that it is totally disingenuous to maintain that FAC is not a local corrosion phenomenon[6]. It is most pronounced and hence most critical in those location were normal flow patterns are is disturbed and where therefore locally high fluid velocities prevail.

In summary it is well established that FILC (FAC) is a chemical phenomenon (dissolution of magnetite), the rate of which is accelerated by locally higher flow rates.

IV. The Erosion Phenomenon in contrast to FILC (FAC)

The question then remains under what conditions will enhanced flow destroy metals, or corrosion product layers, mechanically – i.e. what kind of forces are necessary for this to happen. We will resort to two sets of experiences in order to put this question in perspective. On the one hand it is known, and has become a standard tool in the metal

[4] See for instance: NEC-JH-37, Paper by V.K. Chexal, W.H. Layman, J.S. Horowitz, Tackling the Single Phase Erosion Corrosion Issue. Or Flow Accelerated Corrosion in Power Plants, EPRI Publication; E-4-08
[5] See EPRI Publication E-4-08 pg 7-3
[6] Fitzpatrick Tr 1476 @ 11. (please note that not all versions of the Tr are paginated the same)

working industry that almost any material can be cut with a water jet of high velocity[7]. Water jets are operated from roughly 100 m/sec to 900 m/sec depending on the desired cutting speed and the desired quality of the cut. I have discussed this technology with Professor Dr. Günter Schmitt[8] with whom I have collaborated for quite some time. A review of the pertinent technical and trade literature led us to the conclusion that the minimum necessary flow rate for cutting metal is a jet at 100 m/sec with a near vertical incidence. This number, independently derived, coincides quite well with a number given by L. Piatti nearly 50 years ago at the Swiss Technical Institute of Technology[9], and was mentioned at the hearing on July 24, 2008 by Dr. Horowitz as well[10]. It is at this stage also important to mention that jets of as much as 900 m/sec (mach 3) have been developed to cut metal. While the jet emanates from a nozzle where the flow is parallel to the tubing and nozzle wall it impinges near vertically onto the surface to be cut. Where the flow is parallel to the wall essentially no erosion occurs even at these extreme velocities, while with a vertical incidence of the jet rapid cutting occurs. That the cutting process is indeed mechanical and not corrosive can be seen in fact that metallic particles accumulate in the cutting fluid.

It is therefore not likely that phenomena of this kind occur in high energy fluid piping in the power generation where maximum velocities are of the order of 10 m/sec.

On the other hand it is also known that the compressive strengths of corrosion product layers are of the order of hundreds of mega-Pascals, 10^8 Pa (N/m^2)[8]. It has recently been shown that rapid flow accelerated corrosion occurs at shear stresses of the order of mega Pascals only, and it must then be assumed that the mechanism is based on the accelerated dissolution process rather than mechanical removal of metal or a corrosion product layer.

It is therefore concluded that all corrosion processes occurring in high energy piping under the influence of flow and turbulence are based on the dissolution mechanism of the

[7] Flow International Corporation see www.flowcorp.com see "Water Jet Cutting"
[8] Professor Dr. Günter Schmitt, South-Westfalia University of Applied Sciences, Iserlohn, Germany, See also NEC-RH_03 Ref. 13: R.H. Hausler, G. Schmitt, Hydrodynamic and Flow Effects on Corrosion Inhibition, NACE CORROSION/2004 paper #402
[9] see Tr pg 1479 at 11, (other versions of the transcript may have different pagination)
[10] Transcript pt 4 pg 30 at 7 ctd.

iron oxide, except for some rare occurrences of impingement phenomena and cavitation. The dissolution mechanism under various physical conditions had previously been described in NEC RH-03 and NEC RH-05.

V. Conclusions

The major conclusion from this analysis of the corrosion mechanism in high energy fluid piping systems in power generation is that the prevailing corrosion mechanism is based on the dissolution of the magnetite layer. The dissolution rate is mass transfer and hence flow dependent. This in essence is the consensus one can extract from the papers and publications summarizing the studies which eventually led to the development of Checworks. Even though in the early days of these developments the localized corrosion phenomena were identified as erosion-corrosion this terminology was not based on mechanistic insights.

It makes therefore no sense whatsoever to try and exclude "erosion-corrosion" as used in the past[11)] from being covered by the Checworks model which is then said to be "applicable" to flow assisted corrosion only. FAC, FLIC, and "erosion-corrosion" as understood in the early days, are identical and are local phenomena. The extent of this localized corrosion damage depends entirely on the flow rate and the geometry of the feature which causes the flow disturbance, hence high localized velocities within the turbulences. Since under turbulent conditions the corrosion rate is no longer proportional to the flow rate (NEC RH-03) but varies with an exponent of 2 or larger it is imperative that following the power upgrade Checworks as used at Vermont Yankee needs to be recalibrated. Such recalibration must be part of a responsible AMP.

In light of the above we will now turn to review the steps "the parties stipulated[12)] that during the PEO Entergy proposes to implement with regards to FAC".

[11)] "Tackling the Single Phase Erosion-Corrosion Issue" V.K. Chexal, W.H. Layman, J.S. Horowitz; Paper presented at the American Power Conference, April 18-20, 1988, Chicago Illinois
[12)] Decision pg 103 2nd par

1. *Conduct an analysis to determine the critical locations*. It is truly surprising that this had not already been done and is not an ongoing process as part of the past as well as the future FAC AMP. Nevertheless, the critical locations (localized not general corrosion) cannot be determined unless the predictive power of Checworks is used. There are simply too many locations that are potentially critical. For such an analysis, however, the model needs to be updated and recalibrated using the data gathered post the upgrade as argued above (see also below).

2. *Perform baseline inspections to determine the extent of thinning at these locations*. This activity derives directly from the judicious use of Checworks. Again this is an activity which must be performed regularly in order to improve the predictive power of Checworks because operating conditions may change in the future.

3. *Perform follow-up inspections to confirm the predictions*. This is a routine activity performed whenever and wherever an empirical model is used to predict future events.

It has been said that the effect of the recent EPU on Entergy's FAC analysis for the plant has been reviewed by the NRC staff in its safety evaluation[13]. NEC also has reviewed documents relating to the FAC AMP, specifically Checworks computer printouts which reveal Entergy has not addressed the changes in the plant operating conditions[14] (Appendix A)

The Board found that the Checworks model is only one of several means to select the critical locations for inspections and has a marginal, if any, role in trending wear rate to assess the safety aspects of the plant, or implementing corrective actions. This finding is in direct contradiction to Dr. Horowitz's power point presentation[15]. Furthermore, Checworks is (or is intended to be) the repository of all the wall thickness measurements made at locations extracted from Checworks as critical. Since not all suspected location can be UT'd at any one outage, the correlations and algorithms imbedded in Checworks are used to determine the most probable corrosion rate (trending) of other suspected locations programmed into the software. Therefore, Checworks must be and is considered

[13] Decision page 121 par. 1
[14] Analysis of Data contained in E-4-28. E-4-29 and E-4-30
[15] Power Point presentation by Dr. Horowitz during the Hearing

the central tool of the FAC AMP. Unfortunately, as the Board also found, the model has not been properly updated for the past 6 years and is therefore ineffective as guidance for the next round of inspections.

Similarly, Entergy has indicated that other means in the FAC program are plant and industry experience and engineering judgment[16]. Plant experience presumably is experience at VY as well as other similar plants. The concern here is that lack of understanding of the basic corrosion mechanism underlying the flow accelerated corrosion phenomenon, in particular that the manifestation of FAC is localized, renders interpretation of experiences at other plants or other locations in the same plant rather dubious, and engineering judgment cannot help here either. It should be clear by now that VY personnel does not understand the corrosion mechanisms nor the workings of Checwork as amply demonstrated by the report of an EPRI "Review of the Vermont Yankee Nuclear Plant Flow Accelerated Corrosion Program" of Feb. 28, 2000[17]. In Appendix A we analyze the way Checworks was handled in subsequent years.

In point of fact we cite Mr. Fitzpatrick's statement that VY's FAC predictions have consistently been conservative[18]. Actually all the graphic printouts in E-4-28, -29 and -30 indicate just about as many UT wall thickness measurements to be twice the predicted values as there are half the predictions. Clearly this shows a significant lack of engineering judgment.

An analysis of Checworks data contained in E-4-28, 29, and 30 indicates strongly that purported measurements have in fact been calculated and are not measurements. This is all the more alarming as the NRC staff has audited VY use of Checworks and presumably found it in good order (see discussion in Appendix A)

In light of the above we therefore strongly urge the Board to reconsider their decision and withhold the extension of their operating license until such time that proper independent

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16) Decision pg 108 2nd par

17) EPRI letter to Mr. James Fitzpatrick Feb 28 2000, Ex. E-4-27

18) Decision pg 126, 2nd par

inspection of all critical locations have been performed and the predictive model properly updated.

APPENDIX A

In preparation for the Hearing on Contention 4 on July 24, 2008, I started reviewing the data that had been reported by VY for Chekworks for the 2001, 2003 and 2006 refueling outages. The attached Table will illustrate the result of my reviews. The data are drawn from Entergy Exhibits referred to below[19)].

In the left most column are listed the piping features belonging to the feed water system line identified as "001-16"-FDW-01", along with the relevant "geometry code"[20]. The next columns list the wear rates as of Oct. 28 2001 (E-4-28). These are the average wear rate and the current wear rate. Average would mean, I guess, total metal loss averaged over the duration of service. Current should mean as determined between the previous and the current outage. The same thing is found in the two columns further to the right for the March 28, 2003 outage. One notices that both the average and the current wear rates now are lower. However, the so called line adjustment factor is lower also in 2003. When one uses the ratio of the line adjustment factors between 2001 and 2003 and calculates with this ratio the wear rates for 2003 from the ones in 2001 one finds they agree with the ones "measured" in 2003 by a constant factor of 7 or 6 percent, respectively. That in itself may not be surprising, however, that this difference should be a **constant,** 7 or 6 % for all 19 components listed, accurate to the 3 decimal, is rather incredible. This just does not happen in nature. Rather this suggests to anyone skilled in data analysis that the data were fabricated.

Looking at the data for the 2006 outage one find exactly the same thing, only that now the line correction factor (for the same line) has further decreased. Using the ratios of the line correction factors specified for 2003 and 2006 one finds again that the difference between the 2003 and 2006 data are within 5.5 and 6.5 % for all 19 line features. (again these percentage differences are accurate to the 3rd and 4th decimal. What this means is

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19) These reports are entitled "Vermont Yankee Piping Flow Accelerated Corrosion (FAC) Program EPRI Checworks Wear Rate Analysis Results, Cycles 20 & 21, 22B, and 25., E-4-28, 29, 30

20) The "geometry code" is a specific reference to a feature in the line system with a specific geometry. For specific explanation see NEC JH 037

that the line correction factors for 2003 and 2006 was arbitrarily changed and new total rates and current rates were calculated (not measured) for both 2003 and 2006 outages.

At the bottom of the first table one can find a listing of the "Total Measured Wear" of some of the above objects. That the listing is not complete indicates that not all the objects in the above table had actually been measured as the table may suggest.

However, one finds that between 2001 and 2003 the total measured wear has not changed even by one mil (one thousands of one inch), even though there were measured wear rates of the order of between 10 and 16 mils per year. For 18 months this would calculate out to something like 15 to 24 mils. This magnitude of metal loss should have been detected, even more so as the numbers are apparently given with an accuracy to the third digit. Note however, that in 2006 the total wear has been decreased dramatically. This is the total wear not the incremental wear.

Can anyone believe this data? Apparently the NRC does since the NRC audited Checkworks in 2008[21)]. Even more alarming is the fact the NRC stated that "..its explicitly stated in the LRA the [FAC] program they are currently using is the one they are going to use [for the extended license period].

In my opinion, this analysis strongly suggests, that this plant should be shut down until a complete verifiable baseline will have been reestablished.

[21)] Transcript pg 1576 at 20 (Note: the pagination may vary between transcripts issued at different times)

		Data as reported 25-Oct-01 Lin Adjus Fact 0.891		Data as reported 25-Mar-03 *Line Adjus Factor 0.694*		Calculated Data on basis of ratio of Adj. Factor				Data as reported 28-Sep-06 Line Adjus Factor 0.175		Calculated Data on basis of ratio of Adj. Factor 28-Sep-06 vs 25 Oct-01			
						25-Mar-03	25-Mar-03								
		Ave. Wear Rate mpy	Curr Wear Rate mpy	Ave. Wear Rate mpy	Curr Wear Rate mpy	*Ave. Wear Rate mpy calculated*	*Curr Wear Rate mpy calculated*	Differences between 3/35.03 and 9/28/06		Ave. Wear Rate mpy	Curr Wear Rate mpy	*Ave. Wear Rate mpy calculated*	*Curr Wear Rate mpy calculated*	Differences between 9/28/06 and 3/25/03	
Outlet P-1-1A	31	37.802	29.029	28.571	21.169	29.44398204	22.61069136	3.0%	6%	7.624	5.712	7.20450288	5.33800432	5.5%	6.5%
FD01RD01 L/E	18	17.828	13.003	12.797	9.482	13.8862312	10.12803816	7.8%	6%	3.370	2.559	3.22690922	2.39099424	4.2%	6.6%
FD01RD01 S/E	18	22.289	16.256	16.000	11.855	17.3609046	12.66180022	7.8%	6%	4.213	3.199	4.03458213	2.9893732	4.2%	6.6%
FD01EL01	4	21.988	16.037	15.784	11.695	17.12645567	12.4912211	7.8%	6%	4.157	3.156	3.98011527	2.94902738	4.3%	6.6%
FD01TE05 (U/S)	15	17.828	13.003	12.797	9.482	13.8862312	10.12803816	7.8%	6%	3.370	2.559	3.22690922	2.39099424	4.2%	6.6%
FD01TE05 (D/S)	15	17.828	13.003	12.797	9.482	13.8862312	10.12803816	7.8%	6%	3.370	2.559	3.22690922	2.39099424	4.2%	6.6%
FD01SP01	58	13.074	9.535	9.385	6.954	10.18334007	7.42681257	7.8%	6%	2.471	1.876	2.36653458	1.75353026	4.2%	6.5%
FD01EL02	4	21.988	16.037	15.784	11.695	17.12645567	12.4912211	7.8%	6%	4.157	3.156	3.98011527	2.94902738	4.3%	6.6%
FD01SP02 US	54	19.017	13.87	13.651	10.114	14.81234343	10.80334456	7.8%	6%	3.595	2.729	3.44225504	2.55036023	4.2%	6.5%
FD01SP02 DS	54	19.017	13.87	13.651	10.114	14.81234343	10.80334456	7.8%	6%	3.595	2.729	3.44225504	2.55036023	4.2%	6.5%
FD01EL03	2	21.988	16.037	15.784	11.695	17.12645567	12.4912211	7.8%	6%	4.157	3.156	3.98011527	2.94902738	4.3%	6.6%
FD01SP03 US	52	14.857	10.836	10.665	7.902	11.57211897	8.440161616	7.8%	6%	2.808	2.132	2.68930115	1.99257925	4.2%	6.5%
FD01SP03 DS	52	14.857	10.836	10.665	7.902	11.57211897	8.440161616	7.8%	6%	2.808	2.132	2.68930115	1.99257925	4.2%	6.5%
FD01EL04	2	21.988	16.037	15.784	11.965	17.12645567	12.4912211	7.8%	4%	4.157	3.156	3.98011527	3.01711095	4.3%	4.4%
FD01SP04 US	52	14.857	10.836	10.665	7.902	11.57211897	8.440161616	7.8%	6%	2.808	2.132	2.68930115	1.99257925	4.2%	6.5%
FD01SP04 DS	52	14.857	10.836	10.665	7.902	11.57211897	8.440161616	7.8%	6%	2.808	2.132	2.68930115	1.99257925	4.2%	6.5%
FD01EL05	2	21.988	16.037	15.784	11.695	17.12645567	12.4912211	7.8%	6%	4.157	3.156	3.98011527	2.94902738	4.3%	6.6%
FD01SP05 US	52	14.857	10.836	10.665	7.902	11.57211897	8.440161616	7.8%	6%	2.808	2.132	2.68930115	1.99257925	4.2%	6.5%
FD01SP05 DS	52	14.857	10.836	10.665	7.902	11.57211897	8.440161616	7.8%	6%	2.808	2.132	2.68930115	1.99257925	4.2%	6.5%

Total Meas. Wear

		10/25/2001	3/25/2003	9/28/2006
Outlet P-1-1A	31			
FD01RD01 L/E	18			
FD01RD01 S/E	18			
FD01EL01	4	298	298	79
FD01TE05 (U/S)	15	342	342	122
FD01TE05 (D/S)	15	261	261	133
FD01SP01	58			
FD01EL02	4	455	455	88
FD01SP02 US	54			
FD01SP02 DS	54			74
FD01EL03	2			
FD01SP03 US	52	163	163	48
FD01SP03 DS	52			
FD01EL04	2	102	102	19
FD01SP04 US	52			
FD01SP04 DS	52			14
FD01EL05	2			
FD01SP05 US	52			
FD01SP05 DS	52			

EXHIBIT
RH HR. 2
ASLB 06-849-03 LR

December 4, 2008

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

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

DECLARATION OF RUDOLF H. HAUSLER
IN RESPONSE TO ALSB PARTIAL INITIAL DECISION,
DATED NOVEMBER 24, 2008
(RULING ON CONTENTION 2A. 2B, 3, AND 4)

Rudolf H. Hausler states as follows under penalty of perjury:

1. I have prepared the attached "Memorandum in Response to ALSB Partial Initial Decision" in the above captioned proceeding.

2. The factual statements and opinions expressed in the memorandum are true and correct to the best of my personal knowledge and belief.

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

Executed on December 4, 2008

/Original signed Rudolf H. Hausler/

Rudolf H. Hausler

Rudolf H. Hausler

Notarized:

AMANDA MYERS
Notary Public
STATE OF TEXAS
My Comm. Exp. Feb. 24, 2011

Kaufman County, Texas

UNITED STATES NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

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

NON-DISCLOSURE AGREEMENT

Under penalty of perjury, I hereby certify that: access to Proprietary Documents is provided to me pursuant to the terms and restrictions of the Atomic Safety and Licensing Board's protective order, dated January 12, 2007, in this proceeding; that I have been given a copy and have read said protective order; and that I agree to be bound by it. I understand and agree that Proprietary Documents, their contents, or any notes or other memoranda summarizing or otherwise describing their contents, or any form of information that derives from the Proprietary Documents and copies or discloses the contents of the Proprietary Documents, shall be held in confidence and shall not be disclosed to anyone except in accordance with that protective order. I acknowledge that a violation of this agreement or the protective order, which incorporates the terms of this agreement, constitutes a violation of an order of the Nuclear Regulatory Commission and may result in the imposition of such sanctions as the Board or the Commission may deem to be appropriate.

WHEREFORE, I do solemnly agree to protect such Proprietary Documents, and their contents, as may be disclosed to me in this NRC proceeding, in accordance with the terms of this agreement.

Name (printed): Rudolf H. Hausler

Title: Pres. Corro-Consulta

Employed by or Representing: New England Coalition, Inc.

Signature: Rudolf H. Hausler

Date: 12/13/2008

December 5, 2008

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.)) Vermont Yankee Nuclear Power Station)	Docket No. 50-271-LR ASLBP No. 06-849-03-LR

DECLARATION OF ULRICH K. WITTE
IN RESPONSE TO ALSB PARTIAL INITIAL DECISION,
DATED NOVEMBER 24, 2008
(RULING ON CONTENTIONS 2A, 2B, 3, AND 4)

Ulrich K. Witte states as follows under penalty of perjury:

1. I have prepared the attached "Memorandum in Response to ALSB Partial Initial Decision" in the above captioned proceeding.

2. The factual statements and opinions expressed in the memorandum are true and correct to the best of my personal knowledge and belief.

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

Executed on December 2, 2008

/Original signed Ulrich K. Witte/

Ulrich K. Witte

Notarized:

County of New Haven, Connecticut
Exp. 8-31-2011

Memorandum

I. INTRODUCTION

I have prepared this Memorandum at the request of New England Coalition (NEC) following a detailed review of the Atomic Safety and licensing Board (ASLB) Partial Initial Decision (PID) of November 24, 2008 in the matter of the Entergy Nuclear Vermont Yankee License Renewal Application. My review focused on NEC Contention 4 accepted for litigation and restated by the ASLB in its Order of September 22, 2006, as:

> 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.[1]

I noted that the ALSB furthermore, supplemented the contention and the basis for admitting the contention with "Entergy's plan for managing flow-accelerated corrosion (FAC) in plant piping fails to meet the requirements of 10 C.F.R. § 54.21(a)(3), i.e., 'fails to demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB during the period of extended operations.' "[2]

I also noted, as provided in the September 22, 2006 Order, the ASLB directly quoted from NEC's original petition as part of its opinion to admit the contention including: "NEC takes particular exception to Entergy's proposal to use 'a computer model called CHECWORKS to determine the scope and frequency of

[1] LBP-06-20, Memorandum and Order, (Ruling on Standing, Contentions, Hearing Procedures, State Statutory Claim, and Contention Appeal), dated September 22, 2006, at 107

[2] Id. And, NEC's intervention petition, "Petition for Leave to Intervene, Request for Hearing, and Contentions," dated May 26, 2006 at 18.

inspections of components that are susceptible to FAC.'"[3] The ALSB particularized further and stated that "NEC alleges 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."[4]

The ALSB provided in its Order admitting Contention 4, "For this reason... '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.' "[5]

In its Order and Partial Initial Decision, the ALSB restated Contention 4 as "a safety contention that deals with flow accelerated corrosion (FAC) in the plant piping,"[6] The contention itself as provided below is identical in content:

> Entergy's License Renewal Application does not include an adequate plan to monitor aging of plant piping due to flow accelerated corrosion during the extension period.[7]

Clarification and context o the above broad contention, as provided in the September 22, 2006 ruling by the ASLB and restated above is absent in the Initial Partial Decision.

Although absent, the context and clarification of Contention 4 explicitly quoted by the ASLB is useful so as to clarify the Contention 4—given it particularizes what specific aspects of the Flow Acceleration Program caused NEC to raise this contention. Given the Board entered this language in their admissibility

[3] Id.
[4] Id at 19.
[5] Id.
[6] Order and Initial Partial Decision at 95.
[7] Id.

ruling, in reviewing the ASLB decision, I presume the clarification, particularizing context of the contention remains applicable.

II. BACKGROUND

FAC is by far the single most significant degradation mechanism of piping due to single phase high energy fluids and accounts for 32.5% of failures within a population of 12,000 reactor years domestic with less than 40 years of operation. (see Discussion under Part V, and provided on page 3 of Exhibit NEC - Motion For Reconsideration – No.1)

The decision by the ASLB turned on the Board first settling that CHECWORKS is only one of five methods of selecting and monitoring degradation and is not reliable by itself.[8] The other four methods for selecting and monitoring FAC are essentially elements of engineering judgment. This includes a disciplined, systematic, consistent and procedural method for review by trained engineering staff and effective management oversight to consider and prioritize specific operating history, industry experience, age of new design changes, function changes, known changes in chemistry, and refinement of wall thinning design limits, and other criteria.

However, without agreement on the FAC degradation mechanism, one cannot argue adequacy of the program effectiveness, adequate implementation, and in particular adequate engineering judgment to implement a program whose goal specifically predicts, prioritizes, and monitors the susceptible plant components and piping due to this specific degradation mechanism.

[8] As noted in Part I of this memorandum, this is one of the central questions the Board quoted in admitting Contention 4.

Simply put, one most determine how and why the corrosion is occurring so as to select and monitor the most susceptible wear regions, project wear rates, inspect and intervene prior to catastrophic ruptures. The FAC corrosion model empirically includes approximately eight variables.[9]

Given my experience in development, roll out, implementation, regulatory transparency, with adequate oversight of numerous cross functional engineering programs, the decision by the Board to hold that CHECWORKS was not reliable and use of engineering judgment closed the gap should be carefully reviewed.

The ASLB over simplified when it held CHECWORKS as unreliable, and in addition the ASLB imposed a heavy and unrealistic burden on oversight, management, and disciplined engineering knowledge with substantial experience and expertise for the Licensee to effectively accomplish the FAC Program goals. CHECWORKS cannot be discarded, it does need to be effectively implemented. Once effectively implemented, engineering judgment, while important, is less burdened with distinguishing corrosion phenomena, and ensuring that this program – FAC—only resolves FAC degradation.

Numerous unpredicted failures in the industry start with lack of knowledge of the failure mechanism. Flow Accelerated Corrosion began with a catastrophic failure in 1986, and the regulations were promulgated under 89-07. Unforeseen fire challenges to reactor control at Browns Ferry in 1974, and regulations put in place in 1980 after the unforeseen fire caused loss of about 1600 electrical cables is another example. Boric Acid based corrosion on the Davis Besse reactor vessel head is a

[9] The model is itself, and the actual degradation mechanism are not part of my memorandum, however, are described in Dr. Hausler's affidavit. This memorandum and affidavit does not extend beyond program implementation.

clear example of what happens when the program for monitoring this potential degradation condition is given short shrift, and not the oversight, the priority, and significance warranted against the risks taken.

FAC events continue, with close scrutiny by the industry, because of the complex multivariable degradation mechanism contributing to FAC related wall thinning,

III. SUMMARY

The ALSB settled incorrectly that limits of CHECWORKS as not reliable, and also incorrectly ruled on the unrealistically broad credit given to engineering judgment in determining the FAC Program as currently described to be in place for extended operation is adequate.

If one does not know what one is attempting to evaluate, one cannot conclude either by a well-fleshed out program and with fully pedigreed implementing procedures or with sound engineering judgment as to whether predicting, measuring, and monitoring techniques are adequate.

Given the statements made both in oral testimony, and repeatedly in the proceedings the Licensee has said the FAC Program will comply with the guidance contained in the AMP, and is in fact *currently* in compliance. The Licensee has stated repeatedly that no changes from the current program are planned to ensure FAC management for extended operation.[10]

[10] This statement has been cited beginning the motion for summary disposition, and again in prefiled testimony, and repeatedly during oral testimony.

One cannot conclude the program conforms or that it will comply with relicense rules, and provide an auditable record supporting the fidelity of the program for relicensing given the both the board and the License crediting the current program to establish the adequacy of the program that will be in place for Period of Extended operation .

While testimony provided argues current compliance, the evidence shows a very different conclusion. Of particular relevance is the most recent cornerstone roll up report dated July 7, 2008 regarding the Overall Condition of the Flow Accelerated Corrosion management program. This is report is attached to my affidavit and memorandum as Exhibit NEC MFR No. 3, and discussed below.

IV. STANDARD AND LIMITS OF MY REVIEW:

1. In performing my review, I refer to the ASLB's central issue to settle regarding Contention 4, as provided in their statement:[11]

   > Pursuant to 10 C.F.R. §§ 54.21(a)(3), (c)(1)(iii), Entergy must establish an AMP that is adequate to provide reasonable assurance that the intended function of the piping subject to FAC will be maintained in accordance with the CLB for the PEO. Entergy must demonstrate that its AMP for piping subject to FAC is adequate, and that it satisfies the "reasonable assurance" standard by a preponderance of the evidence[12].

2. The rule of law regarding Motion for Reconsideration is not part of this memorandum. I refer the ASLB to NEC's brief for Motion for Reconsideration to which this affidavit is attached thereof, regarding specific standards and

[11] Order and partial Initial decision, November 24, 2008, at 98.

[12] Zion Station, ALAB-616, 12 NRC at 421.

applicable of law. I proffer my testimony in conformance with these standards as referenced and incorporate them herein.

3. Particular limits of my review as an expert witness in light of specific judicial facts requiring clarification, the Board decision after prefiled testimony, after submittal of NEC's reply, only four calendar days prior to the ASLB hearing of July 21, 2008 through July 24, 2008.[13] This memorandum and my testimony proffers my arguments that honor the late limitations imposed under the standards for reconsideration. The memorandum also clarifies my expertise as exemplified amongst relevant disciplines to establish the effective implementation of the Vermont Yankee Flow Accelerated Corrosion Program. This includes management and implementation of engineering design controls, change controls, and program management expertise to ensure programs that cross functional areas of plant operations such as the Flow Accelerated Corrosion Programs. My professional intent is to provide fully integrated, fully implemented programs that are complaint regulatory requirements and the Licensee's commitments.
4. My expertise is in configuration management, as well as substantial experience in plant licensing and also includes development of standards regarding management of Licensee's license basis contribute to this as well. For example, I was instrumental for the Millstone licensees (in transition at the time) in reestablishing 35,000 docketed regulatory commitments for Units 2, 3. During this initiative I

[13] My credentials and expertise relevant to the factual clarification contained within my testimony, and use of judicial evidence proffered by me is restricted given my expertise, and experience. As noted, it is limited under the Board order to areas of engineering, configuration management, programmatic controls, and functional integration in its order dated July 16, 2008. The petitioner was given four days notice from Oral Hearing, and chose not to motion for reconsideration, or appeal.

contributed to NEI Guidance[14] regarding commitment management prepared during in 1999, and endorsed by NRC Staff on February 22, 2000. A transparent commitment management program (to the regulator as well as each management and staff of each functional area of plant operations) was a key factor in Millstone Units 3 and 2 in regaining the confidence of the regulator and ultimately in granted authorization by the commission to return each Unit to service—after being shut down for more than a year each over loss of control of the license and design basis for each unit.

V. <u>STATEMENTS CONTAINED IN THIS AFFIDAVIT ARE LIMITED AND FOCUS UPON THE BASIS OF THE MOTION FOR RECONSIDERATION RELEVANT CONTENTION 4</u>:

1. After a detailed review of the Initial Partial Decision, dated November 24, 2008, it is my conclusion that the ASLB settled certain judicial facts relevant to the holding and decision incorrectly. The ASLB may have reached the improper holding based upon the current fidelity of the program. Without ambiguity, the Board is settling the contention submitted by the petitioner by wringing out how or if the Licensee has met the following obligation, articulated on page 98:

> Entergy must establish an Aging Management Program that is adequate to provide reasonable assurance that the intended function of the piping subject to FAC will be maintained in accordance with the Current licensing Basis for the Period of Extended Operations. Entergy must demonstrate that its AMP for piping subject to FAC is adequate, and that it satisfies the "reasonable assurance" standard by a preponderance of the evidence.

2. In addressing this broad question, I respond to the above in context of the three questions laid out by the ASLB:

[14] See SECY-00-0045, and NEI-99-04, "Guidelines for Managing NRC Commitments"

i. "Whether the petitioner "asserted that the Entergy's [Aging Management Program] AMP for flow accelerated corrosion, fails to demonstrates that the effects of aging will be adequately managed, [with respect to program scope, program definition, and implementation, oversight, and effectiveness as these things related to operation during the license extension and given the Licensee has explicated stated the present program as implemented not intended to be modified, and stands on its adequacy today as being adequate during the PEO.]"[15]

ii. "Whether the effects of aging from FAC on the intended functions of piping and components will be adequately managed for the [Period of Extended Operation] PEO for Renewal"[16]

iii. "Whether there is reasonable assurance [at the program level, procedure, level, and as implemented, with a view only towards the PEO][17]that the activities authorized by the renewed license will [protect the health and safety of the public, as well as the public's assets, during the PEO][18]

3. Therefore, under the standards, and the limits imposed upon my expertise as it applies to Contention 4, my review consists of a top-down examination of the initial partial decision of the FAC program, its implementation, the record providing adequate oversight including quality assurance, use of cross functional exchange of data as brought forward in both written and oral testimony, including

[15] order and initial statement of position, dated November 24, 2008 at 103, and as required under 10 C.F.R. §54.21(c)(1)(iii)

[16]. Insert is provided to clarify my limits of review. The same insert are be inserted in subsections ii., and iii.

[17] Id.

[18] In accordance with the requirements of the AEA and Part 54, as required by 10CFR § 50.29

examination of precisely what constitutes sufficient engineering judgment, and stops at that boundary. Facts brought forward and conclusions begin by building upon other expert testimony regarding analysis of the precise degradation mechanism and precision regarding the definition of Flow Accelerated Corrosion. With these facts one can formulate them as the bases so as to examine the record as to whether the program was effective implemented, and what does that really mean for settling the broad question confronted by the Board.

4. By this affidavit, together with the affidavit provided by Dr. Hausler, also included as part of the Motion for Reconsideration, I firmly believe the fact errors made either during live testimony, or in prefiled, in testimony, or simply the failure to acknowledge relevant preexisting evidence and the failure to disclose this evidence to the Board and the parties, the testimony provided as of July 24, 2008, led the Board to an incorrect conclusion.

5. I specifically do not allege that any errors of omission, or factual errors, or otherwise were intentional by any party, only to layout the record and ferret out the clarifications so as the Board has an opportunity to reconsider its ruling with all the facts relevant to its decision regardless of the source, and instead where simply incorrect. The result of which was a partial decision that properly deserves considered reexamination. Based on my expert view incorrectly settled the question.

VI. THE DEFINITION OF FLOW ACCELERATED-CORRISON METAL DEGRADATION AS WELL AS ESTABLISHING THE PROGRAM SCOPE ARE EACH FUNDMENTAL ELEMENTS NECESSARY TO EFFECTIVELY IMPLEMENT A COMPREHENSIVE FAC PROGRAM DURING POST EXTENDED OPERATIONS.

1. The definition of Flow Accelerated Corrosion as settled by the Board, and is not the subject of my review. It is an area of controversy, and is also the cornerstone what precisely what program must be in place to manage FAC. As found by Dr. Hausler, the settled the definition was ambiguously and inconsistently defined, where the ALSB first wrestled with the definition as proffered Entergy, NRC Staff, and NEC, however found the Contention 4 was not strictly limited to the effects of chemical wear[19]...but did conclude the FAC is the predominant mechanism for corrosion of flow related wear[20], then added "we found that erosion could contribute."[21] The written and oral transcripts provide numerous expert opinions. The industry has not been consistent with respect to this term, and the record provides conflicting opinions. The precise definition is clarified in Dr. Hausler's affidavit.[22] The number of actual degradation mechanisms leading to pipe thinning can be seen in Exhibit NEC MFR –No. 1. The actual noun names of each are closely similar, but the degradation phenomena is fundamentally different.[23]

[19] At page 109, of the Nov 24th order and initial decision

[20] Id.

[21] Id.

[22] Affidavit of Rudy Hausler, page 2.

[23] Id, at page 2.

2. The ASLB dealt with this premise on an embedded assumption. This assumption needs clarification. While some of these erosion phenomena might result in proper selection of inspection points and base lining as is used under the FAC program at Vermont Yankee (see for example, Exhibit E4-06, provided by Entergy and extensively referenced,) the program for FAC is not designed to select wear points other than those attributed to flow accelerated corrosion. Other degradation consequences are at risk, and should not be credited under the FAC program for predictive wear as being effectively managed. Here the ASLB impermissibly settled the ambiguity of degradation phenomena as not necessary so long as sufficient inspections were performed. What is missed, is selection of proper inspection points, and on what frequency. In order to do this one must have critical skills and procedural guidance on discerning what mechanism applies. Those skills include the basis for discerning other degradation mechanisms and separating them out into the separate programs for maintaining system function, and avoiding failures. The judicial fact as settled leaves this prerequisite requirement unresolved, and settles the fact by implicitly crediting engineering judgment, plant experience, industry experience for selecting grid points for non-FAC degradation. Under Exhibit NEC MFR-No. 2, inspection programs for many of these others degradation mechanisms are described.
3. This is not how the program is established at Vermont Yankee, and would require extensive data and reliable data collection of many operating cycles, and engineering judgment beyond that required for the FAC Program itself. The central element of the holding, is a settling that the degradation did not really matter, so long as inspections were proper, to address not just one phenomena on

Page 3 of Exhibit NEC-MFR-No. 1 but essential all others that are relevant to each system.

4. My own expertise in engineering programs leads me to conclude that crediting the selection and trending criteria for this degradation mechanism is entirely limited to flow-accelerated corrosion, and how the program effectively implements it for PEO. The board reached this decision in part, because the License and NRC Staff were not clear on what degradation was in place for FAC (see Dr. Hausler's affidavit, page 6), and by their own acknowledgement, had allowed the FAC program numerous failures on many technical issues, and on many other technical issues that were essentially improperly categorized as administrative. Exhibit NEC-MFR- No. 3 provides additional clarification on the state of the program effective on as of July 7, 2008.[24]

VII. THE BOARD IMPROERPLY CONCLUDED THE FAC PROGRAM REQUIRED FOR EXTENDED OPERATION WILL BE ADEQUATE BY CREDITING THE CURRENT PROGRAM AS EFFECTIVE.[25]

The ALSB correctly stated that "UT measurements track the total affects of wall thinning mechanisms and cannot easily discriminate between the various mechanisms,"[26] however, while this by itself is true, each degradation mechanism involves different empirical models, different inspection criteria, and are in general outside the scope of FAC program.

[24] I note that this cornerstone roll-up report was prepared in April 2008, updated on July 7, 2008 (two weeks before oral testimony), and to NEC's knowledge has never been provided to the parties. This information was provided to NEC on December, 2008 independently of the proceedings.

[25] See page 114, of the Order and initial decision dated November 24, 2008

[26] See page 109, or the Order and Initial decision dated November 24, 2008

The ASLB resolved the definition is not necessary to its holding, given the program as implemented dealt ambiguously with both corrosion and erosion. In fact by improper selection of inspection points based upon an in incorrect conclusion on what degradation mechanism applies, the model is easily corrupted with wear data that is not meaningful, as data input. I.e. selecting the points based on improper criteria, determine no or insignificant wear, and miss the forest for the trees. In addition, as a result of the assumptions requiring clarification the Board settles definition incorrectly, and moves next to effective program implementation based upon the definition and scope as was incorrectly settled.

The FAC Program cannot be ruled adequate for extended operation if scope and definition are improperly "defined," and settled where as they have been vetted out in the industry. In other words, the ASLB cannot change a well understood definition to something different. The judicial fact was improperly settled.

Flow acceleration corrosion is different from other corrosion erosion phenomena. Inspection programs are also different. See Exhibit NEC-MFR-2, for a detailed discussion on the different approaches for predicting, and monitoring approximately fifteen different degradation mechanisms for pipe and component degradation.

The need for clear and undisputed understanding of scope of the Flow Accelerated Corrosion Management Program is central in establishing fundamental implementation with any amount of confidence. Numerous indicators indicate issues with program implementation that support a program that is sound. A few examples included failure to implement the recommendations from the EPRI review circa 2000, failure to update the model in a timely fashion, and crediting baseline information circa 1995, failure to update the software to the model, failure establish a wear trend consistently during each outage through power uprate and through RFO26, failure to properly address negative time to reach minimum wall thickness and rule out CHECWORK anomalies, as called for under program procedures.[27] Failure to address open Corrective Action Report action items dating as far back as 2003.

VIII. CLARIFICATION AS TO THE EXISTING FAC PROGRAM IMPLEMENTED IS EFFECTIVE FAILED TO CONSIDER KEY EVIDENCE INDICATING THE FLOW ACCELERATION PROGRAM HAD AS OF JULY 7, 2008 SIGNIFICANT IMPLEMENTATION ISSUES.

An examination of the evidence cited on page 113 of the Order and initial decision includes the following conclusion, "the NRC staff specifically reviewed the Entergy's claims regarding its FAC program and found all the program elements conform to the criteria contained in the AMP[28], and that that the corrective actions have been effective in managing FAC at the plant.[29]

I find the statement made in the FSER in error. This may have been a simple oversight, or it may have been based upon the Licensee not disclosing relevant information. The corrective "actions ...[circa 2004] have been effective"[30] implies they are complete.

[27] See for example Exhibit En-06, page 24, step [7], where a structural evaluation is called for in accordance with approved procedures. What was heard in oral testimony, is that engineering judgment is relied upon, and this condition is routinely considered a anomaly of the modeling or software itself.
[28] NUREG-1801, AMP XI.M17,

[29] FSER at 3-16 to 3-17.

[30] Id.

Under the roll-up dated July 7, 2008.and provided as exhibit NEC-MFR-03, I find corrective actions from the four year old CR remain *incomplete*. See page 2.

Overall personnel performance for FAC is RED, monitoring parameters are RED, YELLOW, and RED. This on page 2 only. Also indicated is an Action Plan, and a brand new Condition Report for tracking this item on the same page.

If overall personal performance is RED, and the ALSB concluded that engineering judgment (including careful evaluation of plant experience, industry experience etc) is highly relevant to what makes a program effective and in conformance of quality assurance requirements required under the SER,[31] I am compelled to conclude that the settled fact was incorrect—based upon incomplete disclosure by Entergy, and the silence by Entergy in both oral and written testimony as to this report.

This understandably incorrectly settled judicial fact is on point to the ruling by the ALSB. The corner stone report directly contradicts testimony provided by Entergy[32] and NRC Staff experts.[33]

The cornerstone provides RED finding as of April 10, 2008, stating the "programmatic updates need to be completed."[34]

The same cornerstone report provides an update on open Condition Reports dating back to 2003. A YELLOW finding found on page 5 of 11. Careful

[31] See page 3-143 of the February SER for example for Quality Assurance Requirements for FAC.

[32] See transcript page xx

[33] See transcript page xx

[34] See page 8 of 11 of Exhibit NEC –MFR-3.

examination of the criteria for classifying the finding is two open CRs for more than one year is considered a Red finding. In this case there are two "items" and two CRs, each of which are more than five years old yet remain incomplete.[35]

IX. UPDATING THE [FAC] PROGRAM FOR ADMINSTRATIVE ISSUES HAVE AN IMPACT ON THE FIDELITY OF THE PROGRAM AND CANNOT BE SIMPLY EXCLUDED AS OF NO TECHNICAL SIGNIFANCE—AND INCONSISTENT WITH INDUSTRY PRACTICE AND EPRI RECOMMENDATIONS.

The report provided in Exhibit NEC MFR-3 shows this as one of many findings, yet, given the Quality Assurance requirements of Appendix B, and the consequences of failing to update a program that is required to conform to Appendix B, it controverts the conclusions made by the ASLB regarding whether this program may be credited for extended operation.

The basis of this is Engineering Judgment becomes very difficult without consistent, steady, updates of the model, rebaselining as new components are changed out, or new lines are added, or functional requirements change. The ability to provided sound engineering judgment is severely impaired. Updating of program documents is vague.[36]

I polled a few other Licensees (including the Fleet manager for FAC at one licensee) regarding the effectiveness, and confidence each had on their respective FAC programs, and in particular reliance on CHECWORKS. In discussions with a different Licensee, independent of the Entergy Fleet, the Engineering Programs manager, the Fleet FAC program manager, and the implementing engineer at a similar BWR Mark I design facility provided an overview of their program.

[35] "LO-VYTYLO-2003-00327 ca2, still open, both items due in second half of 2008." Contained on page 5 of 11.

[36] See page 5 of 11 of Exhibit NEC-MFR No.3.

The following was noted. First, the model was kept current. This was done with proceduralized requirements, with strict compliance. CHECWORKS revisions, through 1.0g were completed in a timely manner. Revision 2.2 was made current immediately upon its release. Version 3.0 is going to be installed, given its release only days ago, procedures updated, and the technical changes implemented in a timely manner. Second, the model wear rates were trended over many cycles to establish confidence in the model and to correlate actual UT measured wear data.

Third, anomalies were separated from bona fide threats to wall thinning based upon rigorous adherence to trending wear rates, and other factors such as knowledge of the sister train for example. An unreliable result from CHECWORKS was taken very seriously, and only after systematic review, based upon a record well documented and kept current did the FAC engineer, together with the FAC supervisor resolve (under plant and engineering procedures) whether the data was credible or not. If it was a legitimate prediction of for example negative time to wall thinning prior to the inspection period, QA requirements compelled a condition report be generated. If it was not, the procedures required documentation of resolution of the anomaly.

The statement made repeated in oral testimony, and in prefiled testimony that failure to implement timely updates to models were administrative and not technical did not matter, must be examined in the context of the model history. Without history trending, anomalies are difficult to eliminate as not being legitimate. The panel relied and assumed administrative was not technical as was argued in written testimony. Thus the panel erroneously gave credit to the program during the post uprate cycles in their holding.

In reviewing the record, in light of this comparison, and the testimony provided, Entergy FAC program procedures are substantially more prescriptive and contradict oral and written testimony. For example on page 34 of Exhibit 04-06, the procedure provides specific logic diagrams contained in the program plan for handling wall thinning predictions.

In addition quality assurance requirements are referenced as in the program level procedure are prescriptive, Exhibit E4-06, page 28. §8.2 there are four QAPM requirements. There was no oral testimony brought forward, and I was never asked during oral testimony as to whether quality assurance requirements to the program were being properly implemented, or consistent with the requirements articulated on page 3-143 of the February FSER.

Instead, the ASLB relied on testimony that the program was being properly implemented. Yet it was not, as Exhibit NEC-MFR- No. 1 provides. Entergy Quality Assurance program completed an audit in 2004, and declared the program unsatisfactory. Condition Reports were written yet not remained open for years. Exhibit NEC-MFR-No.1 provides indication that so called administrative open items being open since 2003. Testimony provided indicated they were resolved and closed. Thus the evidence provides to the contrary, as did the resident inspector in February 2008.

The Licensee offered no evidence that the commitments provided on page 28 of Exhibit EN-04-06 are in place, on going, and fully implemented.

X. CONCLUSION

The ASLB ruling including CHECWORKS is not reliable, but that FAC program for PEO is not saved because the four other methods of selecting and monitoring degradation are insufficient. None of the alternatives provide for site specific analytical modeling, trending and wear predicting which considers the numerous variables associated with flow accelerated corrosion wear rates.

Ruling out CHECWORKS as acceptable simply because it was not properly base lined is not the answer. Excluding this software is not the answer. Standing blindly behind four other selection methods is flawed. Entergy clearly overstated the validity of this approach. None of the other four selection criteria establish a singular independent tool, distinct and separate for selecting FAC inspection points, and ranking them independently against the known degradation mechanism, and trending wear rates to avoid rupture. They are essentially elements of engineering judgment. The approach proffered by Entergy is flawed , and not an adequate engineering program controls.

The FAC program can only be effective if the baselines derived from the model are consistently and properly brought current as both the program and the in situ plant configuration evolves. They were not. Heavy reliance on engineering judgment in culling out other resources for selecting, monitoring and predicting FAC failures, when the degradation mechanism itself is not understood is not sufficient to conclude the FAC program will be reliable for post extended operations. Upon my review, I confirmed this with others in the industry.

Without undisputed agreement on the definition of FAC, one cannot conclude there is sufficient engineering judgment to effectively predict FAC related degradation. If the Licensee does not know what he is she is trying to measure, the Licensee cannot conclude the measuring technique or result is adequate.

The Aging Management Program requires a robust, functional, and auditable FAC program. So do regulatory requirements for FAC, and the operating license is conditioned on an effective program. FAC is controlled as an Appendix B program—as delineated in February SER. The program is required to be managed, monitored, controlled, audited, and effective. Reliable indicators including the cornerstone rollup report shows that it is not.

Based upon my review of the judicial facts settled, as provided in the initial partial decision, the failure by the License to bring forward on point evidence, and the errors of fact as proffered, it is my conclusion the flow accelerated program credited by the Licensee as sufficient and for extended operation is incorrect The ALSB inadvertently separated the need for a empirical tool, in this case CHECWORKS, and the necessity of rigorous implementation, *together* with robust oversight, disciplined and knowledge engineering staff to implement the program is required to avoid the 32.5% failure rates of degradation due to FAC.

The evidence specifically provided and cited in this memorandum provides for a different conclusion by the ALSB then what was rendered on November 24, 2008.

Exhibit

NEC-Motion for Reconsideration- No. 1

Fatigue in Operating Nuclear Power Plants Components after 60 years

Steve Gosselin

Pacific Northwest National Laboratory

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stephen.gosselin@pnl.gov

Joint U.S. Nuclear Regulatory Commission (NRC) and U.S. Department of Energy (DOE)

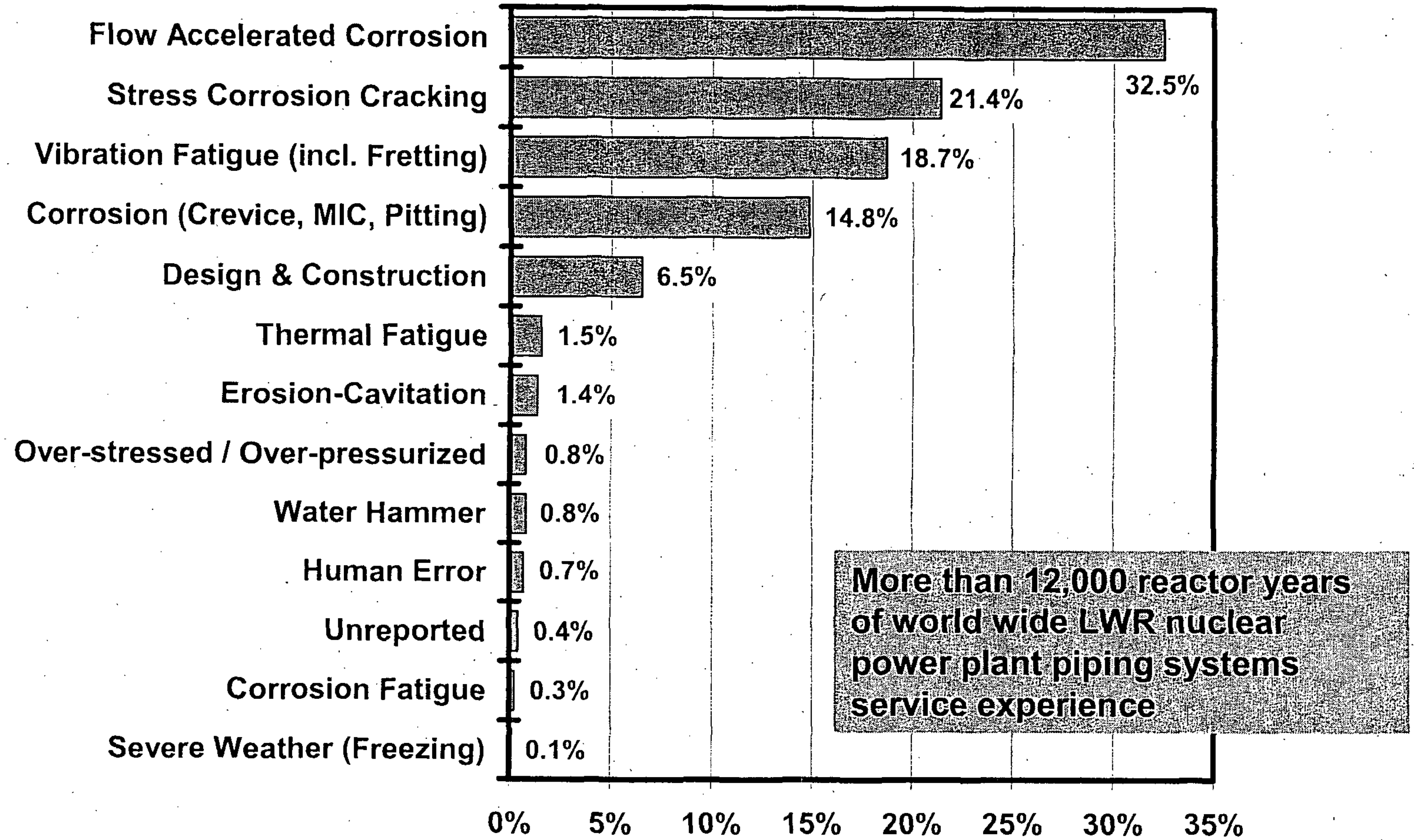
Workshop on U.S. Nuclear Power Plant Life Extension Research and Development Issues

Bethesda, MD, February 19-21, 2008

Summary

- Service Experience
- Component Fatigue Qualification and Serviceability
- Challenges and Directions for the Future
- Questions and Discussion

U.S. Failures by Degradation Mechanisms

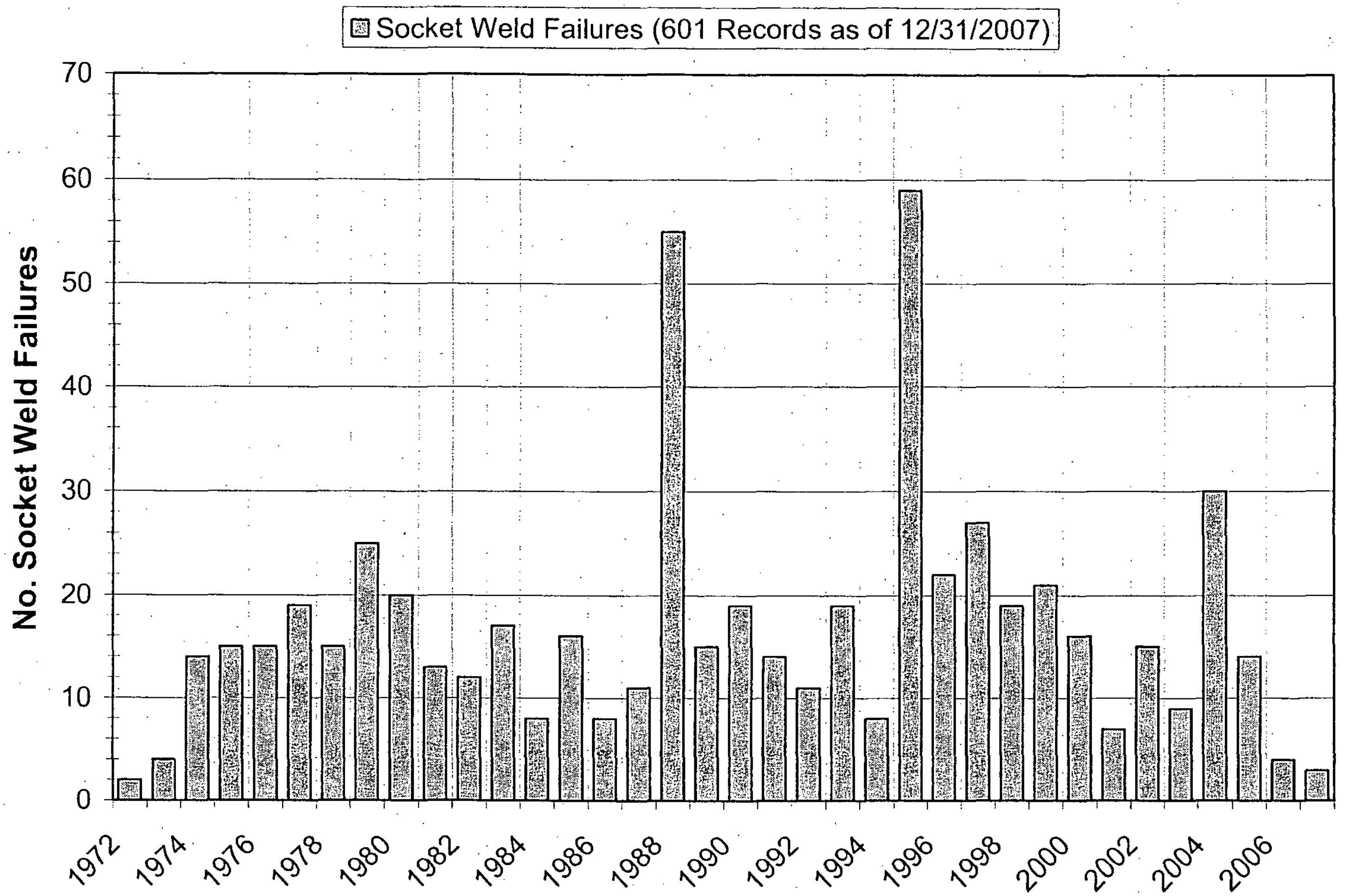


Source: PIPExp Database Data from 1970-2007

Fatigue Failure Experience

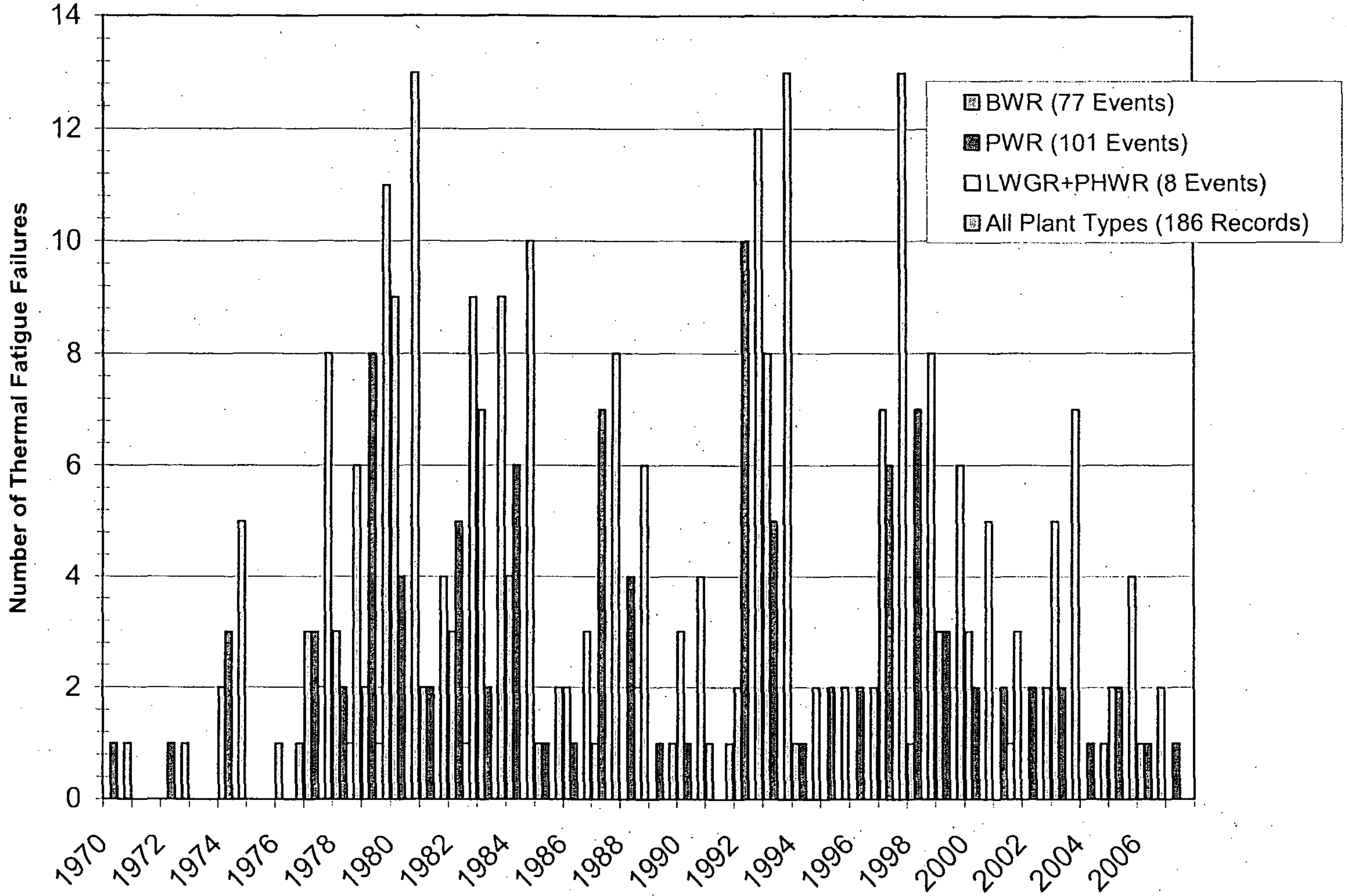
- Fatigue accounts for 21% of all reported failures in domestic operating NPPs
- Vibration Fatigue
  - 90% of the reported fatigue failures
  - Most all in small bore socket weld connections
- Thermal Fatigue
  - 2% of all reported failures
    - Thermal Stratification
    - Turbulent Penetration Effects
    - Hot/Cold Mixing
- Generally the occurrence of these failures has not significantly changed in the last 35 years

Vibration Fatigue Socket Weld Failures



Source: PIPExp Database Data from 1970-2007

Thermal Fatigue Failures

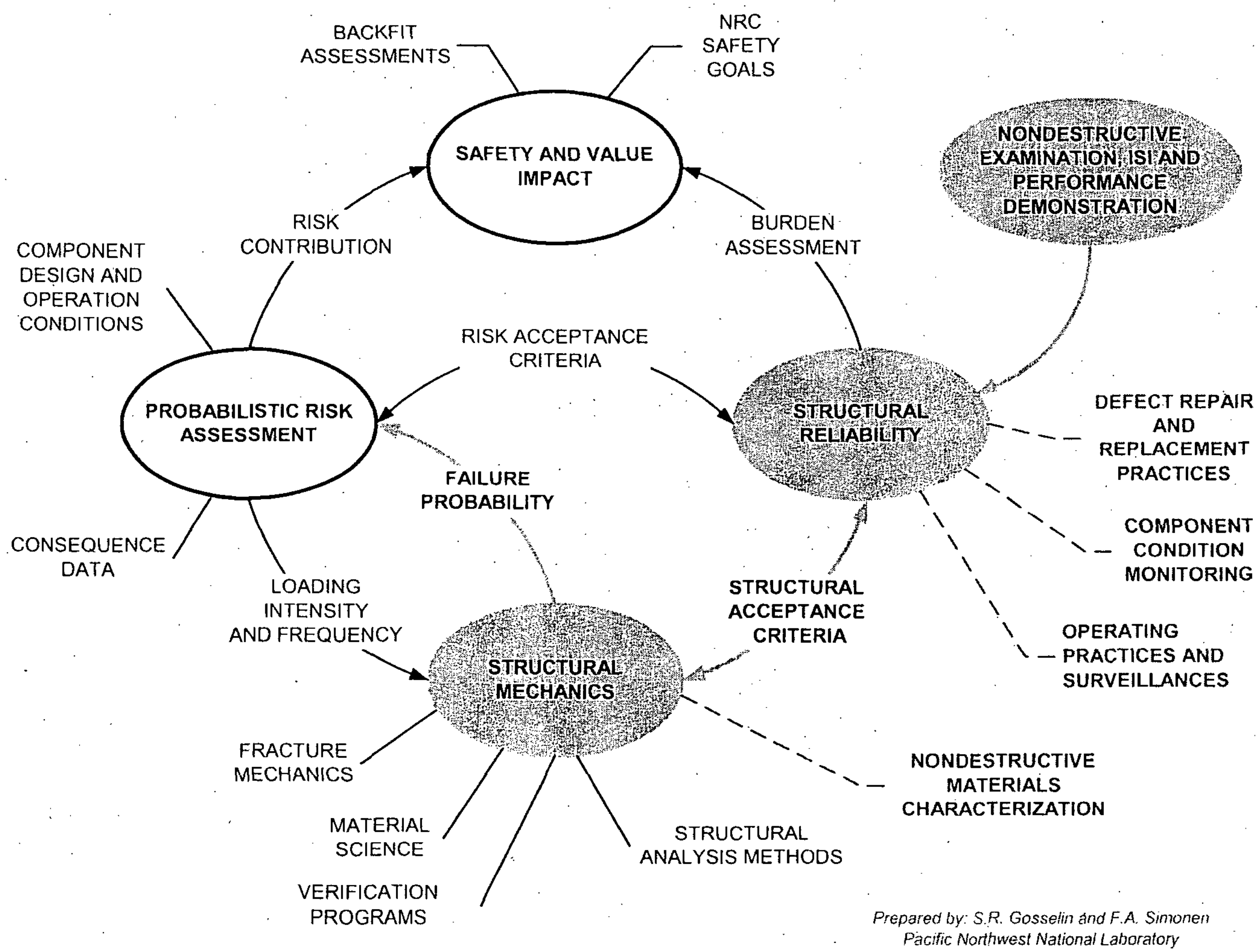


Source: PIPExp Database Data from 1970-2007

Fatigue Qualification and Serviceability

- Component design and operation will be limited to prevent **fatigue crack initiation**
- Component is designed and operated in a manner that will tolerate fatigue accumulation and crack growth without reducing the structural integrity below acceptable limits - **'damage tolerant'**
- Component design and operation will be limited so that component **failure probability/frequency** is within established component reliability goals.

Integrated Integrity Evaluation



Exhibit

NEC-Motion for Reconsideration- No. 2

SETTING THE STANDARD

Inservice Inspection (ISI) and Inservice Testing (IST)

Why test and inspect?

- Functional degradation
  - Active mechanical equipment
- Structural degradation
  - Active and passive mechanical equipment

Inservice Inspection (ISI) and Inservice Testing (IST)

Why test and inspect?

- Prevent structural failure
- Prevent fluid leakage
- Prevent radiation leakage
- Prevent loss of operability

Inservice Inspection (ISI) and Inservice Testing (IST)

Why test and inspect?

- Aging management
  - Monitor degradation
  - Maintain design margins

Active Functions

- Wear
- Corrosion
- Erosion
- Vibration
- Leakage
- Radiation damage
- Thermal aging
- Unusual or unanticipated loads

Passive Functions

- Corrosion
  - General oxidation
  - Pitting
  - Crevices
  - Microbiological
  - Flow-accelerated
  - Erosion/cavitation

ASME
SETTING THE STANDARD

Passive Functions

- Stress corrosion cracking
  - Intergranular
  - Transgranular
  - External Chloride
  - Primary Water

Passive Functions

- Fatigue
  - Mechanical
  - Thermal
  - Corrosion
- Irradiation embrittlement
- Unanticipated events
  - Water hammer
  - Pressurized thermal shock
  - Large seismic event

ASME
SETTING THE STANDARD

Detection of Degradation

How do we detect degradation?

- Establish baseline
  - As early as possible
  - Using inservice methods
  - Update after changes
- Monitor changes
  - Performance testing
  - Nondestructive examination
  - Destructive testing

Performance Testing

- Pumps
  - Vibration
  - Flow rate
  - Differential pressure
  - Bearing temperature
- Valves
  - Stroke time
  - Seat leakage for RCS or containment isolation
  - Relieving Pressure
- Snubbers
  - Range of motion
  - Lockup

Destructive Testing

- Tensile testing
- Impact testing

Nondestructive Testing and Examination

- Chemical analysis
- Volumetric examination
  - Radiographic, ultrasonic, eddy current, acoustic emission
- Surface examination
  - Liquid penetrant, magnetic particle, ultrasonic, eddy current
- Visual examination
- Leak testing

ASME
SETTING THE STANDARD

Design for ISI & IST

- How are inservice inspection and testing considered in design and construction?
- Who is responsible?

Unanticipated Problems

- Lack of accessibility for ISI
- Loss of fracture toughness
- Flow-accelerated corrosion
- Intergranular / primary water stress corrosion cracking (IGSCC/PWSCC)
- Microbiological corrosion (MIC)
- Containment vessel corrosion

Lack of Accessibility

- No automated ISI methods
- ISI needs unknown
- Degradation methods unknown
  - General corrosion
  - Thermal and mechanical fatigue
  - Neutron embrittlement
- 10 CFR 50.55a and Section III & XI revisions have been insufficient
- Designers need to solve

ASME
SETTING THE STANDARD

Loss of Fracture Toughness

- Neutron embrittlement hard to predict
- Vessels with low starting toughness
- Section III & XI revisions provide solution

Flow-accelerated Corrosion

- Unanticipated
- Designer selects materials
- Owners don't want Code or regulatory requirements
- Section XI revisions limited to analytical solutions
- Designers must specify Cr-Mo or PE pipe

Stress-corrosion Cracking

- Unanticipated
- Designer selects materials
- Significant safety issue
  - Challenges leak-before-break assumptions
- Nonlinear propagation rate
- Inadequate NDE
- 10 CFR 50.55a and Section III & XI revisions
- Owners replace or overlay
- Designers must specify resistant materials and configurations

ASME SETTING THE STANDARD

Microbiological Corrosion

- Unanticipated
- Designer selects materials
- No Code or regulatory requirements
- Designers must specify resistant materials

Containment Vessel Corrosion

- Degree unanticipated
- Design issue more than material issue
- 10 CFR 50.55a and Section XI revisions
- Designers must specify resistant materials or coatings or prevent wetting

ASME

SETTING THE STANDARD

Challenges and Directions for the Future

- Environmental fatigue effects make it more difficult to base serviceability on traditional ASME Class 1 analyses
- Synergistic effects of other mechanisms (e.g., corrosion, cast stainless steel thermal embrittlement, etc.)
- Expand application of damage tolerant and PFM methods for component fatigue qualification and fitness for continued service beyond 60 years.
  - Component fabrication and repair welds' flaw size and density distributions and uncertainties
  - Uncertainties associated with: material properties, weld residual stresses, NDE detection and flaw characterization capabilities, crack initiation, and crack growth rates
- Advanced reliability models consider all relevant design, operation and maintenance practices, surveillances, etc, so that fatigue sensitive components will continue to operate with established reliability goals

Exhibit

NEC-Motion for Reconsideration- No. 3

Cornerstone Rollup

Program: Flow Acclerated Corrosion
Plant: Vermont Yankee
Quarter: 2Q08
Last Update: 7/7/2008

Monitored Parameter	Criteria	Color	Total Quality Points	Comments
Overall Program	Green: 110 – 120 White: 85 – <110 Yellow: 75 – <85 Red: <75	White	92	Program technical aspects stable. Personnel will require training and qualification.
Program Personnel Cornerstone	This cornerstone provides an indication of whether or not we have the right personnel with the right skills in the right positions to manage the program.	Red	14	New hire for FAC started 4/14/2008. NDE Level II currently working program and will become backup. Return to Green - 2nd Qtr 2009
Program Infrastructure Cornerstone	This cornerstone provides an indication of the quality of the infrastructure in place to support the program. Infrastructure includes necessary equipment, program procedures, etc.	White	21	Timeliness issues affiliated with the corrective action program and Program updates.
Program Implementation Cornerstone	This cornerstone provides an indication of how well we execute programmatic requirements.	[illegible]	27	Several open items for update remain. No impact on outage scope definition.
Equipment / Related Plant Performance Cornerstone	This cornerstone provides an indication of the health of the components (or other performance indicators impacting plant performance) monitored by the program.	[illegible]	30	None

Rev. 0
Date: 04-25-06

Personnel Performance Cornerstone

Program: Flow Acclerated Corrosion

Plant: Vermont Yankee
Quarter: 1Q08
Last Update: 4/10/2008

Cornerstone Rollup	
Green: 26-30 cornerstone quality points	Red
White: 20-25 cornerstone quality points	
Yellow: 15-19 cornerstone quality points	
Red: <15 cornerstone quality points	

Select Cornerstone Trending	
↑	Up
↔	Stable
↓	Down

Monitored Parameter	Criteria	Result	Relative Value	Quality Points	Comments
Staff Qualification and Experience	Green – Incumbent fully qualified with 3 years or more experience within the program. White – Incumbent fully qualified. Yellow – Incumbent in partially qualified (> or = 25% complete with qualification card.) Red – No incumbent or unqualified incumbent < 25% complete with qualification card.	Red	3	0	New hire in Programs and Components started 4/14. VY is complying with the requirements in EN-DC-329 for PI Corner Stones with RED Indicators, Initiation of an Action Plan and CR for tracking this item.
Bench strength	Green – Backup fully qualified with 3 years or more experience within the program. White – Backup fully qualified. Yellow – Backup in partially qualified (> or = 25% complete with qualification card.) Red – No backup or unqualified backup < 25% complete with qualification card.	Yellow	1	1	NDE Level II in Code Programs will become backup.
Training (CHECWORKS BASIC and ADVANCED Training)	Green: Completed CHECWORKS FAC BASIC and ADVANCED Training. White - Completed CHECWORKS FAC BASIC Training and Qualification Card. Yellow - Incumbent is partically qualified (≥ 25% complete with CHECWORKS Training and Qualification Card) Red - Unqualified	Red	1	0	Incoming Engineer will require training.

Monitored Parameter	Criteria	Result	Relative Value	Quality Points	Comments
Industry Participation (Includes any within the ENS region)	Green – Committee membership, other voting White – Active participation within industry within the past year with active sharing across sites. Yellow – No active involvement over the past year but active involvement within the past two years. Red – Inactive participation.	Green	1	3	CHUG MEMBER: Attended 2008 Summer CHUG Meeting
Program Human Performance (Does not include errors in implementation)	Green – No HPEs over the past 12 months. White – 1 HPE over the past 12 months Yellow – 2-3 HPE over the past 12 months Red – 4 or more HPE over the past 12 months	Green	1	3	None
Owner Availability	Green – Supervisor determines sufficient time is available for proactive program improvements White – Supervisor determines sufficient time allotted for necessary program up keep. Yellow – Supervisor determines insufficient time allotted for long term program up keep. Red – Supervisor determines insufficient time allotted for immediate program needs.	White	2	4	New hire in Programs and Components given sufficient time for upkeep of program
Peer Interaction (Does not include PI worksheet development)	Green – 2 or more peer White – 1 peer meeting/teleconference quarterly Yellow – less than full regional participation for the meeting/teleconference within the quarter. Red – Did not participate in peer meeting/teleconference for the quarter.	Green	1	3	Meeting with Corporate Program Manager (Artie Smith @VY), CHUG Meeting, Conference Call(6/30)
			Total	**14**	

VYCRA0050740

Infrastructure Performance Cornerstone

Program: Flow Acclerated Corrosion

Plant: Vermont Yankee
Quarter: 1Q08
Last Update: 4/10/2008

Cornerstone Rollup	
Green: 26-30 cornerstone quality points	White
White: 20-25 cornerstone quality points	
Yellow: 15-19 cornerstone quality points	
Red: <15 cornerstone quality points	

Select Cornerstone Trending	
↑	Up
↔	Stable
↓	Down

Monitored Parameter	Criteria	Result	Relative Value	Quality Points	Comments
Program Infrastructure CRs (Internal) and External Findings. (External findings are defined as conditions found by independent oversight agencies resulting in A or B level CRs. Oversight agencies include QA [audits], INPO, and NRC.)	Green – (identified within the last two quarters) No A or B level CR AND No external findings AND < 4 C level CRs White – (identified within the last two quarters) No A level CR; AND No external findings; AND < 3 B level CRs; and AND < 6 total B and C level CRs Yellow – (identified within the last two quarters) No A level CRs AND Any of the following 3-4 B level CRs OR 5-15 total B or C level CRs OR 1 external finding. Red – (Any of the following within the last two quarters) Any A level CR OR 5 or more B level CRs OR 15 or more total B or C level CRs OR 2 or more external findings OR Any NRC violation.	[redacted]	2	6	None

Monitored Parameter	Criteria	Result	Relative Value	Quality Points	Comments
Long Range Plan (plan for items requiring significant resources such as outage support requirements, scheduled assessments, program updates, critical infra-structure upgrades, and scheduled component replacements.)	Green – Long range plan in place covering the next 5 years, updated within the last year and with budgetary items IDd in the long range budget. White – Long range plan in place covering the next 3 years, updated within the last year and with budgetary items IDd in the long range budget. Yellow - Foreseeable issues requiring significant resources within the 1 to 3 years not included in the long range plan. Red – Foreseeable issues requiring significant resources within the next 12 months not included in the long range plan. Yellow or Red can be upgraded once adequate plans are in place including funding in budget.	Yellow	1	1	Significant work needed as follows: Program update for verification of modeling software and transition. Small bore report for prioritizing inspections. No rescources due to other station commitments. Long Range Plan needed to provide logic for FAC program updates/upgrades due to P+C Engineer trained in FAC leaving in October 2007.
Open Action Items (Includes ALL CR-CAs, ER post-action items and LO-CAs.)	Green – No due date extensions and no items greater than 6 months old. White – No action items greater than 1 year old. Yellow – Any action item greater than 1 year old. Red – 2 or more CR-CAs and/or ER post-action items (excluding LOs action items) greater than 1 year old.	Yellow	1	1	LO-VTYLO-2003-00327 CA2 and LO-VTYLO-2003-00327 still open. Both items due in 2nd half of 2008
Document / Database Health	Green – No outstanding changes to the program documents (or databases) which impact program performance (e.g. missed commitment, surveillance past due); no outstanding changes for enhancements greater than two quarters old; and use of best-in-practice database or tracking software. White – No outstanding changes to the program documents (or databases) which potentially impact program performance. Yellow – Database compatibility issues OR any outstanding issues with the potential to impact program performance. Red – Any procedural or database issue which directly impacted program performance within the past quarter.	White	3	6	Some updating of Program Documents required, but are administrative in nature. Technical aspects of program are complete.

VYCRA0050742

Monitored Parameter	Criteria	Result	Relative Value	Quality Points	Comments
Test Equipment	Green – Best-in-practice, functional and properly calibrated equipment in the proper numbers to get the job done efficiently.	[redacted]	2	4	None
	White – Equipment functional and properly calibrated in the proper numbers to get the job done efficiently.				
	Yellow – Test Equipment Obsolescence Issues OR Test equipment failure (which did not impact scheduled or required program implementation activity) within the last quarter OR Insufficient equipment available (functional and properly calibrated) for efficient program implementation.				
	Red – Equipment unavailable to support scheduled or required program implementation activity.				
Benchmarks/Self-Assessments	Green: Benchmark or Self-Assement within the last 2 years.		1	3	Independent assessment from Jeff Horowitz (EPRI) Summer '07
	White: Benchmark or Self-Assement within the last 3 years.				
	Yellow: Benchmark or Self-Assement within the last 4 years.				
	Red: No Benchmark or Self-Assessment within the last 4 years.				
			Totals	**21**	

Implementation Performance Cornerstone

Program: Flow Acclerated Corrosion

Plant:	**Vermont Yankee**
Quarter:	**1Q08**
Last Update:	**4/10/2008**

Cornerstone Rollup		Select Cornerstone Trending	
Green: 26-30 cornerstone quality points		↑	Up
White: 20-25 cornerstone quality points		↔	Stable
Yellow: 15-19 cornerstone quality points		↓	Down
Red: <15 cornerstone quality points			

Monitored Parameter	Criteria	Result	Relative Value	Quality Points	Comments
Program Implementation CRs (Internal) and External Findings. (External findings are defined as conditions found by independent oversight agencies resulting in A or B level CRs. Oversight agencies include QA [audits], INPO, and NRC.)	Green – (identified within the last two quarters) No A or B level CR AND No external findings AND < 4 C level CRs White – (identified within the last two quarters) No A level CR; AND No external findings; AND < 3 B level CRs; and AND < 6 total B and C level CRs Yellow – (identified within the last two quarters) No A level CRs AND Any of the following 3-4 B level CRs OR 5-15 total B or C level CRs OR 1 external finding. Red – (Any of the following within the last two quarters) Any A level CR OR 5 or more B level CRs OR 15 or more total B or C level CRs OR 2 or more external findings OR Any NRC violation.	White	1	2	None identified per criterion

VYCRA0050744

Monitored Parameter	Criteria	Result	Relative Value	Quality Points	Comments
Internally Identified Implementation Issues – Other than CRs (Self revealing issues, self assessments' benchmarking, Operating Experience including	Green: None White: Identified issue with action resolved. Yellow: Identified issue less than 1 year old. Red: Any identified issue greater than 1 year old.	Red	1	0	Programmatic updates need to be completed.
Outage Performance Note: Indicator should remain the color until corrective actions are taken to preclude recurrence during the	Green: Met original scope and goals (duration, White: Less than 100% greater than 90% Yellow: Less than 90% greater than 80% Red: Less than 80%	[illegible]	1	3	None
On-line Performance	Green: Met original scope and goals (duration, White: Less than 100% greater than 90% Yellow: Less than 90% greater than 80% Red: Less than 80%	[illegible]	1	3	None
PM's/Surveillance Tasks (window stays the color until the deferred PM's are completed)	Green: No deferrals for the quarter White: Greater than 95% complete for the quarter Yellow: Greater than 90% complete for the quarter Red: Less than 90% complete for the quarter	[illegible]	1	3	None
Other Identified Concerns or Issues (Only captures program concerns that do not fall under other PIs)	Green: No concerns / issues White: Any non-significant concern/issue with action plan Yellow: Any significant concern or issue with action plan or any non significant issue without action plan Red: Any significant issue/concern without action plan	White	1	2	New hire started 4/14/2008.

VYCRA0050745

Monitored Parameter	Criteria	Result	Relative Value	Quality Points	Comments
Implementation resources (i.e. number of qualified personnel)	Green: No identified resource concern White: Identified concern with action plan Yellow: Identified concern without action plan Red: Significant concern without action plan	White	1	2	None
Piping Replacements (Unplanned during cycle or outage)	Green: 0 unplanned pipe or component replacements due to current outage findings. White: 1 unplanned pipe or component replacement due to current outage finding. Yellow: > 1 < 2 unplanned pipe or component replacements due to current outage finding Red - > 2 unplanned pipe or component replacements due to current outage finding. (Note: Color can be up-graded once corrective actions to piping are completed and Program has been corrected to prevent recurrence; i.e., additional exams or exam frequency specified)	[redacted]	2	6	None
Operating Experience	Green: 1 OR less items generated by plant OE department that has not been reviewed. White: 2 items generated by plant OE department that has not been reviewed. Yellow: 3 items generated by plant OE department that has not been reviewed. Red: > 4 items generated by plant OE department that has not been reviewed.	[redacted]	1	3	None
Outage Scope Increase (Unplanned) (PWRs include online inspections in current cycle)	Green: < 10% increase in inspection scope due to inspection findings. White: 10% to < 12% increase in inspection scope due to inspection findings. Yellow: ≥ 13% increase in inspection scope due to inspection findings. Red: > 15% increase in inspection scope due to inspection findings.	[redacted]	1	3	None
			Total	27	

Equipment / Related Plant Performance Cornerstone

Program: Flow Acclerated Corrosion

Plant: Vermont Yankee
Quarter: 1Q08
Last Update: 4/10/2008

Cornerstone Rollup	
Green: 26-30 cornerstone quality points	[redacted]
White: 20-25 cornerstone quality points	
Yellow: 15-19 cornerstone quality points	
Red: <15 cornerstone quality points	

Select Cornerstone Trending	
↑	Up
↔	Stable
↓	Down

Monitored Parameter	Criteria	Result	Relative Value	Quality Points	Comments
Generation Health	Green - No Transients or power reduction resulting from a program issue	[redacted]	2	6	None
	White - No Transients or power reduction resulting from a program issue or component on a quarterly basis				
	Yellow - A " near miss" , transient or a power reduction < 1000 mwhr/qt as a result of a program issue or component				
	Red - A plant trip or significant power reduction > 1000 mwhr/qtr as a result of a program issue of component				
Large Bore Failures (Based on Cycle of operation)	Green: No Large Bore failures in load	[redacted]	4	12	None
	Red: ≥ 1 Large Bore failure resulting in load reduction or safety issues.				
	Note: color can be up-graded once corrective actions to piping are completed and the Program has been corrected to prevent recurrence; i.e.., additional exams or exam frequency specified)				

Monitored Parameter	Criteria	Result	Relative Value	Quality Points	Comments
Small Bore Failures (Based on Cycle of operation)	Green: ≤ 1 Small Bore FAC related failure resulting in a load reduction		2	6	None
	White: < 3 Small Bore FAC related failures resulting in a load reduction or safety issue.				
	Red: > 3 Small Bore FAC related failures resulting in a load reduction or safety issue.				
	(Note: color can be up-graded once corrective actions to piping are completed and the Program has been corrected to prevent recurrence; i.e., additional exams or exam frequency specified)				
Stress Analysis (Cycle of operation including outage)	Green: 1 to 3 detailed stress analysis required.		2	6	None
	White: 3 to 5 detailed stress analysis required.				
	Red: ≥ 6 detailed stress analysis required.				
			Totals	30	

New England Coalition

VT . NH . ME . MA . RI . CT . NY .

POST OFFICE BOX 545, BRATTLEBORO, VERMONT o5302

January 9, 2009

Office of the Secretary
Rulemaking and Adjudications Staff
ATTN: Nancy Greathead
Mail Stop: O-16C1
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Re; New England Coalition's Motion for Reconsideration of a Partial Initial Decision in Docket No. 50-271-LR – ASLBP-08-25 ENTERGY NUCLEAR VERMONT YANKEE, L.L.C., and ENTERGY NUCLEAR OPERATIONS, INC. (Vermont Yankee Nuclear Power Station).

Dear Ms. Greathead,
Dear Rulemaking and Adjudications Staff,

Thank you for providing the opportunity to clarify and correct filing and docketing of the above captioned NEC motion.

You have returned a paper copy of the filing to New England Coalition for NEC's review in order to be certain of what NEC intended to file.

Upon review, we find that the hardcopy filing of the Motion for Reconsideration is essentially correct, except for the inadvertent inclusion of an extra and spurious first page to the motion itself. This page is attached to yellow sheet and returned labeled, NOT INTENTIONALLY INCLUDED, PLEASE DISCARD.

In addition the Certificate of Service contained an error in naming the filing. This has been corrected and a corrected certificate is enclosed.

Original's of NEC's Exhibits A,B, C, and D were returned and found to be correct and are now included. Print outs of E-mail copies of these Exhibits were also returned. They will be resent in a comprehensive resend of the e-mail filing that includes and incorporates the errata e-mail filing of December 19th. You will find that the formatted and marking format of these exhibits will be slightly different than the enclosed hardcopy versions as we do not have the computer capacity to mark image files.

If there are any further questions, please do not hesitate to contact me.

Thank you for your kind assistance in making this filing,

Raymond Shadis
Pro se Representative
New England Coalition
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Edgecomb, Maine 04556
207-882-7801
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From: Raymond Shadis [mailto:shadis@prexar.com]
Sent: Friday, December 19, 2008 12:09 PM
To: 'ask2@nrc.gov'; 'whrcville@embarqmail.com'; 'OCAAmail@nrc.gov'; 'rew@nrc.gov'; 'hearingdocket@nrc.gov'; 'sarah.hofmann@state.vt.us'; 'lbs3@nrc.gov'; 'mcb1@nrc.gov'; 'susan.uttal@nrc.gov'; 'jessica.bielecki@nrc.gov'; 'aroisman@nationallegalscholars.com'; 'zachary.kahn@nrc.gov'; 'Peter.roth@doj.nh.gov'; 'david.lewis@pillsburylaw.com'; 'matias.travieso-diaz@pillsbury.com'; 'Matthew.Brock@state.ma.us'; Travieso-Diaz, Matias F.; Rudolf H. Hausler; Ulrich Witte
Subject: Errata - Wednesday's NEC . Filing in Docket No. 50-271-LR
Importance: High
Attachments:

Witte Notarized Affidavit.pdf

Ulrich K Witte MFR Memorandum FIN.pdf

Final exhibits Witte Affidavit NEC MFR Ex

Final exhibits Witte Affidavit NEC MFR Ex

Final exhibits Witte Affidavit NEC MFR Ex

2008 12-17 MOTION FOR RECONSIDERAT

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of

ENTERGY NUCLEAR VERMONT YANKEE, L.L.C., and ENTERGY NUCLEAR)

December 19, 2008

Docket No. 50-271-LR

ASLBP-08-25

OPERATIONS, INC.

(Vermont Yankee Nuclear Power Station)

ERRATA

NEW ENGLAND COALITION'S MOTION FOR RECONSIDERATION OF THE LICENSING BOARD'S PARTIAL INITIAL DECISION

New England Coalition, Inc ("NEC") has today been made aware that its electronic (e-mail) filing of its Motion for Reconsideration of the Licensing Board's Partial Initial Decision in the above caption matter, transmitted December 17, 2008, contained an omission and several errors (errata). Examination of printed copies of NEC's hardcopy filing reveals that NEC's hardcopy filing did not likely contain any errors.
The omission and errors in the e-mail filing are a result of inadvertent mislabeling of computer files and mistakenly attaching draft instead of final documents.
There are few if any substantive differences, however NEC requests that the Board and the parties substitute the attached electronic files for those e-mailed as attachments on December 17th.
NEC regrets any inconvenience or confusion the e-mail filing error may have caused the Board or the parties.
Attached are the correct intended documents in electronic format (Pdf and MsWord):

(1) New England Coalition's Motion for Reconsideration, the motion itself.
(2) Memorandum of Ulrich Witte
(3) Three Exhibits titled, Exhibit ,NEC –Motion for Reconsideration-No.1, Exhibit ,NEC –Motion for Reconsideration-No.2, and Exhibit ,NEC –Motion for Reconsideration-No.3 ,presented in three attachments (In the e-mail of December 17th, the three exhibits were bundled for in-house review purposes in a single file; Exhibit...No.2 had an extraneous page ("page 9").
(4) Notarized Affidavit of Ulrich Witte.

Said electronic files are dispatched via e-mail at 12 noon, today, December 19, 2008.

Respectfully submitted,

Raymond Shadis
Pro Se representative
New England Coalition
Post Office Box 98
Edgecomb, Maine 04556
207-882-7801

Shadis2prexar.com

New England Coalition

VT . NH . ME . MA . RI . CT . NY

POST OFFICE BOX 545, BRATTLEBORO, VERMONT o5302

December 17, 2008

Office of the Secretary
Attn: Rulemaking and Adjudications Staff
Mail Stop: O-16C1
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

RE: Docket No. 50-271-LR, ASLBP No. 06-849-03-LR, Vermont Yankee Nuclear Power Station

Dear Rulemaking and Adjudications Staff,

Please find enclosed for filing before the Atomic Safety and Licensing Board in the above captioned proceeding:
NEW ENGLAND COALITION, INC.'S MOTION FOR RECONSIDERATION

Thank you for your kind attention,

/RS Raymond Shadis

for New England Coalition, Inc.

Raymond Shadis
Pro Se Representative
Post Office Box 98
Edgecomb, Maine 04556

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

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

CERTIFICATE OF SERVICE

I, Raymond Shadis, hereby certify that copies of NEW ENGLAND COALITION, INC.'S (NEC) MOTION FOR RECONSIDERATION in the above-captioned proceeding were served on the persons listed below, by U.S. Mail, first class, postage prepaid; and, where indicated by an e-mail address below, by electronic mail, on the 17th of December, 2008.

Administrative Judge
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U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
E-mail: ask2@nrc.gov

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Administrative Judge
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Office of the Secretary

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