RAS H-414

DOCKETED
USNRC

April 27, 2009 (8:00am)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

UNITED STATES
NUCLEAR REGULATORY COMMISSION

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

NEW ENGLAND COALITION, INC.'S MOTION FOR LEAVE TO FILE
A TIMELY NEW CONTENTION AND MOTION TO HOLD IN ABEYANCE
ACTION ON THIS PROPOSED CONTENTION UNTIL
ISSUANCE OF NRC STAFF SUPPLEMENTAL SAFETY EVALUATION REPORT

I. INTRODUCTION AND MOTIONS

The New England Coalition, Inc. (NEC) moves, pursuant to 10 C.F.R. § 2.309(f)(2) ;the Atomic Safety and Licensing Board Panel (Board) Partial Initial Decision of November 24, 2008, and the Board's Order of March 9, 2009 (Clarifying the Deadline for Filing New or Amended Contentions), for leave to file a timely new contention addressing Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc.'s ("Entergy") recent reanalysis of environmentally assisted metal fatigue for Recirculation Outlet (RO) and Core Spray (CS) nozzles

In addition NEC respectfully moves the Board to hold in abeyance any action on the motion for leave to file a new or amended contention until such time as the Board and the parties have had an opportunity to review NRC's Supplemental Safety Evaluation report and Audit Summary related to Confirmatory Analyses For The Core Spray And Reactor Recirculation Outlet Nozzles At Vermont Yankee Nuclear Power Station.

II. PROPOSED NEW CONTENTION

Specifically, NEC contends that Entergy has not properly recalculated the Core Spray and Recirculation Outlet nozzle CUFens such that they demonstrate that these important components will not fail during the period of extended operation (i.e., that the calculations produce a value less than unity). Such recalculations involve complex scientific and technical judgments

Template Secy-041

DS-03

The complex scientific and technical judgments employed in Entergy's recalculation of environmentally assisted metal fatigue for Recirculation Outlet (RO) and Core Spray (CS) nozzles, as filed by e-mail on March 10, 2009 and in hardcopy on March 11, 2009, are technically and factually flawed and do not conform to ASME, NRC, or National Laboratory guidance, nor do they fully conform to established engineering practice, or the rules of applied physics. As such Entergy's reanalysis of these pressure boundary components cannot be relied upon for adequate assurance of public health and safety.

III. NEC'S PROPOSED NEW CONTENTION SATISFIES REQUIREMENTS OF 10 C.F.R. § 2.309(F(2)

NEC's proposed new contention is based on documents or other information available at the time of filing; precisely NEC's motion is based upon Entergy's most recent reanalysis of its RO and CS nozzles.

The information upon which the proposed new contention is based was not previously available. The proposed contention is based upon analyses and calculations that were not available until Entergy filed them by e-mail on March 10, 2009 and in hardcopy on March 11, 2009.

The information upon which the proposed new contention is based is materially different than information previously available. Entergy's new analyses of environmentally assisted metal fatigue are materially different from Entergy's previous analyses of this phenomenon reported in Entergy's License Renewal Application Amendments and reviewed in the FSER. Entergy employed the same method as it did in its confirmatory analysis of the feedwater nozzle, however it necessarily exercised its engineering discretion in selecting various input values for geometrically and materially dissimilar components, and produced different results. *See,* Attachment A, Declaration of Dr. Joram Hopenfeld Also, See below, a more complete discussion of applicability of the terms of the Partial Initial Decision constraining permissible subject areas of any new proposed contention.

The proposed new contention has been submitted in a timely fashion based on the availability of the subsequent information. Entergy produced its final reports of this reanalysis and produced them to the Board and to the Parties by E-mail on March 10, 2009 and in hardcopy on March 11, 2009. The Board's

Partial Initial Decision of November 24, 2008 and its Clarifying Order of March 9, 2009 allow 45 days from the time Entergy files its final analysis and calculations of record for the filing of any new contentions that may arise from the parties' review of the final analyses and calculations of record. NEC's Motion for Leave to File a New Contention is therefore, timely.

IV. NEC's PROPOSED NEW CONTENTION SATISFIES 10 C.F.R. § 2.309 (General) ADMISSIBILTY CRITERIA

(A) Statement of Issue of Law or Fact to be Raised, and Brief Explanation of Basis. 10 C.F.R. §§ 2.309(f)(i), 2.309(f)(ii). NEC's Contention 2 (metal fatigue), now held in abeyance, is that critical reactor components may fail due to environmentally assisted metal fatigue during the period of extended operation, and that Entergy has not proposed an adequate aging management plan (AMP)addressing this issue as required pursuant to 10 C.F.R. § 54.21.

Over the course of this proceeding, Entergy has, in lieu of a qualified AMP, submitted two Time Limited Aging Analyses, first employing Green's Function, and the second, a confirmatory analysis, selecting a stress sample (a Feedwater Nozzle) purported to be bounding. NRC Staff accepted a commitment to do confirmatory analysis on two additional components, the RO and CS nozzles prior to expiration of the current license.

Following a technical hearing on the issues, the Board found in its Initial Partial Decision, on November 24, 2009, that with respect to a representative component, a reactor feedwater nozzle, have now been litigated and resolved (decided) in favor of the licensee, and

> ...that Entergy has shown, by a preponderance of the evidence, that its CUFen analyses comply with 10 C.F.R. §§ 54.21(c)(1) and 54.29(a) in all respects, except one. The exception is the CUFen Reanalyses for the core spray nozzle and the reactor recirculation outlet nozzle. The defect in the core spray and reactor recirculation nozzle CUFens is the use of a simplified Green's function methodology that renders them inconsistent with the ASME Code, unable to be validated, and liable to underestimate the nature and extent of metal fatigue at the VYNPS. The current core spray and reactor recirculation nozzles CUFen calculations cannot be the analysis-of-record for these components. In addition, the Board finds that Entergy has failed to show that the Confirmatory CUFen Analysis for the feedwater nozzle necessarily bounds the metal fatigue analyses for the core spray and reactor recirculation nozzles during the period of extended operation.

> The Board also concludes that, as a legal and technical matter, the license renewal cannot be authorized or issued until Entergy either (1) properly recalculates the CS and RR outlet nozzle CUFens such that they demonstrate that these important components will not fail during the PEO (i.e., that the calculations produce a value less than unity), or (2) submits an AMP that demonstrates that aging of these components will be adequately managed during the PEO. Such recalculations (or an adequate AMP) cannot be consigned to some post-hearing activity, because they are a condition precedent to the license, involve complex scientific and technical judgments and discretion, and are not merely ministerial. Thus, the NRC Staff's proposed license condition 4 and Entergy's Commitment 27 do not suffice. Such recalculations (or an adequate AMP) are a pre-requisite to issuance of the license renewal.
>
> The consequence is that the license renewal may be issued only if the above preconditions are met, i.e., our authorization of any license renewal is contingent on these preconditions. Assuming Entergy still wishes to pursue this license renewal, it must (1) recalculate the CUFen analyses for the CS and RR outlet nozzles, in accordance with the ASME Code, NUREG 6583 and 5704, and all other regulatory guidance, (2) resubmit these results to the NRC Staff and serve them on the other parties herein, and (3) either demonstrate that the TLAAs are less than unity or submit an adequate AMP for these components. At that point we presume (but do not and cannot order) that the NRC Staff will evaluate Entergy's submissions. Presumably NEC will do the same

Entergy has now completed a final reanalysis of the impact of environmentally assisted metal fatigue, the results of which purportedly indicate that the RO and CS reactor nozzles are not in fact subject to fatigue failure during the period of extended operation..

However, NEC now contends, as explained in some detail in the attached Declaration of Dr. Joram Hopenfeld (Attached as Exhibit A) that Entergy's submitted recalculations do, "involve complex scientific and technical judgments and discretion, and are not merely ministerial," but are not performed in, "in accordance with the ASME Code, NUREG 6583 and 5704, and all other regulatory guidance. " Thus, Entergy has not, by this flawed reanalysis, demonstrated that the reactor components assessed will not fail due to metal fatigue during the period of extended operation. Nor has it complied with the requirements set forth in Board's Partial Initial Decision. Nor has Entergy credibly demonstrated that its new calculations and analyses for the CS and RO nozzles are consistent with the intent of 10 C.F.R. § 54.21.

B. Scope of the Proceeding and Materiality. 10 C.F.R. §§ 2.309(f)(iii), 2.309(f)(iv). This contention addresses Entergy's confirmatory recalculation and reanalyses of CUFen for the CS and RO nozzles, which Entergy has provided in response to the Board's Partial Initial Decision (relevant excerpt above), and pursuant to 10 C.F.R. § 54.21(c). These are issues within the scope of this proceeding, and material to findings the NRC must make in this matter. *See, 10* C.F.R. § 54.4; *Duke Energy Corp. (McGuire*

Nuclear Station, • Units 1 and 2; Catawba Nuclear Station, Units 1, 2 and 3), 56 NRC 358, 363-64 (2002).

C. Expert or Factual Support. 10 C.F.R. § 2.309(f)(v). This contention is supported by the attached Declaration (NEC Exhibit A) of NEC's expert witness, Dr. Joram Hopenfeld.

In summary Dr. Joram Hopenfeld's Declaration provides a technical critique of Entergy's fatigue reanalysis of the CS and RO reactor nozzles examining the complex scientific and technical judgments and exercised discretion in Entergy's submitted recalculations and finding that they are not performed in, "in accordance with the ASME Code, NUREG 6583 and 5704, and all other regulatory guidance. "

Dr. Hopenfeld points to and explains a selection of Entergy's errors in assumptions, inputs, and the assignment of values going into the calculations, which lead to erroneous calculation results and non-conservative conclusions. Among the errors, to which Dr. Hopenfeld points, are both general errors in engineering principles and practice; and component (RO and CS nozzle) specific errors.

Dr. Hopenfeld is a mechanical engineer with a doctorate in engineering. He has 45 years of professional experience in the fields of instrumentation, design, project management, and nuclear safety, including 18 years in the employ of the U.S. Nuclear Regulatory Commission (NRC). His curriculum vitae was previously filed in this proceeding as an attachment to his declaration in support of NEC's Petition to Intervene.

Dr. Hopenfeld is a "prime source" expert, having assisted in developing and confirming, over his 45 year career with the Atomic Energy Commission, industry, and the U.S. Nuclear regulatory Commission, many of the engineering principles and disciplines; and scientific applications that must be considered with respect to metal fatigue in the subject License Renewal Application.

Dr. Hopenfeld's professional opinion on the matters at hand should be given weight accordingly.

In addition to evidence drawn directly from Entergy's recalculations, Dr. Hopenfeld provides several attachments , including industry comments, and excerpts from various texts and manuals, in support of his declaration

D. Genuine Dispute of Material Law or Fact. 10 C.F.R. § 2.309(f)(vi). The attached Declaration of Dr. Joram Hopenfeld includes ample information to establish a genuine dispute with the Applicant concerning the validity of the CUFen Recirculation Outlet Nozzle and Core Spray Nozzle Reanalysis. NEC is required to make only "a minimal showing that the material facts are in dispute, thereby demonstrating that an inquiry in depth is appropriate." *In Gulf State Utilities Co.,* 40 NRC 43, 51 (1994).

V. ISSUES RAISED IN NEC'S PROPOSED NEW CONTENTION ARE READILY DISTINGUISIBLE FROM ISSUES PREVIOUSLY LITIGATED.

The Partial Initial Decision states,

> If the CUFen analyses are (1) done in accordance with the above stated guidance and the basic approach used in the Confirmatory CUFen Analysis for the FW nozzle, (2) contain no significantly different scientific or technical judgments, and (3) demonstrate values less than unity, then this adjudicatory proceeding terminates. If not, NEC may file a new or amended contention challenging the adequacy of the CUFen calculation.[95]
> AND
>
> [95] NEC may not, however, use any such challenge as an opportunity to rehash or renew any technical challenges that have already been raised and resolved in this proceeding (e.g., dissolved oxygen, outdated equations, etc.), but rather must specifically state how the new analyses are not consistent with the legal requirement and the calculations performed for the feedwater nozzle.

Supported by the Declaration of Dr. Hopenfeld NEC asserts that significantly different scientific or technical judgments were made regarding the geometries, flow characteristics, and material values of the CS and RO nozzles and that these judgments were erroneous and/or non-conservative.

NEC does not seek to "rehash technical challenges that have already been resolved in this proceeding." Dr. Hopenfeld states specifically how new analyses are not consistent with the legal requirement, the feedwater (FW) nozzle calculations, and the guidance cited in the Board's Partial Initial Decision.

Dr. Hopenfeld's Declaration does examine a few concerns that could potentially be construed as outside of the Board's strictures, that is, Declaration does contain some discussion of the necessity of considering local dissolved oxygen (DO) concentrations during transients, and the pipe length, expressed number of diameters, that is required to develop full flow. Both of these issues were roundly discussed in the hearings with respect to the feedwater nozzle analysis.

However, the geometries of the RO and CS nozzles are quite different than the FW nozzle and, as Dr. Hopenfeld explains, require consideration of DO and distance to full flow from perspectives individual distinguishable for the RO and CS nozzles.

The DO and flow discussions are not a major part of NEC's motion. However, as consideration of them is an integral part of the "complex scientific and technical judgments and discretion" and they are, in this case, component specific and distinguishable from FW consideration. Therefore, NEC's Motion should in no way be construed as attempting to "rehash" previously litigated issues.

VI. MOTION TO HOLD IN ABEYANCE ACTION ON THIS PROPOSED CONTENTION UNTIL ISSUANCE OF NRC STAFF SUPPLEMENTAL SAFETY EVALUATION REPORT

Discussion - On February 27th, NEC received a copy of an NRC memorandum (not served in this ASLB Docket) making public NRC's Audit Plan For Review of Confirmatory Analyses For The Core Spray And Reactor Recirculation Outlet Nozzles At Vermont Yankee Nuclear Power Station (February 17, 2009). This memo states, "An audit summary will be placed in the docket and the results of the audit will be documented in a Supplemental Safety Evaluation report related to VY LRA. Both documents are planned to be issued by the end of April 2009."

If NRC performs on schedule, its audit summary and SSER will be issued within the response and reply period for this filing and no delay need be incurred. If it does not perform on schedule it is reasonable to anticipate that little extra time will be needed.

NEC respectfully submits that it is reasonable to anticipate that the NRC Staff's audit summary and SSER will be helpful in both building a record in this docket and helpful to the Board and the Parties in evaluating the merits of NEC's proposed contention. Indeed, the Commission has opined that staff review of an application is a vital aid (at the Commission level) in reaching an informed judgment on the need for a hearing in the public interest.

Carolina Power & Light Co. (Shearon Harris Nuclear Power Plant, Units 1, 2, 3 & 4), ALAB-581, 11 NRC 233,235 (1980), modified, CLI-80-12, 11 NRC 514 (1980).

In addition, the Commission sets as policy/guidance that any evidentiary hearing should not commence before completion of the staff's Safety Evaluation Report (SER) or Final Environmental Statement (FES) regarding an application, unless the presiding officer finds that beginning earlier, e.g., by starting the hearing with respect to safety Issues prior to Issuance of the SER, will indeed expedite the proceeding, taking into account the effect of going forward on the staff's ability to complete Its evaluations In a timely manner.[1]

Similar weight is accorded in an operating license proceeding, where it has been held that summary disposition on safety issues should not be considered or granted until after issuance of the Staff's Safety Evaluation Report . Duke Power Co. (William B. McGuire Nuclear Station, Units 1 & 2), LBP-77-20, 5 NRC 680, 681 (1977).

While these policies and decisions do not specifically address the situation of a pending SSER, they do articulate clearly the Commission's dependence and the ASLB's reliance on the Staff to assist in informing decisions regarding what should or should not go to hearing.

Indeed, had the Staff not elected to accept commitments in place of the confirmatory analyses that are now the subject of NEC's proposed contention, the material in the awaited NRC Staff's SSER would have been in the SER.

Further, while it may be something of a novel idea, intervenors also depend at some level on the NRC Staff's investigations and evaluations in bringing to light considerations otherwise, for lack of discovery, access, and resources, out of reach.[2]

---

[1] STATEMENT OF POLICY ON CONDUCT OF ADJUDICATORY PROCEEDINGS CU-98-12, 48 NRC 18 (1998)[63 Fed. Reg. 41872 (Aug. 5, 1998)]

[2] It is appropriate to require the Staff to release segregable facts on which decisions have been made, even if those facts are contained in predecisional documents. (Vogtle Electric Generating Plant, Units 1 and 2), LBP-94-31, 40 NRC at 142. (1994

VII. NEC HAS CONSULTED OTHER PARTIES

Pursuant to 10 C.F.R. § 2.323(b), NEC has consulted or attempted to consult with all parties to this proceeding via e-mail concerning the Motion for Leave to File a New Contention. Entergy The State of Vermont does not object to the filing of the Motion for Leave to Intervene .Entergy and NRC will respond to its content after reviewing NEC's pleading for consistency with NRC regulation and the ASLBP's Orders of November 24, 2008 and March XX, 2009.. Entergy avows that it will in any case, oppose any such motion. The State of New Hampshire did not respond to inquiries. The State of Massachusetts did not respond to inquiries.

Likewise, albeit, not until the day of filing, NEC has consulted or attempted to consult with all parties to this proceeding via e-mail concerning the Motion to Hold in Abeyance Action on the Proposed Contention. Entergy and NRC Staff will oppose. The State of New Hampshire, the State of Vermont, and the State of Massachusetts did not respond to inquires.

VIII. CONCLUSION

For all of the good reasons above, NEC respectfully requests that the Board grant New England Coalition, Inc.'s Motion for Leave to File A Timely New Contention; admit the Contention, and grant NEC's Motion To Hold In Abeyance Action On This Proposed Contention Until Issuance Of NRC Staff's Supplemental Safety Evaluation Report

Respectfully Submitted,

This 24th Day of April, 2009.

R/S

Raymond Shadis

Pro se representative
New England Coalition
Post Office Box 98,
Edgecomb, Maine 04556
207-882-7801
shadis@prexar.com

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 MOTION FOR LEAVE TO FILE A TIMELY NEW CONTENTION AND MOTION TO HOLD IN ABEYANCE ACTION ON THIS PROPOSED CONTENTION UNTIL ISSUANCE OF NRC STAFF SUPPLEMENTAL SAFETY EVALUATION REPORT 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 24th of April, 2009.

Administrative Judge
Alex S. Karlin, Esq., Chair
Atomic Safety and Licensing Board
Mail Stop T-3 F23
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
E-mail: ask2@nrc.gov

Administrative Judge
William H. Reed
1819 Edgewood Lane
Charlottesville, VA 22902
E-mail: whrcville@embarqmail.com

Office of Commission Appellate Adjudication
Mail Stop: O-16C1
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
E-mail: OCAAmail@nrc.gov

Administrative Judge
Dr. Richard E. Wardwell
Atomic Safety and Licensing Board Panel
Mail Stop T-3 F23
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
E-mail: rew@nrc.gov

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

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

Lloyd B. Subin, Esq.
Mary C. Baty, Esq.
Susan L. Uttal, Esq.
Jessica A. Bielecki, Esq.
Office of the General Counsel
Mail Stop O-15 D21
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
E-mail: lbs3@nrc.gov; mcb1@nrc.gov; susan.uttal@nrc.gov; jessica.bielecki@nrc.gov

Anthony Z. Roisman, Esq.
National Legal Scholars Law Firm
84 East Thetford Road
Lyme, NH 03768
E-mail: aroisman@nationallegalscholars.com
Zachary Kahn
Atomic Safety and Licensing Board Panel
Mail Stop T-3 F23
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
E-mail: zachary.kahn@nrc.gov

Peter C. L. Roth, Esq.
Office of the Attorney General
33 Capitol Street
Concord, NH 03301
E-mail: Peter.roth@doj.nh.gov

David R. Lewis, Esq.
Matias F. Travieso-Diaz
Pillsbury Winthrop Shaw Pittman LLP
2300 N Street NW
Washington, DC 20037-1128
E-mail: david.lewis@pillsburylaw.com
matias.travieso-diaz@pillsburylaw.com

Matthew Brock
Assistant Attorney General
Environmental Protection Division
Office of the Attorney General
One Ashburton Place, 18th Floor
Boston, MA 02108
E-mail: Matthew.Brock@state.ma.us

by: _R/S_______________

Raymond Shadis
Pro se Representative
New England Coalition
Post Office Box 98
Edgecomb, Maine 04556
207-882-7801
shadis@prexar.com

EXHIBIT A

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the matter of	)	
	)	
ENTERGY NUCLEAR VERMONT YANKEE, LLC And ENTERGY NUCLEAR OPERATIONS, INC.	)	Docket No. 50-271-R ASLB No. 06-849-LR
	)	
Vermont Yankee Nuclear Power Station	)	

DECLARATION OF DR. JORAM HOPENFELD
IN SUPPORT OF NEW ENGLAND COALITION'S MOTION
TO FILE A TIMELY NEW OR AMENDED CONTENTION
ON ENTERGY'S FATIGUE REANALYSIS

Q1. Please state your name.

A1. My name is Joram Hopenfeld.

Q2. Have you previously provided testimony in this proceeding?

A2. Yes. New England Coalition, Inc. has retained me as an expert witness. I have provided numerous declarations and testimony in support of New England Coalition, Inc.'s (NEC) contentions throughout this (above captioned) proceeding. I am a mechanical engineer and hold a doctorate in mechanical engineering. My curriculum vitae was attached to my first declaration in support NEC's Petition to Intervene, filed May 26, 2006.

Q2. What is the purpose of your Declaration?

A2. The purpose of my Declaration is to provide technical information in support of NEC's Motion for Leave to File a Timely New or Amended Contention on Entergy's Fatigue Reanalysis.

Q3. Have you reviewed the confirmatory environmentally assisted fatigue (CUF_{en}) analyses on the Core Spray (CS) nozzle and the reactor pressure vessel recirculation outlet (RO) nozzle at the Vermont Yankee Nuclear Power Station, which were performed by

Entergy and submitted on March 10, 2009 in response to provisions set forth in the Atomic Safety and Licensing Board's Partial Initial Decision (Ruling on Contentions 2A, 2B, 3, and 4) of Nov. 24, 2008 ?

A3. Yes. I have reviewed Entergy's analysis as described above.

Q4. What were the major findings of the analyses and do you agree with these findings?

A4. The major findings of the analyses were that the CUFen for both of the above nozzles was less than unity. I do not agree with these conclusions because the analyses were flawed and not conservative. Consequently the analysis does not meet the NRC/ASME guidelines of how the fatigue analysis for plant life extension should be conducted.

Q5. Please explain

A5. Section X.M1 of NUREG 1801 specifies that a license renewal applicant can comply with 10 C.F.R. § 54.21(c)(1)(iii) by applying environmental correction factors to the existing ASME Code fatigue analyses. ASME Code, Subsection NB, Sub article NB-3200 methodology, however, is not prescriptive. The NRC has not published the various approaches that the analyst may take in calculating the CUFens. Consequently all analyses of metal fatigue on reactor internals are heavily dependent on the professional judgment and engineering discretion of the analyst. I am not alone in stressing the importance of the analyst's judgment. When invited by U.S.NRC to comment on the advisability of using Green's function as a time and cost saving measure, nuclear industry respondents, including Entergy's consultants, Structural Integrity Analylists , were almost universal in pointing out that the use of Green's function was less of a determinant than the many points at which engineering discretion must be applied. [1] (Attachment One) I refer the Board to Comment Three in the referenced document:

---

[1] Attachment 1 – Comments on Proposed Generic Communication, "Fatigue Analysis of Nuclear Power Plant Components " U.S. Federal Register, Vol. 73, No. 85, Thursday, May 1, 2008, Notices, p.24094.

> The detailed stress analysis requires consideration of six stress components, as discussed in ANTE Code, Section 111 Subsection NB, Sub-article 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.
>
> <u>ASME Code, Subsection NB, Sub-article 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</u>. 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? [Emphasis added]

When the analyst exercises poor judgment by disregarding common engineering practices and substitutes for them with his own unproven perceptions, his results are not likely to be conservative. Unless the analyst has the required skills and acts in an objective and professional manner the results will be of questionable value. My concerns are primarily with the lack of conservatism in the heat transfer calculations and the use of non conservative oxygen concentrations in the analysis of the CS and RO nozzles.

Q6. Did Entergy introduce a new methodology in their analysis of the CS and RO nozzles?

A6. No, generally speaking the methodology was the same, in fact Entergy stated as much in its confirmatory analysis submittal letter of January 8, 2009.

> ...The methodology applied in the referenced CS and RO confirmatory analyses is in accordance with the approach used in the SIA calculations for the feedwater nozzle that were introduced into evidence in this proceeding, and contains no

significantly different scientific or technical judgments from those used in feedwater nozzle calculations...

Q7. Can the same approach that was used for the feed water (FW) nozzle be repeated for the CS and RO nozzles ?

A7. Absolutely not. The heat transfer coefficients at the boundaries of pipes, elbows and nozzles are very sensitive to the local geometry and the size of the component. For example, the diameters of RO nozzle is more than 2.5 times larger than the diameter of the FW nozzle therefore the heat transfer coefficients both in magnitude and distribution are different in each nozzle. Another major difference relates to the inlet nozzle geometries with regard to how the flow enters the nozzle. The flow field (velocity distribution) at the entrance to the RO nozzle is considerably different than the flow at the entrance to the FW nozzle. These differences have a marked effect on the heat transfer. The fluid flows in a different direction in the RO nozzle than in the FW nozzle. The RO and the CS nozzles and FW nozzle are located at different sections of the reactor vessel and therefore their local coolant chemistries differ during transients. Each component must be examined individually, it is incorrect to claim that the approach that was previously used for the determination of heat transfer coefficients and oxygen concentrations may be universally applied across all the variations of specific local conditions..

Q8. Why is accurate determination of heat transfer coefficients important to the fatigue life of the RO and CS nozzles?

A8. Heat transfer coefficients determine the magnitude and the distribution of the wall temperature during the transients. The local temperature changes during the transients determine the magnitude of the stresses which control fatigue life. There is no disagreement that the fatigue life is very sensitive to even a very small change in the heat transfer coefficient. The use of incorrect heat transfer coefficients would result in invalid fatigue life predictions.

Q9. Why are the equations that were used by Entergy to calculate heat transfer coefficients are not applicable to the RO geometry

A9 In an apparent attempt to reduce the cost of their calculations, Entergy developed an axis-symmetrical model that is based on the supposition that the heat transfer coefficients are constant both in forced and free convection flow. The heat transfer coefficients for forced convection flow were derived from equations which are valid only when the flow inside the pipe is fully developed, i.e. the velocity distribution along the pipe axis is similar at any location along its axis. Because the flow into the RO nozzle enters from the reactor vessel side and the length of the straight upstream piping is zero the flow through the nozzle is not fully developed and therefore the equations used by Entergy grossly misrepresent the actual heat transfer coefficients. Because of the large diameter of the RO nozzle in comparison to the FW nozzle, (36"-26" vs. 9.8") the heat transfer coefficient during natural convection varies considerably more around the circumference of the RO nozzle than around the circumference of the FW nozzle. The model used by Entergy is completely unrealistic and does not represent real life phenomena. Entergy did not justify the assumptions which underlie the axis-symmetrical model and used the heat transfer equations incorrectly.

Q10. Can you please discuss more quantitatively why you believe that the equations used by Entergy to calculate the heat transfer coefficients are incorrect?

A10 Yes. Let me begin with a simplified discussion of forced convection flow. It is a well established fact that when a uniform flow enters a pipe (See A11, below) it takes the distance equivalent to between 25 to 40 diameters, depending on the Reynold's number, for the flow to become fully established. Beyond that distance the flow is similar at any location in the pipe. Because of the analogy between heat transfer and flow friction a length of 25-40 diameters is also required before the heat transfer coefficient in water becomes independent of the axial distance along the pipe. The flow entering the spray nozzle is not uniform, because the upstream elbows cause the flow to enter these nozzles at different axial velocities, (See attachment 4) and therefore it may take even longer to establish a uniform flow.

The flow into the RO nozzle enters from the reactor vessel side where the inlet conditions are entirely different than the inlet conditions at the FW nozzle because the coolant flows is in the opposite direction. The flow through the RO nozzle is commonly classified as boundary layer type flow in a convergent channel. The velocity distribution is considerably different in the convergent channel than the velocity distribution in a pipe of constant diameter, (See Attachment 2). The local velocity profile relates to the local heat transfer coefficient. The equations used by Entergy to calculate the heat transfer coefficient for the RO are applicable to straight a pipe where the flow is fully developed; they are not applicable to a convergent nozzle such as the RO nozzle. Nor are they applicable to the spray nozzle where the flow entering the nozzle is not uniform.

Q11. What approach was used to calculate the heat transfer coefficients for the FW nozzle?

A11. Entergy stated that the flow through the FW nozzle was fully developed **because** they claimed that for that specific geometry of the FW nozzle, five diameters is sufficient for the flow to become fully developed. Entergy's statement clearly differs from all textbooks on thermal hydraulics clearly showing that 25- 40 diameters are required for the flow to become fully developed.

For example, the authoritative Text Book by Schlichting states:

> The length in turbulent flow is considerably shorter than in laminar flow. According to the measurements performed by H. Kirsten its length is about 50 to 100 diameters, but J. Nikuradse determined that the fully formed velocity profile exists already <u>after an inlet length of 25 to 40 diameters</u>.
>
> **'Turbulent Flow Through Pipes'** H. Schlichting, Boundary Layer Theory, 4th Ed. 1960. P 502

In engineering applications where flow meters are used in piping, the flow must be fully developed when it enters the meter. Flow straighteners are inserted upstream of the meter when sufficient long pipe section upstream of the meter is not available.

The length of the required straight pipe section upstream of the straightener depends on the configuration of the upstream component (elbow, valve etc.). By a simple Google search on "flow straighteners," one can quickly obtain a large number of technical articles concerning the required length for the flow to become fully developed. None of these articles, in fact nothing that I can find in the standard literature, leads to the conclusion that 5 diameters are sufficient for the flow to become fully developed. (See also, Attachment Five)

Further, Entergy does not provide any data to support the proposition that 5 diameters is sufficient length for full flow to develop. Even in the unlikely event that such data exist, and even if one were to accept Entergy's contention that there is no difference between flow in a straight pipe and the flow in a convergent nozzle , the flow in the RO nozzle will not be fully developed because the RO nozzle diameter varies between 26 and 36 inches. Since the RO nozzle length is less than 150 inches (30x5), the flow in the RO nozzle can not be fully developed. The large axial and circumferential variation in the heat transfer will affect the maximum stress distribution during transients, leading to a different CUFen than was calculated by Entergy.

To justify not accounting for heat transfer variations due to wall discontinuities along the RO nozzle, Entergy stated without any basis or proof that:

"The effect of non-uniform geometries is judged to be insignificant for flow inside the safe end, because of the smooth transition and small geometry changes"

(See, Entergy Reanalysis Calculation Package- 0801038.304R, Assumption 4 under 3.1)

Entergy's premise that a "smooth transition" assures that there will be no variation in the local heat transfer is absurd. The common criteria for discontinuities are flow separation at the wall which occurs when the local shear force is zero. The formation of vortices increases the local heat transfer coefficient. As an example, the flow a round a sphere exhibits separation and local variation in heat transfer even though the surface of the sphere is 100% smooth. Entergy's description of "smooth transition" is not a relevant engineering description.

Q12. Can you comment the fact that ASLB and the NRC staff apparently found the approach of using a five-diameter length criterion for the establishment of fully developed conditions acceptable when considering the FW nozzle?

A12 The fact that the ASLB and the NRC accepted this approach for the FW nozzle does not change the fact that the RO nozzle require a different analysis because its geometry is completely different then the geometry of the FW nozzle and therefore the assumption of using heat transfer coefficient for a pipe with a uniform diameter like the FW nozzle is simply not applicable.

Q13. How is heat transfer affected by the geometry and the size of the nozzle in natural convection and condensation.

A13. Unlike forced convection, the heat transfer during natural convection and during condensation is governed by gravitational forces and therefore the local heat transfer coefficient varies in the vertical direction as is described by the following equation:

1. $h = C(\mathrm{Gr}\,\mathrm{Pr})^n k/x$ (See, Entergy Reanalysis Calculation Package 0801038.301, page 9)

With the same value of n that Entergy used (n= 0.25), equation <u>1</u> predicts that the heat transfer coefficient would vary with the vertical distance as $1/x^{0.25}$. Accordingly the heat transfer coefficient will vary by a factor of 2.5 $(36/1)^{0.25}$ around the circumference of the RO nozzle, i.e. 240% variation vs. 140% for the FW nozzle. The variation in the heat transfer around the RO nozzle is significantly higher than the variation in the heat transfer around the FW nozzle and can not be neglected.

Q14. How did Energy account for the variation of the heat transfer coefficient ?

A14. Entergy completely ignored the circumferential variation of the heat transfer by using Equation **1** incorrectly. They falsely stated that **x** in equation **1** above is the diameter of the nozzle and thus Entergy has ignored the local variations in the heat transfer coefficient. **X** is clearly not a constant, it is a variable, and it varies with the vertical distance because the flow is driven by gravitational forces. Such a procedure may provide Entergy the answer they seek but it is grossly inconsistent with reality. The difference between a constant and a variable can be learned from any elementary text in mathematics, the two are not interchangeable. **X** in equation **1** can be substituted by the diameter only after integration with respect to **x** to obtain the average heat transfer, (See Attachment Four). Treating x as a constant in the above equation is a major assumption which conspicuously is missing from the list of assumption in the document referenced above. It is clear that the large variation of the heat transfer around the circumference of the nozzle precludes the use of the average heat transfer for the determination of the maximum stress, which is required for the determination of the CUF.

Q15. How does Entergy's approach to calculations of the CUF for the RO nozzle compare to their FW nozzle?

A15. Because the FW nozzle is considerably smaller than the RO nozzle the affect on the maximum stress is less for the FW nozzle. It is clear that even if one could get by with neglecting local variation in the heat input to the FW nozzle, it can not be neglected in the analysis of the RO nozzle. The RO nozzle represents an entirely different geometry than the FW nozzle, therefore it is incorrect to use the same methodology for both the FW and RO to calculate heat transfer coefficients and stress distribution.

Q16 Do you have any comments on Entergy's apparent failure to include realistic consideration of the heat transfer distribution around the nozzles both in forced and natural convection flows?

A16. The heat transfer phenomena which I described above are very fundamental. They are taught at the undergraduate engineering level. Any text on heat transfer, not to mention the hundreds of papers that have been written on the subject provide a thorough discussion of this subject. I have personally conducted studies on flow development in short channels, and there is no question in my mind that Entergy has introduced some novel concepts in heat transfer that have never been documented before.

Given the large amount of information available on this subject I can not believe that Energy is not aware that using average heat transfer coefficients can not be used to calculate maximum stresses during transients.

I have attached data from text books and other sources that clearly contradict Entergy's proposition that the heat flowing into the nozzle wall is uniformly distributed. (Please see, Attachments 2,3,4)

Q17. What do you conclude Entergy should be required to do so as to establish confidence that the heat transfer inputs result in conservative predictions of the fatigue life of the CS and RO nozzles.

A17. I believe that Entergy should be required to demonstrate that the incorrect heat transfer equations that they used actually result in <u>conservative</u> CUFens. This can be done by repeating the calculations with heat transfer equations which are valid for the nozzle geometries and which take into account the local variation in the heat transfer instead of using average values. I believe that such calculations will show that the <u>present results are not conservative</u>.

Q18 Is the methodology of considering the effect of oxygen concentrations for the RO nozzle is the same as the methodology for the FW nozzle

A18. The RO nozzle is different than the FW nozzle because the materials of constructions are different. The RO nozzle is constructed from both stainless steel (safe end) and low alloy steel (forging end). The FW is constructed of carbon steel alone. While in the case of the FW the same oxygen concentrations along the nozzle may be applied, different oxygen concentrations must be applied at the different end of the RO nozzle. On the safe end of the RO nozzle, where stainless steel properties are used, Entergy, disregarding ANL's specification, entered the same oxygen concentrations in the Fen equation as on the other end of the nozzle where Alloy 600 properties are used.

NUREG 6909, Page A.5., states:

> The DO value is obtained from each transient constituting the stress cycle. For carbon and low-alloy steels, the dissolved oxygen content, DO, associated with a stress cycle is the <u>highest</u> level in the transient, and for <u>austenitic stainless steels, it is the lowest oxygen level</u> in the transient. A value of 0.4 ppm for carbon and low-alloy steels and 0.05 ppm for austenitic stainless steels can be used for the DO content to perform a conservative evaluation. [Emphasis added]

<u>Entergy provided no measurements to justify using the same oxygen concentrations for two different materials at approximately the same location in the reactor system</u>

For the CS and RO nozzles, low alloy steel locations, the maximum oxygen concentrations occur at the lowest temperature during the transient because of (1) the inverse solubility relation between oxygen and temperature, and (2) plant data clearly show that the oxygen concentration increases with decrease in temperature during plant startups and shutdowns.

Q22. How would the CUFen be affected if Entergy had used the ANL specified concentration of 0.4ppm?

A22 The CUFen would have been increased by an order of magnitude

Q 23 Did Entergy support their analysis with oxygen measurements during transients for the CS and RO nozzles?

A 23. No, they did not. They only stated that they have done such measurements for the FW nozzle.

Q. 24. Entergy repeatedly states that their fatigue analysis is conservative, does recent industry experience support such statements

A .24. No. Recent discoveries of large cracks in RO nozzles both at the James A Fitzpatrick Nuclear Station (ADAMS Accession Number -ML083300360 "LER 2008-002-00, November 20, 2008) and Oyster Creek Nuclear Generating Station (ADAMS Accession Number ML0090280055 " Submittal of Analytical Evaluation ...") [nozzle indication], January 21, 2009), clearly indicate that Entergy's analysis is not conservative. Entergy excluded the possibility that those nozzles already contain cracks no matter what the source (fabrication, stress corrosion, or fatigue). The presence of such cracks would increase the probability of RO failure under cycling loading.

Q. 25. Does this complete your testimony at this time?

A.25. Yes

DECLARATION

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

Executed this 22nd _day of April, 2009 at Rockville, Maryland.

Joram Hopenfeld

Joram Hopenfeld, PhD

ATTACHMENT ONE (Reformatted for purposes of electronic transmission)

Structural integrity Associates, Inc.

6855 S Havana St. Suite 350
Centennial, CO 80112
Phone: 303-792-0077 Fax: 303-792-2158
imwstruclint.com

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)

(NECO0133599)

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, V ol. 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-1901, Revision 1) and specified in associated documents NUREG/CR-5583 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 NR C 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 ANTE Code, Section 111 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 NR C reviewed all approaches used to assess environmental effects and determined that all other methods are always conservative?

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

Comment 4:

> *The detailed stress analysis requires consideration of six stress components, as discussed in ASME Code, Section ill.Subsection NB, Sub-article 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, I, and r directions may be so chosen that the shear stress components are zero and $0._1$, 6_2, and o are identical to ut, 0'1, and G_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 (G_x, G_y, G_z or G_t, D_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 RI S demonstrated that shear stresses were negligible, and Advisory Committee on Reactor Safeguards (ACRS) testimony earlier this year indicated that the non-conservatism 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 RI S requires the use of all six stress components, why it is acceptable for CLB analyses to not do so and why

the RI S is limited to those select few environmental fatigue evaluations that have used a simplified Green's Function methodology associated with license renewal. Please clarify.

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

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 RI S 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 RI S to avoid misinterpretation.

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

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 NR C 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 NR C 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.

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

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 ill, 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 RI S 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, Sub-article 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.

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

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?

ATTACHMENT TWO

1. **HEAT TRANSFER COEFFICIENTS AT THE ENRANCE SECTION OF A STRAIGHT PIPE.** E. R. G. Eckert and R. Drake, Heat and Mass Transfer 2nd, Ed 1959,

H. Hausen[1] gave the expression for the average Nusselt number:

$$\overline{Nu}_d = 0.116[(Re_d)^{2/3} - 125](Pr)^{1/3}\left[1 + \left(\frac{d}{x}\right)^{2/3}\right]\left(\frac{\mu_B}{\mu_w}\right)^{0.14}$$

where μ_B is the viscosity at bulk liquid temperature and μ_w the viscosity at tube-wall temperature. Apart from the latter, the property values are to be inserted at t_B. This formula takes into account the conditions in the intake region. It also satisfactorily reproduces the values in the transition zone $Re_d = 2{,}300$ to $6{,}000$. This relation is expected to be especially applicable to fluids for which the variation of viscosity is the

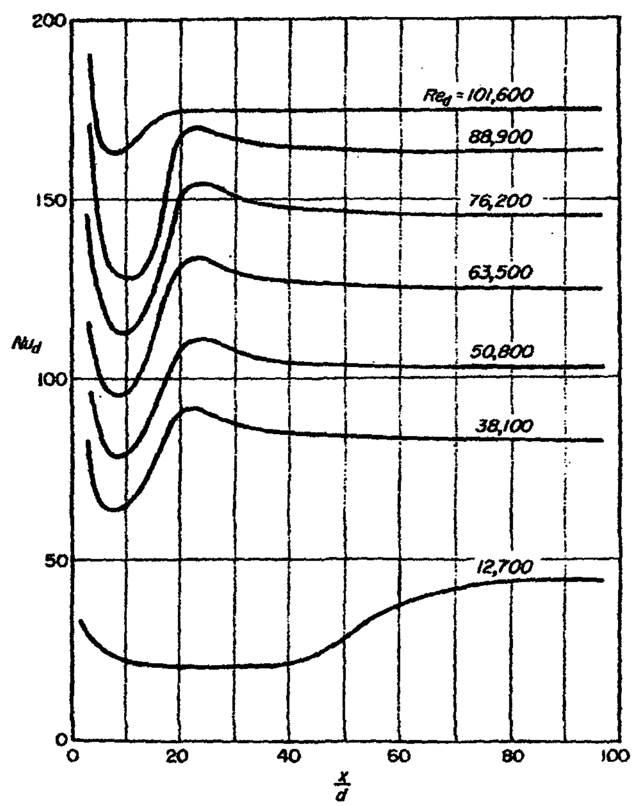
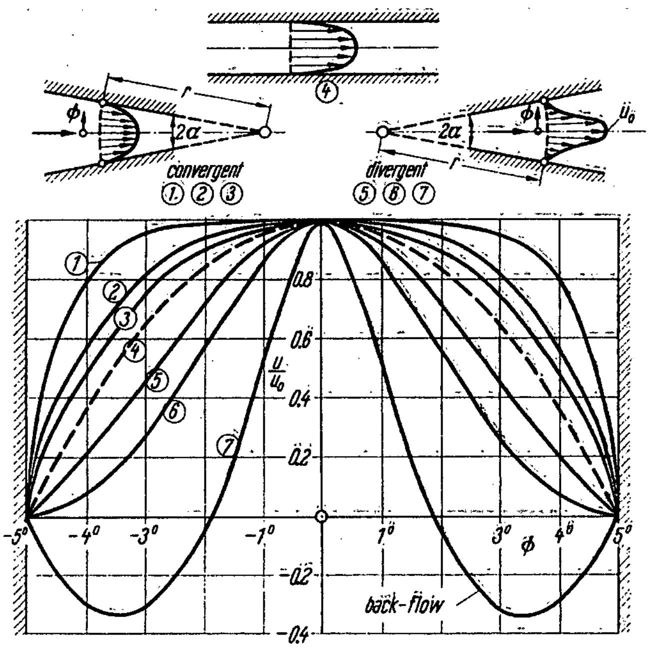


FIG. 8-9. Local Nusselt numbers for flow through a tube near the entrance with simultaneous development of the flow and temperature field. [*From W. Linke and H. Kunze, Allgem. Wärmetech.*, 4:73–79 (1953).]

[1] H. Hausen, *Z. Ver. deut. Ingr., Beih. Verfahrenstech.*, no. 4, 1943, pp. 91–98.

ATTACHMENT THREE

Velocity Distribution in a Convergent Channel (RO Nozzle) In Comparison to a Flow in a Straight Pipe. H. Schlichting, Boundary Layer Theory, 4th Ed. 1960. P 90



Included angle $2\alpha = 10°$
Reynolds number $\bar{R} = u_0 r/\nu$

Convergent channel	Divergent channel
Curve 1: $\bar{R} = 5000$	Curve 5: $\bar{R} = 5000$
Curve 2: $\bar{R} = 1342$	Curve 6: $\bar{R} = 1342$
Curve 3: $\bar{R} = 684$	Curve 7: $\bar{R} = 684$

Curve 4 refers to a channel with parallel walls (Poiseuille's parabolic velocity distribution, *cf.* Fig. 5.1)

Fig. 5.15. Velocity distribution in a convergent and a divergent channel after G. Hamel [10] and K. Millsaps and K. Pohlhausen [21]

(See, Entergy Reanalysis Calculation Package 0801038.304R, Assumption 4 under 3.1)

ATTACHMENT FOUR

Heat Transfer During Natural Convection varies with the Vertical Distance x. Only the average heat transfer coefficient is independent of x. E. R. G. Eckert and R. Drake, Heat and Mass Transfer 2^{nd}, Ed 1959,

From Eq. (11-1) there is obtained

$$q = \frac{2k\vartheta_w}{\delta}$$

On the other hand, the equation by which the film-heat-transfer coefficient is defined reads

$$q = h\vartheta_w$$

Therefore

$$h = \frac{2k}{\delta} \tag{11-8}$$

and in dimensionless form

$$\frac{hx}{k} = \mathrm{Nu}_x = 2\frac{x}{\delta}$$

By introducing the boundary-layer thickness one obtains

$$\mathrm{Nu}_x = 0.508\,\mathrm{Pr}^{1/2}(0.952 + \mathrm{Pr})^{-1/4}(\mathrm{Gr}_x)^{1/4} \tag{11-9}$$

The local film heat-transfer coefficient decreases according to Eqs. (11-6) and (11-8) with increasing distance x. It is inversely proportional to the fourth root of x. By integration over the distance the average heat-transfer coefficient is found to be

$$\bar{h} = \tfrac{4}{3}h$$

This means that the average heat-transfer coefficient of a vertical plate with a height x is $\frac{4}{3}$ the local value at the point x. For ideal gases the relationship $\beta = 1/T$ holds true. As long as the temperature differences are small, the expansion coefficient can be written $\beta = 1/T_0$, where T_0 is the absolute temperature in the gas outside the boundary layer. For air with a Prandtl number Pr = 0.714,

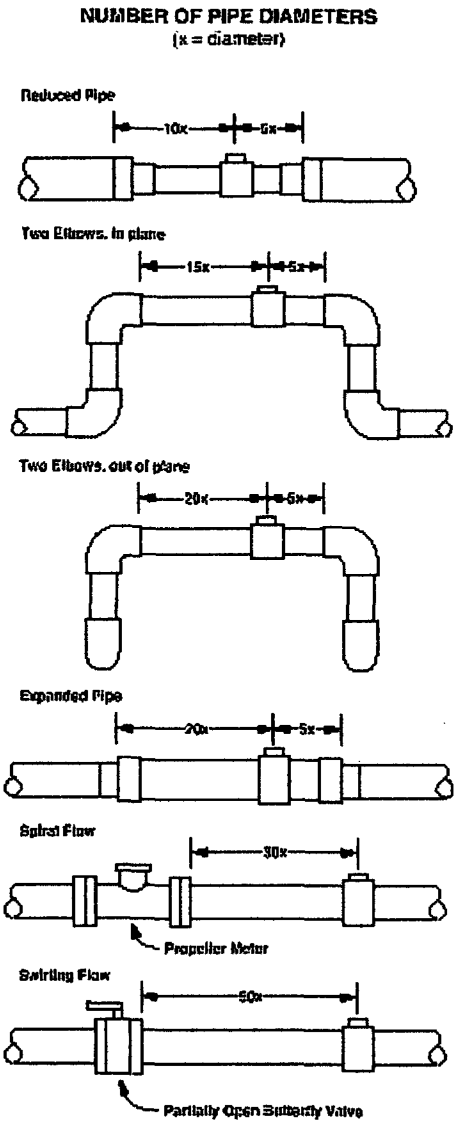
$$\mathrm{Nu}_x = 0.378(\mathrm{Gr}_x)^{1/4} \tag{11-10}$$

For this medium the heat transfer was calculated exactly by E. Pohlhausen in collaboration with E. Schmidt and W. Beckmann.[1] This resulted in a numerical value of 0.360 instead of 0.378 as in Eq. (11-10). Therefore, the approximate treatment agrees quite well with the much longer exact calculation. A comparison between the values calculated here and those measured and calculated by E. Schmidt and W. Beckmann is presented in Figs. 11-2 and 11-3. In the calculations, the property values were introduced at the plate temperature. S. Ostrach[2] has recently solved the laminar free-convection boundary-layer equations for a vertical plate on an electronic computer for several Prandtl numbers.

[1] E. Schmidt and W. Beckmann, *Tech. Mech. Thermodynam.*, 1:1–24 (1930).
[2] S. Ostrach, *Natl. Advisory Comm. Aeronaut. Tech. Note* 2635, 1952.

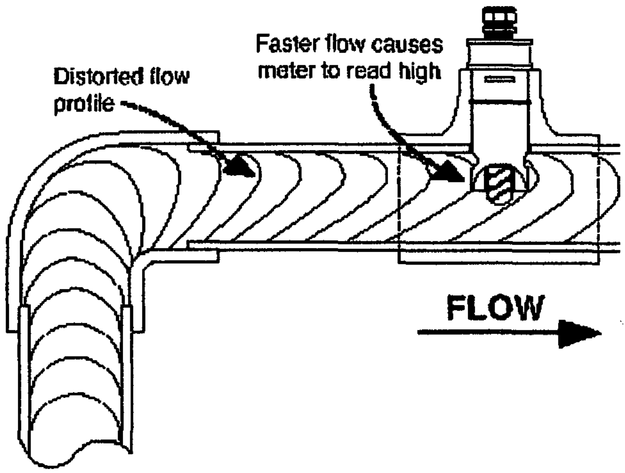
ATTACHMENT FIVE

The fact that more than five diameters is required for the flow to become fully developed is illustrated by the following Global Water Instrumentation, Inc data



How Much Can The Flow Meter's Error Be?

The flow meter's error can be quite large. The error produced by a thermowell installed immediately upstream of a flow meter can be in the range of 5-10%. That of a gate valve or a butterfly valve upstream of a flow meter can be as much as 50-60%! The error produced from a partially closed ball valve can be as much a 50% for flow meters. Chemical injectors can produce significant error in the flow meter reading also. For example, a chlorine injector-diffuser may produce enough entrained undissolved chlorine bubbles to produce an error in the 10- 20% range of the flow meter output.



Good Flow Meter Installations

We suggest at least 10 diameters of straight pipe run upstream and 5 diameters of straight pipe run downstream of any flow meter installation in order to achieve proper accuracy. These are minimum suggested values. As the pipe layout diagram shows, you may need much more straight run prior to the flow meter, under specific circumstances.

New England Coalition

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

POST OFFICE BOX 545, BRATTLEBORO, VERMONT o5302

April 24, 2009

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 Leave to File a Timely New Contention And Motion to Hold in Abeyance Action on This Proposed Contention Until Issuance of NRC Staff Supplemental Safety Evaluation Report

Thank you for your kind attention,

/RS

for New England Coalition, Inc.

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