UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	Docket Nos. 50-247-LR and
	)	50-286-LR
ENTERGY NUCLEAR OPERATIONS, INC.	)	
	)	
(Indian Point Nuclear Generating Units 2 and 3)	)	
	)	October 12, 2012

TESTIMONY OF ENTERGY WITNESSES IAN D. MEW, ALAN B. COX, NELSON F. AZEVEDO, JEFFREY S. HOROWITZ, AND ROBERT M. ALEKSICK REGARDING CONTENTION RK-TC-2 (FLOW-ACCELERATED CORROSION)

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I. WITNESS BACKGROUND

A. Ian D. Mew ("IDM")

Q1. Please state your full name.

A1. (IDM) My name is Ian D. Mew.

Q2. By whom are you employed and what is your position?

A2. (IDM) I am employed by Entergy Nuclear Operations, Inc. ("Entergy"), the Applicant in this matter, as a Senior Engineer in Programs and Components Engineering at Indian Point Nuclear Generating Units 2 and 3 ("IP2" and "IP3," collectively "Indian Point Energy Center" or "IPEC").

Q3. Please describe your educational and professional qualifications, including relevant professional activities.

A3. (IDM) My professional and educational qualifications are summarized in the *curriculum vitae* attached as Exhibit ENT000030. Briefly summarized, I hold a Bachelor of Science degree in Mechanical Engineering from the Polytechnic Institute of New York. I have more than 30 years of experience in the nuclear power industry. For the past 16 years, I have been involved in flow-accelerated corrosion ("FAC") and steam generator program development and inspections, including in-service inspection ("ISI") program development and inspection work throughout the nuclear power industry. I have developed and/or implemented numerous plant design changes, inspection programs, equipment specifications, and operability evaluations of degraded components.

Q4. Please describe the basis for your familiarity with the FAC issues.

A4. (IDM) I have been responsible for FAC issues since 1995. While employed at the New York Power Authority, I was FAC engineer for the James A. FitzPatrick ("JAF") nuclear power plant FAC Program from 1995 to 2004. I first modeled the plant piping systems at JAF using the Electric Power Research Institute ("EPRI") CHECMATE code and selected the piping component inspection locations for the 1996 refueling outage and thereafter. From 2005 to 2007, I also was the Regional Manager and FAC Program Owner for the Entergy northeast fleet, which includes JAF, IPEC, Pilgrim Nuclear Power Station, and Vermont Yankee Nuclear Power Station ("VYNPS"), and charged with standardizing the fleet FAC Programs. From 2004 to 2007, I also served as a peer reviewer for the IPEC FAC engineer. In 2007, I was transferred to IPEC and assumed the role of FAC engineer for both IP2 and IP3.

Before assuming the assignment at IPEC, I was also Chairman of the FAC working group for Entergy's northeast fleet from 2005 to 2007. In that capacity, I developed a long-term "Piping Flow Accelerated Corrosion Inspection Program" ("FAC Program") for Entergy's northeast fleet. This required determining the scope of piping potentially affected by FAC, coordinating the modeling of plant systems using the CHECWORKS code, developing the criteria and procedures for performing FAC inspections, and scheduling and participating in focused self-assessments for each of the Entergy Northeast plants.

My current responsibilities also include reviewing industry operating experience ("OE"), including NRC and Institute of Nuclear Power Operations ("INPO") operating experience with FAC to assess its potential implications for IPEC. As such, I have worked closely with numerous other plant FAC engineers by attending EPRI-sponsored CHECWORKS Users Group ("CHUG") meetings. I have participated either as a team member or team lead in self-assessment audits of FAC programs at Entergy's northeast nuclear power plants.

I have been trained to run the CHECWORKS and FAC Manager software programs that Entergy uses in its FAC Program at IPEC. CHECWORKS Steam Feedwater Analysis 3.0 SP-2 is the version of the CHECWORKS software used at IPEC. FAC Manager is the FAC Program management and trending software developed by CSI Technologies, Inc., a leading firm in FAC consulting that provides FAC analysis and FAC/CHECWORKS-related outage support to Entergy and the nuclear industry. At IPEC, the FAC Manager software is used to calculate wear, wear rate, minimum acceptable wall thickness, remaining service life ("RSL"), and the next scheduled inspection ("NSI").

Based on the above-described experience, I have detailed knowledge of the matters discussed herein that relate to implementation of the FAC Program at IPEC, including the description of that program in the IPEC license renewal application ("LRA") and other related documentation discussed below.

B. **<u>Alan B. Cox (“ABC”)</u>**

Q5. Please state your full name.

A5. (ABC) My name is Alan B. Cox.

Q6. By whom are you employed and what is your position?

A6. (ABC) I am employed by Entergy, the Applicant in this matter, as the Technical Manager of License Renewal. My office is located at Entergy’s Arkansas Nuclear One (“ANO”) facility in Russellville, Arkansas.

Q7. Please describe your educational and professional qualifications, including relevant professional activities.

A7. (ABC) My professional and educational qualifications are summarized in the *curriculum vitae* attached as Exhibit ENT000031. Briefly summarized, I hold a Bachelor of Science degree in Nuclear Engineering from the University of Oklahoma and a Masters of Business Administration (M.B.A.) from the University of Arkansas at Little Rock. I have more than 34 years of experience in the nuclear power industry, having served in various positions related to engineering and operations of nuclear power plants. For example, from 1993 to 1996, I was employed as a Senior Staff Engineer at ANO. From 1996 to 2001, I served as a Supervisor, Design Engineering, at ANO. I was licensed by the NRC in 1981 as a reactor operator and in 1984 as a senior reactor operator for Arkansas Nuclear One, Unit 1. I previously have held a professional engineer’s license in the State of Arkansas.

Since 2001, I have worked full-time on license renewal supporting the integrated plant assessment and LRA development for Entergy license renewal projects, as well as license renewal projects for other utilities. Specifically, as a member of the Entergy license renewal team, I have participated in the development of eight LRAs and in industry peer reviews of at least eleven additional LRAs. I have been a member of the Nuclear Energy Institute (“NEI”)

License Renewal Task Force for approximately ten years, and during portions of that time have served as Entergy's representative on the NEI License Renewal Mechanical Working Group and the NEI License Renewal Electrical Working Group. As a member of the Entergy license renewal team, I have participated in the development of nine LRAs and in industry peer reviews for at least 12 additional LRAs.

Q8. Please describe the basis for your familiarity with the Indian Point Energy Center LRA and FAC Program.

A8. (ABC) As Technical Manager, I was directly involved in preparing the LRA and developing and reviewing Aging Management Program ("AMP") descriptions for IP2 and IP3, including the FAC Program. I also have been directly involved in developing and reviewing Entergy responses to NRC Staff Requests for Additional Information ("RAIs") concerning the LRA and various amendments or revisions to the application (principally as they relate to aging management issues). I also supported Entergy at the related Advisory Committee on Reactor Safeguards ("ACRS") Subcommittee and Full Committee meetings for the IPEC LRA held in March 2009, and in September 2009, respectively. Accordingly, I have personal knowledge of the matters discussed herein that relate to the IPEC FAC Program, including the description of that program in the LRA and other related documentation discussed below.

C. Nelson F. Azevedo ("NFA")

Q9. Please state your full name.

A9. (NFA) My name is Nelson F. Azevedo.

Q10. By whom are you employed and what is your position?

A10. (NFA) I am employed by Entergy, the Applicant in this matter, as Supervisor, of Code Programs, at Indian Point Energy Center in Buchanan, New York.

Q11. Please describe your educational and professional qualifications, including relevant professional activities.

A11. (NFA) My professional and educational qualifications are summarized in the *curriculum vitae* attached as Exhibit ENT000032. I hold a Bachelor of Science degree in Mechanical and Materials Engineering from the University of Connecticut and an M.S. in Mechanical Engineering from the Rensselaer Polytechnic Institute ("RPI") in Troy, New York. In addition, I have received an M.B.A. from RPI.

I have over 30 years of professional experience in the nuclear power industry. During that time, I have held engineering, supervisory, and manager positions with Northeast Utilities for nearly 19 years and Entergy for more than 11 years. Prior to becoming a Manager, I was an Engineer for more than ten years and an Engineering Supervisor for another five at Northeast Utilities which owned and operated the Connecticut Yankee and Millstone Stations. I was responsible for, among other duties, developing the Northeast Utilities' FAC program for the Connecticut Yankee and Millstone Stations and implementing the FAC program for Connecticut Yankee in the early 1990s. This included both working with EPRI to develop the initial program and managing the field activities to implement the required inspections and repairs/replacements. The Northeast Utilities FAC program was one of the first comprehensive FAC programs developed following several pipe failures, and this program later became one of the models used to upgrade FAC programs throughout the industry. As a Department Manager with Northeast Utilities, I managed five engineering sections responsible for implementing numerous engineering programs at the Millstone Nuclear Power Station, including the FAC programs for the three Millstone plants.

Currently, I oversee the IPEC engineering section responsible for implementing American Society of Mechanical Engineers ("ASME") Code programs, including the ISI, inservice testing, FAC, snubber testing, boric acid corrosion control, non-destructive examination, fatigue monitoring, steam generators, buried piping, Alloy 600 cracking, reactor vessel embrittlement, welding, and 10 C.F.R. Part 50, Appendix J containment leakage programs. I am the supervisor of the group responsible for ensuring compliance with the ASME Code, Section XI requirements for repair and replacement activities at IPEC. I also represent IPEC before several industry organizations, including the pressurized water reactor ("PWR") Owners Group Management Committee.

Q12. Please describe the basis for your familiarity with the IPEC license renewal project, including the associated License Renewal Application ("LRA"), with respect to the issues raised in RK-TC-2.

A12. (NFA) As Supervisor of Code Programs at IPEC, I have been responsible for FAC-related issues since January 2001. These activities include the supervision of the IPEC FAC Program, as well as refueling outage-related activities, including inspection location selection, field inspections, evaluation of inspection results, and any necessary repairs and/or replacements.

D. <u>Jeffrey S. Horowitz ("JSH")</u>

Q13. Please state your full name.

A13. (JSH) My name is Jeffrey S. Horowitz.

Q14. By whom are you employed and what is your position?

A14. (JSH) I am an independent consultant hired by Entergy, the Applicant in this matter.

Q15. Please describe your educational and professional qualifications, including relevant professional activities.

A15. (JSH) My professional and educational qualifications are summarized in the *curriculum vitae* attached as Exhibit ENT000033. Briefly summarized, I have more than 40 years of experience in the field of nuclear energy and related disciplines. For over 25 years, I have been an independent consultant specializing in FAC and nuclear safety analysis. My primary client during this time has been the Electric Power Research Institute ("EPRI"). I have consulted on FAC-related issues for Canadian and U.S. utilities that operate nuclear power plants. In Canada, I also have consulted for the CANDU Owners Group. I hold four degrees in Mechanical Engineering: a Bachelor of Science degree from Newark College of Engineering (now known as New Jersey Institute of Technology), and from the Massachusetts Institute of Technology, a Master of Science degree, a Mechanical Engineer degree and a Doctor of Science degree. I am a fellow of the American Society of Mechanical Engineers ("ASME") and the American Nuclear Society ("ANS").

Q16. Please describe the basis for your familiarity with the topic of FAC and how it is addressed in the IPEC LRA.

A16. (JSH) I have personal knowledge of the matters discussed herein that relate to industry and IPEC experience with FAC and the development of programs to predict which plant components may be susceptible to FAC. I recently performed an audit of the IPEC FAC Program documentation in support of my testimony. Horowitz Audit of the IPEC FAC Program (Mar. 2012) (ENT00034). This review was done in support of my testimony in this proceeding. It was performed to: establish an independent baseline of knowledge for me on the specifics of the IPEC FAC program, and establish a record of the features of the IPEC FAC program based on a direct review of related documentation and the CHECWORKS models. I also have reviewed those portions of Entergy's LRA related to FAC and certain other documents discussed below.

Q17. Please describe your initial professional experience with FAC issues.

A17. (JSH) My involvement with the assessment of FAC spans several decades. I first became involved in December 1986, when an elbow in the condensate system of the Surry Unit 2 nuclear power plant failed, causing the release of steam and hot water into the turbine building. As indicated in NRC Bulletin 87-81, post-accident investigations revealed that FAC (known at that time as erosion/corrosion) caused the degradation of the elbow. *See* NRC Bulletin 87-01, Thinning of Pipe Walls in Nuclear Power Plants at 1 (July 1987) (RIV000007). At that time, the U.S. nuclear fleet did not have programs in place to manage single-phase (*i.e.*, water only) piping degradation caused by FAC, though some programs were in place to manage two-phase (*i.e.*, water and steam) piping degradation.

In response to the pipe rupture at Surry in 1986, EPRI committed to develop a computer program to assist utilities in determining the places most susceptible to FAC damage, and thus the key locations to inspect for pipe wall thinning. At EPRI's request, I was the co-developer of the computer program CHEC (Chexal-Horowitz Erosion Corrosion) and demonstrated and provided it to U.S. utilities in 1987. I was responsible for the overall program design, the development of the predictive algorithm, and the implementation of routines to calculate the pH at temperature. I remained the lead technical person in the development of new and revised CHEC-related codes and CHEC's successor computer programs, CHECMATE and CHECWORKS.

In 1989, EPRI replaced CHEC with CHECMATE (Chexal-Horowitz Methodology for Analyzing Two-Phase Environments), which expanded on CHEC's capabilities and featured the first accurate prediction of two-phase FAC. I was responsible for the overall design of the program, the incorporation of routines to do chemical calculations around the power cycle, and technical interface with the contractor who developed the flow analysis routines.

In 1993, EPRI replaced CHECMATE with CHECWORKS (Chexal-Horowitz Engineering Corrosion Workstation). For CHECWORKS I upgraded the chemistry calculations to include the advanced amines and the calculation of the hydrazine distribution in the secondary system of PWRs. Each new version built on the previous program and incorporated user feedback, improvements in software technology, and available laboratory and plant data into the algorithms used in the programs.

CHECWORKS is now used in all U.S. nuclear units, all Canadian nuclear units, and nuclear units in Belgium, the Czech Republic, England, Japan, Korea, Mexico, Romania, Slovenia, Spain, and Taiwan—more than 150 units in total. *See* Douglas Munson, A Brief Overview of FAC Investigations, Experiences and Lessons Learned at 30 (Jan. 2008) (ENT000035).

Q18. Since the events at Surry, how have you remained involved in FAC issues?

A18. (JSH) I have participated in audits of the FAC programs at about 60 nuclear units in the United States and Canada. The primary purpose of these audits is to assess the state of the FAC programs and to provide recommendations for improving those programs.

In 1993, to help utilities improve and standardize FAC programs, EPRI's Nuclear Safety Analysis Center ("NSAC") published NSAC-202L, *Recommendations for an Effective Flow-Accelerated Corrosion Program.* I played a key role in drafting the original version of NSAC-

202L and resolving numerous utility and NRC comments on the draft. Since that time, I have played a significant role in each of the three subsequent revisions to NSAC-202L, which has become the standard-setting document for the conduct of FAC programs in the United States. The NRC has endorsed NSAC-202L. *See* NUREG-1801, "Generic Aging Lessons Learned Report, Rev. 1," at XI M-61 (Sept. 2005) ("NUREG-1801, Revision 1") (NYS00146C).

After developing CHECWORKS, I co-authored three books on FAC and related issues. One book is a compendium of FAC science and experience that is the most complete reference available on the subject of FAC. *See* Bindi Chexal, *et al.*, EPRI, *Flow-Accelerated Corrosion in Power Plants* (1998) ("Flow-Accelerated Corrosion in Power Plants") (ENT00036A-B). The other two books deal with thermal-hydraulic issues. *See* Bindi Chexal, *et al.*, *Pressure Drop Technology for Design and Analysis* (1999); Bindi Chexal, *et al.*, *Void Fraction Technology for Design and Analysis* (1997). I have authored or co-authored about 40 EPRI reports related to FAC and nuclear safety issues. I was the principal investigator and sole author of 29 of those reports, which include an important study of FAC at welds in nuclear piping and preliminary guidance for the protection of piping against damage from erosive forms of attack.

I have made technical presentations at nearly all of the 46 semi-annual CHUG meetings, which typically attract between 50 and 100 utility engineers and station managers. These presentations cover the results of research that I have performed or topics of general interest, including FAC. In addition to making presentations, I have served as session chair and moderated various discussion groups.

I have presented a number of technical papers on FAC, including papers at (1) the "FAC2008" and "FAC2010" conferences held in Lyon, France; (2) the "Water Chemistry of Nuclear Reactors – Chimie 2002" held in Avignon, France (a meeting attended by over 300 international scientists and engineers); (3) ASME Pressure Vessel and Piping Conferences, (4) the NRC Water Reactor Safety Meeting, and (5) other technical meetings.

I also have conducted more than two dozen multi-day training sessions around the world covering FAC and the use of the EPRI computer programs (CHEC, CHECMATE and CHECWORKS). Numerous utility engineers, utility managers, engineers from INPO, and NRC Staff members have attended the training sessions.

In 2008, I provided testimony concerning FAC and CHECWORKS on behalf of Entergy Nuclear Vermont Yankee, L.L.C. and Entergy Nuclear Operations, Inc. in the *Vermont Yankee* license renewal proceeding. On November 24, 2008, a Board ruled that Entergy had demonstrated that its FAC Program for plant piping at VYNPS will adequately manage the effects of aging during the 20-year license renewal period, as required by 10 C.F.R. § 54.21, and that it meets the reasonable assurance standard of 10 C.F.R. § 54.29. *See Entergy Nuclear Vt. Yankee, L.L.C.* (Vt. Yankee Nuclear Power Station), LBP-08-25, 68 NRC 763, 894-95 (2008). In so doing, the Board dismissed a contention, submitted by intervenor New England Coalition ("NEC")—based primarily on testimony from Dr. Joram Hopenfeld—alleging that VYNPS had not "adequately benchmarked" its CHECWORKS model for the change in plant parameters associated with a 20% extended power uprate ("EPU")—a power uprate approximately four to six times larger than the 2004 to 2005 IP2 and IP3 stretch power uprates ("SPUs") cited in Riverkeeper TC-2 and discussed below. *See id.* at 882, 889, 896.

Q19. Please describe the basis for your familiarity with the IPEC license renewal project, including the associated LRA.

A19. (JSH) I have reviewed various materials in preparing this testimony, including those portions of Entergy's LRA for IP2 and IP3 relating to Entergy's evaluation of the effects of

FAC and the FAC Program. In preparing my testimony, I also reviewed the parties' pleadings on RK-TC-2; the Atomic Safety and Licensing Board's ("Board's") July 31, 2008 Memorandum and Order admitting RK-TC-2, *Entergy Nuclear Operations, Inc.* (Indian Point Nuclear Generating Units 2 & 3), LBP-08-13, 68 NRC 43(2008) ("Order Admitting RK-TC-2"); and the exhibits submitted by NYS and Riverkeeper that are relevant to my testimony.

E. **<u>Robert M. Aleksick ("RMA")</u>**

Q20. Please state your full name.

A20. (RMA) My name is Robert M. Aleksick.

Q21. By whom are you employed and what is your position?

A21. (RMA) I am the President and founder of CSI Technologies, Inc., an Illinois Corporation chartered in 1993. CSI Technologies specializes in FAC services and software development, and has assisted IPEC and over 100 other nuclear units with numerous FAC-related projects since 1994. CSI Technologies employs approximately 15 FAC Engineers.

Q22. Please describe your educational and professional qualifications, including relevant professional activities.

A22. (RMA) My professional and educational qualifications are documented in the *curriculum vitae* attached as Exhibit ENT000037. Briefly summarized, I am a published, internationally recognized expert in FAC with over 20 years of engineering and project management experience in the nuclear industry. I have worked on or managed nearly 1,000 FAC-related projects at over 100 nuclear units.

I am the author or co-author of several technical papers on FAC, was the technical editor of the FAC handbook *Flow-Accelerated Corrosion in Power Plants* (ENT00036A-B), and a significant contributor to several important EPRI FAC guidance documents, including NSAC 202L (all revisions) and "Guidelines for Plant Modeling and Interpretation of Inspection Data" (both revisions). I assisted EPRI in preparing training materials for the CHECWORKS software. I have given dozens of industry technical presentations on FAC at the invitation of EPRI.

My field experience includes hands-on FAC duties at over ten nuclear plant outages, as well as serving on a contract basis for six months as the Corporate FAC Engineer for a domestic utility's fleet. In 2010 and 2011, INPO invited me to train their field evaluators in FAC. I have also conducted over twenty detailed (3 to 5 day) FAC training sessions to US nuclear utility personnel since 2002, and have provided similar training and support to utilities in China, Canada, France, Spain, and Belgium. At the invitation of EPRI, in 1999 and 2000 I served as a member of the design review team for the CHECWORKS-Steam Feedwater Application upgrade, a multi-million dollar software development project that resulted in significant changes and improvements to the CHECWORKS code.

I have also completed five years of B.S. studies in Nuclear Engineering at the University of Illinois at Urbana-Champaign, and a one year non-degree certificate program in business administration at the Kellogg Graduate School of Management, Northwestern University.

Q23. Please describe the basis for your familiarity with the topic of FAC and how it is addressed in the IPEC LRA.

A23. (RMA) I have personal knowledge of the matters discussed herein that relate to the industry and IPEC experience with FAC and the development of programs to predict which plant components may be susceptible to FAC. Since 1992, I have prepared, and managed the preparation of, the CHECWORKS models and other key FAC Program documents at IPEC, including work as a contractor to the prior licensee for IP3, the New York Power Authority. Finally, in my role as a FAC consultant, I work closely with nuclear units across the U.S. and overseas on FAC issues every day. The opportunity to work with the FAC programs at over one hundred nuclear units has given me a broad understanding of the tools, technologies, and practices used in this field. I have reviewed various materials in preparing this testimony, including those portions of Entergy's LRA for IP2 and IP3 relating to Entergy's evaluation of the effects of FAC and the FAC Program. In preparing my testimony, I also reviewed the parties' pleadings on RK-TC-2; the Order Admitting RK-TC-2; and the exhibits submitted by NYS and Riverkeeper that are relevant to my testimony.

II. OVERVIEW OF CONTENTION RIVERKEEPER TC-2

Q24. Are you familiar with Riverkeeper contention TC-2, as originally proposed by Riverkeeper?

A24. (IDM, ABC, NFA, JSH, RMA) We are familiar with Riverkeeper's contention TC-2, and have reviewed RK-TC-2 and Riverkeeper's supporting arguments. *See* Riverkeeper, Inc.'s Request for Hearing and Petition to Intervene in the License Renewal Proceeding for the Indian Point Nuclear Power Plant at 15-23 (Nov. 30, 2007) ("Petition"), *available at* ADAMS Accession No. ML073410093; Declaration of Dr. Joram Hopenfeld in Support of Riverkeeper's Contentions TC-1 and TC-2 (Nov. 28, 2007), *available at* ADAMS Accession No. ML073410093; Riverkeeper, Inc.'s Reply to Entergy's and NRC Staff's Responses to Hearing Request and Petition to Intervene at 13-20 (Feb. 15, 2008), *available at* ADAMS Accession No. ML080560247. Riverkeeper alleges, among other things, that Entergy's FAC Program is deficient because it does not adequately address all ten of the program elements for AMPs identified in NRC license renewal guidance, including NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," Rev. 1 (Sept. 2005) ("SRP-LR") (NYS000195). *See* Petition at 16. Furthermore, Riverkeeper contends that Entergy should not rely on CHECWORKS because of allegedly inadequate "benchmarking" of the IP2 and IP3 CHECWORKS models since the SPUs in 2004 to 2005. *Id.*

Q25. Are you familiar with the Contention TC-2, as admitted by the Board on July 31, 2008?

A25. (IDM, ABC, NFA, JSH, RMA) Yes. As admitted by the Board, RK-TC-2 asserts that: (1) Entergy's AMP for IPEC components affected by FAC is deficient because it does not provide sufficient details to demonstrate that the intended functions of the applicable components will be maintained during the period of extended operation ("PEO"); and (2) Entergy's FAC Program relies on the results from CHECWORKS without benchmarking or a track record of performance at uprated IPEC power levels approved by the NRC in 2004 and 2005. *See Indian Point*, LBP-08-13, 68 NRC 176-77.

Q26. Did Entergy file a Motion for Summary Disposition of Contention TC-2?

A26. (IDM, ABC, NFA, JSH, RMA) Yes. We have reviewed Entergy's "Motion for Summary Disposition of Riverkeeper Technical Contention 2 (Flow-Accelerated Corrosion)" (July 26, 2010) ("Entergy's Motion for Summary Disposition"), *available at* ADAMS Accession No. ML102140430, and its supporting attachments. Entergy's motion presented two arguments: first, that the substantial level of detail present in the IPEC FAC Program is based on NRC guidance and satisfies the requirements of 10 C.F.R. Part 54, and second, that Entergy promptly updated its IPEC CHECWORKS models for post-uprate conditions. *See id.* at 15-24.

Q27. Are you familiar with Riverkeeper's response to Entergy's Motion for Summary Disposition of Contention TC-2, and the filings associated with that motion?

A27. (IDM, ABC, NFA, JSH, RMA) Yes. We have reviewed Riverkeeper's "Opposition to Entergy's Motion for Summary Disposition of Riverkeeper Technical Contention

2 (Flow-Accelerated Corrosion)" (Aug. 16, 2010) ("Riverkeeper's Opposition"), *available at* ADAMS Accession No. ML102371214, and its supporting attachments, including the "Declaration of Dr. Joram Hopenfeld" (Aug. 13, 2010) ("2010 Hopenfeld Declaration"), *available at* ADAMS Accession No. ML102371214.

Riverkeeper's primary arguments in opposition to summary disposition, *see* Licensing Board Memorandum and Order (Ruling on Entergy's Motion for Summary Disposition of Riverkeeper TC-2 (Flow-Accelerated Corrosion) at 6-7 (Nov. 4, 2010) ("Ruling on Summary Disposition") (unpublished), were that: (1) CHECWORKS must be "calibrated or benchmarked separately at each individual power plant and recalibrated when plant conditions change," 2010 Hopenfeld Declaration ¶ 9; (2) "CHECWORKS, as used at Indian Point, has not been properly benchmarked and, accordingly, that CHECWORKS predictions of wall thinning at Indian Point have been 'highly unreliable,'" Ruling on Summary Disposition at 6 (*citing* 2010 Hopenfeld Declaration ¶¶ 11, 12); and (3) that "Entergy does not employ any meaningful tools that, separate and apart from CHECWORKS, would sufficiently manage the aging effects of FAC at Indian Point." 2010 Hopenfeld Declaration ¶ 24; *see also* Riverkeeper's Opposition at 19 (making similar statements).

Q28. Are you familiar with the Board's ruling on that motion?

A28. (IDM, ABC, NFA, JSH, RMA) Yes. We have also reviewed the Board's November 4, 2010 Ruling on Summary Disposition. The Board held that there were two genuine issues of material fact raised by Riverkeeper remaining in dispute, which are whether:

> (1) Entergy's AMP for components affected by FAC is deficient because it does not provide sufficient details to demonstrate that the intended functions of the applicable components will be maintained during the extended period of operation; and (2) Entergy's program relies on the results from CHECWORKS

> without adequate benchmarking or a track record of performance
> at IPEC's power uprate levels.

Ruling on Summary Disposition at 8.

Q29. Have you reviewed Riverkeeper's initial written statement of position, prefiled direct testimony, and supporting exhibits concerning RK TC-2, as filed on December 22, 2011?

A29. (IDM, ABC, NFA, JSH, RMA) Yes, we have reviewed the following documents filed by Riverkeeper on December 22, 2011: Exhibit RIV000002, "Riverkeeper Initial Statement of Position Regarding Contention RK-TC-2 (Flow Accelerated Corrosion)" (Dec. 22, 2011) ("Position Statement"); Exhibit RIV000003, "Pre-Filed Written Testimo[n]y of Dr. Joram Hopenfeld Regarding Contention RK-TC-2 – Flow Accelerated Corrosion" (Dec. 21, 2011) ("Hopenfeld Testimony"); Exhibit RIV000004, "Curriculum Vitae for Dr. Joram Hopenfeld" (Dec. 21, 2002); Exhibit RIV000005, "Report of Dr. Joram Hopenfeld in Support of Riverkeeper Contention RK-TC-2 – Flow Accelerated Corrosion" (Dec. 21, 2011) ("Hopenfeld Report"); and Exhibits RIV000006 through RIV000033, as well as NYS000195, NYS000161, NYS00146C, and NYS00147D.

Q30. What other materials have you reviewed in the preparation for your testimony?

A30. (IDM, ABC, NFA, JSH, RMA) We have reviewed numerous documents in preparing this testimony, including, for example, NRC regulations and guidance documents, such as the SRP-LR (NYS000195), NUREG-1801, Revision 1 (NYS00146A-C), the IPEC license renewal application, NRC Staff's August 2009 Safety Evaluation Report and August 2011 Supplemental Safety Evaluation Report, and EPRI guidance documents, such as Nuclear

Safety Analysis Center's ("NSAC's") NSAC-202L-R3, *Recommendations for an Effective Flow-Accelerated Corrosion Program*," at 1-2 (May 2006) ("NSAC-202L-R3") (RIV000012).

Q31. I show you what has been marked as Exhibit ENT000001. Do you recognize this document?

A31. (IDM, ABC, NFA, JSH, RMA) Yes. It is a list of Entergy's exhibits presented for admission into evidence for this contention, and includes those documents which we referred to, used, or relied upon in preparing this testimony, ENT00015A-B, ENT000030 through ENT000089.

Q32. I show you Exhibits ENT00015A-B, and ENT000030 through ENT000089. Do you recognize these documents?

A32. (IDM, ABC, NFA, JSH, RMA) Yes. These are true and accurate copies of the documents that we have referred to, used and/or relied upon in preparing this testimony. In those cases in which we have attached only an excerpt of a document as an exhibit, that is noted on Entergy's exhibit list.

Q33. How do these documents relate to the work that you do as an expert in forming opinions such as those contained in this testimony?

A33. (IDM, ABC, NFA, JSH, RMA) These documents represent the type of information that persons within our fields of expertise reasonably rely upon in forming opinions of the type offered in this testimony. Many are documents prepared by government agencies, peer reviewed articles, or documents prepared by Entergy or the utility industry. We note at the outset that we cannot offer legal opinions on the language of the NRC regulations or adjudicatory decisions discussed in our testimony. However, reading those regulations and decisions as technical statements, and using our expertise, we can interpret what those documents mean for the FAC Program at IPEC.

III. SUMMARY OF DIRECT TESTIMONY AND CONCLUSIONS

Q34. What is the purpose of your testimony?

A34. (IDM, ABC, NFA, JSH, RMA) The purpose of our testimony is to demonstrate that RK TC-2 lacks factual and technical merit. In our professional opinions, Entergy's FAC Program, as set forth in the LRA and reviewed by the NRC Staff, provides reasonable assurance that, for FAC-susceptible components, the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the current licensing basis ("CLB") during the PEO, as required by 10 C.F.R. § 54.21(a)(3).

In particular, our testimony demonstrates that: (1) the FAC Program at IPEC is being carried out consistent with NRC-endorsed industry guidance and industry experience; (2) CHECWORKS is performing satisfactorily at IPEC and is fulfilling its intended role within the broader FAC Program; (3) that the Entergy FAC Program, which includes several tools, including CHECWORKS, provides sufficient details to demonstrate that the effects of aging due to FAC will be managed during the PEO; (4) that in the six years since the SPUs, Entergy has gained more than sufficient post-uprate experience, and, in any case, CHECWORKS does not require the extended additional benchmarking suggested by Riverkeeper; and (5) Riverkeeper's various criticisms of the IPEC FAC Program based on operating experience lack merit.

Q35. Please describe the scope of your testimony.

A35. (IDM, ABC, NFA, JSH, RMA) Our testimony identifies and describes the pertinent portions of the LRA for IP2 and IP3, including Entergy's discussion of the FAC AMP for IPEC. Our testimony addresses the content of Entergy's LRA relative to the guidance for acceptable AMPs provided in NUREG-1801 (NYS00146A-C) for managing aging effects related to FAC. We also discuss, as relevant, supplemental and clarifying information provided by

Entergy to the NRC after submittal of the LRA. And we show that the LRA complies with 10 C.F.R. Parts 50 and 54, is consistent with the guidance for an acceptable AMP in NUREG-1801 and contains sufficient specificity, notwithstanding Riverkeeper's claims to the contrary.

Our testimony will primarily address the content of Entergy's LRA relative to the guidance in NUREG-1801 (NYS00146A-C). Although the NRC Staff has recently issued Revision 2 of NUREG-1801 (in December 2010), Entergy prepared its April 2007 LRA in accordance with the guidance in NUREG-1801, Revision 1. Therefore, unless otherwise noted, references herein to NUREG-1801 are to Revision 1 of that document. We will testify that Entergy's FAC Program, as set forth in the LRA, is consistent with NUREG-1801, and that, in our professional opinion, the program satisfactorily manages the effects of aging due to FAC at IPEC, as required by 10 C.F.R. § 54.21(a)(3). In addition, because it relies on NSAC-202L-R3, the IPEC FAC Program also meets the intent of the new guidance in NUREG-1801, Revision 2. Finally, we will show that CHECWORKS is one tool among many that Entergy uses at IPEC, and that, for certain components, it can reliably be used to determine and prioritize inspection locations.

Q36. Please summarize the basis of your disagreement with the claims made by Riverkeeper and its proffered expert, Dr. Hopenfeld, in RK-TC-2.

A36. (IDM, ABC, NFA, JSH, RMA) Ultimately, Riverkeeper and Dr. Hopenfeld's arguments address two points. Dr. Hopenfeld's first point is that CHECWORKS does not accurately predict FAC wear rates. *See* Hopenfeld Report at 4-18 (RIV000005). His second point is that the only tool in Entergy's FAC Program for IPEC is CHECWORKS. *See id.* at 21-23. Based on this, he concludes that the IPEC FAC Program is inadequate. *See id.* at 25-26.

Neither of these allegations is true. As to the claim that CHECWORKS does not accurately predict FAC wear rates, CHECWORKS performs its intended role within the FAC Program. As we demonstrate below, the CHECWORKS program provides a screening and prioritization function to assist the program owner in ensuring that inspections are focused on the higher-susceptibility locations. CHECWORKS optimizes the selection process for uninspected components within the FAC Program (*i.e.*, components that have not previously been inspected under the FAC Program) by directing more attention towards components with the highest estimated rates of FAC and relatively less attention on components with lower rates of FAC or relatively more certainty in predictions of FAC wear rates. As we will show, CHECWORKS, as one of many elements in the FAC program, adequately performs its intended screening function.

As to the second claim, that CHECWORKS is the only tool in Entergy's FAC Program for IPEC, the available data and documentation which has been provided or is readily available to Riverkeeper clearly demonstrate that CHECWORKS is only one tool within a robust and multi-faceted FAC Program at IPEC. This is the case at any plant that follows the guidelines in NSAC-202L-R3. CHECWORKS is used at more than 150 nuclear units worldwide, and has helped prevent FAC-related accidents since it and its predecessors have been in use.

Finally, Riverkeeper TC-2 presents no valid information to support the claim that the IPEC FAC program is inadequate. Specifically, as we show, the IPEC FAC Program is effective at identifying loss of material due to FAC and providing for corrective action prior to the loss of component intended function.

Throughout our testimony, we respond to Dr. Hopenfeld's statements in his Report as the Report usually includes the most detailed recitation of his position. *Compare* Hopenfeld Report at 14-15 (RIV000005) *with* Hopenfeld Testimony at 10 (RIV000003) (the latter document presenting a truncated version of the information in the Report). Where appropriate, and in the interest of completeness, we will respond to other statements in Dr. Hopenfeld's Testimony or to information in Riverkeeper's other exhibits.

IV. OVERVIEW OF APPLICABLE PART 54 REQUIREMENTS AND GUIDANCE

Q37. Please identify and briefly describe the NRC aging management review ("AMR") requirements applicable to IPEC systems, structures, and components ("SSCs").

A37. (ABC) 10 C.F.R. Part 54 governs the matters that must be considered in a license renewal proceeding. *See Indian Point*, LBP-08-13, 68 NRC at 67-68. Section 54.4 defines the plant SSCs that are within the scope of the license renewal rule based on their intended functions. Section 54.21(a)(1) defines the structures and components subject to AMR as those structures and components that perform an intended function without moving parts or without a change in configuration or properties (*i.e.*, passive structures and components). 10 C.F.R. § 54.21(a)(1).

Q38. What findings must the NRC make to issue a renewed operating license?

A38. (ABC) Section 54.29(a)(1) requires a finding that the applicant has identified and has taken, or will take, actions for managing the effects of aging during the period of extended operation on the functionality of those structures and components identified as subject to AMR under Section 54.21(a)(1). Specifically, the NRC must find that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the plant's CLB during extended operation. *See* 10 C.F.R. § 54.29(a); *see also id.* § 54.21(a)(3). The standard for this demonstration is one of "reasonable assurance." *See* 10 C.F.R. § 54.29(a); Final Rule; Nuclear Power Plant License Renewal; Revisions, 60 Fed. Reg. 22,461, 22,479 (May 8, 1995) (NYS000016) ("the [license renewal] process is not intended to demonstrate absolute assurance that structures or components will not fail, but rather that there is reasonable assurance that they will perform such that the intended functions . . . are maintained consistent with the CLB"). The Commission has recognized that these adverse aging effects generally are gradual and thus can be managed by programs that ensure sufficient inspections and testing. *See* Final Rule; Nuclear Power Plant License Renewal; Revisions, 60 Fed. Reg. at 22,475 (NYS000016) ("Generally, the changes resulting from detrimental aging effects are gradual. Licensees have ample opportunity to detect these degradations through performance and condition monitoring program").

Q39. What guidance has the NRC issued to assist in implementing the requirements of 10 C.F.R. Part 54?

A39. (ABC) The two primary guidance documents issued by the NRC Staff are NUREG-1801 (NYS00146A-C) and the SRP-LR (NYS000195). The latter is primarily NRC Staff guidance for use in its review of LRAs.

Q40. Please describe the function of SRP-LR as it relates to AMPs.

A40. (ABC) The SRP-LR provides guidance to NRC staff for conducting their review of LRAs. It provides acceptance criteria for determining whether the applicant has met the requirements of the NRC's regulations in 10 C.F.R. § 54.21(s*ee* SRP-LR § 3.1.2 (NYS000195)) including acceptable methods for identifying those SSCs within the scope of license renewal that are subject to aging effects, and defining the ten program elements that constitute an effective AMP. One acceptable way to manage a specific aging effect or effects for license renewal is to use an AMP that is consistent with NUREG-1801. *See* SRP-LR § 3.0.1 (NYS000195).

Q41. Please briefly describe the origin and purpose of NUREG-1801.

A41. (ABC) In an NRC Staff paper, SECY-99-148, "Credit for Existing Programs for License Renewal," dated June 3, 1999, the Staff described options for crediting existing licensee programs at operating plants to satisfy the requirements of Part 54. By a Staff Requirements Memorandum dated August 27, 1999, the Commission directed the Staff to develop NUREG-1801, which the Staff first issued in 2001, to document the Staff's evaluation of existing AMPs. NUREG-1801 identifies methods that the NRC has found acceptable for satisfying the requirements of 10 C.F.R. Part 54. An applicant may reference NUREG-1801 in an LRA to show that the programs proposed for the applicant's facility correspond to those reviewed and approved in NUREG-1801. *See* NUREG-1801, Rev. 1, at 3 (NYS00146A).

Q42. Please briefly describe the basic format and contents of NUREG-1801.

A42. (ABC) NUREG-1801 identifies generic AMPs that the NRC Staff has found acceptable for satisfying the requirements of 10 C.F.R. Part 54, based on the experiences and evaluations of existing programs at operating plants during the initial license period. *See* NUREG-1801, Rev. 1, at 1 (NYS00146A). NUREG-1801 includes tables summarizing various structures and components, the materials from which they are made, the environment to which they are exposed, the applicable aging effect (*e.g.*, wall-thinning due to FAC or other forms of corrosion), the AMP found acceptable to manage the particular aging effect in that component, and whether "further evaluation" is necessary. *Id.* at 5. The evaluation results documented in NUREG-1801 indicate that many current term existing licensee programs adequately manage aging effects relevant to particular structures or components for license renewal. *See id.* at 4.

Q43. Does NUREG-1801 specifically address FAC?

A43. (ABC, JSH, RMA) Yes. NUREG-1801 tables identify wall thinning due to FAC as an aging effect requiring management for a number of systems and identify the FAC Program to manage this aging effect. NUREG-1801, Section XI.M17 describes an acceptable FAC program based on the EPRI guidelines in NSAC-202L-R2. *See* NUREG-1801, Rev. 1, at XI M-61 (NYS00146C). While NUREG-1801, Revision 1 references NSAC-202L-R2, the same description generally applies to the more recent update to EPRI's guidance, NSAC-202L-R3.

Q44. What is NSAC-202L-R3?

A44. (ABC, IDM, JSH, RMA) NSAC-202L-R3 is a detailed guidance document that describes the elements of an effective FAC program (Chapter 2), identifies the need for and suggested scope of program implementation procedures and documentation (Chapter 3), recommends specific FAC program tasks (Chapter 4), and explains how to develop a long-term strategy for reducing plant FAC susceptibility (through the use of FAC-resistant materials, optimization of water chemistry, and system design changes) (Chapter 5). *See generally* NSAC-202L-R3 (RIV000012). A FAC program conforming to the EPRI guidelines in NSAC-202L-R3 must include procedures and administrative controls to provide reasonable assurance of the structural integrity of all carbon steel lines containing high-energy fluids (two-phase as well as single-phase). *See* NUREG-1801, Rev. 1, at XI M-61 (NYS00146C); NUREG-1801, "Generic Aging Lessons Learned (GALL) Report, Rev. 2," at XI M17-1 (Dec. 2010) ("NUREG-1801, Revision 2") (NYS00147D). NSAC-202L-R3 is the version that Entergy relies upon in its FAC Program. *See* Entergy, EN-DC-315, Flow Accelerated Corrosion Program, Rev. 6, at 16 (Mar. 1, 2010) ("EN-DC-315") (ENT000038).

Q45. What are the significant differences between NSAC-202L-R2 and NSAC-202L-R3?

A45. (IDM, JSH, RMA) The updated guidance in NSAC-202L-R3 is evolutionary in nature and did not change the document in any fundamental way (*i.e.*, foundational elements. such as the basis for excluding components from examination based on, for example, materials of fabrication, the inspection criteria, or acceptance criteria remain the same). Revision three does, however, provide enhanced guidance in a number of areas, including application of operating experience and how to perform sample expansion if indications of loss of material by FAC or other mechanisms are detected. *See* NUREG-1930, "Safety Evaluation Report [("SER")] Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3," at 3-23 to -24 (Nov. 2009), *available at* ADAMS Accession No. ML093170671 (NYS00326B).

Q46. How does NUREG-1801, Revision 1, Section XI-M17 interface with NSAC-202L-R2?

A46. (ABC, IDM, JSH, RMA) NUREG-1801, Revision 1, Section XI.M17 references NSAC-202L-R2 as describing an effective FAC program. The program predicts, detects, and monitors FAC in plant piping and piping components, such as tees, elbows and reducers. *See* NUREG-1801, Rev. 1, at XI M-61(NYS00146C). Such a program includes the following key actions: (1) conducting an analysis to determine critical locations, for which CHECWORKS is one tool used among many; (2) performing limited baseline inspections to determine the extent of thinning at these locations; and (3) performing follow-up inspections to confirm the predictions, or repairing or replacing components as necessary. NUREG-1801 Section XI.M17 states that the use of a predictive code, such as CHECWORKS, constitutes one aspect of an effective FAC program. *See id*.

Q47. Is there a recent revision to NUREG-1801 that was issued following the preparation and review of the IPEC LRA?

A47. (ABC, IDM, JSH, RMA) Yes. In December 2010, the NRC Staff issued NUREG-1801, Revision 2. This revision was issued three years after the IPEC LRA was submitted, and more than a year after the NRC Staff issued its Safety Evaluation Report on the IPEC LRA in August 2009. Therefore, the IPEC LRA was prepared based on the guidance in NUREG-1801, Revision 1. The NRC Staff reviewed the IPEC LRA against NUREG-1801, Revision 1, found the Entergy FAC Program to be consistent with that document, and based on this finding, concluded that the effects of aging due to FAC will be adequately managed so that the intended functions will be maintained consistent with the CLB for the PEO, as required by 10 C.F.R. § 54.21(a)(3). *See* SER at 3-30 (NYS000326B).

Q48. With respect to the IPEC FAC Program, what are the significant changes in NUREG-1801, Revision 2?

A48. (ABC, IDM, JSH, RMA) The primary difference between the two revisions is that NUREG-1801, Revision 2 permits an applicant to rely on either NSAC-202-L-R2 or -R3 as the basis for its FAC program, while only NSAC-202L-R2 is approved in NUREG-1801. *See* NUREG-1801, Rev. 2, at IX-31 (NYS00147C); NUREG-1801, Rev. 1 at IX-30 (NYS00146C). As explained further below, in the IPEC LRA, Entergy took an exception to the guidance in NUREG-1801, Revision 1, to rely upon the guidance in NSAC-202L-R3 for the FAC Program. *See* NL-07-153, Letter from Fred R. Dacimo, Entergy, to NRC, "Amendment 1 to License Renewal Application (LRA)," Attach. 1, at 46-48 (Dec. 18, 2007) (NYS000159). Because it relies on NSAC-202L-R3, the IPEC FAC Program meets the intent of the new guidance in NUREG-1801, Revision 2.

V. ENTERGY'S AGING MANAGEMENT PROGRAM FOR FAC-SUSCEPTIBLE PIPING AT IPEC

A. Technical Background on FAC

Q49. What is FAC?

A49. (JSH, RMA, IDM) Consistent with NSAC-202L-R3, Entergy's corporate FAC Program, EN-DC-315, Rev. 6, defines FAC as the "[d]egradation and consequent wall thinning of a component by a dissolution phenomenon, which is affected by variables such as temperature, steam quality, steam/fluid velocity, water chemistry, component material composition and component geometry." EN-DC-315 at 6 (ENT000038); *see also* NSAC-202L-R3 at 1-2 (RIV000012) ("FAC leads to wall thinning (metal loss) of steel piping exposed to flowing water or wet steam."). Flow-accelerated corrosion is caused by the combination of the thermodynamic instability (*i.e.*, the tendency to dissolve) of iron oxides under power plant conditions and the presence of a flowing water or water and steam mixture. The flowing stream continually transports iron ions (chiefly ferrous ions, *i.e.*, Fe^{++}) away from the oxide surface resulting in a loss of material from the oxide-covered, steel surface. *See Flow-Accelerated Corrosion in Power Plants* at 2-15 to -18 (ENT00036A). If FAC is not detected, then the piping or vessel walls will slowly become progressively thinner until they can no longer withstand internal pressure and other applied loads.

Although in the past FAC was described as "erosion-corrosion," FAC is a *chemical* corrosion phenomenon that is distinct from other *mechanical* or *erosive* phenomena that may cause pipe wall thinning, such as cavitation, liquid droplet impingement, and solid particle erosion. *See Flow-Accelerated Corrosion in Power Plants* at 2-1 to -8 (ENT00036A). For FAC-susceptible systems, such as the condensate, feedwater, extraction steam and major drain systems, FAC is by far the predominant degradation mechanism. *See* Frank Ammirato, *Status of NDE Research in the US--Contributions of NDE to Reactor Safety and Implementation of NDE Technology* at 3 (May 1999) (ENT000039).

Q50. What variables affect the rate of FAC?

A50. (JSH, RMA, IDM) Wall-thinning caused by FAC occurs under specific water chemistry conditions, thus, the use of proper water chemistry will dramatically reduce the rate of FAC. As explained in *Flow-Accelerated Corrosion in Power Plants*, the major influencing factors can be classified in three categories: (1) hydrodynamic factors, "i.e. flow velocity, pipe roughness, geometry of the flow path, steam quality or void fraction for two-phase flows"; (2) environmental factors, "i.e. temperature, pH, reducing agent and oxygen concentration, oxidation-reduction potential, water impurities"; and (3) metallurgical factors, "mainly the chemical composition of the steel." *Flow-Accelerated Corrosion in Power Plants* at 3-1 to -2 (ENT00036A); *see also* NSAC-202L-R3 at 1-2 (RIV000012) ("[t]he rate of metal loss depends on a complex interplay of many parameters including water chemistry, material composition, and hydrodynamics"). FAC primarily affects carbon steel components in the presence of purified flowing water or wet steam. It does not affect steels containing other fluids, such as oil. Steels containing appreciable amounts of chromium, such as stainless steel, are immune to FAC. *See* NSAC-202L-R3 at 4-3 (RIV000012).

Q51. Dr. Hopenfeld asserts that "FAC includes wall thinning by impingement corrosion, electrochemical corrosion, erosion-corrosion, cavitation-erosion, and chemical dissolution." Hopenfeld Report at 2 (RIV000005). Do you agree?

A51. (JSH, RMA, IDM) Not completely. Dr. Hopenfeld's definition is not a standard industry or academic definition of FAC. The term "flow-accelerated corrosion" was developed in 1992 to unambiguously refer to one particular corrosion phenomenon. As explained in response to Question 49, above, FAC describes the chemical dissolution of the protective oxide surface of carbon and low alloy steels.

Nevertheless, three of the five terms mentioned by Dr. Hopenfeld are synonymous with FAC: "electrochemical corrosion", "erosion-corrosion" and "chemical dissolution," and are therefore encompassed by our testimony on FAC. *See Flow-Accelerated Corrosion in Power Plants* at 2-2 (ENT00036A). "Impingement corrosion" and "cavitation erosion," however, are separate from FAC. Aside from FAC, mechanical or erosive damage to piping surfaces can occur by various means, but as we show throughout our testimony, the FAC Program addresses wall-thinning, whether caused by FAC or not.

As to the remaining two items in Dr. Hopenfeld's definition, "impingement corrosion" is more properly called "liquid droplet impingement erosion." This damage mechanism is *erosion* (a mechanical phenomenon) not *corrosion* (a chemical phenomenon). Liquid droplet impingement erosion does occasionally occur in FAC-susceptible systems. However, unlike FAC, it occurs under off-normal conditions (*e.g.*, downstream of a leaking valve). Also, this mechanism causes small holes in components, as the damaged areas are not widespread. Liquid droplet impingement erosion is included in the IPEC FAC program based on operating experience and engineering judgment; for instance, inspections have been added to the FAC Program based on operating experience with this mechanism.

Finally, "cavitation erosion" is a localized erosion phenomenon resulting from inadequate design. Cavitation erosion rarely occurs in systems susceptible to FAC, and is more properly treated as a design issue, not an aging mechanism. Once cavitation is identified, the situation is normally corrected as part of ongoing operations and maintenance activities. *See Vt. Yankee*, LBP-08-25, 68 NRC at 860, 864; *see also* EPRI, Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools, Rev. 4 at 3-7 (Jan. 2006) (NYS00320A).

Q52. In the same paragraph, Dr. Hopenfeld asserts that, "[i]n many instances both mechanisms [(*i.e.*, physical and chemical degradation)] occur simultaneously." Hopenfeld Report at 2 (RIV000005). Do you agree?

A52. (JSH, RMA) No. Based on our more than 45 years of experience with FAC, this statement is incorrect. Normally, observed damage in FAC-susceptible systems is caused either by a chemical process (*i.e.*, FAC) or an erosive process, but not both. Combinations of mechanisms in FAC-susceptible systems are rare. *See Flow-Accelerated Corrosion in Power Plants* at 2-2 (ENT00036A). In particular, erosion in combination with FAC does not occur in carbon steel piping because the oxide layer that is necessary for FAC cannot form if erosion is occurring. *See id.* With regard to other wall-thinning phenomena, it is important to emphasize that FAC is the predominant degradation mechanism in FAC-susceptible systems. Moreover, operating experience shows that FAC is also responsible for most of the significant failures of FAC-susceptible components. *See*, *e.g.*, Steve Gosselin, Pac. Nw. Nat'l Lab., *Fatigue in Operating Nuclear Power Plants Components After 60 Years* at 3 (Feb. 2008), *available at* ADAMS Accession No. ML080600852 (ENT000040) (presentation slides).

Q53. In his testimony Dr. Hopenfeld states that FAC can occur at a "non-linear rate." Hopenfeld Report at 2 (RIV000005). Do you agree?

A53. (JSH, RMA) No. Dr. Hopenfeld presents no data or technical basis to support his assertion that the rate of FAC can increase over time. In fact, there is ample laboratory testing and field experience to indicate that under constant operating conditions and water chemistry, the rate of FAC is essentially constant over time. *See Flow-Accelerated Corrosion in Power Plants* at B-3 to B-4 (ENT00036B). Notably, Dr. Hopenfeld raised the issue of non-linear FAC wear rates at the *Vermont Yankee* hearing, where the Board disagreed with him, finding that the "FAC wear rates are constant with time, since the variation in wear rates with roughness is small and, given the existing age of the piping, further surface changes are likely to be minimal." *Vt. Yankee*, LBP-08-25, 68 NRC at 892. For the reasons we have just stated, this conclusion applies equally to IPEC.

Q54. Do you agree with Dr. Hopenfeld's stated view that FAC is a "local phenomenon"? Hopenfeld Report at 2 (RIV000005).

A54. (JSH, RMA) No. It would be incorrect to contend that FAC is "a local phenomenon," as though each component were independent from all others as far as the rate of FAC was concerned. *See* Hopenfeld Report at 2 (RIV000005).

While FAC is influenced by local conditions, it is a line-level phenomenon (*i.e.*, when one component is wearing, usually others along the same line are wearing at similar rates, due to the similar operating conditions), except occasionally near geometric discontinuities. *See Flow-Accelerated Corrosion in Power Plants* at 1-5, 4-2 (ENT00036A).

In sum, Dr. Hopenfeld does not correctly describe the causes and process of FAC.

B. The IPEC FAC Program as Described in the LRA

Q55. Please describe how FAC is addressed in the IPEC LRA.

A55. (IDM, ABC, NFA) Chapter 2 of the IPEC LRA summarizes Entergy's detailed assessment to identify those structures and components that require aging management review. Chapter 3 identifies loss of material due to FAC as an applicable aging effect for certain plant components. *See* LRA at 3.3-32 (ENT00015A); *id.* at 3.4-3 to -6 (ENT00015B). The appendices to the LRA contain descriptions of Entergy's FAC Program.

Appendix A presents information required by 10 C.F.R. § 54.21(d) relating to the AMP for FAC that supplements the updated final safety analysis report ("UFSAR") for IPEC. *See* LRA, App. A at A-1 (ENT00015B). Specifically, the supplement to the UFSAR, presented in section A.2 of Appendix A, contains a summary description of the program for managing the effects of aging due to FAC during the PEO. *See id.* at A-24. Appendix A states that this information will be incorporated into the UFSAR following issuance of the renewed operating licenses. *Id.* at A-1.

Appendix B to the LRA describes those AMPs credited for managing aging effects during the PEO. *See* LRA, App. B at B-1 (ENT00015B). Section B.1.15 describes the IPEC FAC Program and indicates that it is consistent with, and takes no exceptions to, the program described in Section XI.M17 of NUREG-1801. *See* NUREG-1801, Rev. 1, at XI.M17 (NYS00146C); *see also* LRA, App. B at B-54 (ENT00015B); NUREG-1801, Rev. 1, at XI M-61 to XI M-62 (NYS00146C). LRA Section B.1.15 further states that the IPEC FAC Program is based on EPRI guidelines for an effective FAC program contained in NSAC-202L-R2, "Recommendations for an Effective Flow-Accelerated Corrosion Program." *See* LRA, App. B at B-54 (ENT00015B).

Q56. Is the IPEC FAC Program consistent with the recommendations in NUREG-1801?

A56. (IDM, ABC, JSH, RMA) Yes. The IPEC FAC Program description in LRA Section B.1.15 incorporates by reference all ten program elements or attributes identified in NUREG-1801, Revision 1, Section XI.M17. This is consistent with the purpose of NUREG-1801, Revision 1 which is to describe AMPs that the NRC Staff has found acceptable for satisfying the requirements of 10 C.F.R. Part 54. *See* NUREG-1801, Rev. 1 at iii. The program elements are: (1) Scope of the Program, (2) Preventive Actions, (3) Parameters Monitored or Inspected, (4) Detection of Aging Effects, (5) Monitoring and Trending, (6) Acceptance Criteria, (7) Corrective Actions, (8) Confirmation Process, (9) Administrative Controls, and (10) Operating Experience. *See* NUREG-1801, Rev. 1 at XI M-61 to -62. Entergy compared the IPEC FAC Program to NUREG-1801, Revision 1, Section XI.M17 with respect to each of the ten program attributes. The results of that evaluation are documented in the original LRA Section B.1.15, which indicates that the IPEC FAC Program includes each of the ten program attributes without exception. The results also are documented in Entergy Engineering Report No. IP-RPT-06-LRD07, "Aging Management Program Evaluation Results – Non-Class 1 Mechanical, Revision 5" at 106-13 (Mar. 18, 2009) ("AMP Evaluation Report") (RIV000014).

Q57. Did Entergy amend its FAC Program description in the LRA as a result of the NRC Staff's review?

A57. (IDM, ABC) Yes. On December 18, 2007, in response to NRC Audit Item 156, Entergy amended the "scope of program" and "detection of aging effects" program elements to identify use of NSAC-202L-R3, dated August 2007, as an "exception" to NUREG-1801 Section XI.M17, which references the prior Revision 2 of NSAC-202L. *See* NL-07-153, Attach. 1, at

46-48 (NYS000159); NRC Audit Report for Plant Aging Management Programs and Reviews at 22-23 (Jan. 13, 2009) ("NRC Audit Report"), *available at* ADAMS Accession No. ML083540662 (ENT0000041). NSAC-202L-R3, Riverkeeper Exhibit RIV000012, incorporates lessons learned following the publication of Revision 2. As stated in NSAC-202L-R3 at 5,

> This revision of NSAC-202L contains recommendations updated with the worldwide experience of members of the CHECWORKS™ Users Group (CHUG), plus recent developments in detection, modeling, and mitigation technology. These recommendations are intended to refine and enhance those of the earlier versions, *without contradiction*, so as to ensure the continuity of existing plant FAC programs.

(Emphasis added).

Entergy did not initially take an exception to NUREG-1801, Revision 1 because, as stated, the implementing guidance in NSAC-202L-R3 remained consistent with the guidance of NSAC-202L-R2.

Q58. What were the results of the NRC Staff's review of the IPEC FAC Program in the LRA?

A58. (IDM, ABC) In its final Safety Evaluation Report, the NRC Staff concluded that the IPEC FAC Program elements are acceptable and consistent with the ten program elements in NUREG-1801, Revision 1, Section XI.M17. *See* SER at 3-22 to -30 (NYS00326B). The Staff also found NSAC-202L-R3 to be an appropriate guidance document for the program. *See id.* at 3-24. The SER states that NSAC-202L-R3 provides enhanced guidance on: (1) applying relevant industry experience and plant-specific experience as an additional basis for selecting and scheduling additional components for UT or RT inspection; (2) expanding samples if relevant indications of loss of material by FAC or other loss of material mechanisms are detected; (3) inspecting in-scope tanks, cross-around piping, and small-bore piping; and (4) applying UT and RT as volumetric inspection techniques for these programs. *See* SER at 3-22 to 3-30

(NYS00326B). Although the NRC Staff later supplemented its SER, the Supplemental SER does not further address the FAC Program. *See* NUREG-1930, Supp. 1, Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3 (Aug. 2011) ("SSER") (NYS000160).

C. Key Attributes of the IPEC FAC Program

Q59. Is the IPEC FAC Program a new program for license renewal?

A59. (IDM, ABC) No. The FAC Program is an established IPEC program, based on a common fleet-wide Entergy program, which Entergy will continue to implement during the PEO. Since establishing its fleet-wide program, Entergy has updated and revised its FAC program documents as appropriate. *See* EN-DC-315 at 13 (ENT000038). In addition, as specified in NSAC-202L-R3, under the FAC Program, Entergy continually draws from extensive industry and IPEC operating experience, including NRC information notices, Bulletins, and Generic Letters, inspection data from recent inspections, and peer reviews. *See* LRA, App. B at B-54 to -55 (ENT00015B); SER at 3-29 to -30 (NYS00326B). The FAC Program also addresses changes in operating parameters, including changes such as those associated with the SPU, as discussed further in response to Questions 90 through 92.

Q60. How was the Entergy FAC Program developed and implemented?

A60. (IDM, ABC, RMA) Initially, Entergy developed its fleet-wide FAC Program in accordance with the elements outlined in the responses to NRC Generic Letter 89-08 and related correspondence. *See* NRC Generic Letter (GL) 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning" (May 2, 1989) (ENT000042); Letter from Murray Selman, Consolidated Edison Co., to William Russell, NRC, Attach. A (Sept. 11, 1987) (Response to NRC Bulletin 87-01 "Thinning of Pipe Walls in Nuclear Power Plants"), *available at* ADAMS Accession No. ML100331261 (ENT000043); Letter from William Josiger, New York Power Authority, to

William Russel, NRC, "NRC Bulletin No. 87-01: Thinning of Pipe Walls in Nuclear Power Plants," Attach. 1 (Sept. 15, 1987) (Response to NRC Bulletin 87-01 "Thinning of Pipe Walls in Nuclear Power Plants"), *available at* ADAMS Accession No. ML100360276 (ENT000044); Letter from Michael Miele, Consolidated Edison Co., to NRC, "Response to Generic Letter GL 89-08, 'Erosion/Corrosion Induced Pipe Wall Thinning'" (July 20, 1989), *available at* ADAMS Accession No. ML100331107 (ENT000045); Letter from John Brons, New York Power Authority, to NRC, "Response to Generic Letter GL 89-08, Erosion/Corrosion Induced Pipe Wall Thinning" (July 21, 1989), *available at* ADAMS Accession No. ML093440934 (ENT000046); Letter from Stephen Bram, Consolidated Edison Co., to NRC, "Response to Request for Additional Information Regarding Response to Generic Letter 89-08, 'Erosion/Corrosion-Induced Pipe Wall Thinning'" (Jan. 17, 1990) (ENT000047).

The program is implemented under a fleet-wide procedure EN-DC-315, Revision 6, "Flow Accelerated Corrosion Program," attached as Exhibit ENT000038, which governs the FAC programs at Entergy's nuclear power plants. In developing EN-DC-315, Entergy reviewed best practices for the FAC Programs at all Entergy sites and guidance from the EPRI CHUG. *See*, *e.g.*, EN-DC-315 at 4 (ENT000038) (citing EPRI, CHUG Position Paper No. 3, *A Summary of Tasks and Resources Required to Implement an Effective Flow-Accelerated Corrosion Program* (June 1999)). EN-DC-315 implements the recommendations of NUREG-1801, Revision 1 (NYS00146A-C) and the more detailed EPRI report, NSAC-202L-R3 (RIV000012).

Q61. What plant components are included in the IPEC FAC Program?

A61. (IDM, RMA) The IPEC FAC Program applies to power piping systems. Engineering and inspection activities are focused on carbon steel plant components susceptible to FAC. *See* EN-DC-315 at 18 (ENT000038). The program requires inspections of single-phase and two-phase piping components in both safety-related and non-safety-related systems. The System Susceptibility Evaluation ("SSE") Reports for IP2, No. 0700.104-02, Rev. 2 (Oct. 14, 2011) ("IP2 SSE Report 0700.104-02") (ENT000048), and for IP3, No. 0700.104-17, Rev. 2 (Oct. 14, 2011) ("IP3 SSE Report 0700.104.17") (ENT000049), provide a comprehensive list of all systems and components covered by the FAC Program at each IPEC unit, and are periodically updated to reflect and address applicable design and operational changes, such as changes in operating conditions due to an SPU. The most recent Steam Feedwater Analysis ("SFA") Reports for IPEC, CHECWORKS SFA Model Calculation for IP2, No. 0705.101-01, Rev. 2 (July 8, 2010) ("IP2 SFA Report 0705.101-01") (ENT000050) and CHECWORKS SFA Model Calculation for IP3, No. 0705.100-01, Rev. 2 (Aug. 2, 2011) ("IP3 SFA Report 0705.100-01") (ENT000051), provide a complete list of CHECWORKS-modeled components for IP2 and IP3, respectively, in Appendices D and E of each report.

Q62. Are all of the plant components covered by the IPEC FAC Program safety-related?

A62. (IDM, RMA, NFA) No. The IPEC FAC Program includes both safety-related piping and non-safety-related power piping. *See* LRA, App. B at B-54 (ENT00015B). The large majority of components in the FAC Program are non-safety-related. *See generally* IP2 SSE Report 0700.104-02 (ENT000048); IP2 Susceptible Non-Modeled (SNM) Report 0700.104-03, Rev. 2 (Oct. 14, 2011) ("IP2 SNM Report 0700.104-03") (ENT000052); IP3 SSE Report 0700.104-17 (ENT000049); IP3 Susceptible Non-Modeled (SNM) Report 0700.104-18, Rev. 2 (Oct. 14, 2011) ("IP3 SNM Report 0700.104-18") (ENT000053).

Q63. On pages 2 and 19 of his Report, Dr. Hopenfeld cites certain ASME (and apparently ANSI) Code provisions. Do these code provisions apply to the IPEC FAC Program?

A63. (IDM, NFA) No. Although Dr. Hopenfeld cites to American National Standards Institute ("ANSI") Code, Section B31.3 and ASME Code Section III on pages 2 and 19 of his Report, neither of those codes apply to the components within the scope of the FAC program at IPEC. The ANSI Section B31.3 Code cited by Dr. Hopenfeld provides the rules for process piping used in petroleum refineries and other industries, but is not used in nuclear applications. Rather, the design rules applicable to the systems covered by the IPEC FAC Program are the rules of the USAS/ANSI B31.1, 1955 Edition (IP2) and the 1967 Edition (IP3). *See generally* ANSI Code, Section B31.1 (ENT000054).

Q64. Does the scope of the IPEC FAC Program include components inside the steam generators?

A64. (IDM, NFA, ABC) No. The IPEC FAC Program does not cover components inside the steam generators, such as the feedwater distribution ring mentioned by Dr. Hopenfeld on page 24 of his Report. Instead, those components are inspected under the Steam Generator Integrity Program, the adequacy of which is unchallenged in this contention. *See* LRA, App. B at B-118 (ENT00015A); *id.* at 3.1-157, 3.1-176 (Tables. 3.1.2-4-IP2, 3.1.2-4-IP3) (ENT00015A). Thus, when Dr. Hopenfeld observes that "Entergy has not provided data on CHECWORKS predictions for components inside the steam generators," he is correct—no such data exist. Hopenfeld Report at 24 (RIV000005).

Q65. Please describe in general terms the inspection methods and the scope of inspections undertaken as part of the IPEC FAC Program.

A65. (IDM, RMA) UT thickness measurements performed in accordance with approved procedures are the primary method used to determine pipe wall thickness. *See* EN-DC-315 at 20 (ENT000038). Radiography and other nondestructive examination techniques are also occasionally employed. *See id.* at 21. The IPEC FAC Program requires that piping and piping component inspections be conducted by personnel who are qualified and certified in accordance with the requirements of the ASME Code. In addition, the IPEC FAC Program provides detailed instructions for selecting components for inspections, evaluating inspection data against specified acceptance criteria, evaluating any components that fail to meet the acceptance criteria, expanding the inspection sample in the case of unexpected thinning, and calibrating the CHECWORKS models. *See* LRA, App. B at B-54 (ENT00015B); EN-DC-315 at 3, 10-28 (ENT000038).

It should be noted that for each component inspected with UT, Entergy typically records dozens or hundreds of individual thickness readings, depending on the size and type of the component being inspected. *See* EN-DC-315 at 19 (ENT000038).

Q66. What other types of details are provided in the IPEC FAC Program's governing document, EN-DC-315?

A66. (IDM, RMA) EN-DC-315 also details the procedures and processes for ensuring that changes or modifications to the plant system configuration or water chemistry are accurately reflected in CHECWORKS. *See* EN-DC-315 at 15-16 (ENT000038). It also delineates the specific responsibilities and qualifications of the FAC Engineer and other key personnel and defines program documentation requirements. *See id*. at 3, 10-28.

Q67. What is the system susceptibility evaluation and how is it used?

A67. (IDM, JSH, RMA) EN-DC-315 and NSAC-202L-R3 describe the process for the initial identification of FAC-susceptible systems. *See* EN-DC-315 at 17 (ENT000038); NSAC-202L-R3 at 4-3 to -5 (RIV000012). The system susceptibility evaluation ("SSE") of plant piping identifies FAC-susceptible piping. Certain systems are excluded from the scope of the FAC Program based on material of construction (*e.g.*, stainless steel is not susceptible to FAC), process fluid (*e.g.*, lube oil systems are excluded), water chemistry (*e.g.*, raw water systems are excluded), operating temperature (*i.e.*, < 200ºF single-phase systems are excluded), and operating time (*i.e.*, systems that operate < 2% of plant operating time are typically excluded). The criteria used in this evaluation are taken from NSAC-202L-R3, as described further below in response to Question 72. This susceptibility evaluation is documented in Entergy reports and revised periodically. Specifically, the SSE reports and susceptible non-modeled reports discussed later in our testimony must be updated every two plant operating cycles, as mandated by EN-DC-315, Revision 6 at 28 (ENT000038). These updates reflect changes in plant configuration, such as modifications, replacement of components with corrosion-resistant material, or changes in operating practice. *See*, *e.g.,* IP2 SSE Report 0700.104-02 (ENT000048); IP2 SNM Report 0700.104-03 (ENT000052); IP3 SSE Report 0700.104-17 (ENT000049); IP3 SNM Report 0700.104-18 (ENT000053).

Q68. For systems that are classified as FAC-susceptible, are different categories of piping evaluated differently under NSAC-202L-R3?

A68. (IDM, JSH, RMA) Yes. Once a piping system, sub-system, or line is classified as "FAC-susceptible," that piping is further subcategorized as either "modelable" or "susceptible non-modeled" ("SNM"). Specifically:

1. Modelable: This category of piping includes the majority of large-bore (nominal pipe size greater than two inches) piping that typically poses the highest consequences of failure. It has known operating conditions and operating times. *See* NSAC-202L-R3 at 4-5 (RIV000012). This category typically includes the major piping systems, including: condensate, feedwater, extraction steam, and major drains.
2. Susceptible Non-Modeled: This category of piping contains all FAC-susceptible piping that, for one reason or another, is not suitable for modeling in CHECWORKS. Unsuitability for modeling may be due to nonstandard configuration, unknown or non-steady-state operating times or conditions, entrained non-condensable gases, or operating parameters outside the range of CHECWORKS validity. Examples of SNM lines include emergency drains, vent lines, and balancing lines. *See* NSAC-202L-R3 at 4-7 to -8. (RIV000012). This category of lines also includes small-bore (nominal pipe size less than or equal to two inches) piping, which is generally less of a safety concern and where replacement may be preferred over inspections. Appendix A of NSAC-202L-R3 provides detailed recommendations regarding the FAC Program for small-bore piping.

Q69. Does the industry guidance specify that inspection locations should be determined in the same manner for both piping categories?

A69. (IDM, JSH, RMA) No. Under the guidance in NSAC-202L-R3, for modeled piping, inspection locations are determined based on CHECWORKS model results, along with trending of past measurements, operating experience (including industry experience), and engineering judgment.

For susceptible non-modeled piping ("SNM"), which includes both large-bore and small-bore piping, inspection locations are selected based on SNM-specific susceptibility rankings, consequences of failure category, operating experience, engineering judgment, and trending of past measurements. *See* NSAC-202L-R3 at 4-7 to 4-8 (RIV000012).

As explained below, experience has shown that this approach has led to effective FAC programs throughout the industry.

Q70. Do we know which piping locations at IPEC are most susceptible to FAC?

A70. (IDM, JSH, RMA) Yes. The piping locations at IPEC that are most susceptible to FAC are locations with two-phase flow and high moisture content, lines which contain saturated liquid that flashes to steam due to changes in pressure and certain areas with high flow velocity and high turbulence. *See* IP2 SNM Report 0700.104-03 at 11-13 (ENT000052); IP3 SNM Report 0700.104-18 at 11-13 (ENT000053). These locations have been confirmed to be the most susceptible based on two decades of inspections performed under the FAC programs at IPEC and other plants. Some examples of the IPEC systems that are most susceptible to FAC are: high pressure extraction steam piping and areas downstream of level control valves in heater and moisture separator reheater ("MSR") drain systems.

Q71. Has Entergy replaced any FAC-susceptible piping components at IPEC?

A71. (IDM) Yes. Certain IPEC piping components which are most susceptible to FAC have been replaced with FAC-resistant materials (*e.g.,* stainless steel, chromium-molybdenum steel). For example, following the detection of wall thinning in the vent chamber drain and high pressure turbine drain components at IP3 in 2005, and in a steam trap pipe at IP2 in 2006

Entergy replaced the affected components. Other examples include the following components that have also been replaced or clad with FAC-resistant materials:

- At IP3, all extraction steam piping between the high pressure turbine and the high-pressure feedwater heaters, with the exception of the turbine exit nozzles and feedwater inlet nozzles were replaced;
- At IP3, the feedwater inlet nozzles were internally clad with stainless steel material;
- At IP3, the steam generator blow down recovery lines were replaced;
- At IP3, portions of the lines from the pressure control valve to the sixth point feedwater heaters (six lines) were replaced; and
- At both units, a large number of FAC-susceptible small-bore piping components, such as the main steam traps and extraction steam trap piping and portions of the high pressure turbine drains were replaced.

Additional details regarding Entergy's replacement of FAC-susceptible materials are provided in Entergy's responses to NRC license renewal aging management audit Items 43 and 49 in the NRC Audit Report at 13-14 & 20-22 (ENT000041), and the NRC Staff's SER at 3-26 to -27 & 3-29 (NYS000326B). The most recent CHECWORKS SFA models also provide, in Appendix E of each report, a list of the modeled piping that has been replaced at IP2 and IP3. *See* IP2 SFA Report 0705.101.01, App. E (ENT000050); IP3 SFA 0705.100-01, App. E (ENT000051).

Q72. How are components selected for FAC inspections at IPEC?

A72. (IDM, RMA) Entergy's criteria for component selection at IPEC for FAC inspection during outages are consistent with those cited in NSAC-202L-R3. *See* EN-DC-315 at 15-19 (ENT000038). For modeled piping, the selections are based principally on: (1) the trending of pipe wall thickness measurements from past outages; (2) predictive evaluations performed using the CHECWORKS code; (3) industry and IPEC-specific operating experience related to FAC; (4) results from other plant inspection programs, such as the preventive maintenance program for moisture separator reheaters, and the thermal performance program; and (5) engineering judgment. *See* NSAC-202L-R3 at 2-3 to -4, 3-2 (RIV000012).

The SNM piping is evaluated for inspection using a similar set of criteria, except that criterion (2), the predictive evaluations from CHECWORKS, is not used, since the piping is not modeled in CHECWORKS. Instead a separate set of SNM rankings, based on operating conditions, consequence of failure, maintenance history, and industry experience is used. Each criterion can be the basis for a decision to select a particular component for inspection.

The planning process for future inspections at IPEC also considers the actual *and* CHECWORKS-predicted margins between nominal wall thickness and minimum required wall thickness and the consequence of failure of a particular component with respect to personnel safety and plant availability. *See* EN-DC-315 at 15-19 (ENT000038).

These criteria are set forth in fleet procedure EN-DC-315, which states that the component selection process should consider:

- Components selected from measured or apparent wear found in previous inspection results; *i.e.*, trending;
- Components ranked high for susceptibility from current CHECWORKS evaluation;
- Components identified by industry and plant operating experience;
- Components selected to calibrate the CHECWORKS models;
- Components subjected to off-normal flow conditions (*e.g.*, normally isolated lines to the condenser in which leakage is indicated from turbine performance monitoring);
- Engineering judgment/other;
- Susceptible piping locations (groups of components) contained in the small-bore piping database, which have not received an initial inspection;
- Piping identified from condition reports/corrective action, work orders (malfunctioning equipment, downstream of leaking valves, etc.); and
- Vessel shells (feedwater heaters, moisture separator re-heaters, drain tanks, etc.).

See EN-DC-315 at 17-18 (ENT000038).

Q73. How is CHECWORKS used at IPEC?

A73. (IDM, RMA) As explained in response to the previous question, consistent with NSAC-202L-R3 (RIV000012) and fleet procedure EN-DC-315 (ENT000038), CHECWORKS is used as one factor—in conjunction with trend data from prior inspections, relevant information from other plant programs, industry and plant operating experience, and engineering judgment—to select and schedule piping components for inspection. The decision to repair or replace piping components at IPEC is based on actual measured wall thicknesses relative to the wall thickness required to carry the piping design loads, *not* on the results of CHECWORKS evaluations. *See* EN-DC-315 at 27 (ENT000038).

Q74. How is industry operating experience used to determine inspection locations?

A74. (IDM, JSH, RMA) Sources of operating experience information include: the information shared by participants at CHUG meetings, Entergy fleet data sources, INPO notices, NRC notices and bulletins, communications with peers at Entergy and other utilities, and Entergy's formal operating experience evaluation program. *See* Entergy, EN-OE-100, Rev. 12, Operating Experience Program (Apr. 15, 2011) (ENT000055). If new operating experience is applicable to IPEC, then Entergy performs inspections based upon it, or, alternatively, documents the reason(s) why an inspection is not required in a particular instance. Conclusions regarding the applicability of operating experience are documented in accordance with Entergy's existing OE Program. *See id*. at 24 (specifying that OE that requires evaluation is documented with a "site-specific condition report, a Nuclear Headquarters condition report, or an OE Written Review depending on the assigned sub-code").

Q75. How is engineering judgment used to determine inspection locations?

A75. (IDM, JSH, RMA) Engineering judgment is the use of sound engineering facts and principles to select the most susceptible components for inspections. Under the IPEC FAC Program, such judgment is exercised by an experienced and qualified engineer. The exercise of engineering judgment necessarily implicates some degree of discretion, within established procedures and guidelines.

Under the FAC Program, engineering judgment is only applied by a qualified FAC Engineer who is familiar with the program requirements and who is experienced and familiar with the selection process. The FAC Engineer is qualified in accordance with Entergy's qualification process. *See* Entergy, FTK-ESPP-G00019, Rev. 2, Nuclear Engineering Qualification Card, Implementing the Flow-Accelerated Corrosion (FAC) Program (ENT000056). Qualifications are issued to engineers who have demonstrated sufficient knowledge and expertise in the field of FAC. *See* EN-DC-315 at 8, 11 (ENT000038) (specifying that a qualified engineer prepares the inspection scope).

Moreover, the FAC inspection plan and the documented bases for the plan for each outage are formally peer-reviewed by another experienced and qualified Entergy FAC engineer through a self-assessment process undertaken prior to every outage, providing further confidence in the adequacy of the engineering judgment exercised under the FAC Program. *See* EN-DC-315 at 17 (ENT000038).

Q76. What proportion of the susceptible lines within the scope of the IPEC FAC Program is modeled in CHECWORKS?

A76. (RMA, JSH, IDM) We have reviewed the most recent System Susceptibility Evaluation ("SSE") reports associated with IP2 and IP3 to determine the proportion of modeled lines. At IP2, 22% of the susceptible lines are modeled in CHECWORKS. *See* IP2 SSE Report 0700.104-02, at 15 (ENT000048) (defining Susceptibility Category abbreviations used in Appendix C); *id*. at App. C (listing all Analysis Lines and identifying whether each is modeled in CHECWORKS, is SNM, or is excluded as non-susceptible). At IP3, the proportion is 20%. *See* IP3 SSE Report 0700.104-17 at 15 (ENT000049); *id.* at App. C (providing similar information). In our experience, this is generally consistent with the proportion of modeled lines throughout the domestic PWR fleet.

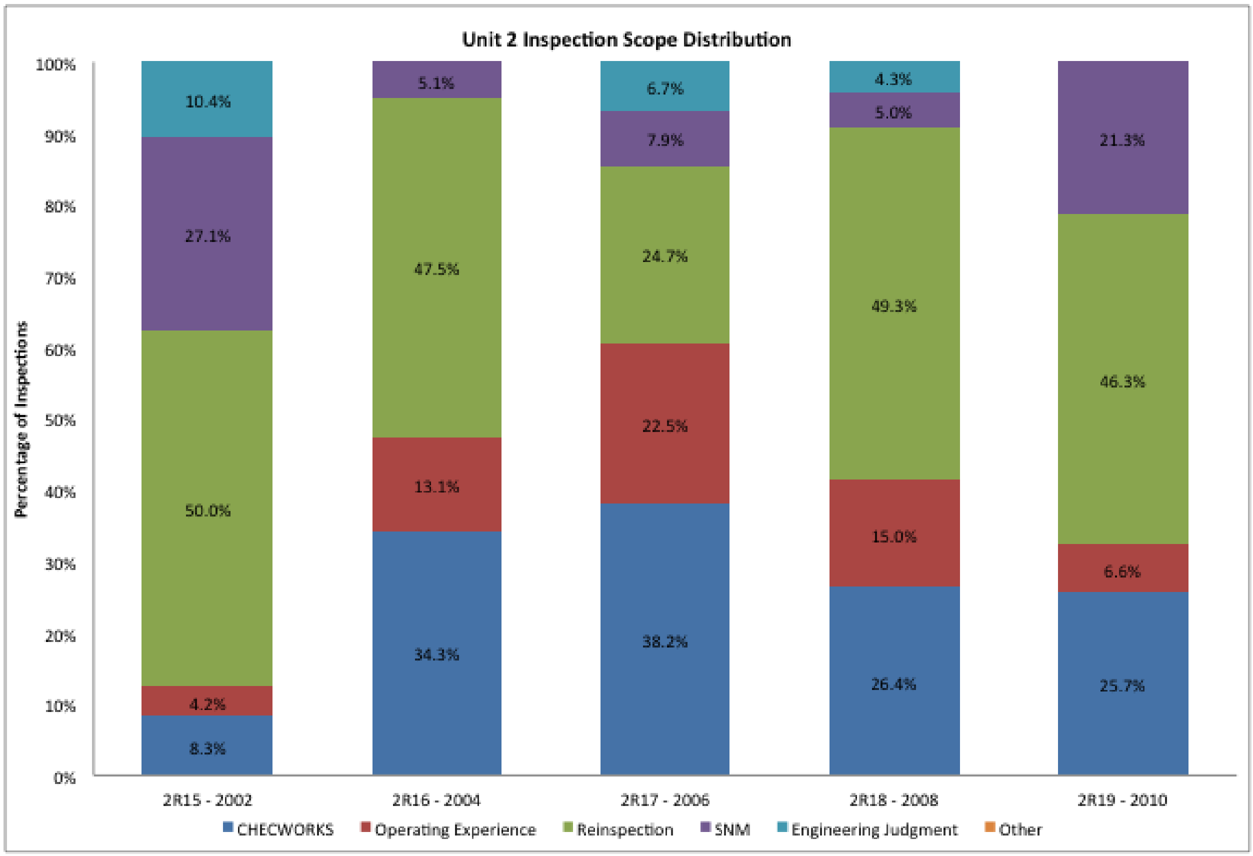
Q77. What proportion of FAC inspection locations are typically chosen by each of the various methods (CHECWORKS initial inspections, reinspections of modeled components, SNM components, engineering judgment, operating experience, etc.)?

A77. (RMA, JSH, IDM) In our experience, newly selected (first-time) inspection locations identified based on predictions from CHECWORKS normally comprise between one-quarter and one-third of FAC inspections in a given outage. This is true for both IP2 and IP3, and in general throughout the U.S. nuclear fleet. For example, during the recent 2R19 outage at IP2 in 2010, new CHECWORKS locations comprised 26% of the total number of inspections. *See* Scope of Flow-Accelerated Corrosion Inspection Points for 2R19 Outage (Apr. 4, 2010) (ENT000057). Reinspections (both modeled and non-modeled) based on trending accounted for 46%, inspections based on operating experience accounted for 7%, and the remaining 21% were inspections of previously uninspected Susceptible Non-Modeled components. *See id*. Similarly, at IP3, in the 3R14 outage in 2007, shortly after the SPU, CHECWORKS locations comprised 36% of the total number of inspections. Reinspections (both modeled and non-modeled) based on trending were 31%, operating experience (including locations selected specifically due to the power uprate) provided 33%, and the remaining 1% were inspections of previously uninspected

Susceptible Non-Modeled components. *See* Scope of Flow-Accelerated Corrosion Inspection Points for 3R14 Outage (Apr. 2, 2007) (ENT000061).

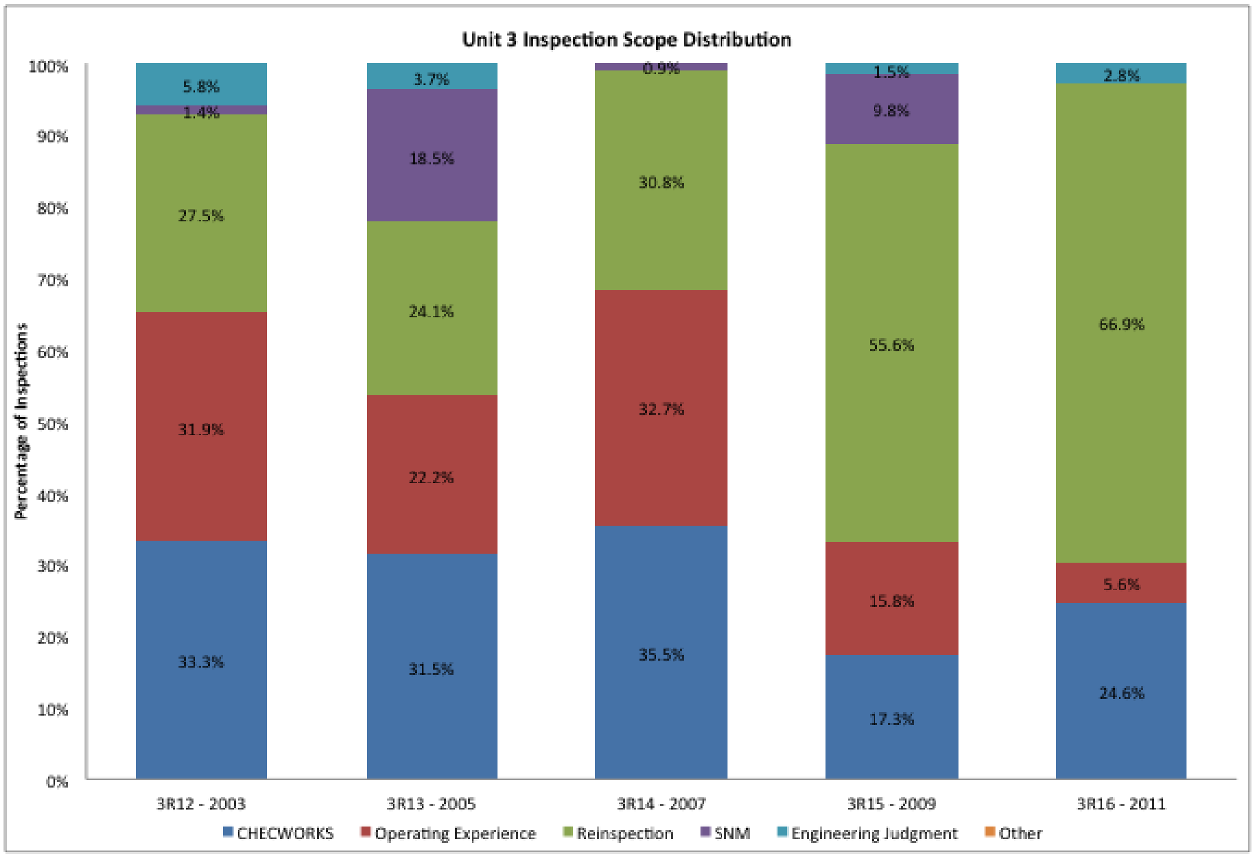
The inspection proportions in other recent IPEC outages were similar. *See*, *e.g.*, Scope of Flow-Accelerated Corrosion Inspection Points for 3R16 Outage (Sept. 19, 2011) (ENT000062) (showing 25% CHECWORKS new inspections, 67% reinspections (both modeled and non-modeled) based on trending, 6% operating experience, and 3% engineering judgment). Figures 1 and 2 below illustrate the inspection distributions from the five most recent outages for both units. These data show that Dr. Hopenfeld is incorrect when he states that "Entergy's method of selecting components for wall measurements and determining the time between successive thickness measurements is primarily based on predictions generated from the computer code, CHECWORKS." Hopenfeld Report at 4 (RIV000005).

Figure 1 – IP2 Inspection Scope Distribution



See Scope of Flow-Accelerated Corrosion Inspection Points for 2R19 Outage (Apr. 4, 2010) (ENT000057); Scope of Flow-Accelerated Corrosion Inspection Points for 2R18 Outage (June 2007) (ENT000058); Scope of Flow-Accelerated Corrosion Inspection Points for 2R17 Outage (Apr. 17, 2006) (ENT000059); Scope of Flow-Accelerated Corrosion Inspection Points for 2R15 and 2R16 Outages (Aug. 2002; June 2005) (ENT000060).

Figure 2 – IP3 Inspection Scope Distribution



See Scope of Flow-Accelerated Corrosion Inspection Points for 3R14 Outage (Apr. 2, 2007) (ENT000061); Scope of Flow-Accelerated Corrosion Inspection Points for 3R16 Outage (Sept. 19, 2011) (ENT000062); Scope of Flow-Accelerated Corrosion Inspection Points for 3R12 and 3R13 Outages (Jan. 2003; Apr. 2005) (ENT000063); Scope of Flow-Accelerated Corrosion Inspection Points for 3R15 Outage (Mar. 2009) (ENT000064).

Q78. Please describe how the results of component inspections are evaluated under the FAC Program, including the criteria for repair or replacement?

A78. (IDM, JSH, RMA, NFA) As stated in Appendix B to the LRA, the IPEC FAC Program is based on NSAC-202L, which specifies criteria for evaluation of inspection results, including the criteria for component repair or replacement. LRA, App. B at B-54 (ENT00015B); *see also* NSAC-202L-R3 at 4-17 to -27 (RIV000012). Entergy fleet procedure EN-DC-315 contains additional details regarding the implementation of NSAC-202L-R3 guidelines. Once the inspection results are obtained, the minimum measured thickness is determined from the matrix of inspection data, this measured thickness is compared to the manufacturer's adjusted nominal thickness (*i.e.*, 87.5% of the nominal wall thickness) to determine if a structural evaluation is required. If the minimum measured wall thickness is less than 87.5% of the nominal thickness, then a location-specific structural evaluation is performed. This evaluation is performed by calculating the required component wall thickness for all component design loading conditions using the required design safety margins. Once the required thickness is determined, then the wear rate is calculated based on the number of years of service, and the amount of wear since the component was first placed in service. The measured wall thickness, along with the wear rate and the required wall thickness are then used to calculate the remaining service life of the component. If the remaining service life is less than one operating cycle, then the component is repaired or replaced prior to returning it to service. If the remaining service life exceeds one operating cycle, then the component is fit to be returned to service. Entergy's procedures EN-CS-S-008-MULTI, Revision 0, "Pipe Wall Thinning Structural Evaluation" (Jan. 1, 2010) ("EN-CS-S-008-MULTI"), attached as Exhibit ENT000065; EN-DC-126, Revision 4, "Engineering Calculation Process" (Jan. 31, 2011) ("EN-DC-126"), attached as Exhibit ENT000066; and EN-DC-315 (ENT000038) are used to perform these evaluations, and describe the process outlined above in more detail.

Q79. What criteria govern expansion of inspections based on measured results?

A79. (IDM, JSH, RMA) If a component is found with a projected wall thickness (*i.e.* projected to the next refueling outage) less than the minimum required wall thickness, then Entergy performs additional inspections of identical or similar components in a parallel or alternate train, as necessary, to determine the extent of thinning. NSAC-202L-R3, Exhibit RIV000012, at pages 4-10 and Section 5.12 of EN-DC-315 describe the sample expansion protocol. EN-DC-315 at 25-26 (ENT000038).

Q80. How is UT data used in this evaluation?

A80. (IDM, RMA, JSH) UT data is normally used to determine a component's minimum wall thickness and to calculate that component's wear rate. These two values are used to calculate the component's projected wall thickness at the next refueling outage. This process of projecting the wall thickness at a future refueling outage based on the measured thickness is often referred to as "trending." Trending is an inherent part of the evaluation described above, in accordance with EN-DC-315, Sections 5.6, 5.9, and 5.10 (ENT000038), and is distinct from CHECWORKS modeling. It is important to emphasize that, contrary to Dr. Hopenfeld's assertion that "for components initially selected for inspection by CHECWORKS, any decisions regarding future inspection scope based on actual pipe wall thickness measurements and wear rate trending of the actual inspection results, necessarily depends upon use of the CHECWORKS computer model," Hopenfeld Report at 21 (RIV000005), *none of the trending calculations described in these sections of EN-DC-315 involve the use of CHECWORKS*. These calculations are performed either by a FAC engineer using the FAC Manager trending software or by a structural design engineer, in accordance with the Entergy calculation procedures cited above.

In brief, trending is accomplished by first determining minimum acceptable wall thickness values, t_{accpt}, through a validated and verified method. *See* EN-DC-315 at 19 (ENT000038). The value is the highest of the following:

- Thickness required to carry the corresponding design loads; or

- Piping replacement criteria of 0.3 nominal thickness (t_{nom}) for Class 1 piping, or 0.2 t_{nom} for Class 2, Class 3, and non-safety related piping.

See id. at 7.

Second, the wear rate is determined by dividing the difference between initial (t_{init}) and measured (t_{meas}) thicknesses by the operating time. *See id*. at 22.

Third, the predicted thickness (t_{pred}) at the time of the next scheduled refueling outage is determined through the following equation, where "Time" is the estimated operating time remaining prior to the next refueling outage:

$$t_{pred} = t_{meas} - \text{Safety Factor x Wear Rate x Time}$$

See id. at 23. In accordance with NSAC-202L-R3, a safety factor of 1.1 is normally used. NSAC-202L-R3 at 4-27 to -28 (RIV000012). The remaining service life ("RSL") is then determined through the following equation:

$$\text{RSL} = (t_{meas} - t_{accpt}) / (\text{Safety Factor x Wear Rate})$$

See id.

This information is then used in the pipe wall thinning evaluation process discussed above. Thus, trending is an aspect of the FAC Program that is entirely independent of CHECWORKS. Moreover, trending is accomplished in the same manner irrespective of the reason for the original inspection (*i.e.*, regardless of whether the component was selected based on CHECWORKS, operating experience, engineering judgment, or other reason).

Q81. How have water chemistry improvements decreased the rates of FAC in US PWRs?

A81. (JSH) As previously noted in response to Question 50, above, pH and oxygen concentration are key influencing factors on FAC. In the late 1980s, when FAC became a prominent issue in the U.S. following the Surry accident, most PWRs were using ammonia as a pH control agent with the room temperature pH in the range of 8.8 to 9.2. *See* EPRI, PWR Advanced Amine Application Guidelines, Rev. 2, at 2-2 (Oct. 1997) (ENT000067). This range was selected for several reasons, a primary reason being concern about copper transport to the steam generators with more alkaline water chemistry. Copper transport was known to cause materials problems in steam generators and in turbines. *See* EPRI, Pressurized Water Reactor Secondary Water Chemistry Guidelines, Rev. 7, at 3-24 (Feb. 2009) (ENT000068).

As explained on pages 5-12 to 5-19 of *Flow-Accelerated Corrosion in Power Plants* (ENT00036B), since the 1980s, there have been two things done to various degrees at all U.S. PWRs:

1. Most plants have removed some or all of the copper alloy components, allowing the use of a higher pH. Generally, the room temperature pH has been increased to around 10.0. Because pH is defined on a logarithmic scale, there is a large decrease in the rate of FAC with the change from a pH of 9.0 to 10.0. *See* NSAC-202L-R3 at 5-3, Fig. 5-4 (RIV000012).
2. Most plants have replaced ammonia with more advanced (i.e., complicated) organic amines – often monoethanolamine ("ETA"). These advanced amines provide greater protection from FAC than ammonia at the same pH, especially in the two-phase portions of systems. *See* NSAC-202L-R3 at 5-4, Fig. 5-3 (RIV000012).

Q82. What water chemistry improvements have been implemented at IPEC?

A82. (IDM, ABC) At IPEC, Entergy has adopted the EPRI water chemistry guidelines. *See* EPRI, Pressurized Water Reactor Secondary Water Chemistry Guidelines, Rev. 7 (Feb. 2009) (ENT000068); *see also* LRA, App. B at B-137 (ENT00015B). Thus, in accordance with the Water Chemistry Control – Secondary Chemistry Specifications at IPEC, Entergy uses an all volatile treatment ("AVT") that includes the addition of monoethanolamine ("ETA") and hydrazine to control pH and oxygen levels at IPEC. *See*, *e.g.*, Entergy Secondary Strategic Water Chemistry Plan, Rev. 3, at 23-24 (Sept. 20, 2011) (ENT000069). Thus, the Water Chemistry Control – Primary and Secondary Program contributes to minimizing FAC at IPEC.

D. <u>Overview of the CHECWORKS Computer Code</u>

Q83. What is CHECWORKS?

A83. (JSH, RMA) CHECWORKS is a computer program designed to assist FAC engineers in identifying and prioritizing potential locations of FAC vulnerability. It is designed for use by plant engineers as a tool for identifying piping locations susceptible to FAC, predicting FAC wear rates, planning inspections, and evaluating inspection data. *See* EPRI, CHECWORKS™ Steam/Feedwater Application Version 3.0 User Guide at 5 (2008) ("CHECWORKS User Guide") (ENT000070).

Q84. How is CHECWORKS used by FAC engineers?

A84. (JSH, RMA) The CHECWORKS user constructs a mathematical model of the FAC-susceptible piping systems, similar in concept to a piping stress model or flow model. The input to the CHECWORKS model includes plant operating parameters such as water chemistry parameters, flow rates, pipe material, operating temperatures and piping configuration, and can be augmented with measured wall thicknesses of modeled components. Based on this input, CHECWORKS predicts the rate of wall thinning and remaining service life on a component-specific basis. CHECWORKS thus allows prediction of wear rates based on changing parameters, such as flow rate, without the need for measured wall thickness values. Entergy, however, routinely augments CHECWORKS predictions with actual measured inspection data.

CHECWORKS predictions of remaining service life for modeled piping are used for scheduling components for inspection and are independent of the remaining service life calculations developed as part of the trending analysis described previously. Thus, for modeled piping, the IPEC FAC engineer could have two independent estimates of remaining service life available for evaluation to determine inspection scheduling; however, measured values are typically used to determine inspection schedules. *See* EPRI, CHECWORKS Steam/Feedwater Application Guidelines for Plant Modeling and Evaluation of Component Inspection Data at 2-1, 2-2 (Nov. 2009) (ENT000071).

Q85. How does CHECWORKS predict wear rates and determine the most susceptible locations for FAC?

A85. (JSH, RMA) CHECWORKS uses two types of evaluations in determining the susceptible locations for FAC and predicting wear rates. The first evaluation, called a "PASS-1 Analysis," is performed to determine predicted wear rates based on plant operating characteristics that do not incorporate measured pipe thicknesses from plant inspections. This evaluation is normally used to generate a list of components for inspection when plant UT thickness data are not available. *See* NSAC-202L-R3 at 4-1 to -2 (RIV000012); EN-DC-315 at 8, 12 (ENT000038).

The second evaluation, called a "PASS-2 Analysis," incorporates measurements from inspections of plant piping and components. The model compares the measured values to the predicted values and adjusts the FAC predictive calculations to more closely approximate measured wall thickness through the use of a "line correction factor" ("LCF"). The LCF is statistically determined to produce a "best estimate" correlation (or "best fit") with the inspection data.

If the predicted wear rate differs from the measured wear rate, then CHECWORKS adjusts the predictions upward or downward to fit the field observations; this is accomplished by multiplying each prediction by the LCF. *See* NSAC-202L-R3 at 4-1 to -2 (RIV000012); EN-DC-315 at 8, 11 (ENT000038); NRC Audit Report at 15 (ENT000041). This calibration, or benchmarking based on measured wall thicknesses, allows for more accurate future predictions of wear rates.

Q86. How do FAC engineers determine whether there is satisfactory correlation between predicted and measured wear rates for a given set of data?

A86. (JSH, RMA, IDM) CHECWORKS results are generally examined on an "Analysis Line" basis. *See*, *e.g.*, IP3 SFA Report 0705.100-01 at 4 (ENT000051) ("Selected lines from the above systems are modeled in the IP3 CHECWORKS Model."). Each Analysis Line is a grouping of components subject to similar operating conditions and water chemistry. An example of an Analysis Line would be the group of components located in the section of the condensate system between feedwater heaters. For each Analysis Line, the main review parameters are the LCF (the statistically derived line correction factor based on the difference between the predicted and measured thickness values within each Analysis Line), the scatter of the data, and the number of inspected components.

The reviewer then classifies each Analysis Line as either "calibrated" or "non-calibrated," based on the criteria in NSAC-202L-R3 at page 4-1 (RIV000012). Lines are considered calibrated if they meet the following criteria: (1) similar operating conditions and water chemistry throughout the Analysis Line; (2) at least five inspected components; (3) a LCF between 0.5 and 2.5; (4) a reasonably good correlation between predicted and measured wall thickness (as judged by the scatter of the results); and (5) sufficient inspection coverage of component geometry types and parallel trains. *See id*.

The recommended LCF range of 0.5 to 2.5 is provided in EPRI Guidance. *See* NSAC-202L-R3 at 4-1 (RIV000012). This range was first proposed for CHECWORKS in the early 1990s when the program was being developed. The range was selected based on consideration of the accuracy of the FAC correlation, the inherent variability of the corrosion process, and the uncertainty of the program inputs. Our experience has shown that this range has met its intended purpose as a screening criterion. We have found that when LCFs are within this range, predictions are reliable and fewer inspections are necessary to ensure that the lines continue to perform their intended functions. *See id.* at 4-7 (RIV000012). These lines are still inspected within the FAC Program, but more inspections are focused on those lines with LCFs outside of the range.

Q87. What is the most common reason for Analysis Lines to be characterized as non-calibrated?

A87. (JSH, RMA, IDM) The most common reason for Analysis Lines to be categorized as non-calibrated is that the wear is minimal. *See* EPRI, CHECWORKS Steam/Feedwater Application Guidelines for Plant Modeling and Evaluation or Component Inspection Data (Nov. 2009) (ENT000071) ("[S]ystems with . . . lower rates of wear . . . tend to have a poorer correlation. This is due to the greater influence of NDE measurement errors and interpretation difficulties compared to the amount of FAC taking place."). Analysis Lines with little or no wear, regardless of the number of inspections, are almost never calibrated. This is because small amounts of wear are difficult to distinguish from measurement uncertainty.

Other reasons for non-calibrated lines can be an insufficient number of inspections, insufficient variety in the types of component geometries examined, and incomplete chemistry data—especially historical chemistry data. In contrast to non-calibration due to low wear, these lines may be wearing at any rate. Therefore, in such situations, additional inspections are usually conducted until the line can be calibrated.

Q88. How does Entergy address non-calibrated lines?

A88. (JSH, RMA, IDM) When an Analysis Line is non-calibrated or the LCF is out of range—unless the cause is very low wear—additional inspections are conducted on these lines or the affected components are repaired or replaced. *See* NSAC-202L-R3 at 4-7 (RIV000012). The IPEC FAC Program therefore addresses LCF values that fall outside the acceptable range and lines that are not calibrated. Indeed, the overall FAC Program focuses inspection activities on these very lines so that inspections are increased, piping is replaced, or other corrective actions are taken.

Q89. What physical phenomena or mechanisms are accounted for in the wear rate predictions provided by CHECWORKS?

A89. (JSH, RMA) As discussed above, in general, FAC is the mechanism accounted for in the CHECWORKS model. As we have previously explained, these are the predominant degradation mechanisms of concern for the FAC-susceptible components analyzed by CHECWORKS.

As explained above, however, the CHECWORKS model is based on empirical data from many plants. Moreover, CHECWORKS is calibrated at individual plants through the PASS-2 analysis, which compares predicted and measured wear rates from UT data. *See* CHECWORKS User Guide at 15-6 (ENT000070). To the extent that plant-specific UT data reflects the effects of degradation mechanisms other than FAC, then after calibration the effects of those mechanisms are accounted for in subsequent wear rate predictions. For modeled lines, however, mechanisms other than FAC are usually negligible.

E. <u>The Impact of Power Uprates on FAC Program Activities</u>

Q90. Does the FAC Program account for changes in plant operating parameters?

A90. (IDM, RMA) Yes. The IPEC FAC Program accounts for changes in plant operating parameters and conditions in calculating FAC-induced wear rate predictions. Specifically, as procedurally required, the CHECWORKS model is updated every cycle with the latest chemistry, operating, and inspection data. *See* EN-DC-315 at 15-16 (ENT000038); NRC Audit Report at 15 (ENT000041). Through this process, changes due to replacement or repair of piping and piping components, adjustments in water chemistry, and changes in operating conditions, such as those due to power uprates, are incorporated into the IPEC CHECWORKS models.

Q91. How has the FAC Program accounted for the 2004 and 2005 stretch power uprates?

A91. (IDM, RMA) The FAC Program recognized the need to consider the impact of the stretch power uprates ("SPUs"). As noted above, the NRC approved SPUs of 3.26% and 4.85% for IP2 and IP3 in October 2004, and March 2005, respectively. Entergy accordingly updated the IP2 and IP3 CHECWORKS models to include SPU operating parameter changes (*e.g.*, flow rates and operating temperatures). Entergy completed these updates to the IPEC CHECWORKS models on March 23, 2005. *See* Indian Point Unit 2, CHECWORKS Power Uprate Analysis, Calc. No. 040711-02 (Mar. 23, 2005) (ENT000072); Indian Point Unit 3, CHECWORKS Power Uprate Analysis, Calc. No. 040711-01 (Mar. 23, 2005) (ENT000073); NRC Audit Report at 15 (ENT000041). The updated CHECWORKS results were used in the inspection planning for the subsequent outages.

The IPEC outage schedules include four IP2 refueling outages and five IP3 refueling outages between implementation of the SPU and the beginning of the PEO, which begins in September 2013 for IP2 and December 2015 for IP3. Consistent with the process described above, Entergy uses the UT inspection results obtained during plant outages to assess the accuracy of the CHECWORKS wear predictions and to improve future CHECWORKS wear predictions.

Entergy has already updated the IP2 CHECWORKS model to incorporate inspection data from the 2R17 (2006), 2R18 (2008), and 2R19 (2010) outages. Similarly, Entergy has updated the IP3 CHECWORKS model to incorporate inspection data from the 3R13 (2005), 3R14 (2007), 3R15 (2009), and 3R16 (2011) outages.

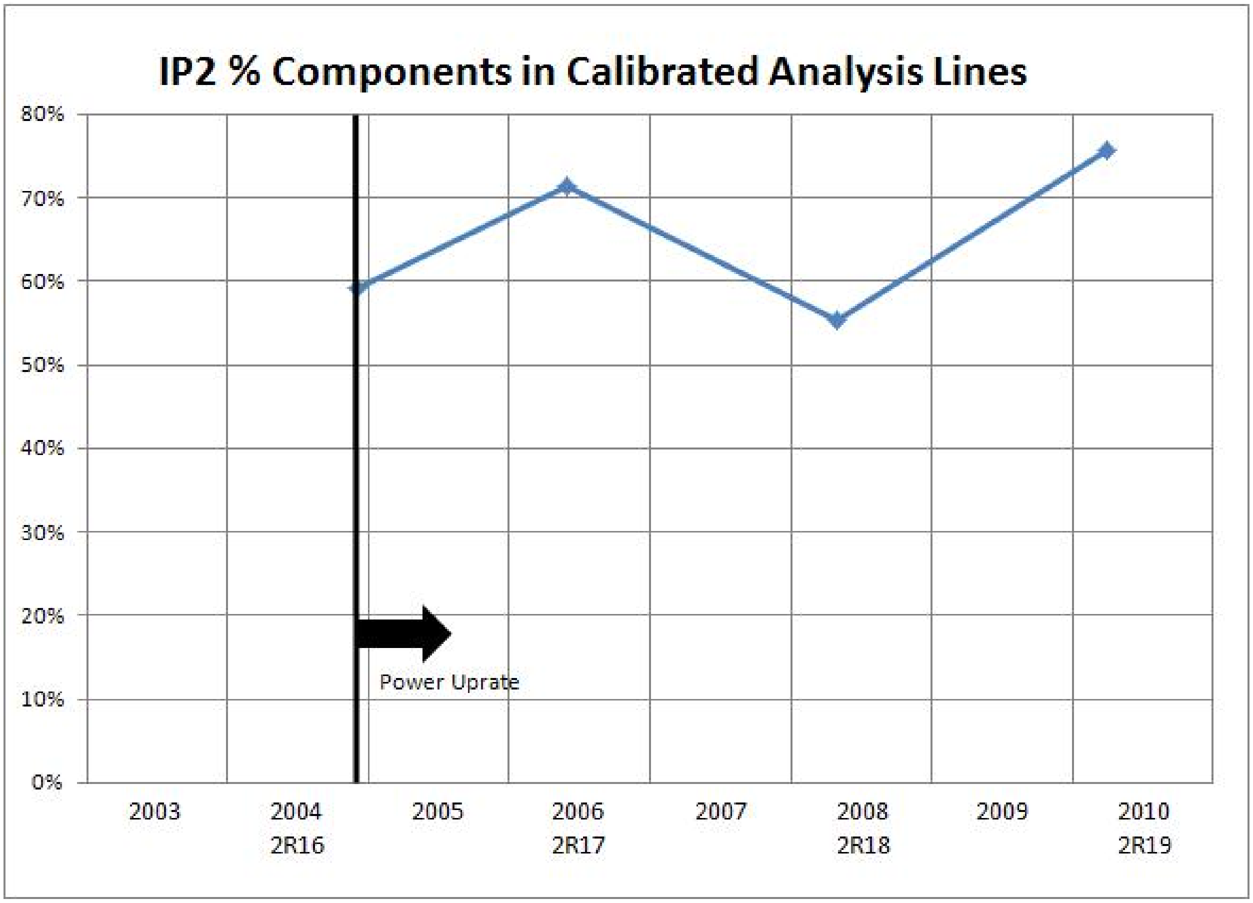
Q92. Since the SPU, how well have predicted and measured wear rates correlated?

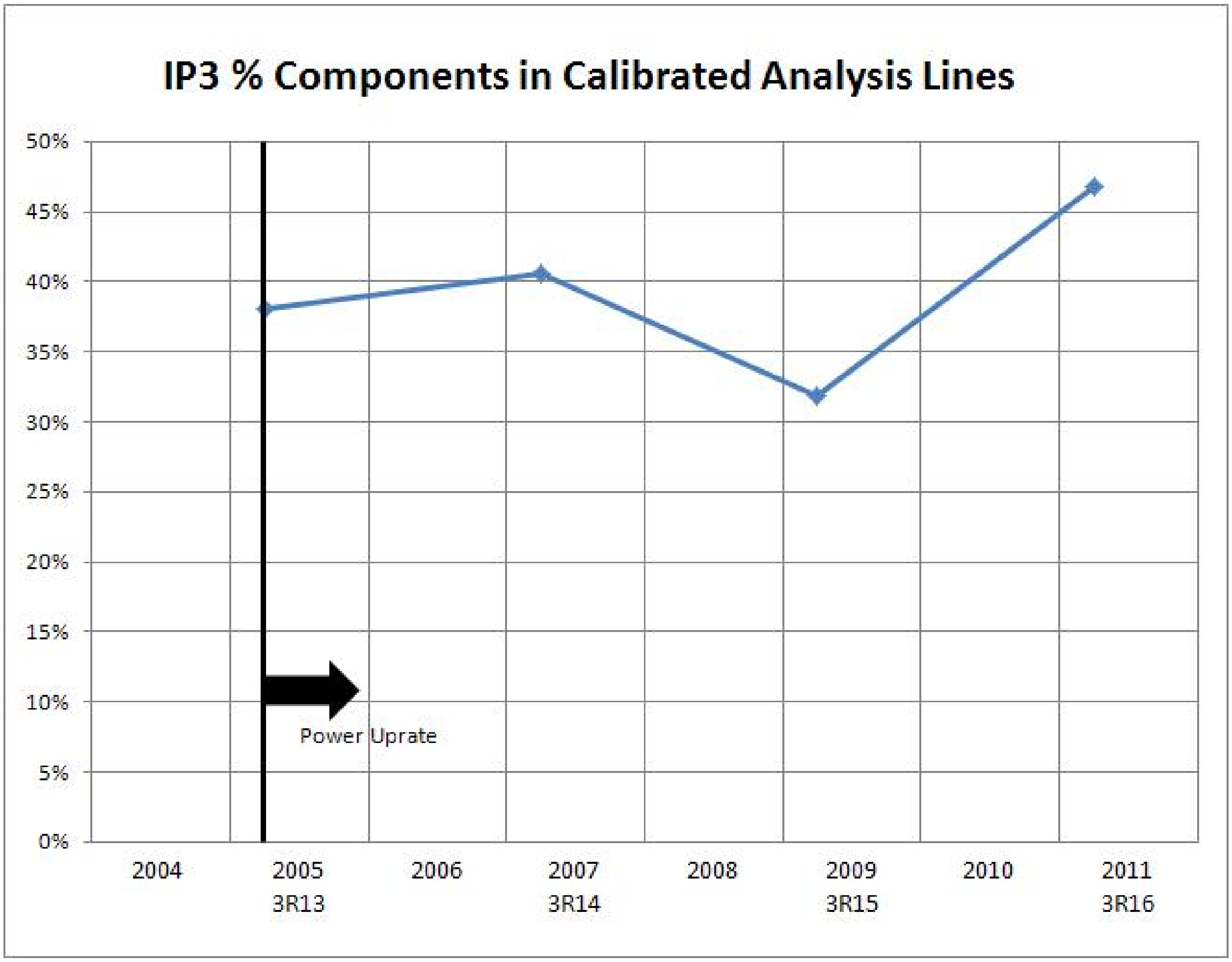
A92. (JSH, RMA) We have compared the most recent PASS-2 outputs with outputs calculated after the outage. Recall that PASS-2 outputs compare the measured wall thickness values to the CHECWORKS-predicted values to calibrate the model. This review includes the CHECWORKS SFA Model Calculation for IP2 dated July 2005, September 2006, November 2008, February 2010, and July 2010 (ENT000050, ENT000074-77), and the CHECWORKS SFA Model Calculation for IP3 dated October 2005, November 2007, February 2010, and August 2011 (ENT000051, ENT000078-80).

Based upon our review of the percentage of lines calibrated and with an acceptable LCF, *see* NSAC-202L-R3 at 4-1 (RIV000012), we conclude that the level of correlation between the CHECWORKS model-predicted wear and the measured wear following SPU implementation at IP2 and IP3 is consistent with the level of correlation prior to SPU. Indeed, the IPEC results are typical in comparison to the other programs we have reviewed throughout our careers. In the reports listed above, an average of approximately 55% of the lines across both plants are calibrated, and the LCFs are in range 70% of the time. CHECWORKS serves its function by focusing the inspections under the IPEC FAC Program on those lines that are not calibrated *or* where the LCF is out of range and, therefore, the rates of FAC are less reliably known.

CHECWORKS continues to effectively support the plant's FAC Program following the SPUs at IPEC. Data from the recent CHECWORKS SFA Model Reports from IPEC (ENT000050-51, ENT000074-80) are illustrated graphically in Figure 3. Figure 3 shows that since the 2R16 and 3R13 outages at IP2 and IP3, respectively (*i.e.*, the outages during which the SPUs were implemented, but which reflect plant operating conditions prior to SPU implementation), the percentage of Analysis Lines calibrated at IPEC has remained relatively constant, so the overall accuracy of the CHECWORKS model was not significantly affected by the SPUs.

Figure 3 – Percentage of Analysis Lines Calibrated Over Time




IP3 % Components in Calibrated Analysis Lines
50%
45%
40%
35%
30%
25%
20%
15%
10%
5%
0%
Power Uprate
2004
2005
3R13
2006
2007
3R14
2008
2009
3R15
2010
2011
3R16

Q93. Why do CHECWORKS results sometimes include negative values of predicted remaining life?

A93. (RMA) CHECWORKS provides numerous outputs. The most important and frequently used is the predicted wear rate, expressed in units of mils (thousandths of an inch) per year. Another CHECWORKS output is the predicted time to critical thickness (TT_{crit}).

TT_{crit} is the time remaining, in hours, until the component reaches the "critical thickness" (that is, the minimum acceptable wall thickness). The critical thickness is calculated based on user input, and TT_{crit} is a function of predicted wear rate, current wall thickness, and critical wall thickness. A negative TT_{crit} value indicates that, in theory, the component reached the critical wall thickness at some point in the past.

TT_{crit} can be negative for a number of reasons. This occurs most commonly where the line is uncalibrated; under these circumstances, the predicted wear rate can be much larger than the actual wear rate (where measurement uncertainty exceeds the very low wear rates). If the predicted wear rate is unrealistically high, TT_{crit} will often be negative.

When considering negative TT_{crit} values, it is important to emphasize that CHECWORKS is primarily used as a ranking tool. It functions to alert the FAC Engineer to locations that require inspection, based on the *relative rankings* provided by CHECWORKS. *See* NSAC-202L-R3 at 4-6 (RIV000012) (recommending that inspections of new lines should be based on "components identified in the wear rankings as having the highest relative wear"). Thus, when negative TT_{crit} values (or any other indication of high *relative* wear) are identified, additional inspections may be conducted, a more precise value of the minimum allowable thickness may be calculated, or the results of a trending analysis may be used to better estimate a component's remaining life. But negative TT_{crit} values are not, standing alone, an indication that the component's actual thickness is below critical thickness.

VI. RESPONSE TO ISSUES RAISED IN RIVERKEEPER TC-2

A. CHECWORKS Is Only One Aspect of the IPEC FAC Program

Q94. Dr. Hopenfeld asserts that Entergy's FAC Program "is largely based on the . . . computer program CHECWORKS to record plant operating experience and predict timing and locations of wall thinning." Hopenfeld Report at 4 (RIV000005). Elsewhere, he asserts that Entergy "primarily" selects inspection locations based on CHECWORKS, *id.* at 4, he describes CHECWORKS as "the predominant feature of the FAC program," *id.* at 23, and makes numerous other assertions along similar lines. Do you agree?

A94. (IDM, RMA, JSH) No. As we have previously explained, CHECWORKS does not play the all-encompassing role within the IPEC FAC Program that Dr. Hopenfeld suggests. As is evident from the data and graphs presented in response to Question 77, CHECWORKS is neither the only method nor the predominant method used at IPEC to select inspection locations. In response to Question 76, we showed that CHECWORKS only models 22% of the susceptible lines at IP2, and 20% of the susceptible lines at IP3. The fact that extensive measured data, above and beyond data from CHECWORKS predictions, is available for use within the FAC Program is evident from Dr. Hopenfeld's own testimony regarding his review of "more than 6,500 data points, Hopenfeld Report at 5 (RIV000005), which he asserts includes "ample post-power uprate data." *Id*. at 20. While Dr. Hopenfeld seems to be under the impression that the only purpose of the thickness data collected under the FAC Program is to calibrate CHECWORKS, this is incorrect. As we have previously explained in response to Question 78, under EN-DC-315, Revision 6 and the EPRI guidance in NSAC-202L Revision 3, upon which the IPEC FAC Program is based, these inspection data are used *directly* to determine remaining component life.

Q95. Dr. Hopenfeld claims that the additional elements of Entergy's FAC Program, apart from CHECWORKS, are not truly independent tools within the FAC Program. *See* Hopenfeld Report at 21 (RIV000005). Do you agree?

A95. (IDM, JSH, RMA) No. As we have just explained, over 75% of the susceptible lines at IPEC are not modeled. Clearly, the elements of the FAC Program that involve these lines are "truly independent" of CHECWORKS. Dr. Hopenfeld's statement is also incorrect for modeled lines. At IPEC and at other plants that follow the guidance in NSAC-202L, as explained above, the following methods are used to select inspection locations: (1) the trending of actual measurements of pipe wall thicknesses from past outages; (2) predictive evaluations performed using the CHECWORKS computer code; (3) industry experience related to FAC; (4) results from other plant inspection programs; (5) engineering judgment; and (6) the SNM rankings. *See* EN-DC-315 at 17-18 (ENT000038). CHECWORKS analyses are only performed on a portion of the large-bore piping. For non-modeled piping, only methods one and three through six are used. For modeled piping, methods one through five are used. Each of the other five methods is independent of CHECWORKS:

- Trending: Once inspection data is taken, it is analyzed using Entergy's formal calculation process. CHECWORKS is *not* used for trending at IPEC. The reason a component was originally selected for inspection is irrelevant after the inspection is performed. Once the data are available, they are trended in every case. *See* EN-DC-315 at 21-23 (ENT000038). Further, as decades of operating experience shows, trending results are an accurate means of determining future wall thickness.

- Industry Operating Experience: Operating experience from all available sources, *i.e.*, CHUG meetings, other Entergy plants, informal communications from other utilities, INPO notices, and NRC communications are reviewed. *See* EN-DC-315 at 17 (ENT000038). If the operating experience applies to IPEC, then the appropriate components are scheduled for inspection. For example, if no inspections have been done on the line in question, or if the line is not calibrated, then the relevant operating experience would generally lead to more inspections. On the other hand, the process of evaluating and incorporating operating experience can involve CHECWORKS. For example, if the line has been inspected and the CHECWORKS results are calibrated, then additional inspections may not be necessary. *See* EN-DC-315 at (ENT000038).
- Other Inspection Programs: Information from other plant activities (*e.g.*, valve maintenance or thermal performance monitoring) is directly used to select inspection locations. For example, if a leaking isolation valve is discovered, it would be appropriate to inspect the piping downstream of the leaking valve. *See* NSAC-202L-R3 at 4-8 (RIV000012) ("Plant experience over the past operating cycle(s) should be reviewed to add components as appropriate to the inspection plan. These locations should include: . . . Components downstream of a leaking steam trap or isolation valve."); EN-DC-315 at 17-18 (ENT000038) ("[T]he criteria for component selection should consider the following: . . . Piping identified from Work Orders (malfunctioning equipment, downstream of leaking

valves, etc.)"). Again, this process of evaluating and incorporating information from other inspection programs is independent of CHECWORKS.

- Engineering Judgment: As also explained in response to Question 75, the experience and judgment of a qualified engineer is also used to select inspection locations. *See* EN-DC-315 at 17-18 (ENT000038).
- SNM Rankings: The SNM rankings cover the FAC-susceptible lines at IPEC that are evaluated independently of CHECWORKS, which, as previously noted, comprise a majority of the lines within the IPEC FAC Program. This evaluation considers operating conditions, maintenance history, operating experience, and consequences of failure to develop a relative ranking of non-modeled lines. From these rankings, inspections are selected. *See* EN-DC-315 at 16 (ENT000038); IP2 SNM Report 0700.104-03 at 7-17 (ENT000052); IP3 SNM Report 0700.104-18 at 7-16 (ENT000053).

Accordingly, trending, industry operating experience, input from other IPEC inspection programs, engineering judgment, and the SNM rankings are all aspects of the FAC Program that are generally independent of CHECWORKS. These six independent input sources for inspection locations, taken together, provide confidence that the selected inspection locations are appropriate.

Q96. Dr. Hopenfeld next states that "Actual pipe wall thickness measurements from past outages are only useful when used in combination with a predictive tool which would prevent the wall thickness of a given component from being reduced to below the minimum design thickness while in service. Accordingly, this is a required input for the

use of CHECWORKS and not a stand-alone 'tool' for component selection." Hopenfeld Report at 21 (RIV000005). Do you agree with this statement?

A96. (IDM, JSH, RMA) No. Any single wall thickness measurement, combined with the original nominal component wall thickness value or another measured value, can be used to project future wall thickness independent of any predictive tool. Through the trending process as we have described in response to Question 80, the results of wall thickness measurements are projected using straight-line extrapolation to provide assurance of continuing compliance with required minimum thicknesses until the next inspection. Thus, thickness measurements are quite useful on their own, independent of CHECWORKS. Thickness data from UT measurements are also used in the CHECWORKS analysis and calibration process, but that does not lessen the value of such data in stand-alone projections.

Q97. In the same paragraph, Dr. Hopenfeld states that "for components initially selected for inspection by CHECWORKS, any decisions regarding future inspection scope based on actual pipe wall thickness measurements and wear rate trending of the actual inspection results, necessarily depends upon the use of the CHECWORKS computer model." Hopenfeld Report at 21 (RIV000005). Do you agree?

A97. (IDM, JSH, RMA) No. As we have previously explained, Dr. Hopenfeld is incorrect because the trending of inspection results is accomplished independent of CHECWORKS.

Q98. Dr. Hopenfeld next suggests that industry and plant operating experience "such as information about wall thinning events" at Indian Point and other plants are "also types of information that feed directly into . . . CHECWORKS," and thus, are not

necessarily independent tools for selecting inspection locations. *See* Hopenfeld Report at 21 (RIV000005). Do you agree?

A98. (JSH, RMA) No. As we have previously explained, industry and plant operating experience directly inform the selection of inspection locations and are not filtered through CHECWORKS. This is obviously true for non-modeled lines, which as we have seen, represent the majority of FAC-susceptible components at IPEC, but is also true for modeled lines. For these reasons, Dr. Hopenfeld's statement regarding the filtering of these types of operating experience through CHECWORKS is baseless.

Q99. While Dr. Hopenfeld concedes that the use of engineering judgment is an independent tool for the FAC Program, he asserts that it is not an effective one by itself. Hopenfeld Report at 22 (RIV000005). How do you respond to his statements regarding engineering judgment?

A99. (JSH, IDM, ABC, RMA) First, engineering judgment is not exercised in isolation. As explained above, Entergy uses experienced engineers with expertise attained through a formal qualification process in the administration of its FAC Program. Engineering judgment is exercised by those qualified and experienced engineers through the specifications of the FAC Program. *See* EN-DC-315 at 17 (ENT000038) (specifying that the FAC Program Engineer prepares an Outage Inspection Plan that is also reviewed by qualified FAC personnel). The exercise of engineering judgment necessarily involves some amount of subjectivity, and certainly cannot by itself substitute for all of the other elements of the FAC Program. But in our opinion, the exercise of engineering judgment by experienced FAC engineers, *as one tool among many* within the FAC Program, is a positive aspect of the program that enhances safety and provides additional confidence in the adequacy of the program beyond the various other, more objective tools. *See* NSAC-202L-R3 at 2-4 (RIV000012). Therefore, while we agree with Dr. Hopenfeld that engineering judgment requires a high degree of knowledge and experience, we do not agree that the exercise of engineering judgment within the IPEC FAC Program is not a systematic or valuable aspect of the program.

Importantly, the use of engineering judgment in this manner at IPEC, as specified in the Entergy corporate FAC Program document, EN-DC-315, Rev. 6 (ENT000038), is consistent with the recommendations of NSAC-202L-R3, and is standard practice throughout the domestic nuclear industry. Thus, Dr. Hopenfeld's broad complaints about the use of engineering judgment, whether as part of the IPEC FAC Program or other engineering programs, are baseless.

In any event, contrary to Dr. Hopenfeld's assertion that engineering judgment is the *only* independent tool for managing FAC other than CHECWORKS, *see* Hopenfeld Report at 21-22 (RIV000005), engineering judgment is the source of a relatively small number of FAC Program inspections, usually no more than 10% of all inspections in any outage, as shown in Figures 1 and 2 (response to Question 77).

Q100. Dr. Hopenfeld breaks down engineering judgment into four elements that he asserts Entergy "does not appear to espouse." ***See*** **Hopenfeld Report at 22 (RIV000005). Please address these points.**

A100. (JSH, IDM, ABC, RMA) We address each of these four items in turn and explain why each, contrary to Dr. Hopenfeld's claims, is present in the IPEC FAC Program.

- "<u>good documentation of historical FAC assessments</u>": As we explain later in this testimony in response to Question 129, the IPEC CHECWORKS databases contain historical inspection data. Contrary to Dr. Hopenfeld's statements, no inspection data has been lost.

- "good communication between the organization that conducts analytical assessments and plant operators": Although Dr. Hopenfeld asserts the presence of "anomalies" and an apparent lack of communication, he does not provide any basis for this allegation. Dr. Hopenfeld does not identify any specific anomalies that he believes were not properly communicated between Entergy and CSI.
- "knowledge of FAC assessment methods": As we have explained, the IPEC FAC Program is managed by experienced and qualified engineers who are knowledgable of FAC assessment methods.
- "knowledge or [sic] risks and consequences": Every Analysis Line within the scope of the IPEC FAC Program is assessed for its level of risk (consequences of failure and level of susceptibility). For CHECWORKS-modeled lines (*i.e.*, high-energy large-bore lines like feedwater), all are considered "high-consequence" (that is, failure would pose a risk to human safety, plant shutdown, equipment damage, derate, or other negative event) and they are treated accordingly. Susceptible non-modeled lines are also evaluated for risk. *See* EN-DC-315 at 16 (ENT000038). Each SNM line is categorized as either "high-consequence" or "low consequence" and prioritized for inspection accordingly. This is spelled out in considerable detail in the SNM reports provided to Riverkeeper. *See* IP2 SNM Report 0700.104-03 at 7 (Section 5 defining "Category 1" as "High Consequence" and "Category 2" as "Low Consequence" and section 5.1 defining

criteria for "Category 1" and "Category 2" piping) (ENT000052); IP3 SNM Report 0700.104-18 at 7 (providing the same information) (ENT000053).

Thus, we disagree with Dr. Hopenfeld when he states that the "documents and information provided by Entergy do not reflect adequate consideration of inspection priorities of FAC-susceptible components relative to the safety risks posed due to a FAC-related failure." Hopenfeld Report at 23 (RIV000005).

B. <u>CHECWORKS Is Performing Well at IPEC</u>

Q101. Referencing his prior definitions of FAC, Dr. Hopenfeld states that "CHECWORKS also fails to meet the guidance of the GALL Report because it does not ensure that all forms of FAC will be adequately managed." Hopenfeld Report at 19 (RIV000005). Do you agree?

A101. (JSH, RMA) No. As we have previously shown, not all of the erosive mechanisms to which Dr. Hopenfeld refers are actually forms of FAC. *See* Response to Question 51. Nevertheless, the FAC Program, including CHECWORKS to some degree, addresses those mechanisms to the extent they are significant. Inspection locations for erosive mechanisms are selected based on operating experience, information from other IPEC programs (*e.g.*, thermal performance has identified a leaking valve), and engineering judgment. *See* EN-DC-315 at 18 (ENT000038). Thus, the IPEC FAC program does address these other erosive mechanisms.

Q102. Is CHECWORKS designed to provide a best estimate or a bounding prediction of wear rates due to FAC?

A102. (JSH, RMA) CHECWORKS is designed to provide *best-estimate* wear rates due to FAC, through the PASS-2 analysis described above. Consequently, after calibration half of the inspected components' wall thicknesses will be greater than predicted and the other half less than predicted. This is consistent with Dr. Hopenfeld's observations that CHECWORKS predictions are "*non*-conservative" 40 to 60% of the time. *See*, *e.g.*, Hopenfeld Report at 15 (RIV000005).

Q103. Why is it appropriate for CHECWORKS to follow a best-estimate approach rather than a bounding approach?

A103. (JSH, RMA) It is appropriate for CHECWORKS to follow a best-estimate approach, because, as explained in response to Question 93, above, CHECWORKS is primarily a ranking tool. It functions to assist the FAC Engineer in identifying inspection locations, based on the *rankings* provided by CHECWORKS. *See* NSAC-202L-R3 at 4-6 (RIV000012). Shifting the best estimate approach to a bounding analysis would not assist or improve the ranking process. Entergy uses CHECWORKS to help focus its attention on those components that may be experiencing wear. Then, based on actual, measured data, the appropriate corrective action is taken.

Moreover, a best-estimate approach provides the additional, and considerable, benefit of providing information about the degree of confidence we have in the results for any given Analysis Line. The best-estimate approach provides that through calibration. Systems where the FAC wear correlates well with the predictions will require relatively fewer inspections; more inspections can then be directed towards other lines. This approach therefore allows for optimizing inspections, thus decreasing overall risk.

Q104. Dr. Hopenfeld asserts that his analysis of Entergy's data shows that the IPEC CHECWORKS model "is certainly not currently benchmarked" and that this fact "renders CHECWORKS an ineffective tool for detecting and managing FAC at Indian

Point during the PEO." Hopenfeld Report at 5 (RIV000005); *see also id.* at 6-8 (*citing* RIV000016A-B). Please discuss the exhibits Dr. Hopenfeld highlights in this report.

A104. (IDM, JSH, RMA) The graphs presented in exhibits RIV000016A and RIV000016B are partial results of the PASS-2 analyses for some of the recent SFA reports. Again, PASS-2 analyses compare the measured wall thickness values to the CHECWORKS-predicted values to calibrate the model. *See* NSAC-202L-R3 at 4-1 (RIV000012). The graphs plot the CHECWORKS predictions versus the inspection results, and they are used as inputs to determine whether or not an Analysis Line is calibrated. They are also used to identify specific outlying components within an Analysis Line which may require additional investigations even if the overall Analysis Line is within specifications.

Q105. Do the graphs selected by Dr. Hopenfeld provide an accurate overall representation of the analyses conducted?

A105. (IDM, JSH, RMA) No. Dr. Hopenfeld's selected graphs purportedly show a deficiency in CHECWORKS, by providing examples with poor agreement between wear predictions and actual measurements. In each report, there are counter-examples of good agreement between predictions and measurements. *See*, *e.g.*, IP2 SFA Report 0705.101-01, App. J, Plots J.8, J.16, J.22, J.26, J.32, J.39 (ENT000050); IP3 SFA Report 0705.100-01, App. J, Plots J.6, J.9, J.18, J.28, J.29 (ENT0000051) (each showing good correlation with LCF values in range of EPRI recommendations). As we have explained throughout this testimony, the process described in NSAC-202L, Revision 3, for judging whether or not Analysis Lines are "calibrated" programmatically accounts for lines with less accurate predictions. Under the IPEC FAC Program, lines which are judged to be non-calibrated will generally have more inspections devoted to them. Thus, Entergy recognizes that for certain components predictions are not as reliable, and in those cases actual inspections will be necessary to determine current wall thickness.

As a result, we disagree with Dr. Hopenfeld's opinion that these graphs show CHECWORKS is not properly benchmarked. He has focused on an unrepresentative sample of CHECWORKS results. Contrary to Dr. Hopenfeld's statements, more than 20 years experience with CHECWORKS and predecessor programs has shown that the approach of using CHECWORKS, *together with the other recommendations of NSAC-202L*, has been very effective in virtually eliminating FAC failures in analyzed piping. In addition, the recent IPEC FAC reports listed above show that there are no unique problems with the implementation of CHECWORKS at IPEC, and instead, the performance of CHECWORKS at IPEC is as good as or better than elsewhere in the industry.

Q106. Throughout his Report, Dr. Hopenfeld asserts that the plotted points on these graphs are widely scattered. In his view, the data "clearly demonstrate[ ] that the CHECWORKS model employed at Indian Point cannot predict FAC to any degree of accuracy or precision." Hopenfeld Report at 8 (RIV000005). Do you agree?

A106. (IDM, JSH, RMA) Dr. Hopenfeld's observation that the points on his selected graphs that we discussed in response to the previous question are in some cases widely scattered are true. However, the fact that there are uncalibrated Analysis Lines does not support Dr. Hopenfeld's conclusion that CHECWORKS "cannot predict FAC to any degree of accuracy or precision." Hopenfeld Report at 8 (RIV000005). CHECWORKS can and does predict FAC to a high degree of accuracy in many situations. NSAC-202L-R3 and the IPEC FAC Program recognize the benefits and limitations of CHECWORKS, and consequently have different processes in place to guide inspection location selection, including in those instances where the correlation between predicted and measured values is lacking.

In short, the mere fact that some of the graphs selected by Dr. Hopenfeld show scatter between measurements and predictions demonstrates only that the calibration is poor for those Analysis Lines. In the IPEC CHECWORKS calculations, there are many other graphs that show relatively good correlation, including the examples we have referenced in response to Question 106.

Q107. How does the FAC Program address Analysis Lines that fall outside the desired line correction factor ("LCF") range?

A107. (JSH, IDM, RMA) As we explained previously in response to Questions 86-88, Analysis Lines with an LCF that is out-of-range are addressed on a case-by-case basis, either through increased inspections, piping replacement, or other corrective actions as appropriate, just as uncalibrated Analysis Lines are.

Q108. Dr. Hopenfeld also questions whether a LCF within the range of 0.5 to 2.5 is acceptable. *See* Hopenfeld Report at 8 (RIV000005). In your opinion, is this an acceptable range?

A108. (JSH, RMA) Yes. As we have explained, decades of experience by hundreds of users has shown that when LCFs are within this range CHECWORKS is an effective tool in providing relative ranking of susceptible locations. Moreover, this range is defined by EPRI in NSAC-202L-R3 at 4-1 (RIV000012), a document endorsed by the NRC Staff, not only by Entergy, as Dr. Hopenfeld suggests. *See* SER at 3-24 (NYS00326B). If the LCFs fall outside of this range—or even close to this range—then additional inspections are likely required to improve the characterization of the line. In summary, the LCF is one tool used by Entergy to verify that the CHECWORKS program is performing its intended function, and to identify those instances where additional inspections are required to better characterize localized conditions.

Q109. Dr. Hopenfeld concludes that there is "a complete lack of correlation between predictions and measurements, indicating a very poor predictive accuracy of the CHECWORKS model at Indian Point." Hopenfeld Report at 13 (RIV000005). In support, he states that the CHECWORKS predictions are "highly non-conservative" 40 to 60% of the time. *Id.* Do you agree with Dr. Hopenfeld's conclusions?

A109. (JSH, RMA) No. In our opinion, there is adequate correlation for CHECWORKS to assist Entergy in prioritizing its inspections within the FAC program. FAC is a complex phenomenon and good correlation will not be achieved for all Analysis Lines. The FAC Program accounts for this by focusing more inspection resources on Lines with poor correlation. As explained above, CHECWORKS and its calibration process are designed to provide a best estimate of FAC wear rates. Consequently, CHECWORKS is logically expected to overpredict the wear rate 50% of the time and underpredict the wear rate 50% of the time. Dr. Hopenfeld essentially confirms this with his broad estimate that CHECWORKS underpredicts the FAC wear rate 40% to 60% of the time. *See* Hopenfeld Report at 13 (RIV000005).

Q110. Dr. Hopenfeld also criticizes Entergy's CHECWORKS model reports for failing to provide a single measured value for every predicted point. *See* Hopenfeld Report at 6 (RIV000005). How do you respond?

A110. (JSH, RMA) The reason that there are several points at the same value of predicted wear is that separate components in the same Analysis Line will be predicted to experience the same FAC wear rate. In the instances Dr. Hopenfeld cites (Flow-Accelerated Corrosion Program CHECWORKS Analysis Enhancement, Rev. 0 (00130-TR-001) at I-350, J-

15 (Dec. 2000) (RIV000016A)), CHECWORKS returned identical wear predictions for multiple components with identical inputs. For example, a line might have 5 to 10 elbows. If they are all identical (same geometry, size, operating conditions, chemistry), CHECWORKS will, appropriately, report identical predicted wear rates for each. However, the measured results will not be the same for each individual component. This can occur for different reasons, such as differing amounts of trace chromium in different components, uncertainties in component initial thickness, or uncertainties in NDE measurements.

Q111. Dr. Hopenfeld next turns to the lines labeled "+50%" and "-50%" that appear on the graphs in Exhibits RIV000016A and RIV000016B. *See* Hopenfeld Report at 6 (RIV000005). What is the significance of these lines?

A111. (JSH, RMA) The +/- 50% lines are there for the convenience of the reader of the report, to provide consistent reference points across different graphs. Contrary to Dr. Hopenfeld's suggestion, they are not used by FAC engineers to determine whether any CHECWORKS prediction is "accurate" or "acceptable." Hopenfeld Report at 7 (RIV000005). As we have previously explained, the PASS-2 analysis process focuses on whether the LCF is in range and whether a line is calibrated.

Q112. Dr. Hopenfeld refers to a 1988 paper (*Tackling the Single-Phase Erosion Corrosion Issue* (RIV000017)), where EPRI allegedly asserted "that the computer model was 'predicting erosion-corrosion rates within a +/-50% band.'" Hopenfeld Report at 7 (RIV000005) (*quoting Tackling the Single-Phase Erosion Corrosion Issue* (RIV000017)). He also refers to another EPRI report (EPRI, CHECWORKS Steam/Feedwater Application Guidelines for Plant Modeling and Evaluation of Component Inspection Data (Sept. 2004)

(RIV000018)), in which Entergy purportedly indicates its "belief" that CHECWORKS predictions have a precision of +/-50%. How do you respond?

A112. (JSH, RMA) We would note the vintage of Dr. Hopenfeld's references, the first of which (RIV000017) refers to CHEC rather than CHECWORKS, and provides superseded information, as does the second (RIV000018). The modern approach to the PASS-2 analysis, which is used in the more recent IPEC CHECWORKS reports, provides more valuable information about the goodness-of-fit of the program's predictions, accounting for more complex situations and variables than did the original CHEC program. Finally, we note that Dr. Hopenfeld incorrectly ascribes the statements in Exhibit RIV000018 to Entergy, when this is an EPRI document.

Q113. To illustrate the "safety consequences of using CHECWORKS," Dr. Hopenfeld provides a hypothetical example of what might happen if CHECWORKS underestimates wear rates. *See* Hopenfeld Report at 14 (RIV000005). How do you respond?

A113. (JSH, RMA) Dr. Hopenfeld's hypothetical speculates that Entergy may forego inspections because CHECWORKS may underestimate the time to critical thickness. *See* Hopenfeld Report at 14 (RIV000005). In his hypothetical, he misstates the way the FAC Program uses the information provided by CHECWORKS. Entergy does not inspect components based solely on CHECWORKS predictions of the time to critical thickness. Instead, as we have explained in response to Question 93, CHECWORKS is used to rank components and assist in the selection of inspection locations based on *relative* predicted wear rates. Even modeled components, moreover, can be selected for inspection for many reasons independent of CHECWORKS. Once components are inspected, the re-inspection and replacement decisions are based on the trending of measurements, not the CHECWORKS predictions. So his hypothetical is a simplistic—and incorrect—illustration of how CHECWORKS is used. In any event, Dr. Hopenfeld has himself reviewed thousands of inspection data points, *id*. at 5, showing actual measurements obtained under the FAC Program. Each of these data points is a single value derived from numerous UT thickness measurements on a single component. Given the "ample" (in Dr. Hopenfeld's words) existing data set for actual inspections, *id*. at 20, we are confident that IPEC will not experience the hypothetical scenario that Dr. Hopenfeld postulates.

Q114. In discussing selected graphs, Dr. Hopenfeld concludes that CHECWORKS has a "highly erratic predictive behavior" and is therefore "useless." Hopenfeld Report at 13 (RIV000005). Do you agree with Dr. Hopenfeld's conclusion?

A114. (IDM, JSH, RMA) No. CHECWORKS results are used as just one input into the process of identifying inspection locations. If a given Analysis Line is not calibrated—as is the case in some of the Analysis Lines represented by the graphs Dr. Hopenfeld has selected—then the FAC engineer will consider the calibration status in selecting inspection locations. This is not a deficiency in the FAC Program. More than 20 years of world-wide experience with CHECWORKS and predecessor programs has shown that the approach of using CHECWORKS, together with the other recommendations of NSAC-202L, has been effective in virtually eliminating FAC failures in analyzed piping. Therefore, we disagree with Dr. Hopenfeld's statement that CHECWORKS is "useless" as a tool for identifying locations for inspections.

On the contrary, as we have shown, CHECWORKS is universally accepted in the United States as the best available analytical tool for prioritizing inspections for a FAC program, is used for the same purpose in many other countries, and has been endorsed by the NRC Staff.

C. <u>CHECWORKS Does Not Require Extended Benchmarking Following A Power Uprate</u>

Q115. Dr. Hopenfeld asserts that "CHECWORKS is based on empiricism, i.e. statistics, rather than on a theoretical model. Rather than being based on a mechanistic model, CHECWORKS is solely based on a collection of selective data which represents only a fraction of the total flow area. As a result, the CHECWORKS computer model is not reliable unless it is adequately benchmarked for each component." Hopenfeld Report at 3 (RIV000005). How do you respond?

A115. (JSH, IDM, ABC, RMA) While we find it difficult to understand the meaning of Dr. Hopenfeld's reference to "a fraction of the total flow area," we agree that CHECWORKS is based on empirical data. Many common engineering phenomena and are described by relationships that are based on empirical data. Furthermore, while empirically based, the CHECWORKS wear rate algorithm employs well-known engineering principles from fluid mechanics, mass transfer and unit operations.

And, as we have previously discussed in response to Questions 105 and 106, there is no need for extended post-SPU calibration or benchmarking over a period of multiple years or outage cycles *before* CHECWORKS can be used as part of the FAC Program. CHECWORKS is properly performing its intended role within the IPEC FAC Program by identifying Analysis Lines with the combinations of material, chemistry, and flow conditions that are most likely to experience FAC—and those lines where the rate of FAC cannot be accurately modeled. In fact, one of the goals of developing CHECWORKS was to allow the user to predict the impact of changes of input parameters (*i.e.*, operating conditions and water chemistry parameters).

Consistent with NSAC-202L-R3 and with EN-DC-315, CHECWORKS predictions are compared with measurements at each individual plant. Whether this process is called "refinement" or "calibration" or "benchmarking" is a matter of semantics. This process, the CHECWORKS PASS-2 analysis, is applied for each Analysis Line; not for individual components. Further, the degree of benchmarking he describes—particularly when he refers to the need for CHECWORKS to be "calibrated continuously," Hopenfeld Report at 19 (RIV000005)—goes beyond what is reasonable. Dr. Hopenfeld raised a similar point in the *Vermont Yankee* hearing, and the Board there correctly concluded that, even following a 20% power uprate, "…no further benchmarking is needed since the plant is operating within the range of plant parameters used in benchmarking the model." *Vt. Yankee*, LBP-08-25, 68 NRC at 890. The Board went on to say, "We find that NEC's experts may be misunderstanding the purpose of CHECWORKS in the FAC Program in their attempt to use continuous benchmarking of the model to predict absolute wear." *Id*. at 891. As we have explained, the same conclusions apply to IPEC.

Q116. Is Dr. Hopenfeld correct when he alleges, on pages 3 to 4 of his Report (RIV000005), that, at least for some components, it would take as many as 10 to 15 years following the SPUs at IP2 and IP3 in October 2004, and March 2005, respectively, to "benchmark" CHECWORKS for use at IPEC?

A116. (JSH, IDM, RMA) No. CHECWORKS was designed, and has been shown, to accommodate changes in chemistry, flow rate and other operating conditions that may be associated with power uprates, without inspection data from multiple outages.

For example, a recent study examined the impact of SPUs and EPUs of up to 20% on the FAC programs of 22 U.S. nuclear units and found that, for all of the units, the CHECWORKS predictions reasonably matched actual inspection conditions after the power uprates. *See* EPRI, Plant Engineering: Impact of Electric Power Uprates on Flow-Accelerated Corrosion (July 2011)

(ENT000081). The study concluded that, based on user experience, CHECWORKS properly accounts for the change in FAC wear rates that occur due to power uprates.

Further, post-SPU inspection data have been added to the CHECWORKS database. As noted previously, comparison of the measured wear and CHECWORKS model-predicted wear indicates a level of correlation following SPU implementation that is consistent with the level of correlation at IPEC before power uprates.

Moreover, Dr. Hopenfeld's theory would lead to the incorrect conclusion that CHECWORKS would be unusable as an aid for selecting inspection locations for a decade or more after any power uprate or after inception of a FAC Program. Dr. Hopenfeld previously raised the theory that 10 to 15 years of benchmarking was needed at Vermont Yankee (after a 20% power uprate) during the hearing on that plant's license renewal application, but the Board rejected this claim as "unreasonable and not defensible in light of the goal of CHECWORKS to merely identify locations for plant inspections." *Vt. Yankee,* LBP-08-25, 68 NRC at 890. For the reasons we have provided in response to this question, we agree with the *Vermont Yankee* Board and believe the same conclusion applies to IPEC.

Q117. Did the NRC Staff raise a question regarding "benchmarking" of CHECWORKS in an RAI to Entergy?

A117. (JSH, IDM, ABC) Yes. In response to an NRC Staff request for additional information, Entergy explained why the validity of the CHECWORKS model does not depend on benchmarking against plant-specific measured wear rates of components operating under SPU conditions. *See* NL-08-004, Letter from Fred R. Dacimo, Entergy, to NRC, "Reply to Request for Additional Information Regarding License Renewal Application (Steam Generator Tube

Integrity and Chemistry)," Attach. 1, at 3 (Jan. 4, 2008) (ENT000082). As the RAI response explained:

> In its use throughout the industry, the CHECWORKS model has been benchmarked against measurements of wall thinning for components operating over a wide range of flow rates. Consequently, the validity of the model does not depend on benchmarking against plant-specific measured wear rates of components operating under SPU conditions. In addition, by the time IPEC enters the period of extended operation (in the year 2013), inspection data under SPU conditions will have been obtained. These additional data sets, when added to the CHECWORKS database, will result in more refined wear rate predictions. Since the previously most susceptible locations have been replaced, wear rates are low. Due to the low wear rates, the small changes in operating parameters due to SPU, and the relatively short time since SPU, changes to wear rates since SPU will be very small. The accuracy of the model is not expected to change significantly due to the SPU.

Q118. Do you agree with Entergy's conclusions in its RAI Response?

A118. (JSH, RMA) Yes. The conclusion set forth in Entergy's RAI response is consistent with our experience and professional opinion. Moreover, as we have previously explained in response to Questions 91 and 92 above, the available data now confirms the expectations identified in the RAI response: the validity of the IPEC CHECWORKS model did not change significantly due to the SPU.

Q119. What conclusions did the NRC Staff reach with regard to the need to benchmark CHECWORKS following the SPU?

A119. (IDM, ABC, JSH) In its SER, the NRC Staff concluded that CHECWORKS is "a self-benchmarking" computer code. SER at 3-28 (NYS00326B). As the Staff notes, "the applicant uses the actual UT inspection results to confirm the predictive modeling of the CHECWORKS™ analyses and to perform re-baselined CHECWORKS™ predictive analyses." *Id.* at 3-29. We agree.

Q120. Does CHECWORKS have a successful record of performance at plants that have undergone power uprates?

A120. (JSH) Yes. As we have explained, CHECWORKS is used by more than 150 nuclear units worldwide, including all U.S. units. CHECWORKS has a successful performance record at IPEC and other nuclear plants after implementing EPUs much larger than the SPUs performed at IPEC in 2004 and 2005. As also discussed above in response to Question 116, a recent EPRI study concluded that CHECWORKS could properly model changes in FAC wear rates resulting from EPUs as large as 20%. This is significantly larger than the SPUs at IPEC.

(RMA) In my experience with approximately 20 CHECWORKS EPU and SPU models, the results have been and continue to be effective in predicting post-SPU changes in wear rates and locations attributable to operating conditions.

Q121. What U.S. plants have recently undergone significant power uprates?

A121. (JSH, RMA) Since 2001, the NRC has approved numerous EPUs exceeding 15 percent: Clinton (20%), Dresden Unit 2 (17%), Dresden Unit 3 (17%), Duane Arnold (15.3%), Ginna (16.8%), Point Beach 1 (17%), Point Beach 2 (17%), Quad Cities Unit 1 (17.8%), Quad Cities Unit 2 (17.8%), and Vermont Yankee (20%). *See* Approved Applications for Power Uprates (Oct. 28, 2009), http://www.nrc.gov/reactors/operating/licensing/power-uprates/status-power-apps/approved-applications.html (ENT000083). The same website shows that there are many other plants that have received smaller magnitude uprates in the past decade.

Q122. How has CHECWORKS performed at those plants?

A122. (JSH, RMA) As we have previously explained, EPRI has studied CHECWORKS' performance following significant power uprates, and concluded that CHECWORKS has performed well. *See* EPRI, Plant Engineering: Impact of Electric Power Uprates on Flow-

Accelerated Corrosion at 4-1 (July 2011) (ENT000081). There have been no reported failures in any major steam or feedwater system piping component at any of these plants, each of which has continued to use CHECWORKS since implementation of their respective EPUs. *See id.* at 3-2.

D. The *Vermont Yankee* Board Previously Rejected Many of Dr. Hopenfeld's Various Theories Regarding FAC

Q123. Were you a witness in the Vermont Yankee license renewal hearing, at which Dr. Hopenfeld also testified?

A123. (JSH) Yes.

Q124. Does the Vermont Yankee plant implement the same Entergy fleet-wide FAC Program that is in place at IPEC?

A124. (JSH, IDM) Yes. At the time of the *Vermont Yankee* hearing in 2008, and today, all Entergy plants implement the same fleet-wide procedure, EN-DC-315, which, along with NSAC-202L-R3, establishes the FAC Program. In 2008, there was an earlier version of EN-DC-315 in place, but it was not materially different from the current version. Entergy was relying on NSAC-202L-R3 at the time of the *Vermont Yankee* hearing. In other words, the Entergy program that the *Vermont Yankee* Board found acceptable in 2008 is the same program at issue here.

Q125. At the *Vermont Yankee* hearing, did the intervenor (with Dr. Hopenfeld as an expert witness) raise many of the same issues that Dr. Hopenfeld now raises in his testimony on this contention?

A125. (JSH) Yes. The intervenor's claims in the *Vermont Yankee* hearing included: (1) that CHECWORKS needed extended benchmarking following a power uprate; (2) that CHECWORKS and the FAC Program did not address physical erosion; and (3) that the FAC program relied solely on CHECWORKS. The *Vermont Yankee* Board rejected each of those claims. *See Vt. Yankee*, LBP-08-25, 68 NRC at 889-94.

Q126. Dr. Hopenfeld distinguishes the experience at Vermont Yankee from IPEC by referring to the relatively small size of the Vermont Yankee plant. *See* Hopenfeld Report at 19-20 (RIV000005). Do you agree with this distinction?

A126. (JSH, IDM, RMA) No. The absolute size (*i.e.*, the power output) of a plant is not strongly related to the rates of FAC seen in the various parts of a plant. The main difference between large and small plants will be the mass flow rate through the various systems. Normally, when plants are designed, the architect engineering firm will maintain the same design velocity in the various systems. Although a large power plant will have a higher mass feedwater flow rate than a smaller plant, it will typically have either a larger number of, or larger-sized, pipes. As a result, the velocities (and hence the rate of FAC as we have previous explained in response to Question 50) will remain about the same in different size light water reactors.

Q127. Dr. Hopenfeld also distinguishes the Vermont Yankee example because that plant is a boiling water reactor, while the IPEC plants are pressurized water reactors, which he asserts to be "significantly" more susceptible to FAC. Hopenfeld Report at 20. Do you agree?

A127. (JSH, RMA) No. We disagree with the statement that PWRs are "significantly" more prone to failures from wall thinning due to FAC than BWRs. This statement was true when the issue of FAC emerged after the Surry accident in 1986. However, since that time, the water chemistry practices in PWRs have improved tremendously. *See*, *e.g.,* EPRI, PWR Advanced Amine Application Guidelines, Rev. 2, at 1-1 (Oct. 1997) (ENT000067). We have observed that the rates of FAC in PWRs have been reduced significantly. Thus, throughout the nuclear industry in general, the rates of FAC in the two types of reactors are now roughly comparable.

Also, the data in NUREG/CR-6936 at 5.25 that Dr. Hopenfeld relies upon does not show that PWRs are "significantly" more susceptible than BWRs. Consider the column in Table 5.15 labeled "1988-2005 Through-Wall." *See* NUREG/CR-6936, "Probabilities of Failure and Uncertainty Estimate Information for Passive Components – A Literature Review" at 5.25 (May 2007) (RIV000023). Note that although there are more failures in the PWR row of table 5-15 of this document, there are also considerably more PWRs than BWRs, so that the difference on a per plant basis is insignificant. Further, the data set evaluated in NUREG/CR-6936 includes the late 1980s and the 1990s, before the major improvements in PWR water chemistry took place.

Q128. Next, Dr. Hopenfeld states that post-uprate data was not yet available at the time of the *Vermont Yankee* hearing, but that the available data for IPEC shows that CHECWORKS is not adequately benchmarked. *See* Hopenfeld Report at 20 (RIV000005). Do you agree?

A128. (JSH, RMA) No. This claim is premised on the validity of Dr. Hopenfeld's prior claims regarding IPEC FAC Program data. We have previously shown that his claims are incorrect in response to Questions 105 and 106.

Q129. Dr. Hopenfeld claims that the *Vermont Yankee* Board found "prolonged benchmarking" to be unnecessary because the plant had the benefit of data since 1989. Hopenfeld Report at 20 (RIV000005). In contrast, he states that "Entergy has no CHECWORKS related documentation related to Indian Point Unit 2 generated prior to the year 2000" and that Entergy did not provide any CHECWORKS documentation for IP3 generated prior to 2001. *Id.* at 5; *see also id.* at 20. Because of this, Dr. Hopenfeld

asserts that "more than half of the overall amount of CHECWORKS-related data and documentation has been lost." Hopenfeld Report at 22 (RIV000005). Do you agree with Dr. Hopenfeld's description of the CHECWORKS data available to the IPEC FAC Program?

A129. (IDM, NFA) No. As Entergy explained in its Answer to Riverkeeper Inc.'s Motion to Compel Disclosure of Document (Aug. 13, 2010) ("Answer to Motion to Compel"), Entergy disclosed hundreds of FAC-related documents to Riverkeeper, including all available CHECWORKS reports from IP2 (including reports dating back to 2000) and IP3 (including reports dating back to 1999). Nevertheless, Riverkeeper sought the disclosure of all CHECWORKS reports from IP2 and IP3, dating back to the inception of the IPEC FAC Program. Answer to Motion to Compel at 4.

(IDM, NFA, RMA) Riverkeeper's request sought disclosure of CHECWORKS *reports*. The inspection *data* collected during the outages has been incorporated into and remains available in the IPEC CHECWORKS models. Therefore, the current models include data from inspections, operating data, and chemistry data dating back to original development decades ago. For IP3, the earliest inspection data is from 3R8 in April 1992. This is clearly shown in the SFA model reports that Entergy disclosed to Riverkeeper. *See*, *e.g.*, IP3 SFA Report 0705.100-01, at 21 (ENT000051); *id.* App. F (Section 5.5 discusses contents of Appendix F, Appendix F lists all inspections included in the CHECWORKS model back to 3R8 in 1992). For IP2, the most recent SFA model reports list all inspections conducted since 2R16 in 2006. *See* IP2 SFA Report 0705.101-01, at 17 & App. F (ENT000050). Data collected prior to that outage, however, is still reflected in the CHECWORKS model for IP2. *See* Indian Point Unit 2, CHECWORKS SFA Model Calculation No. 050714b-01, Rev. 0, § 3 (July 5, 2005) (ENT000074) ("The

CHECWORKS model reflects plant design and operation through Refuel Outage 16. All historical records (i.e. inspections, replacements, water chemistry, power levels, etc.) through Refuel Outage 16 were included in this analysis.").

Q130. The final distinction that Dr. Hopenfeld seeks to draw between IPEC and Vermont Yankee is that at Vermont Yankee, the Board found that "only a small fraction of inspection locations were based on the use of CHECWORKS." Hopenfeld Report at 20 (RIV000005). In contrast, he asserts that the IPEC FAC Program relies "integrally" on CHECWORKS to determine inspection locations. *Id.* Later in his Report, Dr. Hopenfeld asserts that Entergy's FAC Program "primarily" relies on CHECWORKS, with no other meaningful independent tools. *See*, *e.g.*, Hopenfeld Report at 26 (RIV000005). How do you respond?

A130. (IDM, ABC, RMA, JSH) This assertion is incorrect. The *Vermont Yankee* Board found the Entergy corporate FAC Program to be acceptable, and IPEC follows the same program. *See* EN-DC-315 at 3 (ENT000038). Under that program, as we have previously demonstrated, CHECWORKS is only one of several tools used to select inspection locations. As Riverkeeper points out, in *Vermont Yankee* the Board found, based on Entergy's testimony, that "only one-third of the inspection locations were based on the results from CHECWORKS." *Vt. Yankee*, LBP-08-25, 68 NRC at 881; *see also* Position Statement at 31 (RIV000002). As we have previously shown, at IPEC, between one-quarter and one-third of inspection locations are based on CHECWORKS, as is clearly shown in documents Entergy has disclosed to Riverkeeper. *See*, *e.g.,* Scope of Flow-Accelerated Corrosion Inspection Points for 3R14 Outage (Apr. 2, 2007) (ENT000061). This is consistent with the data from Vermont Yankee.

E. Riverkeeper's Criticisms Based on Selected IPEC Operating Experience Lack Merit

Q131. On page 17 of his Report, Dr. Hopenfeld describes Entergy's 2008 Operating Experience Review Report IP-RPT-06-LRD05, Rev. 3 (Entergy Engineering Report, Operating Experience Review Report, IP-RPT-06-LRD05, Rev. 3 (2008) ("Entergy Op Ex Rev Report") (RIV000024)) as documenting "numerous leaks and reports of excessive wall thinning" between 2001 and 2006. Hopenfeld Report at 17 (RIV000005). According to Dr. Hopenfeld, this document shows that "CHECWORKS has not been successful at preventing FAC-related occurrences." ***Id.*** **Do you agree with this description?**

A131. (IDM, ABC) No. Most of the items listed in Exhibit RIV000024 are unrelated to FAC. Most of the FAC-related entries are condition reports documenting wall thinning revealed during inspections conducted under the FAC Program. Those examples illustrate no deficiency in the FAC Program. To the contrary, they illustrate the successful operation of the program in identifying loss of material due to FAC for correction prior to a loss of component intended function. Dr. Hopenfeld, however, draws no distinctions between events involving FAC-susceptible systems or other systems; nor does he acknowledge that most of the FAC-related items listed in his exhibit are program successes, not deficiencies.

Regarding the leaks over the five-year period covered by RIV000024, there were no more than 16 instances of FAC-related leaks identified within the extensive secondary-side or balance-of-plant piping systems covered by the IPEC FAC Program at the two units. RIV000024 also shows a declining trend in deficiencies after Entergy purchased IP2 and IP3 in 2001 and 2000, respectively, as there were 13 leaks across both plants in 2001 and 2002, but only 3 in the next three years. Entergy documented each of these events for corrective actions under the IPEC corrective action program. None of these items occurred on safety-related piping; they occurred on balance-of-plant piping, and therefore these examples have low or no nuclear safety significance. Only *one* of the 16 leaks was in piping modeled with CHECWORKS. Entergy's discovery, documentation, and correction of these low-safety-significance items do not show any deficiency in the FAC Program, much less a deficiency with CHECWORKS.

Similarly, Dr. Hopenfeld points to a two-part list of condition reports that Entergy disclosed to Riverkeeper, spanning the years 1995 to 2008. *See* Hopenfeld Report at 17 (RIV000005) (*citing* Entergy Wall Thinning Condition Report Part 1 (Sept. 2008) (RIV000026); Entergy Wall Thinning Condition Report Part 2 (Sept. 2008) (RIV000027)). Of the 171 entries in this report, 103 document wall thinning or inspections in FAC-susceptible systems. Of those 103 entries, 101 documented the results of inspections under the FAC Program or other programs. These events, therefore, do not indicate any deficiency in the FAC Program.

Q132. In the same paragraph, Dr. Hopenfeld similarly describes a series of Entergy Condition Reports and other documents as demonstrating deficiencies in CHECWORKS or the FAC Program. Do you agree?

A132. (IDM, ABC) No. The first example was the discovery of a pinhole leak in the crossunder piping leading to the IP2 Moisture Separator Reheater "MSR-21A" in 2001. *See* Entergy Condition Report, CR-IP2-2001-10525 (Oct. 31, 2001) (RIV000028). Although this leak was attributed to FAC, it does not reflect on CHECWORKS's predictive value because the cross-under piping segment at issue is an example of large bore susceptible-non-modeled piping. This piping was known to be susceptible to FAC, and the piping was weld overlaid with stainless steel to prevent wear, except for certain areas near expansion joints, where this could not be done. The pinhole leak was discovered within a small section of piping that was not clad with stainless steel. Because this was a pinhole leak, this event was of negligible safety significance as the affected pipe section remained structurally capable of performing its intended functions and did not pose a danger to plant personnel.

The second example is wall thinning that occurred in a pipe downstream of the main stream traps. *See* Daily DER, DER-01-01522 (Apr. 25, 2001) (RIV000025). This event was one of the 16 FAC-related items that are listed in RIV000026, and addressed in response to Question 131, above. While the thinning was attributable to FAC, like the first example, this pipe is susceptible-non-modeled (*i.e.*, it is not modeled by CHECWORKS). The wall thinning was noted during a routine inspection and the component was replaced. Further inspections revealed that other components on the line were not leaking, but another inspection was scheduled for the next outage, nonetheless. In other words, this event was used *directly* as operating experience input into the selection process for FAC inspections.

The third example is a small leak found downstream of level control valve 1104 leading to a feedwater heater at IP3 in 2006. *See* Entergy Condition Report, CR-IP3-2006-02270 (July 23, 2006) (RIV000029). The Entergy Condition Report documents the corrective actions taken, including the repair of the leak and additional FAC inspections conducted as part of the extent of condition review. *See generally* CR-IP3-2006-02270 (RIV000029). Thus, this event was also used as operating experience to inform the FAC inspection scope for the next outage.

Taken together, the events described in Riverkeeper's exhibits—all of low nuclear and industrial safety significance, and all properly addressed through the IPEC FAC and corrective action programs—simply do not show that CHECWORKS is failing to perform its intended function at IPEC, or that there is any deficiency in the IPEC FAC Program. In particular, the purpose of the FAC Program is to provide reasonable assurance that systems and components will continue to perform their intended functions, and none of Riverkeeper's examples involve an event where such a function was compromised. It is also incorrect to suggest that the reasonable assurance standard in 10 C.F.R. § 54.29 demands perfection or the absence of any leaks—however small—at all times during the plant's operating history. *See also* NSAC-202L-R3 at 1-3 (RIV000012). Instead, the FAC Program is designed to specify inspections, evaluations, and corrective actions to provide reasonable assurance that the plant will continue to be operated in accordance with the CLB through the PEO.

Q133. Based on these exhibits, Dr. Hopenfeld concludes that "the number of failures at Indian Point appear to be significantly higher than in the rest of the industry where only 15 leaks were reported in the 18 month period between January 2000 and July 2001." Hopenfeld Report at 18 (RIV000005). Please respond to his conclusion.

A133. (IDM, ABC) His conclusion is based on an illogical and inappropriate comparison. Dr. Hopenfeld appears to base his conclusion on the entry from the Entergy OE report for CR-IP3-2001-03019 (RIV000024). That CR documents industry OE from an INPO OE report. It specifically says "[b]etween January 1, 2000 and July 1, 2001, the industry reported approximately 15 events involving leakage from reactor coolant system (RCS) piping, penetrations, or components. Events reported included leakage from control rod drive housings, hot leg nozzles, etc." Entergy Op Ex Rev Report at 43 (RIV000024). Clearly, these 15 events involve RCS components rather than components susceptible to FAC. Dr. Hopenfeld's conclusion that IPEC has a significantly higher rate of "FAC-related failures" than in the rest of the industry based on a comparison to the number of RCS leaks throughout the industry is baseless. *See* Hopenfeld Report at 18.

Q134. Dr. Hopenfeld also asserts that "the NRC Staff in this license renewal proceeding has also questioned Entergy regarding several incidences of component wall

thinning that were below minimum acceptable levels," referring to the transcript of a meeting of the ACRS. Hopenfeld Report at 17 (RIV000005). How do you respond to this statement?

A134. (IDM, ABC) While the NRC Staff did "question" Entergy regarding certain incidences of wall thinning discovered through FAC Program activities, Entergy answered those questions to the satisfaction of the NRC Staff, and to the satisfaction of the ACRS. For example, when thinning was discovered in the IP3 vent chamber drain pipe during refueling outage 13, Entergy took appropriate corrective actions. The ACRS testimony shows that Entergy replaced the affected piping, performed additional inspections to determine the extent of condition for similar piping, and continued the inspection sample expansions until no additional components were detected with significant wear. *See* ACRS Meeting Tr. at 91-92 (Sept. 10, 2009) (RIV000030). These activities, as described, exemplify the successful implementation of EPRI guidance for an effective FAC program. Accordingly, the ACRS transcript reveals an example of the effectiveness of the IPEC FAC Program, rather than a deficiency.

F. Riverkeeper's Criticisms Based on Selected Operating Experience at Other Facilities Lack Merit

Q135. In his Report, Dr. Hopenfeld asserts that "[n]umerous instances of undetected FAC have previously resulted" in failures, citing events at the Surry, San Onofre, Fort Calhoun, and Mihama plants. Hopenfeld Report at 3 (RIV000005). Does this operating experience indicate any problems with CHECWORKS and its ability to perform its intended function at IPEC?

A135. (JSH) No. The examples Dr. Hopenfeld cites are in many cases quite dated. The industry has made great strides since the 1980s in addressing FAC, including the development of NSAC-202L and CHECWORKS, both of which are now approved by the NRC Staff for use in addressing FAC. Notably, none of the examples that Dr. Hopenfeld cites involved plants that were properly implementing either NSAC-202L or CHECWORKS.

(JSH) Thus, the operating experience cited by Dr. Hopenfeld does not indicate any problems in the use of CHECWORKS as part of a FAC Program, at IPEC or anywhere else. The Surry plant had no FAC program when the 1986 pipe rupture event occurred—well before CHECWORKS had been written and the EPRI guidance had been published. In fact, the Surry accident resulted in the writing of CHEC, the first EPRI computer program used to predict FAC. *See* NSAC-202L-R3 at 1-1 (RIV000012).

(JSH) At San Onofre, the cited events occurred within the plant's steam generators. CHECWORKS was not used to analyze these components (and, even today, it is not commonly used to analyze piping components located within equipment such as steam generators).

(JSH, RMA) The 1997 pipe rupture at Fort Calhoun was not due to any defect in CHECWORKS or EPRI guidance. Specifically, although Fort Calhoun had used CHECWORKS, it made several errors, including a data input error that severely biased the model and was a direct cause for omitting inspection of the failure location. *See* NRC – Information Notice–IN 97-84, Rupture in Extraction Steam Piping as a Result of Flow-Accelerated Corrosion (Dec. 11, 1997) (ENT000084).

(JSH, RMA) Finally, the Mihama plant had a FAC program that did not use CHECWORKS or EPRI guidance (*i.e.*, NSAC-202L) when the 2004 incident occurred.

(JSH, RMA) The Entergy FAC Program uses experienced and qualified engineers who are primarily assigned to the FAC Program, and incorporates specified peer reviews, independent assessments, and self-assessments, in order to minimize the potential for the types of problems experienced at Fort Calhoun and Mihama. For example, based on the Mihama operating experience, Entergy performed specific additional FAC inspections. *See* Entergy Corrective Action LO-NOE-2006-00611-CA13, at 3 (July 2006) (ENT000085) ("The piping components upstream and downstream of the boiler feed pump suction RO's [restricting orifice], similar to the failure location at Mihama, were inspected during 2R16 and 3R13. No wall thinning was evident from any of the inspections.").

Q136. With respect to the events at Mihama, Dr. Hopenfeld states that the pipe rupture at that plant "occurred at low velocity in a straight pipe, downstream of an orifice, and CHECWORKS does not model such flow situations correctly." Hopenfeld Report at 17 (RIV000005). Do you agree with that statement?

A136. (JSH) No. Dr. Hopenfeld correctly describes the location of the Mihama rupture, but not CHECWORKS' capabilities. In developing CHECWORKS, the development team used a great deal of laboratory data from the United Kingdom and France. In particular, some of the single-phase data from the UK showed the FAC rate downstream of an orifice, such as Mihama. These data were used extensively in developing the algorithm used in CHECWORKS. Thus, CHECWORKS is capable of correctly modeling the flow conditions that existed at Mihama.

Q137. Dr. Hopenfeld also asserts that NUREG/CR-6936, PNNL 16186, "Probabilities of Failure and Uncertainty Estimate Information for Passive Components – a Literature Review" (May 2007) (RIV000023) ("NUREG/CR-6936"), further demonstrates CHECWORKS' "limited effectiveness to predict wall thinning." Hopenfeld Report at 16 (RIV000005). Does this document indicate any deficiency in CHECWORKS?

A137. (JSH) No. The data on which NUREG/CR-6936 relies (draft NUREG-1829) includes worldwide operating experience that encompasses plants that have never used CHECWORKS or did not use CHECWORKS for the entire period of time specified in Table

5.15. *See* NUREG/CR-6936, Tbl. 5.15, at 5.25 (RIV000023); NUREG-1829, Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process (NUREG-1829) - Draft Report for Comment, App. D at D-15 (June 2005), *available at* ADAMS Accession No. ML051520574 (ENT000086). Furthermore, the data provide no indication of how many of the failures occurred in lines that were modeled in CHECWORKS.

Furthermore, rather than indicating deficiencies in FAC programs, NUREG/CR-6936 states that Table 5.15 "shows the pre-1987 and post-1987 service experience as an indication of the *effectiveness* of FAC mitigation programs implemented by the industry in the aftermath of lessons learned from FAC-induced pipe failures at Trojan in 1985 and Surry Unit 2 in 1986." NUREG/CR-6936 at 5.25 (emphasis added) (RIV000023). It further states that "[t]he cause and effect of FAC is well understood, and the industry has implemented FAC inspection programs, as well as piping replacement using FAC-resistant materials such as stainless steel, carbon steel clad on the inside diameter with stainless steel, or chrome-molybdenum alloy steel." *Id.* NUREG/CR-6936 thus indicates that industry has made significant progress in addressing FAC.

Q138. Dr. Hopenfeld next asserts that "pipe thinning events have occurred in recent years (and since the publication of NUREG/CR-6936) at numerous nuclear power plants across the United States, including Duane Arnold, Hope Creek, Clinton, Braidwood, LaSalle, Peach Bottom, Palo Verde, Palisades, Catawba, Calvert Cliffs, Kewaunee, Browns Ferry, ANO, and Salem." Hopenfeld Report at 16 (RIV000005). Given that NUREG/CR-6936 was issued in 2007, please respond.

A138. (JSH, RMA) Dr. Hopenfeld does not explain what he means by "pipe thinning events," or "numerous leaks and pipe ruptures," as he states later in that paragraph in his Report. It is also not clear what he means by "recent years," and, most importantly, he provides no citation or reference to any specific event. Therefore, we cannot comment on these nonspecific claims.

Q139. In its Motion for Summary Disposition, Entergy stated that there has been no "major FAC-caused pipe rupture in a U.S. nuclear unit for more than 10 years." Applicant's Motion for Summary Disposition of Riverkeeper Technical Contention 2 (Flow-Accelerated Corrosion) at 23 (July 26, 2010), *available at* ADAMS Accession No. ML102140430. Dr. Hopenfeld takes issue with this statement as "purely circumstantial," and that it "does not change the reality that FAC that violates applicable standards has been well-documented across the industry, clearly demonstrating that CHECWORKS has not been successful." Hopenfeld Report at 16-17 (RIV000005). How do you respond?

A139. (JSH, RMA) Dr. Hopenfeld does not dispute that there have been no "major FAC-caused pipe ruptures in a U.S. nuclear units in more than ten years." While, Dr. Hopenfeld provides no facts to support the alleged "well-documented" counter-examples, it appears that he is referring to numerous examples of the IPEC FAC Program identifying wall thinning leading to piping replacement or repair prior to loss of the component intended function. As we have previously discussed in response to Questions 131 to 132, above, these are examples of the effective performance of the FAC Program; not demonstrations of any lack of success.

Q140. What is the significance of the ACRS subcommittee meeting transcript for the Waterford EPU cited on page 16 of Dr. Hopenfeld's Report?

A140. (RMA, JSH) The pertinent portions of the transcript simply confirm that licensees use CHECWORKS as one tool among many to determine inspection priorities, confirm CHECWORKS wear-rate predictions through inspection data, and make appropriate adjustments to their CHECWORKS models based on that data.

(RMA) For example, at the same meeting, I explained as follows:

> Some runs [*i.e.*, Analysis Lines] results are imprecise and some more precise. And we look at both accuracy and precision. Programmatically we account for that, that reality, by treating those runs that have what we call well calibrated results, *i.e.*, precise and accurate results coming out of the model that are substantiated by observations, we treat those piping segments differently programmatically than we do areas where the model is less good. If the model results do not correlate well with reality, different actions are taken primarily increased inspection coverage to increase our level of confidence that those systems can continue to operate safely.
>
> In addition to the CHECWORKS results many other factors are considered to assure that the piping retains its integrity, chief among these are industry experience as exchanged through the EPRI sponsored CHUG group. Plant experience local to Waterford in this case. And the FAC program owner maintains an awareness of the operational status of the plant so that, for example, modifications or operational changes that occur are taken into account in the inspection of the secondary [side] FAC susceptible piping.

ACRS Subcommittee on Thermal-Hydraulic Phenomena Meeting Tr. at 245-46 (Jan. 26, 2005), *available at* ADAMS Accession No. ML050400613 (ENT000087).

(JSH, RMA) These practices are reflected in Entergy fleet procedure EN-DC-315 and the NSAC-202L-R3 guidelines upon which that procedure is based. In approving the Waterford EPU discussed during the January 2005 ACRS subcommittee meeting, the NRC noted that the licensee had submitted a comparison of predicted wall thickness versus measured wall thickness of sample piping, and that "the CHECWORKS prediction at Waterford 3 has been demonstrated to be adequate." Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 199 to Facility Operating License No. NPF-38, Entergy Nuclear Operations, Inc., Waterford Steam Electric Station, Unit 3, Docket No. 50-382, at 19 (Apr. 15, 2005), *available at* ADAMS Accession No. ML051030068 (ENT000088). Thus, while Dr. Hopenfeld points to Dr. Ford's *questions* regarding CHECWORKS, he ignores the expert response provided and the ACRS's ultimate conclusion. The ACRS reached a similar conclusion in this proceeding. *See* SER at 5-2 (NYS00326E).

G. Riverkeeper's Programmatic Challenges to the FAC AMP Lack Merit

Q141. Dr. Hopenfeld concludes his Report by claiming that, to comply with NUREG-1801 and the SRP-LR, with respect to "the method for determining component inspections, frequency of such inspections, and attendant criteria for component repair and replacement . . . Entergy cannot simply rely on procedural documents which depend upon the proper use of CHECWORKS." Hopenfeld Report at 25 (RIV000005). As a result, he asserts that Entergy's FAC AMP is deficient. How do you respond to this conclusion?

A141. (IDM, ABC, JSH, RMA) Entergy does not "simply rely on procedural documents which depend upon the proper use of CHECWORKS." Hopenfeld Report at 25 (RIV000005). As indicated in the LRA, the IPEC FAC Program is consistent with the recommendations in NUREG-1801, Revision 1, at XI M-61 (NYS00146C). *See* LRA, App. B at B-54 (ENT00015B). It also meets the intent of the guidance in NUREG-1801, Revision 2, as we have previously explained in response to Question 48. It is a comprehensive detailed program defined by the guidance of NSAC-202L-R3. As explained throughout our testimony, the program entails much more than CHECWORKS.

H. Riverkeeper's Assorted New Challenges to the FAC Program Lack Merit

Q142. In Section VI of Dr. Hopenfeld's report, he states that "undetected FAC during the extended operating terms at Indian Point also poses a risk of loss of coolant accidents ('LOCA') in violation of NRC's General Design Criterion ('GDC') 4, which requires plant structures, systems and components be able to 'accommodate the effects of . . . loss of coolant accidents' and 'be appropriately protected against dynamic effects . . .

that may result from equipment failures and from events and conditions outside the nuclear power unit." Hopenfeld Report at 24 (RIV000005). How do you respond?

A142. (NFA, ABC) FAC affects secondary plant systems, not primary systems, which are fabricated from or clad with FAC-resistant material such as stainless steel. To the extent that Dr. Hopenfeld's argument can be construed as an allegation that FAC-susceptible components could fail as the result of a LOCA, this argument presupposes a deficiency in the FAC Program, which is not the case. And as we have previously noted in response to Question 62, the vast majority of FAC-susceptible components are non-safety-related components that perform no accident mitigation function. As has been shown throughout this testimony, the FAC Program provides reasonable assurance that components within its scope will continue to perform their intended functions throughout the PEO.

Q143. Next, he asserts that Entergy "has failed to consider how the uncertainty related to pipe wall thickness at Indian Point will affect the integrity of components under transient loads other than plant transients, such as earthquakes and station blackouts." Hopenfeld Report at 25 (RIV000005). How do you respond?

A143. (NFA, ABC) As with the previous issue, these arguments all assume a deficiency in the FAC Program, which does not exist. As has been shown throughout this testimony, the FAC Program provides reasonable assurance that components within its scope will continue to perform their intended functions throughout the PEO. When FAC Program inspections reveal wall thinning, that data is evaluated against the appropriate design loading conditions, including seismic loads, if applicable. When a component is found with insufficient wall thickness, that component is repaired or replaced prior to returning it to service. Finally, the basis for Dr. Hopenfeld's allegation regarding SBO is unclear. An SBO event involves a loss of station alternating current (AC) power. The secondary side of the plant cannot operate without AC power. Consequently, most of the systems susceptible to FAC are shutdown and at greatly reduced system pressures during an SBO event. Dr. Hopenfeld does not articulate any theory of why he believes that FAC-susceptible components will fail during an SBO event.

Q144. Finally, he claims that "Entergy has not considered how the operation of Indian Point with such large uncertainties about pipe wall thicknesses will affect the likelihood of components succumbing to the effects of metal fatigue." Hopenfeld Report at 25 (RIV000005). How do you respond?

A144. (NFA, ABC) As an initial matter, Dr. Hopenfeld's references to "large uncertainties" presuppose a deficiency in the IPEC FAC Program, which as shown throughout this testimony, is not the case. In any event, the requirements for fatigue design in the ASME Code, Section III that require the calculation of cumulative usage factors apply to the reactor coolant and other primary systems, but do not apply to the secondary plant systems that are susceptible to FAC. The rules of the ANSI B31.1 Code used to design the secondary, balance of plant systems do not require CUF calculations. *See* NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," at 2-1 (NYS000355). Instead, ANSI B31.1 uses lower stress allowables to accommodate up 7000 cycles. *See id.* Satisfying the ANSI B31.1 design rules ensures that appropriate margins exist against fatigue cracking. Since the IPEC FAC Program requires that any wall thinning below 87.5% of nominal thickness detected during inspections be evaluated per the system design requirements, including the rules of ANSI B31.1, *see* EN-DC-315 at 21 (ENT000038); EN-CS-S-008-MULTI at 4 (ENT000065), the program ensures that the ANSI B31.1 stress allowables are maintained, thereby adequately protecting components against fatigue cracking.

As to the potential for FAC to affect primary plant components that are clad with stainless steel, Dr. Hopenfeld has conceded that this is not a concern. *See* Riverkeeper, Inc. Opposition to Entergy's Motion in Limine to Exclude Portions of Pre-filed Testimony, Expert Report, Exhibits, and Statement of Position for Contention NYS-26B/RK-TC-1B (Metal Fatigue) Attach. 1 (Feb. 17, 2012) (Declaration of Joram Hopenfeld ¶ 21 ("I am well aware that stainless steel is not affected by flow accelerated corrosion.")).

V. CONCLUSION

Q145. Please summarize your testimony and the bases for your conclusion that Riverkeeper TC-2 lacks factual and technical merit.

A145. (JSH, IDM, ABC, RMA, NFA) Riverkeeper TC-2 lacks merit for the following principal reasons:

- The IPEC FAC Program is consistent with the ten program elements recommended in NUREG-1801, Revision 1, Section XI.M17, and meets the intent of NUREG-1801, Revision 2, Section XI.17, without exception.

- Entergy fleet procedure EN-DC-315, *Flow Accelerated Corrosion Program*, implements the recommendations of NUREG-1801 and the more detailed EPRI guidelines contained in NSAC-202L-R3.

- The decision to repair or replace piping and piping components at IPEC is not based solely or directly on CHECWORKS predictions, but on measured wall thickness of the relevant components, or specific operating experience.

- CHECWORKS is only one of several independent tools and processes used at IPEC to assist Entergy in selecting component locations for inspection in order to avoid the potential adverse effects of FAC. It is used in conjunction with other sources of information, such as the trending of pipe wall thickness measurements from past inspections; industry operating experience related to FAC, data from other plant inspection programs, condition reports, and engineering judgment.

- The CHECWORKS predictive algorithms are based on laboratory data and operating data from many plants. As such, CHECWORKS has been "benchmarked" against measurements of wall thinning for components operating over a wide range of plant operating parameters. CHECWORKS was designed to, and does in fact, account for changes in plant chemistry, flow rate, and other operating parameters.

- There is no need to further "benchmark" CHECWORKS against plant-specific measured wear rates of components operating under post-SPU conditions. Input of new plant characteristics associated with a power uprate (*e.g.*, flow rate and temperature) is all that is required and has been done for IPEC.

- CHECWORKS continues to perform well at IPEC under post-SPU conditions. Comparison of the measured wear and CHECWORKS model-predicted wear indicates a level of correlation following SPU implementation that is consistent with the level of correlation prior to the SPU implementation. The performance of CHECWORKS at IPEC indicates that the program is properly performing its intended function as one aspect of the overall IPEC FAC Program, by focusing the attention of the IPEC FAC Program on piping lines where the rates of FAC are higher or less predictable.

- None of the documents cited by Riverkeeper, including IPEC condition reports, ACRS testimony, or other industry operating experience suggest that there is a substantial deficiency in the IPEC FAC Program. A review of these documents shows the effectiveness of a FAC program implemented in accordance with EPRI guidance, as is the case at IPEC.

- Entergy's FAC Program therefore provides reasonable assurance that, for FAC-susceptible components, the effects of aging will be adequately managed so that their intended functions will be maintained consistent with the CLB for the PEO, as required by 10 C.F.R. § 54.21(a)(3).

Q146. Does this conclude your testimony?

A146. (JSH, IDM, ABC, RMA, NFA) Yes.

Q147. In accordance with 28 U.S.C. § 1746, do you state under penalty of perjury that the foregoing testimony is true and correct?

A147. (JSH, IDM, ABC, RMA, NFA) Yes.

Executed in accord with 10 C.F.R. § 2.304(d)

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