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

Before Administrative Judges:

Alex S. Karlin, Chairman
Dr. Anthony J. Baratta
Lester S. Rubenstein

In the Matter of

ENTERGY NUCLEAR VERMONT YANKEE
L.L.C.
and
ENTERGY NUCLEAR OPERATIONS INC.

(Vermont Yankee Nuclear Power Station)

Docket No. 50-271-OLA

ASLBP No. 04-832-02-OLA

February 26, 2007

INITIAL DECISION
(Ruling on NEC Contention 3)

TABLE OF CONTENTS

I.	INTRODUCTION.....	1
II.	BACKGROUND.....	2
	A. Procedural History.....	2
	B. Witnesses.....	10
	1. Entergy Witnesses.....	10
	2. NRC Staff Witnesses.....	12
	3. NEC Witness.....	16
III.	GOVERNING LEGAL STANDARDS.....	17
IV.	FINDINGS OF FACT.....	19
	A. Basic Factual Findings.....	19
	1. Definitions and Basic Concepts.....	19
	2. Staff Guidance Relating to Large Transient Testing.....	27
	B. Factual Findings on Key Contested Issues.....	32
	1. Assertions of Parties - Overview.....	32
	2. Contested Issue One - Existence of new thermal-hydraulic phenomena and/or new system interactions.....	34
	a. Key Evidence Presented.....	34
	b. Board Findings.....	38
	3. Contested Issue Two - Adequacy of computer stress analysis.....	40
	a. Key Evidence Presented.....	40
	b. Board Findings.....	42
	4. Contested Issue Three - Adequacy of ODYN code benchmarking for pressurization transients for steady state operation.....	42
	a. Key Evidence Presented.....	42
	b. Board Findings.....	47
	C. Ultimate Factual Finding - Need for Large Transient Testing.....	49
V.	CONCLUSIONS OF LAW.....	53
VI.	ORDER.....	54

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I. INTRODUCTION

This initial decision concerns an application submitted by Entergy Nuclear Vermont Yankee L.L.C. and Entergy Nuclear Operations, Inc. (collectively, Entergy) to amend the operating license for the Vermont Yankee Nuclear Power Station (VYNPS) in Windham County, Vermont. The proposed amendment, if approved, would authorize a 20-percent increase in the maximum power level of the plant from 1,593 megawatts thermal (MWt) to 1,912 MWt (referred to as an extended power uprate or EPU). The proposed amendment would also authorize certain associated changes to the technical specifications for the VYNPS. The New England Coalition (NEC), an environmental organization, challenged the application, asserting that the license amendment should not be granted unless "large transient testing" is required as a

license condition.¹ After considering the evidence and arguments, we conclude that Entergy has met its burden of showing that it is not necessary to perform the testing proposed by NEC in order to satisfy the relevant legal requirement -- 10 C.F.R. Part 50, Appendix B, Criterion XI -- and thus deny NEC's contention.

II. BACKGROUND

A. Procedural History

In September 2003, Entergy submitted its EPU application to the Commission to amend the VYNPS operating license.² On July 1, 2004, the Commission issued a notice of consideration of issuance of the proposed amendment and opportunity for a hearing. 69 Fed. Reg. 39,976 (July 1, 2004). In response, NEC and the Department of Public Service of the State of Vermont (State) filed timely petitions to intervene, each requesting admission as a party to any proceeding concerning Entergy's application.³ NEC initially proposed seven contentions, and the State proposed five.

On September 14, 2004, this Board was established to rule on the petitions and to preside over any adjudicatory proceeding in connection with Entergy's license amendment

¹ "Large transient tests" are tests intended to demonstrate that the plant will operate in accordance with design specifications both during normal steady state conditions and, to the extent practical, during and following anticipated operational occurrences, such as main steam isolation valve (MSIV) closures and a generator load rejections (GLRs). NRC Staff Testimony of Richard B. Ennis, Steven R. Jones, Robert L. Pettis Jr., George Thomas, and Zeynab Abdullahi Concerning NEC Contention 3 (May 17, 2006) at 7 (fol. Tr. at 1383) [Ennis et al. Direct Testimony for NRC Staff].

² Letter from Jay K. Thayer, Entergy Site Vice President, to U.S. Nuclear Regulatory Commission, Document Control Desk, Vermont Yankee Nuclear Power Station License No. DPR-28 (Docket No. 50-271) Technical Specification Proposed Change No. 263 Extended Power Uprate (Sept. 10, 2003), ADAMS Accession No. ML032580089 [Application].

³ New England Coalition's Request for Hearing, Demonstration of Standing, Discussion of Scope of Proceeding and Contentions (Aug. 30, 2004) [NEC Petition]; Vermont Department of Public Service Notice of Intention to Participate and Petition to Intervene (Aug. 30, 2004).

application. 69 Fed. Reg. 56,797 (Sept. 22, 2004). On November 22, 2004, the Board found that both parties had standing and admitted two NEC contentions and two State contentions. LBP-04-28, 60 NRC 548, 553-54, 558-564, 571-73 (2004). As originally admitted, the four contentions read as follows:

State Contention 1: Entergy has claimed credit for containment overpressure in demonstrating the adequacy of ECCS pumps for plant events including a loss of coolant accident in violation of draft General Design Criteria 44 and 52 and therefore Entergy has failed to demonstrate that the proposed uprate will provide adequate protection for public health and safety as required by 10 C.F.R. § 50.57(a)(3).

State Contention 2: Because of the current level of uncertainty of the calculation which the applicant uses to demonstrate the adequacy of ECCS pumps, the Applicant has not demonstrated that the use of containment overpressure to provide the necessary net positive suction head for ECCS pumps will provide adequate protection for the public health and safety as required by 10 C.F.R. § 50.57(a)(3).

NEC Contention 3: The license amendment should not be approved unless Large Transient Testing is a condition of the Extended Power Uprate.

NEC Contention 4: The license amendment should not be approved because Entergy cannot assure seismic and structural integrity of the cooling towers under uprate conditions, in particular the Alternate Cooling System cell. At present the minimum appropriate structural analyses have apparently not been done.

Id. at 580.

Subsequently all of the admitted contentions except for NEC Contention 3 were settled, withdrawn, or otherwise resolved. State Contentions 1 and 2 were settled. On May 2, 2006, the State filed a notice of withdrawal and request for dismissal of the two contentions which indicated that the State and the Applicant had “agreed to a mutually satisfactory resolution of the issues raised by the State in this proceeding.”⁴ The State subsequently modified this notice

⁴ Notice of Withdrawal and Request for Dismissal of Contentions of the Vermont Department of Public Service (May 2, 2006) at 1.

to conform to the requirements of 10 C.F.R. § 2.338(g) and (h) regarding the form and content of settlements.⁵ The Board approved the modified settlement agreement and dismissed the State's two contentions on June 23, 2006.⁶ Accordingly, the State was no longer a party to this proceeding.

NEC Contention 4 was resolved in a different manner. NEC Contention 4 was originally a "contention of omission," *i.e.*, a contention alleging that the application was deficient because it failed to include (omitted) some necessary element.⁷ Original contention 4 was dismissed as moot on September 1, 2005, on the ground that Entergy had cured the omission by performing a structural and seismic analysis of the cooling towers under EPU and submitting the report thereon. LBP-05-24, 62 NRC 429, 433 (2005). On September 21, 2005, NEC filed a new Contention 4 challenging the adequacy of Entergy's structural and seismic analysis.⁸ The Board admitted this contention on December 2, 2005. LBP-05-32, 62 NRC 813, 826 (2005). However, on August, 10, 2006, NEC withdrew new Contention 4, eliminating it from this proceeding.⁹

⁵ Amended Notice of Withdrawal and Request for Dismissal of Contentions of the Vermont Department of Public Service (May 9, 2006).

⁶ Licensing Board Memorandum and Order (Approving Settlement Agreement, Granting Dismissal of Contentions, and Accepting Withdrawal of Vermont Department of Public Service) (June 23, 2006) (unpublished).

⁷ "There is . . . a difference between contentions that merely allege an 'omission' of information and those that challenge substantively and specifically how particular information has been discussed in a license application. Where a contention alleges the omission of particular information or an issue from an application, and the information is later supplied by the applicant or considered by the Staff in a draft EIS, the contention is moot." Duke Energy Corp. (McGuire Nuclear Station, Units 1 and 2; Catawba Nuclear Station, Units 1 and 2), CLI-02-28, 56 NRC 373, 382-383 (2002).

⁸ New England Coalition's Request for Leave to File a New Contention (Sept. 21, 2005).

⁹ NEC noted that its expert had "conclude[d] Contention 4 is largely satisfied in that [the
(continued...)]

The procedural history of NEC Contention 3, the only remaining contention, is straightforward.¹⁰ On December 2, 2005, Entergy filed a motion for summary disposition of Contention 3 which attempted to refute the technical material submitted in support of the contention by NEC's expert.¹¹ The Board denied the motion, stating that weighing the affidavits of competing experts "is not appropriate at the summary disposition stage" of the proceeding. LBP-06-05, 63 NRC 116, 125 (2006).

On March 10, 2006, during a telephone conference with the parties, a disagreement arose with respect to the scope of Contention 3, *i.e.*, confusion as to specific tests that were meant to be included under the rubric of "large transient testing." Tr. at 819-20. The parties were instructed to submit briefs on this issue, and on April 17, 2006, the Board ruled that "the scope of NEC Contention 3 is limited to two large transient tests: the main steam isolation valve [MSIV] closure test and the turbine generator load rejection [GLR] test."¹² The Board noted that testimony and other evidence to be submitted in connection with NEC Contention 3 should be limited to these two tests. *Id.* at 3.

⁹(...continued)
relevant] omissions and flaws have largely been remedied by extra examinations, analyses, and inspections, particularly evidenced in recent and supplemental Entergy documentation." New England Coalition's Notice of Withdrawal of its Contention Regarding Inadequate Analysis of the Vermont Yankee Alternate Cooling System Performance Under Conditions of Extended Power Uprate (Aug. 10, 2006) at 2.

¹⁰ Additional contentions proposed by NEC in two separate motions filed in 2006 were rejected as untimely. *See* LBP-06-14, 63 NRC 568 (2006); Licensing Board Memorandum and Order (Ruling on Admissibility of Additional NEC Contention and on Request to Supplement Additional Contention) (July 7, 2006) (unpublished).

¹¹ Entergy's Motion for Summary Disposition of New England Coalition Contention 3 (Dec. 2, 2005).

¹² Licensing Board Memorandum and Order (Clarifying the Scope of NEC Contention 3) (Apr. 17, 2006) at 2 (unpublished).

Meanwhile, during the pendency of this adjudicatory proceeding, Entergy's EPU license amendment application was being reviewed and processed by the NRC Staff and by the Advisory Committee on Reactor Safeguards (ACRS). On November 2, 2005, the Staff published its Draft Safety Evaluation Report (DSER) concerning the requested amendment.¹³ On November 15-16 and 29-30, 2005, the ACRS Subcommittee on Power Upgrades held meetings to receive input from the Applicant, the Staff, and members of the public on Entergy's EPU amendment application. On December 7, 2005, the full committee of the ACRS held public meetings on the application and on January 4, 2006, the ACRS sent a letter to the Commission recommending approval of Entergy's EPU application.¹⁴ On January 27, 2006, the Staff published its Environmental Assessment and Finding of No Significant Impact concerning the proposed VYNPS EPU license amendment in the Federal Register.¹⁵ On March 2, 2006, the Staff issued its Final Safety Evaluation Report (FSER) for the VYNPS EPU license amendment,¹⁶ along with a Finding of No Significant Hazards Consideration.¹⁷

¹³ Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. ___ to Facility Operating License No. DPR-28, Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc., Vermont Yankee Nuclear Power Station, Docket No. 50-271 (Nov. 2, 2005), ADAMS Accession No. ML053010167.

¹⁴ See Letter from Graham B. Wallis, Chairman, ACRS, to Nils J. Diaz, Chairman, NRC, (Jan. 4, 2006) (Entergy Exh. 22) (recommending approval of Vermont Yankee EPU).

¹⁵ Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc., Vermont Yankee Nuclear Power Station; Final Environmental Assessment and Finding of No Significant Impact Related to the Proposed License Amendment To Increase the Maximum Reactor Power Level, 71 Fed. Reg. 4614 (Jan. 27, 2006).

¹⁶ Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 299 to Facility Operating License No. DPR-28, Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc., Vermont Yankee Nuclear Power Station, Docket No. 50-271 (Mar. 2, 2006) [FSER]. The proprietary version of the FSER was introduced at the hearing as Staff Exhibit 1P, and a redacted, non-proprietary version was introduced as Staff Exhibit 2.

¹⁷ Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc., Notice
(continued...)

In Section 2.12 of the FSER, the NRC Staff evaluated Entergy's proposed EPU testing program and concluded that it was acceptable and that large transient testing was not required. Staff Exh. 1P, 2 at 260-74. The Staff thus agreed with the recommendation of the ACRS. See Entergy Exh. 22 at 1, 4. The Staff therefore issued the requested license amendment, which was effective immediately, concurrently with the FSER. 71 Fed. Reg. at 11,682.

Given the Staff's actions, Entergy was entitled to implement the EPU immediately, with our adjudicatory hearing to be held later. See 10 C.F.R. § 50.91(a)(4). NEC objected. On March 3, 2006, the Commission denied NEC's request that the license amendment be stayed pending the completion of evidentiary hearings on the requested amendment.¹⁸ The Commission noted, however, that its denial of NEC's stay request did not constitute an expression of the Commission's views on the validity of the amendment, CLI-06-8, 63 NRC at 238 n.9, and that "[i]f the Board determines after full adjudication that the license amendment should not have been granted, it may be revoked (or conditioned)." Id. at 238. Thus, the adjudicatory process continued.

Pursuant to our scheduling orders and 10 C.F.R. § 2.332(d), the issuance of the FSER on March 2, 2006, triggered the filing of evidence and other events leading to the evidentiary hearing.¹⁹ On May 17, 2006, the three remaining parties (NEC, Entergy, and the Staff) filed

¹⁷(...continued)
of Issuance of Amendment to Facility Operating License and Final Determination of No Significant Hazards Consideration, 71 Fed. Reg. 11,682 (March 8, 2006).

¹⁸ CLI-06-8, 63 NRC 235 (2006). See also 10 C.F.R. § 50.58(b)(6) ("The staff's [no significant hazards consideration] determination is final, subject only to the Commission's discretion, on its own initiative, to review the determination.").

¹⁹ Licensing Board Order (Initial Scheduling Order) (Feb. 1, 2005) at 3-4 (unpublished); Licensing Board Order (Revised Scheduling Order) (Apr. 13, 2006) at 3-4 (unpublished).

their initial statements of position²⁰ and written direct testimony and exhibits regarding the merits of NEC Contention 3.²¹ On June 14, 2006, all parties filed rebuttal statements of position²² and NEC and Entergy submitted additional written testimony and exhibits.²³ Meanwhile, on June 5, 2006, the Board ordered the parties to supplement their exhibits by submitting to the Board any documents that were relied upon by the parties' experts in their written testimony, but that were not included as exhibits at the time the testimony was submitted.²⁴ Entergy and the NRC Staff submitted these supplemental exhibits on June 19, 2006.²⁵ Pursuant to a notice in the Federal Register, 71 Fed. Reg. 19,549 (Apr. 14, 2006), the Board held meetings in Brattleboro, Vermont, on June 26 and 27, 2006, where members of the

²⁰ New England Coalition's Statement of Position (May 17, 2006); Entergy's Initial Statement of Position on New England Coalition Contention 3 (May 17, 2006) [Entergy Statement of Position]; NRC Staff's Initial Statement of Position Concerning NEC Contention 3 (May 17, 2006).

²¹ Prefiled Written Testimony of Dr. Joram Hopenfeld Regarding Contention 3 (May 17, 2006) (fol. Tr. at 1510) [Hopenfeld Direct Testimony for NEC]; Testimony of Craig J. Nichols and José L. Casillas on NEC Contention 3 – Large Transient Testing (May 17, 2006) (fol. Tr. at 1175) [Nichols/Casillas Direct Testimony for Entergy]; Ennis et al. Direct Testimony for NRC Staff.

²² New England Coalition's Response to the Statements of Position of Entergy and NRC Staff (June 14, 2006); Entergy's Rebuttal Statement of Position on New England Coalition Contention 3 (June 14, 2006); NRC Staff's Response to the Initial Statements of Position Filed by Other Parties (June 14, 2006).

²³ Declaration of Dr. Joram Hopenfeld in Support of New England Coalition's Response to the Statements of Position of Entergy and NRC Staff (June 14, 2006) (fol. Tr. at 1510) [Hopenfeld Rebuttal Testimony for NEC]; Rebuttal Testimony of Craig J. Nichols and José L. Casillas on NEC Contention 3 – Large Transient Testing (June 14, 2006) (fol. Tr. at 1177) [Nichols/Casillas Rebuttal Testimony for Entergy].

²⁴ Licensing Board Order (Regarding Submission of Supplemental Documents) (June 5, 2006) (unpublished).

²⁵ Entergy's Supplement to Direct Testimony on NEC Contentions 3 and 4 (June 19, 2006); NRC Staff's Supplement to its Initial Testimony Concerning NEC Contentions 3 and 4 (June 19, 2006).

public made oral limited appearance statements.²⁶ On September 12, 2006, the Board, accompanied by representatives of the three parties, conducted a site visit of the VYNPS in order to view plant components relevant to NEC Contention 3.²⁷

On September 13 and 14, 2006, the Board conducted the evidentiary hearing on NEC Contention 3 at the Windham County Courthouse in Newfane, Vermont. Pursuant to our order of December 16, 2004, the evidentiary hearing was held in accordance with 10 C.F.R. Part 2, Subpart L. LBP-04-31, 60 NRC 686, 706 (2004). This was the first Subpart L evidentiary hearing held since the Commission substantially amended the adjudicatory hearing regulations in 2004.²⁸ Because some of the exhibits submitted by Entergy and the NRC Staff were claimed to be proprietary and privileged, it was necessary to hold a short (less than one hour) closed session of the evidentiary hearing on September 14, 2006. Pursuant to our March 1, 2005, protective order,²⁹ only representatives who had signed a non-disclosure agreement were allowed to attend this short proprietary session.³⁰

²⁶ The Board has also received and considered a number of written limited appearance statements.

²⁷ Licensing Board Order (Scheduling Site Visit and Evidentiary Hearing (July 28, 2006) at 1 (unpublished); Licensing Board Order (Site Visit and Evidentiary Hearing Administrative Matters))Aug. 24, 2006) at 1 (unpublished).

²⁸ See Final Rule, Changes to Adjudicatory Process, 69 Fed. Reg. 2182 (Jan. 14, 2004).

²⁹ Licensing Board Order (Protective Order Governing Non-Disclosure of Proprietary Information) (Mar. 1, 2005) (unpublished).

³⁰ The representatives of NEC declined to sign the non-disclosure agreement and therefore were not allowed into the proprietary session. Tr. at 1007. However, a representative of the State (formerly a party) signed the non-disclosure agreement and, given that no party objected, was allowed to attend the September 14, 2006 proprietary session. Letter from Sara Hofmann, Vermont DPS, to the Board (Sept. 6, 2006) (requesting permission to attend at proprietary session); Tr. at 1121-22. A redacted version of the transcript of the proprietary session was later made available to NEC and the public. Licensing Board Order (Transmitting Redacted Version of Transcript from Proprietary Session (Oct. 12, 2006) (unpublished).

B. Witnesses

During the evidentiary hearing on NEC Contention 3 a total of eight witnesses appeared on behalf of Entergy, the Staff, and NEC. Some of the witnesses were fact witnesses, and all of them also provided some opinion testimony. All of the witnesses were found to be qualified to present their testimony on the matters they addressed. As previously stated, written direct testimony was submitted for all of the parties witnesses and written rebuttal testimony was submitted by the Entergy and NEC witnesses. All of the witnesses also provided oral testimony in response to questioning by the Licensing Board.

1. Entergy Witnesses

Entergy presented a panel consisting of two witnesses in support of its license amendment application.³¹ They were: (1) Mr. Craig J. Nichols, an Electrical Engineer, who was Entergy's Project Manager for the VYNPS EPU and who was the manager responsible for implementing the Vermont Yankee EPU; and (2) Mr. José L. Casillas, a Mechanical Engineer, who is the Plant Performance Consulting Engineer in the Nuclear Analysis group of the Engineering organization of the General Electric (GE) Nuclear Energy Company, LLC, and is responsible for boiling water reactor (BWR) plant performance design and analyses, including evaluations in support of EPU applications . Nichols/Casillas Direct Testimony for Entergy at 1-3.

Entergy witness Craig Nichols received a Bachelor of Science degree in Electrical Engineering from Northeastern University. Id. at 2; Resume of Craig Joseph Nichols, Entergy Exh. 1, at 2. Mr. Nichols has over twenty years of professional experience working in various technical and managerial capacities at VYNPS. Nichols/Casillas Direct Testimony for Entergy

³¹ See Nichols/Casillas Direct Testimony for Entergy; Nichols/Casillas Rebuttal Testimony for Entergy.

at 1-2. As Entergy's Project Manager for the Vermont Yankee EPU, Mr. Nichols was responsible for managing all engineering, analysis, modifications, implementation, and fiscal aspects of the EPU. Id. In this regard, he was responsible for overseeing the plant modifications needed to implement the upgrade and the performance of the technical evaluations and analyses required to demonstrate Vermont Yankee's ability to operate safely under uprate conditions. He is familiar with Vermont Yankee's operating history, current plant operations, and the anticipated operating conditions after the uprate. Id. at 2-3. The Board found Mr. Nichols to be qualified as an expert witness on the subject of BWR operation and the response of BWRs to transients. In addition, Mr. Nichols served as a fact witness with regard to the VYNPS EPU, the justification that Entergy submitted to the NRC Staff to show that large transient testing was not needed, the plant modifications at VYNPS associated with the EPU, and the history of some of the prior transients at VYNPS.

Entergy witness José L. Casillas received a Bachelor of Science degree in Mechanical Engineering from the University of California, Davis. Id. at 3; Resume of José L. Casillas, Entergy Exh. 2. Mr. Casillas is the Plant Performance Consulting Engineer in the Nuclear Analysis group of the Engineering organization of General Electric (GE) Nuclear Energy, which is a consultant to Entergy. At GE Nuclear Energy, Mr. Casillas is responsible for BWR plant performance design and analyses, including evaluations in support of EPU applications and the development and application of computer codes used to predict BWR plant performance. Nichols/Casillas Direct Testimony for Entergy at 3. Mr. Casillas has over thirty-two years of direct technical experience working in all aspects of plant performance at GE Nuclear Energy, including transient analysis. Mr. Casillas is familiar with the analytical codes used to predict BWR plant response to operational transients and with the industry experience regarding the response of BWRs to large transients. Id.; Entergy Exh. 2. He presented testimony which

addressed, inter alia, industry experience regarding the response of BWRs to large transients. The Board found Mr. Casillas to be qualified as an expert witness on the subjects of BWR plant system performance evaluation, BWR transient and loss of coolant accident (LOCA) analysis, and thermal hydraulic design and evaluation of BWR fuel. In addition, the Board found that Mr. Casillas is familiar with industry experience regarding the response of BWRs to large transients.

2. NRC Staff Witnesses

The NRC Staff presented a panel consisting of five witnesses concerning this contention. These were: (1) Mr. Richard B. Ennis; (2) Mr. Steven R. Jones; (3) Mr. Robert L. Pettis, Jr.; (4) Mr. George Thomas; and (5) Ms. Zena Abdullahi. Ennis et al. Direct Testimony for NRC Staff.

NRC Staff witness Richard B. Ennis is employed by the NRC as a Senior Project Manager in the Division of Operating Reactor Licensing in the NRC's Office of Nuclear Reactor Regulation (NRR). Mr. Ennis served as the Senior Project Manager for the Staff's review of the Vermont Yankee EPU. As part of his official responsibilities, he coordinated the Staff's evaluation of the Vermont Yankee EPU, assisted in preparation of the Staff's DSER for the EPU application, and coordinated the Staff's preparation of the FSER. Mr. Ennis received a Bachelor of Science degree in Electrical Engineering from Bucknell University and has over twenty-eight years of engineering experience in the nuclear power industry, including project management, design and licensing basis documentation, nuclear facility design verifications and modifications, software development and validation, and instrument setpoint and loop uncertainty calculations and methodologies. Id. at 1-2, 4; Ennis Professional Qualifications (fol. Tr. at 1383) at 1. The Board found Mr. Ennis to be qualified as an expert witness on the subjects of Entergy's EPU license amendment application, NRC regulatory requirements and guidance pertaining to BWR EPU applications, and the bases for Staff approvals of licensee

requests for exceptions to large transient testing in EPU applications. We note that some of his testimony was also as a fact witness.

NRC Staff witness Steven R. Jones is employed by the NRC as a Senior Reactor Systems Engineer in the Division of Engineering, NRR, and served as Acting Chief of NRR's Balance of Plant Branch. As such, he is responsible, inter alia, for evaluating the functional requirements, design, and performance of auxiliary, support, and mechanical systems other than those directly associated with the nuclear steam supply system (i.e., balance of plant systems – the main steam and turbine, feedwater and condensate, diesel generator support, auxiliary feedwater, spent fuel pool cooling, circulating water, open and closed cycle cooling water, and reactor coolant leakage detection systems) for both current and planned nuclear plants. Ennis et al. Direct Testimony for NRC Staff at 1-2. As part of his official responsibilities, Mr. Jones supervised the Staff's safety review of balance of plant systems, to evaluate the effects of the proposed EPU on such systems; these include the condensate, feedwater, main steam, main turbine, and turbine bypass systems that are involved in the plant's response to transients, as described in Sections 2.5 and 2.12 of the Staff's FSER (Staff Exhs. 1P and 2). Id. at 4. Mr. Jones received a Bachelor of Science degree in Marine Engineering from the United States Naval Academy and has over twenty years of experience in nuclear engineering and regulation, including experience as a Senior Resident Inspector. Jones Professional Qualifications (fol. Tr. at 1383) at 1-2. The Board found Mr. Jones to be qualified as an expert witness on the subject of the impacts of EPU operation on balance of plant systems, NRC regulatory requirements and guidance pertaining to BWR EPU applications, and the Staff interpretation as to the need for large transient testing in connection with such applications, as pertinent to balance of plant systems.

NRC Staff witness Robert L. Pettis, Jr., is employed by the NRC as a Senior Reactor Engineer in the Division of Engineering, NRR. As such, he is responsible for the technical review of several EPU and license renewal amendment requests. As part of his responsibilities, Mr. Pettis was responsible for evaluating the power ascension and testing plan section of the Vermont Yankee EPU application. He coordinated the Staff's review of the overall power uprate testing program of the Vermont Yankee EPU application, including preparation of Section 2.12 in the Staff's FSER. Ennis et al. Direct Testimony for NRC Staff at 1-3. Mr. Pettis received a Bachelor of Science degree in Civil Engineering and a Master of Science Degree in Civil Engineering from Northeastern University. He has over thirty years' engineering experience in the commercial nuclear power industry, including significant experience in the following areas: engineering management; technical writing; nuclear facilities audits, inspections, and design verifications; structural engineering and design; software quality assurance, verification and validation; EPU reviews; and professional engineer reviews of ASME Class I component supports. Pettis Professional Qualifications (fol. Tr. at 1383) at 1. The Board found Mr. Pettis to be qualified as an expert witness on the subjects of NRC regulatory requirements and guidance pertaining to nuclear power plant operational testing and of the Staff interpretation as to need for large transient testing in connection with BWR EPU applications.

NRC Staff witness George Thomas is employed by the NRC as a Senior Reactor Systems Engineer in the Division of System Safety, NRR. As such, he is responsible for reviewing and evaluating design, process design parameters, and performance of reactor thermal-hydraulic systems for BWR designs, including advanced reactor designs and combined operating licenses associated with the reactor coolant system and normal and emergency core cooling systems under steady state, transient, and accident conditions. In addition, he is

responsible for reviewing the analysis of anticipated operational occurrences, postulated accidents, and actual operating experience from the viewpoint of systems operation and transient dynamics; and he conducts evaluations of the effects of changes to licensed thermal power, license renewal, and other technical specification changes related to BWR reactor systems. Ennis et al. Direct Testimony for NRC Staff at 2-3. As part of his responsibilities, Mr. Thomas conducted the reactor systems review of the transient analyses submitted by Entergy for the Vermont Yankee EPU, including preparation of Section 2.8.5 in the Staff's FSER. Id. at 4. Mr. Thomas received a Bachelor of Science degree in Physics from Kerala University (India), and he has over thirty-seven years of BWR experience including twenty-six years at the NRC. His experience includes a broad range of functions related to the design, engineering, testing, operations, and evaluation of BWR systems. Thomas Professional Qualifications (fol. Tr. at 1383) at 1-2. The Board found Mr. Thomas to be qualified as an expert witness on the subjects of BWR thermal-hydraulic system performance, the dynamics of BWR transients, and the analysis of transients related to BWR reactor systems.

NRC Staff witness Zena Abdullahi is employed by the NRC as a Senior Reactor Systems Engineer in the BWR Systems Branch of the NRR Division of System Safety. As such, she is responsible for evaluating the impacts of proposed license amendments on reactor response during steady state, transient and accident conditions. Her areas of responsibilities include evaluating design basis safety analyses supporting BWR operation (e.g., reactor fuel and core performance, transients, emergency core cooling system (ECCS) LOCAs, and instabilities), the capabilities of reactor safety coolant systems (e.g., ECCS, reactor core isolation cooling (RCIC)) to perform their safety functions, and the adequacy of nuclear monitoring and safety system actuation and trip setpoints during steady state, transient and accident conditions. Ennis et al. Direct Testimony for NRC Staff at 2-3; Abdullahi Professional

Qualifications (fol. Tr. at 1383) at 1. Ms. Abdullahi conducted the Staff's review of the analytical methods used in the Vermont Yankee EPU application to perform the reactor neutronic and thermal-hydraulic analyses, as described in Section 2.8.7 of the Staff's FSER. Ennis et al. Direct Testimony for NRC Staff at 5. Ms. Abdullahi received a Bachelor of Science degree in Mechanical Engineering from the University of California, Davis, and a Master of Science degree in Mechanical Engineering from the University of Maryland. She has over thirteen years' experience at the NRC and in the nuclear power industry, including considerable experience in evaluating nuclear reactor core and fuel performance during steady state, transient, and accident conditions. Abdullahi Professional Qualifications at 1. The Board found Ms. Abdullahi to be qualified as an expert witness on the subjects of neutronic and thermal-hydraulic analyses.

3. NEC Witness

NEC presented one witness, Dr. Joram L. Hopenfeld, in support of its contention.³² Dr. Hopenfeld has had forty-four years of professional experience, which has included the publication of fourteen papers in peer-reviewed journals. Dr. Hopenfeld has designed and conducted tests related to thermal hydraulics, materials/coolant compatibility, and reactor safety. During his career Dr. Hopenfeld worked for the NRC, where he was responsible for a test program designed to benchmark thermal hydraulic codes for pressurized water reactor nuclear reactors. Dr. Hopenfeld received a Bachelor of Science degree, a Master of Science degree, and a Ph.D. in Engineering from the University of California, Los Angeles, with emphasis in fluid flow, heat transfer and electrochemistry. Hopenfeld Direct Testimony for NEC

³² Hopenfeld Direct Testimony for NEC; Hopenfeld Rebuttal Testimony for NEC.

at [unnumbered] 1-3. The Board found Dr. Hopenfeld to be qualified as an expert witness on the subject of thermal-hydraulic analyses.³³

III. GOVERNING LEGAL STANDARDS

Several regulations apply to the testing of nuclear power plants, and thus govern our consideration of NEC's contention that the EPU should not be granted unless large transient testing is imposed as a license requirement. First, 10 C.F.R. § 50.34(b)(6)(ii) specifies that each applicant for a license to operate a power plant must submit a final safety analysis report (FSAR) that includes the "[m]anagerial and administrative controls to be used to assure safe operation" set forth in the "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," Part 50 Appendix B.³⁴ An application to amend a license and authorize an extended power uprate is subject to the same considerations. 10 C.F.R. § 50.92(a). Second, 10 C.F.R. § 50.54(a)(1) states that "[e]ach nuclear power plant . . . subject to the quality assurance criteria in appendix B of this part shall implement, pursuant to § 50.34(b)(6)(ii) of this part, the quality assurance program described or referenced in the [FSAR]."

The third and most substantive element in NRC's regulatory structure is Appendix B to Part 50, which prescribes the quality assurance program (QAP), including testing, that must be implemented at each nuclear power plant. Specifically, Appendix B states:

Nuclear power plants . . . include structures, systems, and components [SSCs] that prevent or mitigate the consequences of postulated accidents and that could

³³ In its original petition, NEC supported the admission of NEC Contention 3 with a declaration by its consultant, Mr. Arnold Gundersen. NEC Petition, Exh. D, Declaration of Arnold Gundersen in Support of Petitioners' Contentions (Aug. 30, 2004). However, in the evidentiary phase of this proceeding, NEC did not submit written direct or rebuttal testimony from Mr. Gundersen and he was not available for questioning by the Board in the oral hearing on September 13-14, 2006. Therefore, Mr. Gunderson's declaration is not part of the evidence in this proceeding.

³⁴ 10 C.F.R. § 50.34(b)(6)(iii) also requires that the FSAR include the applicant's "[p]lans for preoperational testing and initial operations."

cause undue risk to the health and safety of the public. This appendix establishes quality assurance requirements for the design, construction, and operation of those [SSCs]. The pertinent requirements of this appendix apply to all activities affecting the safety-related functions of those [SSCs] . . . [including] testing.

* * * *

XI. Test Control

A test program shall be established to assure that all testing required to demonstrate that [SSC] will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. The test program shall include, as appropriate, proof tests prior to installation, preoperational tests, and operational tests during nuclear power plant or fuel reprocessing plant operation, of [SSCs]. Test procedures shall include provisions for assuring that all prerequisites for the given test have been met, that adequate test instrumentation is available and used, and that the test is performed under suitable environmental conditions. Test results shall be documented and evaluated to assure that test requirements have been satisfied.

10 C.F.R. Part 50, Appendix B, Criterion XI (emphasis added).

Thus, Criterion XI requires that each nuclear power plant have a QAP with a test program that includes “all testing required to demonstrate” that SSCs will perform satisfactorily in service. We note that this regulation is somewhat vague – Criterion XI requires “all testing required” to demonstrate that the SSC will perform “satisfactorily.” Exactly what is “required” and “satisfactory” is not specified. Nevertheless, this is the legal standard that the Board must use in resolving NEC Contention 3. More specifically, the legal standard for determining whether the VYNPS EPU amendment should be approved is whether, in the absence of the two large transient tests sought by NEC (the MSIV closure test and the generator load rejection test), Entergy’s EPU test program complies with Criterion XI, i.e., “assures that all testing required to demonstrate that SSC will perform satisfactorily in service is identified and performed.” Entergy, as the applicant, has the burden of persuasion on this issue. 10 C.F.R. § 2.325; 69 Fed. Reg. 2182, 2213 (Jan. 14, 2004).

IV. FINDINGS OF FACT

A. Basic Factual Framework and Staff Approach

The issue before this Board - whether the two large transient tests sought by NEC (the MSIV closure test and the generator load rejection test) should be required as part of the VYNPS EPU – requires an understanding of several basic and uncontested terms. Thus, this section briefly discusses the meaning of such terms as “transient,” “MSIV transient,” and “MSIV transient test.” Likewise, an understanding of the key issue is significantly enhanced by a review of the basic guidance that the Staff uses when it considers the need for large transient testing. While this guidance is not controlling on the Board, it is helpful and relevant in understanding this case. Accordingly, this section IV.A provides some of the uncontested basics and background concerning this proceeding.

1. Definitions and Basic Concepts

a. “Transient”

Although it is commonly used in the NRC regulations and the nuclear industry, NRC regulations do not define the term “transient.” The NRC webpage states that a “transient” is “[a] change in reactor coolant system temperature and/or pressure due to a change in power output of the reactor.”³⁵ This description is useful, but not entirely correct. This Board uses the term “transient” to include a change in any reactor or reactor cooling system parameter, not just the temperature and/or pressure, and not just those “due to a change in the power output of the reactor.” Transients can be caused by (1) adding or removing neutron poisons, (2) increasing or decreasing the electrical load on the turbine generator, or (3) accident conditions. Id. The non-transient mode of operation is referred to as “steady state operation,” which is the absence of change in the conditions within the reactor and reactor cooling systems.

³⁵ U.S. NRC Glossary at <http://www.nrc.gov/reading-rm/basic-ref/glossary/transient.html>.

Transients can be, and often are, “anticipated operational occurrences” which are “conditions of normal operation.” See 10 C.F.R. Part 50, Appendix A, General Design Criteria for Nuclear Power Plants, Definitions and Explanations. Normal operations include startups and shutdowns as well as power changes and steady state operation. Anticipated operational occurrences are defined as “those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include, but are not limited to, loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power.”³⁶ Criterion 10 of Appendix A states that “[t]he reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.” 10 C.F.R. Part 50, Appendix A, Criterion 10.

The two transients that the parties agreed are of concern here are an inadvertent closure of the main steam isolation valves (MSIV closure transient) and a generator load rejection (GLR) transient.³⁷

b. “MSIV Transient”

An MSIV closure transient, or simply “MSIV transient,” is a transient involving the sudden closure of the main steam isolation valves. “Main steam isolation valves,” or MSIVs,

³⁶ Id. See also Office of Nuclear Reactor Regulation, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800, Draft Revision 0 (Dec. 2002) at 14.2.1-16 (Entergy Exh. 4) (also using the term “anticipated transients” for such occurrences).

³⁷ Licensing Board Order (Clarifying the Scope of NEC Contention 3) (Apr. 17, 2006) at 3 (unpublished).

are the valves that are intended to isolate the steam system “inside” the reactor containment³⁸ from the steam system “outside” the reactor containment. In the case of VYNPS, there are eight MSIVs. Nichols/Casillas Direct Testimony for Entergy at 9. These valves serve a safety function in the event of fuel failure by preventing fission products from the fuel inside of the reactor from being released into the steam system outside of the reactor containment. See 10 C.F.R. Part 50, Appendix A, Criterion 54.

In an MSIV transient, something triggers at least two of the eight MSIV to close. When the two valves are about 10 percent closed, the reactor control system automatically initiates the sudden shutdown of the reactor by rapidly inserting the control rods into the reactor. Tr. at 1181-83. In short, the MSIV closure triggers a sudden reactor shutdown, or “SCRAM.”³⁹ Because the SCRAM signal is initiated based on the position of the stem of the MSIV (two valves at 10 percent closed), the SCRAM is referred to as a “position” SCRAM. Tr. at 1180. At the initiation of the position SCRAM, when only two valves are 10 percent closed, all of the valves are still essentially fully open.⁴⁰

Once the MSIV closure triggers the SCRAM, the remaining MSIVs close fully in about three to five seconds. This isolates the reactor core, causing the pressure in the reactor to

³⁸ The reactor containment is a gastight shell or other enclosure around a nuclear reactor intended to confine fission products that otherwise might be released to the atmosphere in the event of an accident. U.S. NRC Glossary at <http://www.nrc.gov/reading-rm/basic-ref/glossary/containment-structure.html>.

³⁹ Nichols/Casillas Direct Testimony for Entergy at 9. A SCRAM is “[t]he sudden shutting down of a nuclear reactor, usually by rapid insertion of control rods, either automatically or manually by the reactor operator. May also be called a reactor trip. It is actually an acronym for ‘safety control rod axe man,’ the worker assigned to insert the emergency rod on the first reactor (the Chicago Pile) in the U.S.” U.S. NRC Glossary at <http://www.nrc.gov/reading-rm/basic-ref/glossary/scram.html>.

⁴⁰ The “10 percent” closure refers to the position of the stem of the valve at 10 percent, not to 10 percent of the sealing surface of the valve. Given the shape of the valves, even when the stem is at the 10 percent position, the valve remains open at much greater than 90 percent.

increase and resulting in an increase in moderator density.⁴¹ The pressure increases until the effects of inserting the control rods, which shuts down the reactor, are able to offset any increase in power caused by the increase in reactor pressure. Tr. at 1183; see also Nichols/Casillas Direct Testimony for Entergy at 8-9. After SCRAM, operators would open safety relief valves to further control pressure and use the high pressure emergency core cooling system to control primary system pressure and to remove residual heat from the system. Eventually, the reactor core isolation cooling system would be used to provide finer control of pressure until the system pressure was low enough for the residual heat removal system to be used and normal shutdown conditions achieved. Tr. at 1187-88.

Of the two transients considered here, the MSIV transient is the more severe operational transient from the standpoint of increased pressure on the nuclear reactor systems. See Nichols/Casillas Direct Testimony for Entergy at 9. During the MSIV transient, and subsequent SCRAM, there is an increase in the reactor vessel pressure on the order of 50 to 100 pounds per square inch gauge (psig).⁴² The goal is to avoid a pressure increase that is large enough to reach the design pressure of the system and to avoid causing the American Society of Mechanical Engineers (ASME) code safety valves to open. Tr. at 1191-92. These pressure relief valves open when the system pressure reaches the 1375 psig limit, which is 110 percent of the reactor vessel design pressure of 1250 psig.⁴³

⁴¹ A moderator is a material, such as ordinary water, heavy water, or graphite, that is used in a reactor to slow down high-velocity neutrons, thus increasing the likelihood of fission. U.S. NRC Glossary at <http://www.nrc.gov/reading-rm/basic-ref/glossary/moderator.html>.

⁴² Tr. at 1188-89. Pounds per square inch gauge, or psig, is equal to pounds per square inch absolute minus approximately fifteen pounds.

⁴³ Tr. at 1192. Because the proposed uprate is a constant pressure power uprate, there is no change in the system design pressure or safety valve opening pressures.

Although MSIV transients are usually unintentional events, they occasionally occur. An MSIV transient is therefore classified as an “anticipated operational occurrence,” and the regulations require that nuclear power stations be designed and built to withstand them. 10 C.F.R. Part 50, Appendix A, Criterion 10. When an MSIV transient occurs, the reactor operator is required to analyze what happened and how the reactor systems responded and performed, and to report to the NRC. 10 C.F.R. § 50.73(a)(2)(iv)(A)-(B) (requiring submission of Licensee Event Reports (LERs) following such occurrences).

c. “MSIV Transient Test”

An “MSIV transient test,” also commonly referred to as an “MSIV closure test,” is an intentional triggering of an MSIV transient to determine how the reactor, reactor control system, and steam system will perform in the event of an MSIV transient. It is intended to demonstrate that the systems behave as expected in the event of an inadvertent MSIV closure transient, to check the MSIVs for proper operation, and to determine how long it takes for the MSIVs to close when the reactor is at full power. Nichols/Casillas Direct Testimony for Entergy at 9. When an MSIV transient test is performed, the operator issues a signal that causes all eight MSIVs to close from full power. Id. at 8-9. For safety reasons however, the MSIV transient test is conducted without defeating the plant’s safety systems.⁴⁴ Because the MSIV transient test results in a position SCRAM, it serves to confirm that (1) the signals to shut down the reactor are issued, (2) the safety systems respond as intended, and (3) the relief valves operate as expected.⁴⁵ The MSIV transient test is a type of large transient test that is typically performed during initial startup testing of every boiling water reactor. See Section IV.A.2 herein.

⁴⁴ Id. at 9. See also Tr. at 1193-94, 1399- 1402.

⁴⁵ Nichols/Casillas Direct Testimony for Entergy at 9; Tr. at 1196.

d. "Generator Load Rejection (GLR) Transient"

A generator load rejection (GLR) transient is a transient that occurs when, for any reason, the electrical output from the nuclear power plant's generators suddenly has no place to go. Tr. at 1219. This could occur if there is a break in the electrical power lines exiting the generators, or if a transformer immediately downstream of the generator malfunctions. In response, the steam flow control valves on the turbines (the turbines drive the generators) close in approximately 100 to 200 milliseconds, thereby initiating a reactor SCRAM. Tr. at 1256-58. As the turbine control valves close, the path of the steam through the turbine to the condenser begins to close as well, and the turbine bypass valves begin to open. VYNPS has ten such turbine bypass valves arranged in two banks, which, at the power uprate conditions, are capable of handling 86 percent of the full steam flow. Tr. at 1219-20. Because of the fast closure of the turbine control valves, a pressure wave travels backwards, into the reactor, causing a pressurization whose magnitude is related to the difference between the steam that goes into the condenser via the bypass valves and the steam produced by the reactor. Tr. at 1256-58. The reactor thermal power will rise as the increase in pressure causes an increase in density of the steam-water mixture in the reactor core and in the moderator density, and will continue to do so until the control rods are fully inserted. After that occurs, the reactor power and pressure will decrease. Depending on the amount of pressure produced by the reactor power increase, a relief valve may or may not open. If the reactor power starts to decrease fast enough, there will be only a very small pressure rise in the reactor, and the pressure will be controlled by the bypass valves. Tr. at 1256-58. Compared to an MSIV transient, the peak pressure increase in a GLR transient is lower. Tr. at 1259-60.

Like MSIV transients, GLR transients occasionally occur and are classified as "anticipated operational occurrences" that nuclear power stations must be designed and built to

withstand. 10 C.F.R. Part 50, Appendix A, Criterion 10. Similarly, GLR transients must be analyzed and reported to the NRC. 10 C.F.R. § 50.73(a)(2)(iv)(A)-(B).

e. "GLR Transient Test"

A GLR transient test is a test of the reactor, reactor control system, steam bypass system, and steam systems of a nuclear power plant that is performed intentionally by closing the turbine control valves so that the test would progress as if it occurred during normal plant operations. Tr. at 1262. As discussed below, for safety reasons, in a GLR transient test no attempt is made to defeat operation of the bypass valves, because such a test would evaluate the outer limits of the system and thus be a design basis or bounding transient.

Nichols/Casillas Direct Testimony for Entergy at 10; Tr. at 1222-23, 1262-63.

f. "Design Basis Transient Analysis"

It is important to distinguish between transients, transient tests, design basis transients, and design basis transient analyses. Anticipated transients are events that, though unintended, are expected and may occur from time to time at a nuclear power station. Although transients are inadvertent, examination of them can yield valuable data. In contrast, transient tests are planned events that are conducted without bypassing the necessary and appropriate safety systems of the nuclear reactor. The entire purpose of such tests is to gather valuable data.

Design basis transients are different. Under NRC regulations, each nuclear reactor must be designed to withstand certain challenging conditions or events, such as certain earthquakes and certain large pipe break LOCAs. The collection of specified events on which a reactor design is based constitute part of what is termed the reactor's "design basis." An event that would challenge the maximum limits of a reactor's design basis is termed a "design basis transient." For obvious safety reasons, NRC does not require or allow licensees to conduct actual design basis tests, i.e., tests that would reach maximum limits of an operating nuclear

reactor's design. Instead, NRC requires licensees to perform computer analyses of what would happen if a design basis transient happened at their reactor. These computer analyses are called "design basis transient analyses."

In the case of MSIV transients, the actual test of the operating nuclear reactor (i.e., the MSIV transient test) is intentionally less challenging than the MSIV design basis transient computer analysis. The MSIV design basis transient analysis assumes that the SCRAM signal from the valve position indicators fails (i.e., the signal to shut down the reactor fails) and the reactor SCRAMs on high neutron flux level. Tr. at 1192-93. For the purposes of vessel pressurization, the MSIV design basis transient analysis is considered more severe (i.e., "bounding") than what would be allowed in any actual MSIV transient test or is likely to occur during an unintentional MSIV transient. This is because the failure of the position indicator during an MSIV closure would result in a much greater power excursion and a larger pressure increase than would otherwise occur. Tr. at 1192-93. See also Nichols/Casillas Direct Testimony for Entergy at 16, 20-21.

Similarly, in GLR design basis transient analysis, the computer simulation assumes that the bypass valves do not open. Such a postulated event is referred to as a "Generator Load Rejection from High Power Without Bypass" (GLRWB). Nichols/Casillas Direct Testimony for Entergy at 9-10. A GLRWB, where the bypass/relief valves do not open, would result in a far more severe transient than would otherwise be experienced during actual plant operations. Tr. at 1222-23. Such a postulated design basis transient provides a more severe challenge to the fuel than one in which the turbine bypass valves are assumed to operate properly. Nichols/Casillas Direct Testimony for Entergy at 10. Like the design basis MSIV transient analysis, a GLRWB would never form the basis for an actual test since it would pose a major threat to the plant. Id.

2. Staff Guidance Relating to Large Transient Testing

Although the legal standards governing the Board's decision are set forth in Section III above, the NRC Staff has issued certain guidance documents relevant to the need for, and value of, large transient testing for nuclear reactors. While these regulatory guides and Staff review plans are worth noting, they do not have the force of law and are not binding on our determination as to whether Entergy's testing program satisfies the legal standard in 10 C.F.R. Part 50, Appendix B, Criterion XI. See Curators of the University of Missouri (TRUMP-S Project), CLI-95-8, 41 NRC 386, 397 (1995).

The first example of relevant Staff guidance is NRC Regulatory Guide (RG) 1.68, Initial Test Programs for Water-Cooled Nuclear Power Plants.⁴⁶ As its name implies, RG 1.68 describes the general scope and depth of initial test programs that the NRC Staff has found acceptable during the review of initial operating license applications. Appendix A of RG 1.68 describes a set of tests that the Staff requires at the initial start-up of a nuclear plant to demonstrate that it will operate in accordance with design specifications both during normal steady state conditions and, to the extent practical, during and following anticipated operational occurrences. The MSIV transient test and the GLR transient test are both included in Appendix A of RG 1.68. Ennis et al. Direct Testimony for NRC Staff at 7; Entergy Exh. 4 at 1.68-18.

The second relevant document contains regulatory guidance for EPU's. Known as RS-001, "Review Standard for Extended Power Uprates," this document was developed primarily to increase the standardization and effectiveness of EPU reviews performed by the NRC Staff.⁴⁷ RS-001 provides the Staff's reviewers with references to existing review criteria (i.e., applicable

⁴⁶ NRC Regulatory Guide (RG) 1.68, Initial Test Programs for Water-Cooled Nuclear Power Plants, Revision 2 (Aug. 1978) (Staff Exh. 4).

⁴⁷ See Office of Nuclear Reactor Regulation, Review Standard for Extended Power Uprates, RS-001, Rev. 1 (Dec. 2003) (Staff Exh. 5).

Standard Review Plan (SRP) sections, branch technical positions, information notices and bulletins, generic letters, NUREGs, industry standards, applicable generic topical reports, etc.) and includes a template safety evaluation. Safety evaluation template Section 2.12, Power Ascension and Testing Plan, indicates that the acceptance criteria for a proposed EPU test program are based on Criterion XI. Ennis et al. Direct Testimony for NRC Staff at 8; see also Nichols/Casillas Direct Testimony for Entergy at 8.

As indicated in RS-001, Matrix 12, specific review criteria and NRC Staff guidance for assessing the extent of testing necessary for EPU applications is described in a third document, the Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants.⁴⁸ The relevant portion of this document is SRP Section 14.2.1, Generic Guidelines for Extended Power Uprate Testing Programs. Subsection III.A, Review Procedures, of SRP Section 14.2.1, provides staff guidance for a comparison of the proposed EPU test program to the initial plant test program. Subsection III.B of the SRP provides guidance for a review of EPU post-modification testing requirements. Attachment 2 to SRP Section 14.2.1, entitled “Transient Testing Applicable to Extended Power Uprates,” provides a generic listing of transient tests, drawn from RG 1.68, that the Staff indicates are the “typical transient testing acceptance criteria and functions important to safety associated with these anticipated [EPU]

⁴⁸ Office of Nuclear Reactor Regulation, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800, Draft Revision 0 (Dec. 2002) (Entergy Exh. 4). Entergy’s EPU application, submitted in 2003, was prepared by the Applicant and was reviewed and approved by the Staff, in accordance with the regulatory guidance contained in the December 2002 draft of this document. Entergy’s conformance with the draft guidance was addressed in the parties’ testimony and in this decision. Nonetheless, we note (as did the Staff) that in August 2006, the Staff’s draft guidance was superseded by the issuance of a final version of Section 14.2.1 (Generic Guidelines for Extended Power Uprate Testing Programs). See Ennis et al. Direct Testimony for NRC Staff at 8 n.5. We overruled NEC’s objection to the Staff’s reference to this fact in its testimony; as we observed, the revised guidance was not introduced as evidence and it does not affect our decision. See Tr. at 1381-83.

events.” Entergy Exh. 4 at 14.2.1-7. The two large transient tests that are the subject of the contention before us, the MSIV transient test and the GLR transient test, are included in Attachment 2 and are listed therein as “Dynamic Response of Plant to Automatic Closure of All Main Steam Isolation Valves,” id. at 14.2.1-18, and “Dynamic Response of Plant for Full Load Rejection,” id. at 14.2.1-17, respectively. Ennis et al. Direct Testimony for NRC Staff at 8-9; see also Nichols/Casillas Direct Testimony for Entergy at 8.

Under SRP Section 14.2.1, however, the Staff allows licensees to propose an EPU test program that does not include all of the large transient testing that would otherwise be required by Subsections III.A and III.B of SRP Section 14.2.1. Subsection III.C of Section 14.2.1, Use of Evaluation to Justify Elimination, states:

In certain cases, the licensee may propose an EPU test program that does not include all of the power-ascension testing that would normally be required by the review criteria of Sections III.A and III.B above. The licensee shall provide an adequate justification for each of these normally required power-ascension tests that are not included in the EPU test program.

Id. at 14.2.1-7. The SRP specifies that “[i]f a licensee proposes to not perform a power-ascension test that would normally be required . . . the [Staff] reviewer should ensure that the licensee provides an adequate justification” and goes on to list the following seven “factors” that the reviewer should consider:

- previous operating experience;
- introduction of new thermal-hydraulic phenomena or identified system interactions;
- facility conformance to limitations associated with analytical analysis methods;
- plant staff familiarization with facility operation and trial use of operating and emergency operating procedures;
- margin reduction in safety analysis results for anticipated operational occurrences;
- guidance contained in vendor topical reports; and
- risk implications.

Id. at 14.2.1-7 to 10; Ennis et al. Direct Testimony for NRC Staff at 9.

In summary, the Staff's regulatory guidance usually requires that large transient testing, including the MSIV closure transient test and the GLR transient test, be performed as part of an EPU, but also allows an applicant to propose, on a case-by-case basis, an EPU test program that does not include such large transient testing.

NRC recently rejected an industry request for a generic exemption from large transient testing for BWR EPU license applicants. Ennis et al. Direct Testimony for Staff at 10, 16. Two topical reports submitted by General Electric Company (GE), the nuclear steam supply system vendor for VYNPS, are of interest here. First, GE submitted General Electric Licensing Topical Report ELTR-1, Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate, to NRC.⁴⁹ The NRC Staff approved this report – Topical Report ELTR-1 – and issued it in February 1999. Ennis et al. Direct Testimony for Staff at 9. Topical Report ELTR-1 provides generic guidelines for BWR EPUs. GE's Topical Report ELTR-1 specifies, at Section 5.11.9 and Appendix L.2.4, that an MSIV transient test will be performed for any EPU greater than 10 percent and a GLR transient test will be performed for any EPU greater than 15 percent. Id. at 9-10. Topical Report ELTR-1 was based on the assumption that the maximum reactor operating pressure would be increased under EPU conditions. Id. at 10.

Subsequently, GE developed a different approach to uprating reactor power in BWRs that does not increase the maximum reactor operating pressure. This approach is described in GE Licensing Topical Report NEDO-33004-A, Constant Pressure Power Uprate [CPPU].⁵⁰ The

⁴⁹ See Ennis et al. Direct Testimony for Staff at 9; Entergy Exh. 25 at 3 (referencing Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate, NEDC-3242P-A (Feb. 1999) [ELTR-1]).

⁵⁰ GE Licensing Topical Report NEDO-33004-A, Constant Pressure Power Uprate, Revision 4 (July 2003) (Entergy Exh. 25 (non-proprietary version) or 30P (proprietary version)).

CPPU approach forms the basis for the Vermont Yankee EPU application. Ennis et al. Direct Testimony for NRC Staff at 10; Nichols/Casillas Direct Testimony for Entergy at 5.

In the CPPU topical report, GE proposed that if an EPU used the constant pressure approach, it should be relieved or exempt from performing the large transient tests (e.g., MSIV closure and GLR tests) – which are otherwise required under Topical Report ELTR-1 (where the pressure is assumed to increase). In support of this proposed generic exemption, GE provided a generic justification for not performing these tests and concluded that they are not needed to demonstrate the safety of plants implementing a CPPU. Ennis et al. Direct Testimony for NRC Staff at 10.

The NRC Staff reviewed and approved the CPPU topical report, as described in a Safety Evaluation (CPPU SE) released in March 2003.⁵¹ However, the Staff rejected GE's proposed generic exception of CPPUs from MSIV transient and GLR transient testing. Ennis et al. Direct Testimony for NRC Staff at 10, 16. Instead, the Staff concluded that it would continue to consider the need to conduct these tests on a plant-specific basis. In evaluating GE's generic justification to dispense with the two large transient tests, the Staff considered: (1) the modifications made to the plant for a CPPU that are related to the two tests; (2) component and system level testing that will be performed, either as part of the licensee's power ascension and test plan or to meet technical specification surveillance requirements; (3) past experience at other plants; and (4) the importance of the additional information that could be obtained from performing the two tests with respect to plant analyses. Id. at 10. The Staff stated that it was "developing guidance to generically address the requirement for conducting large transients

⁵¹ Safety Evaluation by the Office of Nuclear Reactor Regulation, GE Nuclear Energy Licensing Topical Report, NEDC-33004P, Revision 3, "Constant Pressure Power Uprate" (Mar. 31, 2003) [CPPU SE]. This document is incorporated into Entergy Exh. 25 at 3-87(non-proprietary version) and in Entergy Exh. 30P at the same page numbers (proprietary version).

tests in conjunction with power uprates,” adding that “information obtained from the MSIV closure and generator load rejection tests could be useful to confirm plant performance, adjust plant control systems, and enhance training material.” Id.; see also CPPU SE § 10.5.9. Finally, the CPPU SE indicated that, for BWRs using the CPPU approach, licensees may request plant-specific exemptions from the need to conduct the large transient tests in EPU situations. Ennis et al. Direct Testimony for NRC Staff at 10-11; CPPU SE §§ 10.5.8, 10.5.9.

B. Factual Findings on Key Contested Issues

The Board now turns to the specific contested issues in this proceeding. The basic facts are that Entergy asserts, pursuant to Subsection III.C, SRP 14.2.1, that there is no need to perform the MSIV transient test or the GLR transient test. The Staff agrees and so states in its FSER. NEC objects. And now this Board must decide whether an MSIV transient test and a GLR transient test are “required to demonstrate that the structures, systems and components [of the VYNPS on the reactor at the uprated conditions] will perform satisfactorily in service.” 10 C.F.R. Part 50, Appendix B, Criterion XI.

1. Assertions of Parties - Overview

Entergy and the Staff assert that an MSIV transient test and a GLR transient test are not “required to demonstrate that [the VYNPS] will perform satisfactorily in service” at the uprated power because operational experience shows that the effects of large transients on the VYNPS at EPU conditions can be predicted analytically, on a plant-specific basis, without the need for actual transient testing. They base this argument on (a) the similarity of the pre-EPU and post-EPU VYNPS design configuration and system functions; (b) the results of past transient testing at the VYNPS and other BWRs, and the plant’s response to unplanned transients; (c) confirmation that the results of transient computer simulations are consistent with, and bound,

the experience from actual transients; and (d) the experience with unplanned transients at other plants that have been granted an EPU. Nichols/Casillas Direct Testimony for Entergy at 4.

In contrast, NEC asserts that Entergy's rationale is technically unsound because it is based on three unsubstantiated propositions. The first of these allegedly unsound propositions is that none of the plant modifications introduces new thermal-hydraulic phenomena or system interactions during or as a result of the transients introduced. The second is that the computer simulations or analysis performed accurately predict the plant response during a large transient. The third allegedly unsound proposition is that the computer simulations of the transients that were done for VYNPS were performed using General Electric's NRC-approved transient analysis computer code "ODYN," which NEC asserts is problematic.⁵²

Dr. Hopenfeld asserts that each of these propositions is flawed because new and unexpected effects could occur during large transients due to the numerous system and component modifications made for the power uprate. He specifically cites the changes that were made to the steam dryers. Hopenfeld Direct Testimony for NEC at 6. With regard to the computer simulations, Dr. Hopenfeld states that Entergy has not provided a discussion showing why its simulations can be used as a substitute for transient testing. Id. at 5. Finally, NEC's witness asserts that Entergy does not state how the ODYN code was benchmarked against experiments for pressurization transients or for steady state operation. Id.

In subsections IV.B. 2, 3, and 4 we evaluate each of NEC's three main arguments in turn. We find some merit in portions of NEC's concerns. In subsection IV.C however, we turn

⁵² Hopenfeld Direct Testimony for NEC at 4. "The 'ODYN code' is the One Dimensional DYNAMIC (ODYN) Core Transient Model, which is a General Electric licensing code designed to simulate selected fast transients in boiling water reactors. . . . ODYN has been approved by the NRC for application to transients such as feedwater controller failure - maximum demand; pressure regulator failure - closed; generator load reject; turbine trip; MSIV closure; loss of auxiliary power - all grid connections; and MSIV closure with position switch failure (MSIV flux scram)." Ennis et al. Direct Testimony for NRC Staff at 17.

to the ultimate factual issue and conclude that although there are some questions about the benchmarking of the ODYN code, other more important factors, such as industry and VYNPS operating experience, provide assurance that large transient testing of the VYNPS at uprate conditions is not required.

2. Contested Issue One - Existence of new thermal-hydraulic phenomena and/or new system interactions

a. Key Evidence Presented

NECs first argument is that, in order to justify the exemption from the MSIV transient test and the GLR transient test, Entergy needs to show that “[n]one of the plant modifications that have been or will be made for the EPU will introduce new thermal hydraulic phenomena, nor will there be any new system interaction during or as a result of the analyzed transients introduced,” and that Entergy has failed to do so. Hopenfeld Direct Testimony for NEC at 4. Dr. Hopenfeld stated, for example, that, because of the increased flow velocity at EPU conditions, steady state temperature and pressure fluctuations will increase the fatigue usage factors of the steam dryer, leading to a cumulative usage factor (CUF) that could be above the allowable limit of 1.⁵³ In discussing industry experience, he argued that Entergy’s reference to several BWR reactors that have undergone transients, and for which Entergy claimed that no new phenomena have been exhibited, is insufficient. Id. at 5. Dr. Hopenfeld asserted that Entergy has not provided any analysis to indicate why these results are applicable to the VYNPS at the EPU conditions. Hopenfeld Rebuttal Testimony for NEC at 5. To make a valid comparison between the experience at other reactors and what is expected to occur at the VYNPS under transient conditions, Dr. Hopenfeld said, Entergy must show by actual computer

⁵³ Id. at 6. The cumulative usage factor (CUF) can be defined as the number of actual events divided by the maximum number of allowable events of that type. The ASME Boiler and Pressure Vessel Code invoked by 10 C.F.R. § 50.55a(c) limits the value of the CUF to 1 or less.

analysis – including calculation of the stresses on key components – that the reactor experience referenced by Entergy is of sufficient relevance to support an exemption to the transient testing requirement. Id. at 14.

Entergy disagrees, arguing that assurance that operations at EPU will not introduce new thermal hydraulic phenomena or unexpected system interactions is provided by (1) the behavior of similar BWRs at EPU conditions, (2) the behavior of the VYNPS after it was physically modified for the EPU but prior to implementation of the actual uprate, (3) the system and component testing performed by Entergy during normal operations, and (4) the similarities between pre- and post-EPU plant design and configuration. Entergy Statement of Position at 9-15; Nichols/Casillas Direct Testimony for Entergy at 18-26.

With regard to industry experience, Entergy referred to thirteen BWRs (asserted to be similar to the VYNPS) that have implemented EPUs and noted that none of the eleven EPUs that occurred in the United States have been required to perform large transient testing. Entergy Statement of Position at 9. In particular, Entergy's witnesses pointed to two of these BWRs – the two-unit Brunswick plant⁵⁴ and the Hatch plant⁵⁵ – to support the proposition that large transient testing is not required under Criterion XI “to demonstrate the [VYNPS at EPU] will perform satisfactorily in service.” Entergy Statement of Position at 9-11; Nichols/Casillas Direct Testimony for Entergy at 18-20. In an exhibit to its testimony, Entergy compared a

⁵⁴ Brunswick consists of two reactors, Units 1 and 2, that are located near Southport, North Carolina. Unit 1 has an electrical output of 872 MWe, was manufactured by General Electric, and is a BWR 4 with a Mark I containment. Unit 2 has a slightly lower electrical output of 811 MWe but is otherwise the same as Unit 1. See <http://www.nrc.gov/info-finder/reactor/bru2.html>.

⁵⁵ Hatch consists of two reactors, Units 1 and 2, that are located near Baxley, Georgia. Unit 1 has an electrical output of 856 MWe, was manufactured by General Electric, and is a BWR 4 with a Mark 1 containment. Unit 2 has an electrical output of 870 MWe and is also a General Electric BWR 4 with a Mark 1 containment. See <http://www.nrc.gov/info-finder/reactor/hat2.html>.

number of parameters for Brunswick and VYNPS (including power density, relief capacity and bypass capacity) and asserted that the facilities are similar in all significant respects that bear on large transient performance. Entergy Exh. 38 at 1. For example, Entergy noted that the Brunswick units are both BWR 4s with Mark 1 containments – as is the VYNPS. Entergy’s witnesses asserted that a comparison of the design-important parameters for the Brunswick and VYNPS plants show that they are similar in the parameters that would affect the large transient performance of the plants, for example, power density and steam relief and bypass capacities. Nichols/Casillas Direct Testimony for Entergy at 6; Entergy Exh. 3. Mr. Nichols and Casillas further stated that in the fall of 2003, Brunswick Unit 2, which was granted a 120-percent EPU, experienced an unplanned generator turbine trip transient⁵⁶ when it was at 115.2 percent of its original licensed thermal power (OLTP) and that no anomalies or unanticipated plant behavior or phenomena occurred. Nichols/Casillas Direct Testimony for Entergy at 19.

Entergy also made reference to Unit 2 of the Hatch plant, another BWR 4 with a Mark 1 containment system similar to the VYNPS, which experienced an MSIV closure from 113 percent of the OLTP. Id. at 18. The operators of Hatch reported that all of the Hatch systems functioned as expected. Id.; Entergy Exh. 10. Entergy’s witnesses concluded that the absence of anomalies or unexpected phenomena during the post-uprate unplanned transients at Brunswick and Hatch supports the conclusion that the VYNPS should also perform as predicted during uprated conditions. Nichols/Casillas Direct Testimony for Entergy at 20.

With regard to the operational experience of VYNPS itself, Mr. Nichols and Casillas testified that five large transients occurred between 1991 and 2005 while the VYNPS was operating at full pre-EPU power levels, including two that occurred “after most of the

⁵⁶ A “generator turbine trip transient” is a transient whose triggering event is different from that of a GLR transient, but which proceeds in the same manner as a GLR transient.

modifications associated with the EPU were already implemented,” and that VYNPS experienced no significant anomalies during these transients. Nichols/Casillas Direct Testimony for Entergy at 23. Entergy’s witnesses also stated that there are great similarities between VYNPS’s pre and post-EPU plant design and physical configuration and concluded that none of the EPU changes will introduce new thermal-hydraulic phenomena or new system interactions. Id. at 24-25.

The Staff agrees with Entergy on this matter. Staff witnesses stated that the information submitted by Entergy, including the operating experience at Hatch Units 1 and 2, support the proposition that the EPU-related modifications at VYNPS will not introduce new operating phenomena or anomalies. Ennis et al. Direct Testimony for NRC Staff at 12. For example, the Staff witnesses stated that, after uprate the Hatch plant experienced a Unit 1 turbine trip transient in 2000, a Unit 1 GLR transient in 2001, and a Unit 2 GLR transient in 1999, and that these transients produced no anomalies or unexpected phenomena. Id. In sum, the Staff witnesses stated that they reviewed Licensee Event Reports (LERs) concerning transients at other BWR units operating at EPU levels, looking specifically for examples of new phenomena, different responses in the modified systems, or any unusual behavior that could be attributed to the increased steam flow or feed flow. See Ennis et al. Direct Testimony for NRC Staff at 13. The Staff witnesses stated that they did not observe any such abnormal behavior, nor did they see any modifications to the VYNPS that were inconsistent with the modifications implemented at other facilities. Id.

Ms. Abdullahi of the Staff also testified that, for overpressure protection, the most important plant parameter in an MSIV transient is safety relief valve (SRV) capacity. Tr. at 1471. In contrast, for a GLR transient, she said the most important parameter is bypass capacity. Tr. at 1473. Ms. Abdullahi stated that the Staff examined the similarities of Brunswick

and the VYNPS to determine if the performance of the two plants during MSIV transients and GLR transients would be similar, and concluded that they would. With regard to the MSIV transient, Ms. Abdullahi stated the SRV capacity for the VYNPS (at uprate) is 60 percent, which is similar to, and more conservative (i.e., safer) than the 56 percent SRV capacity for the Brunswick plant (at uprate). Tr. at 1471-72. With regard to the GLR transient, Ms. Abdullahi testified that the VYNPS has a bypass capacity of 86 percent of rated steam flow (at uprate), which is similar to and more conservative than the 69 percent bypass capacity of Brunswick Unit 2 (at uprate). Tr. at 1473. According to Ms. Abdullahi, these comparisons suggest to the Staff that the VYNPS has sufficient relief valve and bypass capacity in the event of an overpressure transient such as an MSIV closure or a GLR transient. Tr. at 1471-74.

b. Board Findings

The Board finds that the comparisons and similarities between the Brunswick BWRs and the VYNPS are persuasive. Both Brunswick and VYNPS are BWR 4s with Mark 1 containments, and they have similar power densities. Since both transients under consideration are pressurization transients, it is particularly important that the VYNPS has slightly greater relief capacity than Brunswick (60 percent for the VYNPS and 56 percent of total steam flow at uprated conditions for Brunswick). For the GLR transient, the higher steam bypass capacity for the VYNPS (86 percent) compared to Brunswick (60 percent) provides an even greater margin of assurance. Since the relief and steam bypass capacities to a large extent determine how a plant performs during a pressurization transient, the Board finds that Brunswick and the VYNPS would be expected to respond in a similar manner to either an MSIV closure transient or a GLR transient, with VYNPS having a somewhat greater safety margin in both instances.

This finding is further supported by the testimony regarding the actual behavior of the VYNPS during recent GLR transients. As Mr. Nichols pointed out, the transients at the VYNPS

in 2004 and 2005 occurred after most of the modifications associated with the EPU were already implemented, including the new high pressure turbine rotor, main generator stator rewind, the new high pressure feedwater heaters, condenser tube staking, an upgraded isophase bus duct cooling system, and condensate demineralizer filtered bypass.

Nichols/Casillas Direct Testimony for Entergy at 23. He added that VYNPS's performance during these transients, including that of the modified components, demonstrated that the EPU modifications do not introduce new hydraulic phenomena or significantly affect the plant's response during transient conditions. Id. Although these transients occurred at the original license power (or below) and not at the uprated conditions, they took place after most of the uprate modifications were completed. No anomalies were observed during the VYNPS transients.

While neither Entergy nor the Staff provided detailed comparisons of Hatch and the VYNPS, they did note that both are BWR 4s with Mark 1 containments and would thus be expected to behave in a similar manner during large transients. Hatch showed no anomalous behavior during an MSIV closure at uprated conditions.

Based on the testimony and exhibits concerning the operating experiences at Hatch and Brunswick under uprate conditions, the similarities between those plants and the VYNPS, and the transients and events that have occurred at the VYNPS, including two that occurred after most of the uprate modifications were made at VYNPS, albeit prior to the implementing the actual power increase, the Board finds that there is reasonable assurance that the operation of VYNPS at uprated conditions will not introduce new thermal hydraulic phenomena or system interactions that would occur during an MSIV transient or GLR transient.

3. Contested Issue Two - Adequacy of computer stress analysis
 - a. Key Evidence Presented

One argument put forth by NEC concerning the computer analysis centers on the allegation that although General Electric's ODYN code is able to predict the maximum system pressure during a transient, it fails to predict the stress or vibration levels in individual components. Hopenfeld Rebuttal Testimony for NEC at 4, 7. Dr. Hopenfeld asserted that the applied structural stresses and allowable stresses ultimately determine whether a given component performs satisfactorily in service, and thus that ODYN's focus on the maximum pressure alone is insufficient to assure system performance. Id. at 7-8. Dr. Hopenfeld also asserted that "[t]he frequency and amplitude of the vibrations as well as the component's natural frequency, which is affected by temperature and temperature gradients, for example, govern failure of components from vibrations." Id.

Dr. Hopenfeld testified that he was concerned that resonance vibrations of high amplitude could be excited during a transient. Tr. at 1517. If a given component is already weakened and has used up its fatigue cycles, he claimed, the component would already be at its endurance limit for fatigue prior to the stresses imposed by a transient. Or, if there is stress corrosion and the components are already cracked, then the resonant vibration could potentially cause a problem such that the component would not fulfill its design requirement. Tr. at 1516.

Mr. Casillas, testifying for Entergy, acknowledged that the ODYN computer analysis of large transient tests focuses on the peak vessel pressure and does not analyze other loads or stresses on individual components. Nichols/Casillas Direct Testimony for Entergy at 15-16. He asserted, however, that the peak vessel pressure analysis was appropriate to confirm that the reactor components and vessel meet the loads used in their design. Id. at 16-17. Mr. Nichols

pointed out that there was a whole section of structural analysis performed for the power uprate, covering steady state, transients, and accident loads. Tr. at 1576.

Mr. Ennis, speaking for the Staff, also acknowledged that ODYN does not do a calculation of the stress in a component or what is commonly referred to as a “stress analysis.” Tr. at 1482-83 He asserted, however, that a stress analysis for important components was done using other acceptable methods, as outlined in the constant pressure power uprate (CPPU) safety evaluation. Id. (citing CPPU SE § 3.2). For example, Mr. Ennis testified that the Staff review found that General Electric had calculated the stresses for the ASME base load code cases, and those calculations include transient conditions, as well as other conditions such as seismic. Tr. at 1481-82. He stated that the methodology used was consistent with the CPPU SE, that the stresses would remain within acceptable limits, and structural integrity would be maintained under EPU conditions, including transients. Ennis et al. Direct Testimony for Staff at 11-12; Tr. at 1482-83. He added that section 3.2 of the CPPU SE discusses the stress analysis of the reactor pressure vessel and its internals, and that section 3.4 discusses piping systems and associated components. Tr. at 1481-83; see also CPPU SE at §§ 3.2, 3.4. Mr. Ennis further testified that a stress analysis was performed for the steam dryers, including stress under transient and steady state conditions, even though they are not ASME components.⁵⁷ Mr. Ennis stated that the results of the analysis predicted that the structural integrity of the steam dryer and of piping system and components would be maintained under repeated loading conditions. Tr. at 1484.

⁵⁷ Tr. at 1486. “ASME components” is a term of art that refers to those components required by 10 C.F.R. § 50.55a(c) to meet the requirements of Class 1 components in Section III of the ASME Boiler and Pressure Vessel Code.

b. Board Findings

Because it was acknowledged that General Electric's ODYN computer code does not do a component stress analysis, the Board finds that NEC is correct that the ODYN code, by itself, is inadequate to determine the structural integrity of the components at steady state and during transients at the uprated power. As the Staff testified, however, the ODYN code was not used by itself. Additional stress analysis, as outlined in the CPPU SE, was done to determine the stress levels in various critical components, including the steam dryers, and the results were acceptable. The Board therefore finds that the stress analysis performed in accordance with ASME-accepted analysis⁵⁸ methods on the steam dryer and on the ASME components, in conjunction with the ODYN computer analysis, provided adequate assurance of safe operation after the uprate and is therefore acceptable.

4. Contested Issue Three - Adequacy of ODYN code benchmarking for pressurization transients or for steady state operation

a. Key Evidence Presented

NEC's expert, Dr. Hopenfeld, is critical of Entergy's use of the General Electric ODYN computer code, asserting that such a computer code must be validated (i.e., "benchmarked") by comparing its predictions with data from well-instrumented prototype components. Hopenfeld Direct Testimony for NEC at 5. Dr. Hopenfeld stated that, if such validation or benchmarking is not done, the predictions of the code may result in significant errors in values calculated by the code, for example in the values of the parameters that determine the transfer of heat. Id. at 6. Knowing the uncertainty in a code's predictions, which is to say how much error there might be in the calculation, is essential to understanding the capability of the code to estimate whether the component will fail under uprated conditions. Dr. Hopenfeld testified that, when Entergy

⁵⁸ Tr. at 1486. Acceptable methods are discussed in the ASME Boiler and Pressure Vessel Code, the use of which is required by 10 C.F.R. § 50.55a(c).

discussed the benchmarking of the ODYN code, Entergy (1) provided no comparison of experimental data with code predictions, (2) did not describe in sufficient detail how the code was qualified, and (3) failed to state that the ODYN code was benchmarked for pressurized transients and for steady state operations. Id. at 5-7. Dr. Hopenfeld asserted that Entergy must provide the public with an analysis of the key assumptions underlying use of the code. Id. at 5.

Dr. Hopenfeld further pointed out that because computer codes such as ODYN incorporate certain simplifications to describe transient behavior, their validity is limited to those cases in which the code was benchmarked by comparison with real-world data. Id. He further testified that, because of those simplifications, a computer code such as ODYN has a limited range of validity, i.e., such codes can predict outcomes very accurately under a certain set of boundary conditions, yet they will be very inaccurate in predicting the outcome under different boundary conditions.⁵⁹ Dr. Hopenfeld asserted that it is not the amount of conservatism that is important, but rather the understanding of the reasons for any discrepancy between the experimental data and the code predictions. Hopenfeld Rebuttal Testimony for NEC at 9.

According to Dr. Hopenfeld, neither Entergy nor the Staff discusses the specific test data, particularly the Peach Bottom turbine trip data,⁶⁰ that was compared to the ODYN predictions to validate the code. Hopenfeld Rebuttal Testimony for NEC at 6. NEC's expert also stated that neither Entergy nor the Staff explain why the predicted peak reactor pressure

⁵⁹ Declaration of Dr. Joram Hopenfeld Supporting New England Coalition's Response to ENVY's Motion for Summary Disposition (Dec. 21, 2005) at 3 (incorporated by reference into Hopenfeld Direct Testimony for NEC at 7).

⁶⁰ Peach Bottom Unit 2 is a General Electric BWR 4 with a Mark 1 containment. <http://www.nrc.gov/info-finder/reactor/pb2.html>. The Peach Bottom turbine trip tests refer to a series of three turbine trip experiments performed at Peach Bottom Unit 2 in April 1977. General Electric Licensing Topical Report, Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors, Vol. 1 (Aug. 1986) at II-39 to II-40 (Entergy Exh. 26).

calculated by the ODYN code for the Peach Bottom turbine trip experiment exceeded the measured experimental data. Id. at 9. Dr. Hopenfeld acknowledged that, to understand the validity of the code predictions, one need not review or be interested in the specific mathematical techniques in the ODYN code or in any proprietary data. Rather, he declared that it would be sufficient to be able to determine, from information Entergy should be supplying, how accurately ODYN can predict the experimental measurements from the Peach Bottom experiment. Id. For example, he testified that Entergy should compare the ODYN code predictions of core exit pressure rise, pressure oscillations, and water levels to the measured values from the turbine trip tests at Peach Bottom. Id. at 9-10.

Mr. Casillas, testifying for Entergy, disagreed, asserting that the ODYN code accurately models BWR vessel physical components, mechanical equipment functions, and control systems and accurately predicts the nuclear thermal-hydraulic phenomena. Nichols/Casillas Direct Testimony for Entergy at 12- 13. He stated that “[t]he simulation involves describing the actual physical plant in the model (i.e., volumes, flow paths, resistances), establishing the desired operating conditions (i.e., water level, power, pressure) and introducing a disturbance (i.e., valve closure, pump trip, control action).” Id. at 13. Based on the physical model correlations, Mr. Casillas concluded that the ODYN code accurately predicts the plant response behavior. Id.

Mr. Casillas further asserted that GE has benchmarked the ODYN code “against all significant plant transients including turbine trips (equivalent in its effects to a generator load rejection test) and main steam valve isolation events.” Id. at 14. He stated that the turbine trip data were obtained from the Peach Bottom and Swiss KKM plants, and that the MSIV closure data were obtained from the Hatch plant. Id. Mr. Casillas further declared that the Peach Bottom turbine trip tests date back to the late 1970s and form the initial benchmark for

pressurization transients and uncertainty margins for the ODYN code. Id. According to him, all subsequent advanced versions of the ODYN code have been assessed against these tests and continue to form the basis for the code's accuracy. Id. at 14-15. He stated that “the current version of the ODYN code continues to accurately predict the overpower magnitude and slightly overpredict the overpressure magnitude vis-a-vis the Peach Bottom tests.” Id. at 15.

Mr. Casillas testified that an earlier version of the ODYN code, the 05 version, was qualified (i.e., benchmarked) by GE against MSIV transient data from a cycle one test at the Hatch nuclear power plant that occurred in 1983. Tr. at 1330; Nichols/Casillas Direct Testimony for Entergy at 14. Mr. Casillas stated that there are two important parameters in an MSIV closure - pressure and water level. Tr. at 1602 (redacted version). According to him, GE compared the water level and pressure that were predicted by ODYN (05 version) against the actual water level and pressure that occurred during the Hatch test and concluded that the ODYN (05 version) code was accurate and conservative in its prediction of peak pressure and water level during an MSIV transient. See id. at 1602-05. Mr. Casillas pointed out, however, that NRC accepted the ODYN code based on the Peach Bottom tests/benchmark and that the Hatch benchmark was not part of that acceptance. Tr. at 1330.

Mr. Casillas acknowledged that current code validation practice requires that one perform representative transients that one intends to analyze using the subject code, and that this approach is substantially different from what was done with ODYN for an MSIV transient. Tr. at 1333-35. In the case of ODYN, he pointed out that GE includes the Peach Bottom turbine trip transient in its suite of code comparisons used to benchmark the code for licensing applications, but does not include an MSIV closure such as the Hatch test. Tr. at 1329-30.

Mr. Casillas also acknowledged that in addition to benchmarking the code, an analyst must ensure that the plant model represents the subject reactor. He testified that GE indeed

uses a design procedure whereby the inputs are verified to ensure that they reflect the characteristics of the subject plant. Tr. at 1352-53. He indicated that the procedure tells the analyst how to nodalize⁶¹ the model, where the nodes should be, how big or how small they need to be, and how many of them are needed. Tr. at 1353. Mr. Casillas explained that the plant data used to develop the ODYN plant model is taken from drawings, system settings and set points, and plant dimensions. Id. He stated that the designer runs some stability tests and some model comparisons, including the steady state condition predicted by the model with the plant conditions. Id. But, he acknowledged that no comparisons are made between the ODYN code that uses the plant model and data from actual transients. Tr. at 1354. Once completed, the model is checked by an independent verifier. Tr. at 1355.

The NRC Staff witnesses Zeynab Abdullahi and George Thomas pointed out that ODYN “has been approved by the NRC for application to transients such as . . . generator load reject, turbine trip; [and] MSIV closure.” Ennis et al. Direct Testimony for Staff at 17. They testified further that the qualification process for the code included “quantifying the accuracy of the code’s predictions” and comparing ODYN predictions with real-world occurrences and with the predictions of other models. Id. at 18-20. After the Staff initially approved it, Ms. Abdullahi and Mr. Thomas stated that the ODYN code was assessed against actual transients in plants at EPU conditions, and the model has performed properly in these circumstances. Id. at 21-24. According to Ms. Abdullahi and Mr. Thomas, these tests “provide reasonable assurance that

⁶¹ As the Board understands it, the term “nodalize” refers to the fact that, once a geometric model has been created, a procedure is used to define and break up the model into smaller elements called nodes. The computer model is defined by a geometric mesh or network of these nodes. The nodes represent the regions or volumes where the physical parameters of interest such as pressure and temperature are calculated. The nodes are defined by a numbering scheme that allows reference to be made to the parameters of interest at specific locations in the model.

use of the ODYN code will acceptably simulate plant response to limiting pressurization response.” Id. at 23-24.

b. Board Findings

It is the Board’s conclusion that, as Entergy has acknowledged, current code validation practices require that a code be benchmarked or compared against all transients of interest. The transients of interest here (i.e., the ones that are the subject of NEC Contention 3) are the GLR transient and the MSIV transient. In the case of GLR transients, the Board finds that the GE benchmarking of the ODYN code against data from the Peach Bottom turbine trip experiment satisfied the benchmarking requirement for GLR transients because, for this purpose, we consider turbine trip transients equivalent to GLR transients.

With regard to MSIV transients, however, even Entergy's witnesses admit that GE does not routinely benchmark or do a comparison of versions of the ODYN code with plant data from an MSIV closure. For the model of the plant used with the ODYN code, Entergy witnesses explained that the model is checked using a validation procedure where the input data is confirmed by an independent verifier. Tr. at 1353. The model is then used with the ODYN code to calculate the pressures, temperatures and other reactor and reactor system characteristics of the plant while at steady state conditions. The results are compared with actual plant data to validate the plant model. Tr. at 1353-54. Entergy witnesses stated, however, that no transients are analyzed using the plant data to benchmark the plant model. Tr. at 1354.

As was noted above, the Board finds that the method used by GE to benchmark the ODYN code for steady state and for a GLR transient are adequate to calculate reactor pressure for a GLR transient because each version of the code is checked against a test suite that includes the Peach Bottom turbine trip transient. We find that the methods used to benchmark

the ODYN code for an MSIV transient are not adequate, however, because data from such a transient is not in the test suite used to assess each version of the ODYN code. We also find that the plant models are not adequately verified because the verification process does not include checking the models' ability to replicate anything other than steady state conditions.

While thus concluding that the ODYN code benchmarking for MSIV transients could be improved, we do not agree, based on the evidence before us, that this deficiency alone is a sufficient basis for resolving this challenge to Entergy's EPU request in NEC's favor. As the record before us amply demonstrates, actual operational experience, rather than the ODYN code, is the important factor in determining what testing is needed to assure safe operation under uprated conditions.

While any benchmarking deficiencies relative to the ODYN code thus are not determinative in our decision on NEC contention 3, we do note that there have been a number of improvements made (and assessed) to the ODYN code since it was originally approved for use in licensing by the NRC in 1981.⁶² Code development and verification and bench-marking techniques have evolved over the years and currently are relatively sophisticated when compared to those in use when ODYN was approved. Understanding the inherent uncertainties in the various models internal to the code is especially important where safety margins are reduced, as in the case of power uprates. Consequently, if continued regulatory use of ODYN is contemplated, the Board encourages the Staff to take a fresh look at the code's components and their uncertainties to see if a reassessment of the ODYN code using modern methods is warranted.

⁶² See Letter from Robert L. Tedesco, Assistant Director for Licensing, NRC, to Dr. G.G. Sherwood, Manager for Safety and Licensing, General Electric Co., Acceptance for Referencing General Electric Licensing Topical Report NEDO-24154/NEDE-24154P (Feb. 4, 1981) (incorporated into Entergy Exh. 26 at 3).

C. Ultimate Factual Finding

As framed by the three specific objections raised by NEC in Contention 3, the ultimate factual and legal issue in this case may be summarized as whether, under all of the facts and circumstances presented in the record, Entergy has adequately demonstrated that the VYNPS structures, systems and components will perform satisfactorily under uprated conditions, without the need for an MSIV transient test or a GLR transient test. Entergy, not NEC or the Staff, bears the burden of proof on this question.

Our consideration of this issue begins with the proposition that the NRC Staff's guidance contemplates that, as a general rule, a MSIV transient test and a GLR transient test should be performed prior to an EPU. Entergy Exh. 4 (SRP Section 14.2.1). However, the Staff guidance also provides a mechanism whereby the EPU applicant can submit a case-by-case justification as to why such testing is unnecessary. The guidance specifies seven factors that should be considered in determining whether these large transient tests are needed. See discussion supra at 31. And while the Staff guidance is not binding on the Board, we find it provides a set of reasonable and useful factors to consider.

In this case, Entergy and the Staff followed the approach outlined in SRP 14.2.1. Entergy's EPU application included a request and justification as to why large transient testing should not be required. Entergy Exh. 5. Entergy's justification covered six of the seven factors laid out in SRP 14.2.1. Id. The Staff reviewed this request and determined that it should be granted, i.e., "that there is reasonable assurance that the VYNPS SSCs will perform satisfactorily in service under EPU conditions." Entergy Exh. 7 (FSER at 271). NEC disagreed, and filed the instant contention, raising the specific issues and challenges set forth above.

As an initial matter in resolving NEC Contention 3, the Board attempted to understand the basis for the Staff's conclusion that the MSIV and GLR transient tests were unnecessary.

All of the filings and the exhibits including those taken from the FSER (1) recited the factors identified in SRP Section 14.2.1, (2) repeated Entergy's statements and justifications, and (3) summarily concluded, with virtually no explanation, that the Staff believed that Entergy had satisfied the guidance and that it should be exempt from large transient testing.⁶³ Given this information, NEC's position was not entirely surprising. Our concern was further fueled by the fact that, although the Staff previously denied GE's request for generic exemption from large transient testing in EPU situations, and instead required that a case-by-case justification be presented, in reality the Staff has granted every case-by-case exemption that has ever been requested (all fifteen). Tr. at 1454. Of great concern was the Staff's failure to explain, until questioned by the Board, the logic used in reaching the conclusion that large transient testing was not necessary at VYNPS.

As it turned out, however, during the evidentiary hearing both Entergy and the Staff provided persuasive testimony and evidence supporting the proposition that the MSIV and GLR transient tests are not required to demonstrate that the VYNPS structures, systems and components will perform satisfactorily under the uprated conditions. In this regard, Entergy and the Staff provided ample evidence that the industry operating experience at analogous BWR plants indicated that large transient testing at VYNPS under uprated conditions is not needed. They discussed the thirteen BWR plants that have implemented EPUs and focused specifically on the Hatch and Brunswick units, explaining the substantial similarities between those facilities and the VYNPS. Nichols/Casillas Direct Testimony for Entergy at 18-20; Ennis et al. Direct

⁶³ The Staff's explanation of its conclusion seemed to be a generic one. "From the EPU experience referenced by the licensee, it can be concluded that large transients, either planned or unplanned, have not provided any significant new information about transient modeling or actual plant response." Entergy Exh. 7 (FSER at 271). We do not know what to make of this rationale, given that the Staff previously rejected GE's attempt to obtain a generic exemption from large transient testing in a CPPU EPU.

Testimony for Staff at 12. This testimony included evidence that the performance of the Hatch and Brunswick plants under MSIV and GLR transients has been satisfactory with no anomalies or unexpected thermal-hydraulic phenomena. Nichols/Casillas Direct Testimony for Entergy at 18-20; Entergy Exh. 38; Ennis et al. Direct Testimony for Staff at 12. Likewise, Ms. Abdullahi of the Staff testified that “everything happened as designed, and as expected” during the MSIV closure event (at 113 percent uprate) at Hatch and the turbine trip event (at 120 percent uprate) at Brunswick. Tr. at 1434. Ms. Abdullahi emphasized, rightly we believe, that empirical operating experience, not ODYN, is the most important factor in evaluating what testing, if any, is necessary to assure that the VYNPS will perform safely at uprate conditions. Tr. at 1433-35.

The Board is also impressed that the operating experience at VYNPS, the nature of the modifications made at the plant as a part of the EPU, and the component testing, all indicate that the EPU will not introduce new thermal or hydraulic phenomena that warrant conducting MSIV or GLR transient tests. In this regard, Mr. Jones, testifying for the Staff, stated they considered four factors in evaluating the delta of the EPU. Tr. at 1427. First, Mr. Jones stated that the Staff evaluated the scope of the modifications. Tr. at 1428. According to him, there were twenty modifications, the most important of which were listed by Entergy in Exhibit 39. Tr. at 1426; Staff Exh. 2 at 273. In assessing the impact of the modifications on a GLR transient, Mr. Jones asserted that very few that would alter the response of either the turbine bypass system or the feed and condensate system to a GLR transient. Tr. at 1428. Second, Mr. Jones said that the Staff looked for any indication that there would be new thermal-hydraulic phenomena that would affect the response of the VYNPS to a GLR. He stated that the Staff’s conclusion that no such phenomena can be identified is based on several LERs from other plants that have experienced load rejection transients at extended power uprate conditions. Tr. at 1428. Third, Mr. Jones stated that the Staff considered the recent experience at the VYNPS

after many of the modifications for the uprate had been made, including the load rejection event occurred at the VYNPS in June 2004. Id. Although the 2004 event occurred at a lower power than that of the uprate, it occurred after many of the balance of plant modifications for the uprate had already been implemented. Mr. Jones stated that no unusual behavior was observed as a result of this event. Tr. at 1428. Fourth, Mr. Jones declared that the Staff considered the power ascension test program, which included extensive monitoring of the plant under steady state conditions as well as during a slow power ascension. Tr. at 1428-29.

He added that the Staff also considered the separate effects tests, such as the technical specification test that checks the feedwater isolation if the reactor vessel is overfilled and the tests of other systems that would be implemented as part of the post-modification EPU testing. Tr. at 1429. Finally, Mr. Jones cited the condensate and feedwater test that was implemented as part of the license condition, asserting that it demonstrated, again, the proper integrated performance in the feedwater and condensate systems to a transient. Tr. at 1429.

On the basis of the foregoing, and the entire record herein, the Board finds that the industry experience at the Hatch and Brunswick plants, as well as prior experience at VYNPS, has shown no abnormal behavior or evidence of fuel damage as a result of the transients experienced. Further, although it occurred before the uprate was completed, the 2004 transient at the VYNPS also provides reassurance that transient testing is not required because most of the EPU modifications were already in place at the time. Most fundamentally, the Board agrees with the NRC Staff's assertion that industry operating experience, not code predictions, should be the major factor in this type of decision. Furthermore, although the ODYN code predictions were not the major determination in the Staff's decision, or in ours, the Board notes that the predictions of the ODYN code are consistent with the observed transient behavior of Hatch, Brunswick, and the VYNPS despite the apparent lack of adequate benchmarking.

The Board finds that the industry experience cited by the Staff and applicant, as well as the transient experienced at the VYNPS, provides an adequate basis for us to conclude that it is not necessary to perform an MSIV closure test or a generator load rejection test to satisfy the regulatory requirements described in Section III of this order.

V. CONCLUSIONS OF LAW

Criterion XI of 10 C.F.R. Part 50, Appendix B and 10 C.F.R. § 50.54(a)(1) require that each nuclear power plant implement a quality assurance program that includes “all testing required to demonstrate that the structures, systems and components will perform satisfactorily in service.” It is the burden of the EPU applicant, Entergy, to show that its QAP testing program meets this criterion. Here, the New England Coalition asserts that large transient testing – specifically a main steam isolation valve transient test and a generator load rejection test – are needed to demonstrate that the VYNPS will perform satisfactorily in EPU service. The NRC Staff and the Advisory Committee on Reactor Safeguards considered the matter and concluded that such large transient testing is not required.⁶⁴

As stated above, the Board is persuaded by the evidence presented, particularly the industry experience cited by the Staff and Entergy and the transient experienced at the VYNPS, that a main steam isolation valve closure test or a generator load rejection test are not necessary to assure safe operation of the VYNPS after its extended power uprate. Accordingly, we conclude that Entergy’s quality assurance program satisfies Criterion XI and 10 C.F.R. § 50.54(a)(1) by providing “all testing required to demonstrate that the structures, systems and components will perform satisfactorily in service.” Thus, NEC Contention 3 is resolved in favor of Entergy.

⁶⁴ Entergy Statement of Position; see also Entergy Exh. 22.

VI. ORDER

For the foregoing reasons it is hereby ordered that NEC Contention 3 is resolved in favor of the applicant, Entergy. This initial decision shall constitute the final decision of the Commission forty (40) days from the date of its issuance, unless, within fifteen (15) days of its service, a petition for review is filed in accordance with 10 C.F.R. §§ 2.1212 and 2.341(b).⁶⁵ Filing of a petition for review is mandatory for a party to exhaust its administrative remedies before seeking judicial review. 10 C.F.R. § 2.341(b)(1).

It is so ORDERED.

THE ATOMIC SAFETY
AND LICENSING BOARD⁶⁶

/RA/

Alex S. Karlin (Chairman)
ADMINISTRATIVE JUDGE

/RA/

Anthony J. Baratta
ADMINISTRATIVE JUDGE

/RA/

Lester Rubenstein
ADMINISTRATIVE JUDGE

Rockville, Maryland
February 26, 2007

⁶⁵ Pursuant to 10 C.F.R. § 2.1207(a)(3)(iii), the Board, by separate order, is providing to the Commission's Secretary all questions submitted by the parties under 10 C.F.R. § 2.1207(a)(3)(i)-(ii).

⁶⁶ Copies of this memorandum and order were sent this date by Internet e-mail transmission to representatives for (1) licensees Entergy Nuclear Vermont Yankee, L.L.C. and Entergy Nuclear Operations, Inc.; (2) intervenor New England Coalition of Brattleboro, Vermont; and (3) the NRC Staff.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of)
)
ENTERGY NUCLEAR VERMONT YANKEE L.L.C.) Docket No. 50-271-OLA
and ENTERGY NUCLEAR OPERATIONS, INC.)
)
)
)
)
(Vermont Yankee Nuclear Power Station))

CERTIFICATE OF SERVICE

I hereby certify that copies of the foregoing LB INITIAL DECISION (RULING ON NEC CONTENTION 3) (LBP-07-02) have been served upon the following persons by deposit in the U.S. mail, first class, or through NRC internal distribution.

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Docket No. 50-271-OLA
LB INITIAL DECISION (RULING ON
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Office of the Secretary of the Commission

Dated at Rockville, Maryland,
this 26th day of February 2007