May 17, 2006

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
ENTERGY NUCLEAR VERMONT YANKEE, LLC and ENTERGY NUCLEAR OPERATIONS, INC.	)	Docket No. 50-271-OLA
	)	ASLBP No. 04-832-02-OLA
(Vermont Yankee Nuclear Power Station)	)	

NRC STAFF TESTIMONY OF RICHARD B. ENNIS, STEVEN R. JONES, ROBERT L. PETTIS JR., GEORGE THOMAS, AND ZEYNAB ABDULLAHI CONCERNING NEC CONTENTION 3

Q1. Please state your names, occupations, and by whom you are employed.

A1(a). My name is Richard B. Ennis (RBE).[1] I am employed by the U.S. Nuclear Regulatory Commission ("NRC" or "Commission") as a Senior Project Manager in the Division of Operating Reactor Licensing, Office of Nuclear Reactor Regulation ("NRR"), in Rockville, MD. A statement of my professional qualifications is attached hereto.

A1(b). My name is Steven R. Jones (SRJ). I am employed by the NRC as a Senior Reactor Systems Engineer in the Division of Systems Safety, NRR, in Rockville, MD. A statement of my professional qualifications is attached hereto.

A1(c). My name is Robert L. Pettis, Jr. (RLP). I am employed by the NRC as a Senior Reactor Engineer in the Division of Engineering, NRR, in Rockville, MD. A statement of my professional qualifications is attached hereto.

[1] In this testimony, the sponsor of each numbered paragraph is identified by his or her initials; no such designation is provided for paragraphs that are sponsored by all witnesses.

A1(d). My name is George Thomas (GT). I am employed by the NRC as a Senior Reactor Systems Engineer in the Division of Safety Systems, NRR, in Rockville, MD. A statement of my professional qualifications is attached hereto.

A1(e). My name is Zeynab Abdullahi (ZA). I am employed by the NRC as a Senior Reactor Systems Engineer in the Division of Safety Systems, NRR, in Rockville, MD. A statement of my professional qualifications is attached hereto.

Q2. Please describe your current responsibilities.

A2(a). (RBE) I currently serve as the Senior Project Manager for the NRC Staff ("Staff"), concerning the extended power uprate ("EPU") license amendment for the Vermont Yankee Nuclear Power Station ("Vermont Yankee" or "VYNPS"). I am currently responsible for NRC headquarters coordination and communication of technical issues related to the Vermont Yankee EPU.

A2(b). (SRJ) I am responsible for evaluating the functional requirements, design, and performance of auxiliary, support and balance of plant systems (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. I also evaluate design features and methods for protection of essential systems and components from the effects of internal and external flooding, internally and externally generated missiles, and postulated pipe breaks outside containment. In addition to evaluating licensing actions, I provide technical expertise for inspections, operational event reviews, and policy activities in the assigned areas of review responsibility.

A2(c). (RLP) I am currently responsible for the technical review of several EPU and license renewal amendment requests. As part of my responsibilities, I have been responsible for evaluating the power ascension and testing plan section of the Vermont Yankee EPU application.

A2(d). (GT) I am currently responsible for reviewing and evaluating design, process design parameters, and performance of reactor thermal-hydraulic systems for boiling water reactor ("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. I am also responsible for reviewing the analysis of anticipated operational occurrences, postulated accidents, and actual operating experience from the viewpoint of systems operation and transient dynamics. My duties also include reviews and evaluations of the effects of changes to licensed thermal power, license renewal, and other technical specification changes related to BWR reactor systems.

A2(e). (ZA) I am currently responsible for evaluating the technical merit of applications requesting changes to the operation of nuclear power plants, regarding the impacts of the proposed changes on reactor response during steady state, transient and accident conditions. My areas of responsibilities include evaluating design basis safety analyses supporting BWR plants' operation (*e.g.*, reactor fuel and core performance, transients, emergency core cooling system ("ECCS") loss of coolant accidents ("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.

Q3. Please explain what your duties have been in connection with the NRC Staff's review of the application of Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (collectively, "Entergy" or "Applicant") for an EPU license amendment for Vermont Yankee.

A3(a). (RBE) As part of my official responsibilities as the Senior Project Manager for the Staff's review of the Vermont Yankee EPU, I was the principal point of contact for NRR activities related to the EPU amendment. In addition, I coordinated the Staff's evaluation of the Vermont Yankee EPU and assisted in preparation of the Staff's draft Safety Evaluation for the EPU application ("Draft SE"), issued to the Advisory Committee on Reactor Safeguards ("ACRS") in October 2005 (Revision 0), and to the public in November 2005 (Revision 1); and I coordinated the Staff's preparation of the Final Safety Evaluation for the EPU application ("Final SE"), issued on March 2, 2006.[2]

A3(b). (SRJ) As part of my official responsibilities, I supervised the Staff's safety review of mechanical systems other than those directly associated with the nuclear steam supply system (*i.e.*, "Balance-of-Plant" systems) related to the Vermont Yankee EPU application; these include the condensate, feedwater, main steam, main turbine, and turbine bypass systems that are involved in the plant's response to transients. My supervisory role included verifying that the Staff developed safety conclusions which were adequately supported by the Applicant's responses to Staff requests for additional information and the Staff's technical evaluation of the effects of the proposed EPU on Balance-of-Plant systems. These technical reviews are described in Sections 2.5 and 2.12 of the Staff's Draft SE and Final SE.

A3(c). (RLP) As part of my official responsibilities, I coordinated the NRC 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 Draft SE and Final SE.

A3(d). (GT) As part of my official responsibilities, I conducted the reactor systems review of the transient analyses submitted by the Applicant for the Vermont Yankee EPU, including preparation of Section 2.8.5 in the Staff's Draft SE and Final SE.

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[2] *See* "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 229 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).

A3(e). (ZA)) As part of my official responsibilities, I 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. This review is discussed in Section 2.8.7 in the Staff's Draft SE and Final SE.

Q4. What is the purpose of this testimony?

A4. The purpose of this testimony is to provide the NRC Staff's views with respect to NEC Contention 3, challenging the Applicant's justification for not performing large transient testing as a condition of the EPU license amendment.

Q5. Are you familiar with NEC Contention 3?

A5. (RBE, RLP, GT, SRJ) Yes. As admitted by the Licensing Board's Memorandum and Order of November 22, 2004, NEC Contention 3, states as follows:

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

Further, we have reviewed the Declaration of Arnold Gundersen ("Gundersen Declaration") filed in support of this contention as part of NEC's Request for Hearing dated August 30, 2004. As discussed in the Licensing Board's Memorandum and Order of April 17, 2006, we understand that two tests, the main steam isolation valve ("MSIV") closure test and the generator load rejection test, are embraced within the scope of this contention.

Q6. Please identify the bases alleged by Mr. Gundersen in his Declaration filed in support of this contention.

A6. (RBE, RLP, GT, SRJ) Mr. Gundersen asserted that the Applicant's plan to not perform large transient testing at EPU conditions "cannot be justified as good engineering practice nor is it in accord with Staff positions interpreting NRC regulation." Gundersen Declaration at 3. He also states that he "disagee[s] with and dispute[s] the assumptions and reasoning Entergy musters" to support not performing large transient testing. *Id.* Specifically, Mr. Gundersen took issue with certain statements made in Attachment 7 to the EPU application, entitled “Justification for Exception to Large Transient Testing.” As we understand his concerns, Mr. Gundersen asserted, in essence, that: (1) the Applicant’s citation of operational experience in the nuclear industry does not justify taking an exception to performing large transient testing for Vermont Yankee at EPU conditions; (2) Vermont Yankee’s own experience with generator load rejections at 100% of the original licensed power level does not demonstrate that there will be adequate plant performance during transients at EPU conditions; and (3) periodic testing of systems, structures, and components (”SSCs”), during steady-state plant operation, does not confirm performance characteristics of the SSCs required for appropriate transient response. Gundersen Declaration at 4-5.

In addition, Mr. Gundersen asserted that “Entergy ignores the NRC Staff’s decision in the case of the Duane Arnold EPU application.” Gundersen Declaration at 4. In particular, the declaration quotes from an NRC request for additional information, dated May 9, 2001, to the Duane Arnold licensee which states, in part, that:

> The NRC-approved ELTR-1 [General Electric Licensing Topical Report NEDC-32424P-A, “Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate” (ELTR1), dated February 1999] requires the MSIVC [main steam isolation valve closure] test to be performed if the power uprate is more than 10% above previously recorded MSIV closure transient data. The topical report also requires the GLR [generator load rejection] test to be performed if the uprate is more than 15% of previously recorded data.

Q7. Did you review other information submitted by NEC pertaining to this contention, in addition to Mr. Gundersen’s Declaration?

A7. (RBE, RLP, GT, SRJ) Yes. We also reviewed NEC’s Answer to the Applicant’s Motion for Summary Disposition, dated December 26, 2005; and NEC’s Answer to Entergy’s Statement of Material Facts Regarding NEC Contention 3, dated December 22, 2005, including the Declaration of Dr. Joram Hopenfeld.

Q8. Please identify the Commission's requirements and guidance pertaining to whether large transient testing is required or should be performed to support plant operations at EPU conditions.

A8. (RLP, SRJ) Testing requirements are derived from the quality assurance program that is incorporated into the operating license for each reactor pursuant to 10 C.F.R. § 50.34(b)(6)(ii) and implemented pursuant to 10 C.F.R. § 50.54(a). In accordance with 10 C.F.R. Part 50, Appendix B, Criterion XI, the quality assurance program must include a test program to assure that testing necessary to provide reasonable assurance that SSCs will perform satisfactorily in service is identified and performed.

Most necessary testing is performed at the component or system level, but initial test programs include integrated transient tests. Commission guidance for initial plant testing is discussed in NRC Regulatory Guide ("RG") 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants," Revision 2, dated August 1978. The RG 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 acceptable 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 MSIV closure and generator load rejection tests.

NRC regulatory guidance for EPUs is contained in RS-001, "Review Standard for Extended Power Uprates,"[3] which was developed primarily to increase the standardization and effectiveness of EPU reviews performed by the NRC Staff. This review standard provides the Staff's reviewers with references to existing review criteria (*i.e.*, applicable Standard Review Plan ("SRP") sections, branch technical positions, information notices and bulletins, generic

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[3] Office of Nuclear Reactor Regulation, "Review Standard for Extended Power Uprates," RS-001, Rev. 1 (Dec. 2003) (ADAMS Accession No. ML033640024) (NRC Staff Exhibit A).

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 NRC’s acceptance criteria for a proposed EPU test program are based on the requirements of 10 C.F.R. Part 50, Appendix B, Criterion XI.

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 NUREG-0800,[4] SRP Section 14.2.1, “Generic Guidelines for Extended Power Uprate Testing Programs,” Draft Revision 0, dated December 2002. Subsection III.A, “Review Procedures,” of SRP Section 14.2.1, provides procedures for a comparison of the proposed EPU test program to the initial plant test program. Subsection III.B provides procedures for a review of EPU post-modification testing requirements. Attachment 2 to SRP Section 14.2.1 provides a generic listing of transient tests drawn from RG 1.68 that are typically included in initial plant test programs that may be affected by modifications associated with an EPU. The two large transient tests that are the subject of this contention, MSIV closure and generator load rejection, are included in Attachment 2 and are listed therein as “Dynamic Response of Plant for Full Load Rejection,” and “Dynamic Response of Plant to Automatic Closure of All Main Steam Isolation Valves,” respectively.

Under SRP Section 14.2.1, licensees may propose an EPU test program that does not include all of the power ascension testing (including large transient testing) that would be identified by application of the review procedures in 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,” provides for such proposals and lists the following factors to be considered when assessing the adequacy of the licensee’s justification:

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[4] Office of Nuclear Reactor Regulation, NRC, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” NUREG-0800 (NRC Staff Exhibit B).

- 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.

SRP Section 14.2.1, at 7-10.

Q9. Please identify any previous NRC accepted Staff positions or guidance for EPUs relative to large transient testing for boiling water reactors.

A9 (RBE, RLP, GT, SRJ) The NRC Staff has approved General Electric Licensing Topical Report ELTR-1, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate"; following NRC approval, ELTR-1 was issued in February 1999. Topical report ELTR-1 provides generic guidelines for BWR EPUs. Section 5.11.9 and Appendix L.2.4 of ELTR-1 state that: (1) a MSIV closure test, equivalent to that conducted in the initial startup testing, will be performed if the power uprate is more than 10% above any previously recorded MSIV closure data; and (2) for uprates of more than 15%, a generator load rejection test, equivalent to that conducted in the initial startup testing, will be performed if the power uprate is more than 15% above any previously recorded generator load rejection transient data.

The approach described in ELTR-1 was based on the assumption that the maximum reactor operating pressure would be increased under EPU conditions. GE subsequently developed a different approach to uprating reactor power in BWRs that does not increase the maximum reactor operating pressure. This approach, which is the basis for the Vermont

Yankee EPU, is described in GE Licensing Topical Report NEDC-3300P-A, Revision 4, dated July 2003, "Constant Pressure Power Uprate ["CPPU"]."

The NRC Staff has reviewed and approved the CPPU topical report, as described in a Safety Evaluation ("CPPU SE") dated March 31, 2003.[5] As discussed in Section 10.5.2 of that SE, in the CPPU topical report, GE proposed that large transient tests (MSIV closure and generator load rejection) included in topical report ELTR-1 not be performed for CPPU type uprates. 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. In evaluating GE's generic justification to not perform 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. The conclusions in the Staff's CPPU SE Section 10.5.9 stated, in part, that the Staff has previously accepted not performing large transient tests on a plant-specific basis and that the Staff was developing guidance to generically address the requirement for conducting large transients tests in conjunction with power uprates. Therefore, the Staff stated that it was not prepared at that time to accept GE's generic proposed elimination of large transient tests for CPPU type uprates. The conclusions in the CPPU SE also stated that the Staff finds 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. Finally, the

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[5] *See* "Safety Evaluation by the Office of Nuclear Reactor Regulation, GE Nuclear Energy Licensing Topical Report, NEDC-33004P, Revision 1" (March 31, 2003). Sections 3.4 and 10.5 of this SE are provided as NRC Staff Exhibit C.

CPPU SE indicated that the Staff will continue to consider, on a plant-specific basis, the need to conduct these tests.

Q10. What impact do these NRC positions have on Entergy's EPU amendment request relative to the proposed elimination of large transient tests?

A10. (RBE, RLP, GT, SRJ) Entergy provided a plant-specific justification to not perform large transient testing in Attachment 7 of its Application, dated September 10, 2003; and it subsequently updated Attachment 7 in Supplement 3 to the EPU amendment request, dated October 28, 2003. Additional information was provided in Supplements 23, 28, and 30, dated February 24, April 22, and August 1, 2005, respectively.

Based on the Staff's CPPU SE, dated March 31, 2003, for BWRs utilizing the CPPU approach, licensees may provide plant-specific information to justify not performing the full load rejection and MSIV closure transient tests. The Vermont Yankee EPU is based on the CPPU approach, and, as part of its application, Entergy provided plant-specific information to justify not performing these tests for Vermont Yankee. Consistent with the guidance provided in SRP Section 14.2.1, the Staff found that the performance of those large transient tests was not necessary to demonstrate that SSCs important to safety would perform acceptably in service. This conclusion was based on the scope of the post-modification and power ascension test programs, previous operating experience, the lack of significant new thermal-hydraulic phenomena associated with a constant-pressure power uprate, conformance with limitations associated with analytical analysis methods, and the absence of a significant change in the results of safety analyses.

Q11. Do you agree with NEC's and Mr. Gundersen's assertion that the Applicant's citation of operational experience elsewhere in the nuclear industry does not support an exception to performing large transient testing for Vermont Yankee at EPU conditions?

A11. (RLP, SRJ) No.

Q12. Please provide the basis for this conclusion.

A12. (RLP, SRJ) In accordance with Subsection III.C of SRP Section 14.2.1, industry operating experience is one consideration licensees may use to support an exception to certain EPU power ascension tests. The Applicant submitted information to the Staff citing both industry experience and Vermont Yankee plant-specific experience. The most relevant industry experience was that of Hatch Units 1 and 2 in 1998. In that case, the Staff granted an EPU without requiring the performance of large transient testing. Both Vermont Yankee and Hatch are BWR/4 designs with Mark I containments. Subsequent to its uprate, Hatch Unit 1 experienced a turbine trip in 2000 and a generator load reject event in 2001. Hatch Unit 2 experienced an unplanned event that resulted in a generator load reject from 98% of uprated power in 1999. These events produced no anomalies in the plant's response. This outcome supports the conclusion that extended power uprates at facilities of similar design are unlikely to produce new or unexpected phenomena in response to anticipated transients.

Q13. Do you agree with NEC's and Mr. Gundersen's assertion that Vermont Yankee's own experience with generator load rejections at 100% of the original licensed power level does not support the conclusion that there will be adequate plant performance during transients at EPU conditions?

A13. (RLP, SRJ, GT) No.

Q14. Please provide the basis for this conclusion.

A14. (RLP, SRJ, GT) In addition to the discussion of industry operating experience in A12 above, the licensee also cited plant-specific experience at Vermont Yankee, which included several generator load rejections from 100% of the original licensed thermal power.
A generator load rejection in 2004 occurred after many physical modifications supporting the power uprate, including modifications to the main turbine and main feedwater system, had been implemented. No significant anomalies were seen in the plant's response to these events. In

addition, the licensee stated that past transient and safety analyses correlate closely to results from actual transients. Thus, the Vermont Yankee operating experience supports the conclusion that the plant will respond as designed to transients at EPU conditions by demonstrating that many physical modifications supporting the uprate have not adversely affected the transient response and by validating analytical methods used to predict plant response with those modifications in place.

Q15. Do you agree with NEC's assertion that periodic testing of SSCs, during steady-state plant operation, does not confirm performance characteristics of SSCs required for appropriate transient response?

A15. (RBE, RLP, SRJ, GT) No. As described in the Staff's Final SE for the Vermont Yankee EPU, the purpose of the EPU test program is to demonstrate that SSCs will perform satisfactorily in service at the proposed EPU power level. The test program also provides additional assurance that the plant will continue to operate in accordance with design criteria at EPU conditions. Final SE at 260.

Technical Specification ("TS") surveillance testing of SSCs performed during steady-state conditions confirms performance of SSCs required for appropriate transient response. TS surveillance testing is conducted pursuant to 10 C.F.R. § 50.36 ("Technical Specifications"). Under that regulation, TSs are required to include items in the following five specific categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation ("LCOs"); (3) surveillance requirements; (4) design features; and (5) administrative controls.

As discussed in 10 C.F.R. § 50.36(c)(2)(ii), a TS LCO must be established for each item meeting one or more of the following criteria:

Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 4: A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Criteria 2 and 3 relate, in part, to functional performance of SSCs necessary to demonstrate that the plant response to transients is as assumed in the associated safety analyses. Consistent with the requirements in 10 C.F.R. § 50.36(c)(3), TS surveillance testing (*e.g.*, component testing, trip logic system testing, and simulated actuation testing) assures that TS LCOs are met. When an LCO is met, the associated SSC is considered to be operable. In this regard, the Vermont Yankee TSs define "operable" as follows:

> A system, subsystem, division, train, component or device shall be operable or have operability when it is capable of performing its specified safety function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal or emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

Vermont Yankee TSs at 2.

If a SSC is determined to be operable during TS surveillance testing, that determination provides assurance that the SSC is capable of performing its specified safety functions as assumed in the plant safety analysis. For example, the reactor protection system instrumentation that is relied on to mitigate large transients by providing a reactor scram (*i.e.*, MSIV closure, turbine control valve fast closure, and turbine stop valve closure) is tested quarterly, assuring it will carry out its safety function in the event of a large transient.

Based on the above, periodic testing of SSCs during steady-state plant operation can confirm performance characteristics of SSCs required for appropriate transient response. We therefore disagree with NEC's assertion concerning the periodic testing of SSCs during steady-state plant operation.

Q16. Administrative Judge Baratta has inquired as to how the calculations of mechanical stress on various components during a transient under EPU conditions were performed, and whether they account for stresses experienced during the transient. *See* Tr. 902, 903-04. Please address this question.

A16. (RBE) The Staff's SE for the CPPU topical report, dated March 31, 2003, discusses an acceptable methodology for evaluating the stresses on various components subject to increased loadings due to power uprate conditions for plants using the CPPU approach to power uprate (such as Vermont Yankee). Specifically, Section 3.2 of the CPPU SE discusses the reactor pressure vessel and its internals, and Section 3.4 discusses piping systems and the associated components (*e.g.*, nozzles, anchors, pumps, valves, supports). As described in those sections of the CPPU SE, the loads and load combinations are evaluated for normal operation, upset (*i.e.*, transient), emergency, and faulted conditions to determine if the calculated stresses and fatigue usage factors are less than the code allowable limits. For Vermont Yankee, specific details regarding how stresses were analyzed for the reactor pressure vessel and its internals (other than the steam dryer) are discussed in Section 2.2.3 of the Staff's Final SE. Specific details regarding how stresses were analyzed for the steam dryer are discussed in Section 2.2.6 of the Staff's Final SE. Section 2.2.2 of the Staff's Final SE discusses how stresses were analyzed for piping systems and components.

Q17. Please provide your views regarding Mr. Gundersen's assertion that "Entergy ignores the NRC Staff's decision in the case of the Duane Arnold EPU application."

A17. (RBE, RLP) As discussed above in the answer to Question 6, Mr. Gundersen's assertion is based on material quoted from an NRC Staff request for additional information ("RAI") that was sent to the licensee for the Duane Arnold plant on May 9, 2001. The RAI cites the NRC-approved General Electric licensing topical report (ELTR-1), which was referenced in the Duane Arnold EPU License Amendment Request. As described in A9 above, ELTR-1 includes certain conditions where an MSIV closure test and a generator load rejection test should be performed as part of the EPU test program.

As discussed above, the Vermont Yankee EPU is based on the GE CPPU topical report, which had stated on a generic basis that large transient tests were not necessary for EPUs using the CPPU approach. In the CPPU SE dated March 31, 2003, the NRC Staff did not accept the generic determination that no large transient testing was necessary, but specified that test programs for EPUs using the CPPU approach would be evaluated on a plant-specific basis. That is the approach followed by Vermont Yankee here. For the Vermont Yankee EPU amendment request, Entergy provided a plant-specific justification to not perform large transient testing in Attachment 7 of its Application, dated September 10, 2003, which Entergy subsequently updated in Supplement 3 to the Vermont Yankee EPU amendment request, dated October 28, 2003.

In summary, the Vermont Yankee EPU is based on the CPPU approach, pursuant to which BWR licensees may justify not performing large transient testing by providing certain plant-specific information. As part of its Application, Entergy provided plant-specific information to justify not performing these tests for Vermont Yankee. The approach of ELTR-1 with respect to this issue therefore does not apply, and NEC's reference to the Staff's review of the Duane Arnold EPU application, which was based upon the ELTR-1 topical report is irrelevant.

Q18. In his Declaration of December 21, 2005, Dr. Hopenfeld challenged Entergy's use of the ODYN code as part of its rationale for not performing large transient testing.

Specifically, Dr. Hopenfeld stated a concern regarding whether the ODYN code was properly benchmarked for the transients and plant conditions applicable to the Vermont Yankee EPU. What is the "ODYN code"?

A18. (ZA, GT) 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 of boiling water reactors. The ODYN model consists of an integrated one-dimensional reactor core model which is coupled to the recirculation and major system control models. The recirculation and control systems are integrated in the code to analyze the plant's response to fast pressurization transients. ODYN also models the steamline pressure wave response in the main steam line, which includes modeling of the main turbine control valves and bypass valves.

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).[6] ODYN is also approved for other anticipated operational occurrences ("AOO") events such loss of feedwater heating, pressure regulator failure - open, recirculation flow decrease events, recirculation flow increase events and increase in coolant inventory events.

Q19. Does Entergy's EPU application rely on analyses performed with ODYN as part of its justification for not performing large transient testing?

A19. (ZA, GT) Yes. As part of the NRC-approved standard reload process for BWRs, Vermont Yankee analyzed the limiting transients for each fuel cycle using ODYN.

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[6] *See* Letter from Robert L. Tedesco (NRC) to Dr. G. G. Sherwood (General Electric Co.), dated February 4, 1981, and enclosed "Safety Evaluation for the General Electric Topical Report Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors, NEDO-24154 and NEDE-24154-P" (June, 1980).

ODYN uses plant-specific core neutronic and thermal-hydraulic conditions as inputs. As part of its justification for not performing large transient testing, Entergy stated that the MSIV closure pressurization transient analysis (that bounds the load reject without bypass pressurization event) had been performed at Vermont Yankee for EPU conditions using the ODYN code. Entergy stated that the analyses assumed worse conditions than would be experienced during an actual transient. The results of the analyses showed that the response of the plant to this bounding transient to be acceptable.

Q20. How was the ODYN code qualified?

A20. (ZA, GT) The NRC Staff's approval of ODYN included evaluation of the performance of the code's analytical models by quantifying the accuracy of the code's predictions (*e.g.*, uncertainties) to be accounted for in the transient simulation. Some of the ODYN analytical models evaluated include: the recirculation loop model, the control systems model, the steam separator model, the upper plenum, vessel dome and bulkwater model, the steam line core thermal-hydraulic model (e.g., drift flux and mechanistic boiling), the core physics model, and the fuel heat transfer model.

The Staff compared specific models in ODYN against separate effects test data. The specific model assessment included code-to-code comparisons (e.g, ODYN thermal-hydraulic model against 3-D core simulator), ODYN comparisons to plant measurement data and separate effect test data. These assessments were used to establish the potential uncertainties and biases associated with the specific models in order to account for any potential under-predictions or conservatism in the code's simulation of plants' transient response. The Staff's assessment of ODYN also included comparisons of the code's predicted integral response against the integral test data (*e.g.*, three Peach Bottom Unit 2 ("PB-2") transient tests and one Muehleberg Nuclear Power Plant ("KKM") transient test), discussed below.

Finally, the Staff's assessment of ODYN included comparisons of its simulation of specific transients against the predictions of independent confirmatory analyses (BNL-TWGL and RELAP-3B). The confirmatory codes were benchmarked against the PB-2 transient test. The Staff evaluated differences between the PB-2 transient test results and the ODYN predictions. Based on the confirmatory analyses/ODYN code-to-code comparisons and the comparisons of ODYN predictions against the integral test data, the Staff quantified the uncertainties in ODYN's predictions that must be accounted for in the simulations of the plants' transients. The Staff found the use of ODYN acceptable for performing design bases transients, in a safety evaluation issued in 1981. *See* footnote 5 above. In November 1985, the Staff approved an updated version of ODYN that incorporated improvements in the specific models stemming from some of the differences observed in the PB-2 integral tests comparisons.[7]

Q21. Please describe the integral tests performed to validate ODYN.

A21. (ZA, GT): As stated above, integral tests were performed at Peach Bottom Unit 2 and at KKM, a Swiss BWR. The integral tests that were performed were as follows.

Peach Bottom 2 Integral Test: In April 1977, three integral tests were performed at PB-2. The PB-2 integral tests involved turbine trip transients with the turbine valve fast closure scram disabled. The tests were initiated from power levels of 47.4, 61.6 and 69.1 percent, and core flow rates of 100, 82.1 and 100 percent, respectively. Each transient test was initiated from a different control rod pattern and the results were compared against the axial power distribution shift in order to assess the one-dimensional nuclear model. One of the PB-2 tests included a control rod pattern selected to assess the ODYN model's capability to simulate the core wide radial power distribution effect. For each of the transient tests, the turbine stop valve

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[7] *See* Letter from Cecil O. Thomas (NRC) to J. S. Charnley (General Electric Co.), dated November 5, 1985 ("Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A, Rev. 6, Amendment 11, 'General Electric Standard Application for Reactor Fuel' (GESTARII))."

scram was disabled and the reactor scrammed on high neutron flux. This is a conservative test because the delayed scram results in higher power response in comparison to the plant power response for direct stop valve closure scram. This was done to obtain transient results comparable in severity to licensing analyses.

KKM Integral Test: Integral tests were also performed at the Muehleberg Nuclear Power Plant (KKM), a BWR located in Switzerland. KKM is smaller than Peach Bottom Unit 2, and has a unique steamline/turbine configuration. The KKM plant has two turbines and two sets of steamlines with a reheater line in each steamline. These differences require spatial modeling considerations for the ODYN simulations. Consequently, GE developed a special version of ODYN that models the KKM configuration. In addition, KKM differs from domestic BWRs in terms of measurement capability and actuation of the turbine stop valve and bypass. Again, the ODYN model and valve actuations were adjusted in order to simulate KKM valve actuations. The KKM turbine trip transient was initiated from 77% power at 86.5% core flow. The reactor was at end-of-cycle ("EOC"), all rods out, conditions. The KKM transient test resulted in milder pressurization response than the PB-2 tests; accordingly, most of the ODYN code validations that have been performed use the PB-2 tests.

Q22. Please describe the conclusions of the Staff's ODYN code assessment.

A22. (ZA, GT) The Staff compared the integral test results for key parameters against the ODYN predictions. In addition, the Staff evaluated the adequacy of the ODYN 1D Thermal-Hydraulic model against the integral tests by evaluating the Local Power Range Monitor (LPRM) flux reading and power distribution at a given axial location against the predictions. The change in critical power ratio ("ΔCPR") values predicted by ODYN for a given test were compared against the ΔCPR obtained in the integral tests by using the measured core parameters. The measured jet pump ΔP, measured pressure, and the measured power during the transient were used to predict the ΔCPR of the test.

The Staff found that the code demonstrates good prediction against existing test data obtained from separate effects and integral plant tests (*e.g.,* PB-2 and KKM). Comparisons of the PB-2 and KKM integral test data against the ODYN predictions indicate that the results are within the calculated uncertainties. The Staff found that the ΔCPR calculation from the ODYN code set is neither conservative nor non-conservative, but that it predicts the available data well and within the expected uncertainty range. Further, based on the Peach Bottom tests, the Staff determined that ODYN is a "best estimate" code for ΔCPR calculations.

Subsequent to the initial comprehensive assessment of the ODYN performance, GE incorporated improved analytical methods and revised specific models that provided input to ODYN. *See* footnote 6 above. The "improved" ODYN code set comparisons against the PB-2 tests yielded closer predictions to the test results than the original comparison. As specific input models are revised or improved, the fuel vendor has assessed the code against the original PB-2 test data in order to confirm that the code's performance is acceptable.

Q23. After its initial validation, was ODYN assessed against an EPU plant transient response?

A23. (ZA, GT) Yes. Several domestic BWRs that have implemented extended power uprates have experienced transient events; in addition, a foreign plant, Liebstat ("KKL") that had undergone an EPU performed large transient tests. In all transient events and tests at the EPU power levels, the plants responded as expected, without indicating any significant changes in the fidelity of the analytical models and codes at the EPU conditions. A review of these events was provided to the NRC by Exelon Generation Company, LLC ("Exelon"), in a letter supporting the EPU applications of the Dresden and Quad Cities plants, submitted in

May 2001.[8] The following discussion summarizes the information provided by Exelon in its letter of May 18, 2001.

Exelon indicated that Hatch Units 1 and 2 implemented an EPU that was 13% above the original licensed thermal power ("OLTP").[9] The licensee (Southern Nuclear) did not perform large transient testing. However, in 1999, while Hatch Unit 2 was operating at 98% of the uprated power level, it experienced a generator load reject transient event. The licensee compared the plant process and measurement data during the transient against ODYN predictions. The key parameters compared were reactor pressure, neutron flux, heat flux and changes in the reactor water level. Exelon concluded that for these key parameters that are important to the transient response, the recorded values were less than or equal to the values predicted by ODYN.

Exelon further indicated that Liebstat ("KKL"), a European BWR, also underwent transient testing as part of its uprate implementation plan. The plant was uprated in phases, with testing at the uprated conditions conducted: (1) in 1998, at 10.5% above OLTP, (2) in 1999, at 13% above OLTP, and in 2000 at 16.7% above OLTP. A turbine trip test was performed at 10.5% above OLTP. During the KKL testing, the following key parameters and system and actuation setpoint characteristics were monitored: reactor power, reactor vessel and turbine steam flow, reactor vessel and turbine pressure, effectiveness of the reactor recirculation runback, effectiveness of the selected rod patterns, and modified turbine control

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[8] Letter from R. M. Krich, Exelon Generation Co., LLC, to NRC, "Additional Testing Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station and Quad Cities Nuclear Power Station," RS-01-104 (May 18, 2001). This letter was cited in the Staff's approval of the Dresden power uprate applications. *See* "Safety Evaluation by the Office of [NRR] Related to Amendment No. 191 to Facility Operating License No. DPR-19, and Amendment No. 185 to Facility Operating License No. DPR-25, Exelon Generation Company, LLC, Dresden Nuclear Power Station Units 2 and 3, Dockets No. 50-237 and 50-249" (Mar. 2, 2006), at 90-98.

[9] Hatch Units 1 and 2 implemented a 5% stretch power uprate in 1995 and an 8% extended power uprate in 1998, therefore the total is approximately 13% above OLTP.

valve response characteristics. In its May 18, 2001 letter describing the results of these tests, Exelon reported a close match between the predicted ODYN calculations and the measured plant response.

In addition to the Hatch and KKL experience cited above, on January 30, 2004, after implementation of its EPU, Dresden Unit 3 experienced an inadvertent turbine trip event, with the plant operating at 95.4%. Exelon performed a comparison of the actual Dresden plant response to the ODYN predicted response for a large transient with similar initial conditions and equipment availability. In a July 2005 document (GE-NE-0000-041-1254-R0), Exelon compared the ODYN predicted plant response against the actual plant response values. Exelon stated that the predicted trends and timing of the ODYN response were consistent with the actual trends and timing experienced in the plant response for key plant parameters such the neutron flux, reactor peak pressure and reactor vessel level. Exelon concluded that the Dresden 3 turbine trip comparisons demonstrate that ODYN as used for reload licensing analyses for plants that have undergone an EPU will conservatively predict the overpressure and minimum critical power ratio response.

Although the Staff has not reviewed the benchmarking performed for the transients at Hatch and Liebstat (KKL) discussed in Exelon's report, a preliminary assessment of Exelon's July 2005 evaluation of the Dresden 3 turbine trip indicates that, overall, the ODYN predictions appear to be generally consistent with the timing and trends of the plants' instrumentation readings. Specifically, for the key parameters important in pressurization response, the ODYN predictions are consistent with measured data. Further, other EPU plants which were analyzed with ODYN, that experienced transient events, have responded as analyzed, indicating no significant change in the overall accuracy of the ODYN code. Therefore, comparisons of ODYN against plant data at EPU conditions provide reasonable assurance that use of the ODYN code will acceptably simulate plant response to limiting pressurization response, in terms of peak pressure and change in the MCPR.

Q24. Having reviewed the assertions presented by Mr. Gundersen in support of NEC Contention 3, have you reached a conclusion as to the issue of whether the Applicant has adequately justified not performing large transient testing at EPU conditions?

A24. Yes. As discussed above in A8, 10 C.F.R. Part 50, Appendix B, Criterion XI, requires a licensee to establish a written test program to demonstrate that SSCs will perform satisfactorily in service. In accordance with Criterion XI, the test program is required to include testing necessary to provide reasonable assurance that SSCs will perform satisfactorily in service following the EPU; an EPU test program, however, is not required to include the performance of any specific test. Entergy has provided acceptable information regarding its startup test program, and its relationship to the proposed EPU power ascension test program, which provides adequate justification for not performing the two large transient tests addressed in NEC Contention 3. Based upon our review of the contention and the bases offered in support thereof, and our review of the Vermont Yankee EPU application, and supplements thereof, we are satisfied that the Applicant has adequately justified not performing the MSIV closure test and generator load rejection test at Vermont Yankee under EPU conditions. Further, we have concluded that the Vermont Yankee EPU testing program satisfies the requirements of 10 C.F.R. Part 50, Appendix B, Criterion XI.

Q25. Does this conclude your testimony?

A25. Yes.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
ENTERGY NUCLEAR VERMONT YANKEE LLC and ENTERGY NUCLEAR OPERATIONS, INC.	)	Docket No. 50-271-OLA
	)	ASLBP No. 04-832-02-OLA
(Vermont Yankee Nuclear Power Station)	)	

AFFIDAVIT OF ZEYNAB ABDULLAHI
CONCERNING NEC CONTENTION 3

I, Zeynab (Zena) Abdullahi, being first duly sworn, do hereby aver that my statements in the foregoing "NRC Staff Testimony of Richard B. Ennis, Steven R. Jones, Robert L. Pettis, Jr., George Thomas, and Zeynab Abdullahi Concerning NEC Contention 3" are true and correct to the best of my knowledge, information and belief.

/Original Signed by/
____
Zeynab (Zena) Abdullahi

Subscribed and sworn to before me
this 17th day of May, 2006

/Original Signed By/
____
Notary Public

My commission expires: March 1, 2010

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
ENTERGY NUCLEAR VERMONT YANKEE LLC and ENTERGY NUCLEAR OPERATIONS, INC.	)	Docket No. 50-271-OLA
	)	ASLBP No. 04-832-02-OLA
	)	
(Vermont Yankee Nuclear Power Station)	)	

AFFIDAVIT OF RICHARD B. ENNIS
CONCERNING NEC CONTENTION 3

I, Richard B. Ennis, being first duly sworn, do hereby aver that my statements in the foregoing "NRC Staff Testimony of Richard B. Ennis, Steven R. Jones, Robert L. Pettis, Jr., George Thomas, and Zeynab Abdullahi Concerning NEC Contention 3" are true and correct to the best of my knowledge, information and belief.

/Original Signed By/

____
Richard B. Ennis

Subscribed and sworn to before me
this 17th day of May, 2006

/Original Signed By/
____
Notary Public

My commission expires: March 1, 2010

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
ENTERGY NUCLEAR VERMONT YANKEE	)	Docket No. 50-271-OLA
LLC and ENTERGY NUCLEAR	)	
OPERATIONS, INC.	)	ASLBP No. 04-832-02-OLA
	)	
(Vermont Yankee Nuclear Power Station)	)	

AFFIDAVIT OF STEVEN R. JONES
CONCERNING NEC CONTENTION 3

I, Steven R. Jones, being first duly sworn, do hereby aver that my statements in the foregoing "NRC Staff Testimony of Richard B. Ennis, Steven R. Jones, Robert L. Pettis, Jr., George Thomas, and Zeynab Abdullahi Concerning NEC Contention 3" are true and correct to the best of my knowledge, information and belief.

____
Steven R. Jones

Subscribed and sworn to before me
this 17th day of May, 2006

____
Notary Public

My commission expires: _____________

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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ENTERGY NUCLEAR VERMONT YANKEE LLC and ENTERGY NUCLEAR OPERATIONS, INC.	)	Docket No. 50-271-OLA
	)	ASLBP No. 04-832-02-OLA
(Vermont Yankee Nuclear Power Station)	)	

AFFIDAVIT OF ROBERT L. PETTIS, JR.
CONCERNING NEC CONTENTION 3

I, Robert L. Pettis, Jr., being first duly sworn, do hereby aver that my statements in the foregoing "NRC Staff Testimony of Richard B. Ennis, Steven R. Jones, Robert L. Pettis, Jr., George Thomas, and Zeynab Abdullahi Concerning NEC Contention 3" are true and correct to the best of my knowledge, information and belief.

/***Original Signed By***/
____
Robert L. Pettis, Jr.

Subscribed and sworn to before me
this 17th day of May, 2006

/Original Signed By/
____
Notary Public

My commission expires: March 1, 2010

UNITED STATES OF AMERICA
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ENTERGY NUCLEAR VERMONT YANKEE LLC and ENTERGY NUCLEAR OPERATIONS, INC.	)	Docket No. 50-271-OLA
	)	ASLBP No. 04-832-02-OLA
(Vermont Yankee Nuclear Power Station)	)	

AFFIDAVIT OF GEORGE THOMAS
CONCERNING NEC CONTENTION 3

I, George Thomas, being first duly sworn, do hereby aver that my statements in the foregoing "NRC Staff Testimony of Richard B. Ennis, Steven R. Jones, Robert L. Pettis, Jr., George Thomas, and Zeynab Abdullahi Concerning NEC Contention 3" are true and correct to the best of my knowledge, information and belief.

/Original Signed By/
____
George Thomas

Subscribed and sworn to before me
this 17th day of May, 2006

/Original Signed By/
____
Notary Public

My commission expires: March 1, 2010

Ms. Zena Abdullahi
Senior Reactor Systems Engineer
Nuclear Regulatory Commission
Nuclear Reactor Regulation

EXPERIENCE

SENIOR REACTOR SYSTEMS ENGINEER **May 1997 - Present**
BWR Systems Branch *Rockville, Maryland*
Division of System Safety

- Responsible for evaluating technical merit of license applications requesting changes to operation of Boiling Water Reactors, including extended power uprate and operation at expanded operating domains. Review the impact of the proposed changes to the design bases safety analyses supporting the plants operation during steady state, transient and accident conditions. Principle topics of responsibilities include: core and fuel performance, ECCS-LOCA, instability, ATWS, safety system performance, neutron monitoring system actuation and trip setpoints.

- Responsible for evaluating topical reports submitted by fuel vendors. Topical reports present the analytical methods (core physics, fuel behavior, core thermal-hydraulic) used to perform the safety analyses, or describe the generic guidelines and the scope of analysis that would be provided in plant-specific applications that will implement proposed changes to nuclear plants licensed operation (e.g., power uprates, changes in operating power/flow operating domain, cycle length extensions, single recirculation loop operation).

- Responsible for leading and/or participating in technical audits of the analytical methods and calculations supporting generic topical report or plant-specific licensing applications.

- Responsible for communicating technical issues and staff positions with both internal and external stakeholders (NRC, fuel vendors, Owners Groups, ACRS).

GENERAL ENGINEER **1995 - 97**
Nuclear Reactor Regulation *Rockville, Maryland*

- Completed two years of development training program which includes rotational assignments and technical training. Rotational assignments included Project Licensing Directorate *(e.g., licensing project management)*, License Renewal and Standardization, Mechanical

Engineering and Civil Engineering Branch *(e.g.,In-service inspection reviews)* and Office of Nuclear Material Safety and Safeguards-Office *(e.g.,Thermal analysis of Fuel transportation*). Rotated to Pilgrim Nuclear Power Station Site.

STRESS ANALYST **1987 - 88**
Bechtel Eastern Power *Gaithersburg, Maryland*
Mechanical and Processing Division

- Performed stress analysis for TVA's Watts Bar Power Plant.

- Analysis of the effects of pressure, deadweight, thermal expansion, thermal transients loads, thermal anchor movement, design basis accident inertia and movement loads, seismic anchor movements, seismic and hydrodynamic loading.

- Proposed, initiated and completed a set of special calculations for the project procedure manual (Design Code: ASME, Section III; TPIPE)

EDUCATION

M.S. MECHANICAL ENGINEERING (Fluid and Energy Systems) ***1995***
University of Maryland, College Park

- Graduate work included courses in computational fluid dynamics, compressible fluid flow, incompressible fluid flow, multi-phase flow and heat transfer, advanced topics of thermal science (computational two-phase flow), advanced convection heat transfer, advanced conduction and radiation heat transfer and combustion

- Independent research on “Comparisons of Predictions of Different Critical Power Correlations .” Conducted literature review on critical power flux phenomena. Modified CANAL, a program used in the analysis of fluid flow and heat transfer in the core of boiling water reactors; wrote subroutines that predict the occurrence dryout and locations using ten dryout correlations. Compared experimental data (e.g.,Oak Ridge National Laboratory) of dry out locations to the predicted values and locations for a given operating conditions.

B.S. Mechanical Engineering **1987**
University of California, Davis

NRC TRAINING

- Power Plant Engineering (E-100);
- GE Nuclear Technology (R-200B, R-304B, R-504B, and R-624B)
- Reactor Safety (R-800);
- GE Maintenance Overview (R-802),
- Containment Thermal-Hydraulic Review and Analytical Techniques,
- Probability and Statistics (P-102),
- PRA Basics for Regulatory Applications (P-105),
- Applied Statistics,
- RELAP-Novice User Workshop,
- Inspecting for Performance (G-303),
- Fundamentals of Inspection (G-101),
- OSHA Indoctrination (G-111),
- Site Access Training (H-100), M
- Motor Valve Actuators (E-112),
- NRC: What it is and What it Does,
- The Regulatory Process,
- Multi-phase Flow and Heat Transfer for Industrial Applications,
- Finite Element Analysis: Heat Transfer & Fluid Flow Applications:,
- Station Nuclear Engineer
- TRACE

RICHARD B. ENNIS
Statement of Professional Qualifications

CURRENT POSITION:

Senior Project Manager	Division of Operating Reactor Licensing, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Rockville, MD

EDUCATION:

B.S. in Electrical Engineering, Bucknell University, 1977

SUMMARY:

Over 28 years engineering experience in the commercial nuclear power industry. Significant experience in the following areas:

- Project Management
- Technical Writing
- Design & Licensing Basis Documentation
- License Renewal
- Nuclear Facilities Audits and Design Verifications
- Design Modifications
- Instrument Setpoint and Loop Uncertainty Calculations & Methodologies
- Software Development, Quality Assurance, and Verification & Validation

EXPERIENCE:

U.S. Nuclear Regulatory Commission, Project Manager, 1998 - Present

Project Manager in the Office of Nuclear Reactor Regulation. Serve as headquarters focal point for technical review coordination, information and communication on issues concerning assigned nuclear power plants. Responsibilities include coordination, review, and preparation of safety evaluations, environmental evaluations and other documentation to support the licensing activities for the plant. Also serve as lead project manager for special projects. Assignments have included the following:

- Lead Project Manager, Vermont Yankee Extended Power Uprate (10/05 - Present)
- Project Manager, Vermont Yankee Nuclear Power Station (12/03 - 10/05)
- Project Manager, Millstone Nuclear Power Station, Unit 2 (3/02 - 12/03)
- Project Manager, Hope Creek Generating Station (3/98 - 6/00, 11/00 - 3/02, 5/03 - 9/03)

- Lead Project Manager, Steam Generator Action Plan (11/00 - 6/01).
- Lead Project Manager, Indian Point Unit 2 Steam Generator Tube Failure Lessons-Learned Task Group (6/00 - 11/00).

<u>Scientech, Inc., Senior Engineer, 1997 - 1998</u>

Worked as a contractor for Baltimore Gas and Electric Company in the Calvert Cliffs Nuclear Power Plant (CCNPP) Life Cycle Management Group. Prepared technical reports for the CCNPP license renewal application to the NRC in accordance with 10 C.F.R. Part 54. Reports prepared for the Radiation Monitoring System, Chemical and Volume Control System, Saltwater System, Electrical Cables Commodity Evaluation, Instrument Lines Commodity Evaluation, Intake Structure, and Turbine Building.

<u>TENERA, Inc., Project Manager/Senior Engineer, 1988 - 1996</u>

Responsibilities included technical consulting, project management, budget and schedule control, marketing and business development, and preparation of proposals. Also served as corporate Configuration Control Manager (CCM) for development of computer software applications. CCM responsibilities included ensuring that software life cycle activities were implemented in accordance with quality assurance (QA) requirements. Managed and provided engineering support for numerous projects as described below.

- Commonwealth Edison Company - Managed and performed a license conformance review at the LaSalle plant that included developing plant licensing and design basis requirements from the UFSAR and reviewing these requirements against design documents and procedures (e.g., operations, maintenance, engineering) to ensure that the plant was operating within its design and licensing basis.
- Commonwealth Edison Company - Performed design basis verification for the Auxiliary Power System for Zion Station Units 1 and 2, and Standby Gas Treatment System for Dresden Station.
- Nebraska Public Power District - Authored the Reactor Protection System (RPS), Standby Liquid Control System, and Neutron Monitoring System design basis documents for Cooper Nuclear Station. Also performed design basis verification for the Reactor Protection, DC Electrical, Diesel Generator, Standby Liquid Control, Neutron Monitoring, and Control Rod Drive systems.
- Northern States Power Company - Performed reactor trip instrument setpoint calculations for Prairie Island Units 1 and 2.
- Northeast Utilities - Authored the RPS Equipment Coefficients Methodology for Millstone Unit 2. Also performed fuel reload analysis for fuel cycle 13.

- New York Power Authority, Consolidated Edison Company - Authored engineering evaluations and documents related to Electrical Separation for the FitzPatrick and Indian Point Unit 2 nuclear plants. Work included preparation of Electrical Separation Design Criteria documents, justifications for cable separation anomalies, review of cable and raceway installation standards, fault current analysis, and preparation of a training package.

- Philadelphia Electric Company - Authored the Regulatory Guide 1.97 Post-Accident Monitoring design basis documents for Limerick Generating Station and Peach Bottom Atomic Station.

- Florida Power and Light Company - Co-authored the RPS Equipment Coefficients Methodology for St. Lucie Unit 1. Also performed calculations to verify the methodology and performed fuel reload analysis for fuel cycles 12, 13, and 14.

- Portland General Electric Company - Performed audit of the setpoint control program for Trojan Nuclear Plant.

- Washington Public Power Supply System - Performed system review (mini-SSFI) of Process Radiation Monitoring System for WNP-2.

- Southern California Edison Company, Arizona Public Service Company, Baltimore Gas and Electric Company, Northern States Power Company, Wisconsin Public Service Corporation - Developed QA computer software applications for San Onofre Nuclear Generating Station, Palo Verde Nuclear Generating Station, Calvert Cliffs Nuclear Power Plant, Prairie Island Nuclear Generating Station, and Kewaunee Nuclear Power Plant. Software packages included instrument-related databases and reports, setpoint calculations, instrument calibration scaling, head correction calculations, and insulation resistance calculations. Work included full life cycle development of QA Verified and Validated (V&V) software applications in IBM PC DOS and Windows environments.

- Consolidated Edison Company - Prepared design modification package for Emergency Diesel Generator Building HVAC System for Indian Point Unit 2.

- System Energy Resources, Inc. - Performed FSAR review and updates for Grand Gulf Nuclear Power Station.

<u>Bechtel Power Corporation, 1977 - 1988</u>

Assignments were as follows:

- Instrument and Controls Group Leader and Electrical/Control Systems Deputy Supervisor, Davis-Besse Nuclear Power Station, Gaithersburg, MD (4/85 - 11/88). Supervised Electrical/Control Systems group (approximately 40 engineers). Coordinated and reviewed design work including revision and issue of the following types of documents: specifications, control board layouts, loop diagrams, instrument installation details, tubing isometrics, instrument index, setpoint index, P&ID's, electrical schematics, connection diagrams, safety evaluations and conceptual designs. Responsible for design and specification of instrumentation and controls equipment. Responsible for preparing schedules, man-hour estimates, and staffing requirements.

- Results Engineering Group Leader, Wolf Creek Generating Station, New Strawn, KS (1/83 - 4/85). Supervised instrument and controls engineers in Results Engineering group (approximately 10 engineers). Coordinated all work related to generation of Wolf Creek instrument calibration documents. Reviewed instrument calibration data and prepared setpoint calculations. Generated startup field reports and processed instrument change requests. Reviewed startup test procedures and test results and wrote engineering procedures. Coordinated with instrument and controls maintenance group and startup group to support component and system tests.

- Instrument and Controls Group Leader, Grand Gulf Nuclear Power Station, Gaithersburg, MD (1/81 - 1/83). Supervised instrument and controls engineers in systems group (approximately 6 engineers). Coordinated and reviewed design work including logic diagram, loop diagram, and P&ID revisions; instrument calibration data; and design changes to comply with new licensing requirements.

- Instrument and Controls Engineer, Grand Gulf Nuclear Power Station, Gaithersburg, MD (7/79 - 1/81). Designed logic, loop and level settings diagrams. Prepared instrument calibration data and wrote instrument purchase specifications and evaluated bids. Prepared stress and seismic calculations and resolved startup field reports and field change requests.

- Instrument and Controls Engineer, Davis-Besse Nuclear Power Station, Gaithersburg, MD (7/77 - 7/79). Designed logic diagrams and prepared control valve specifications. Completed valve data sheets and ran computer program for instrument index updating.

Steven R. Jones

Statement of Professional Qualifications

EXPERIENCE:

Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission

Acting Chief, Balance of Plant Branch: November 2004 - Present

Supervised the safety review of mechanical systems other than those directly associated with the nuclear steam supply system, which are referred to as “Balance-of-Plant” systems. In this capacity, I have supervised the NRC Staff’s technical review of Balance-of-Plant systems review activities related to operating reactor license amendment requests (e.g., power uprate license amendment requests), aging management program scope for license renewal, design certification of new reactor designs, and operating experience analysis and resolution of associated generic safety issues.

Senior Reactor Systems Engineer: August 2001 - Present
Reactor Systems Engineer: October 1990 - June 1997

Performed evaluations of significant changes in design or operational limits and other technical issues related to secondary safety systems at commercial nuclear power plants, with a focus on service water cooling systems, power conversion systems, compartment transient analysis, spent fuel storage, and control room habitability. Assessed system capability and potential system failure modes. Reviewed system design to verify compliance with NRC regulations, applicable regulatory guidance, and industry standards. Evaluated technical safety issues involving spent fuel cooling and other secondary safety systems, and presented briefs regarding resolution of these issues to NRC senior management, the NRC Chairman, and advisory committees. Evaluated research reports related to secondary safety systems and recommended direction for future research activities.

NRC Region I, U.S. Nuclear Regulatory Commission

Senior Resident Inspector / Resident Inspector: June 1997 - July 2001

Planned and led implementation of the resident inspection program at Millstone Unit 2 under the revised Reactor Oversight Program. Monitored plant management performance and the conduct of operational, maintenance, and engineering activities at the unit with respect to the maintenance of reactor safety and compliance with NRC regulations. Evaluated the capability of important structures, systems, and components to perform their functions under limiting design conditions, based on mechanical design, fluid dynamics, heat transfer, electrical circuit analysis, control systems, and other technical considerations. Verified that the physical condition, maintenance practices, and operating procedures were consistent with maintaining the reliability of associated structures, systems, and components in performing their design functions. Used knowledge of risk analysis and the NRC's Significance Determination Process to evaluate several inspection findings involving degraded performance of essential mitigating systems. Analyzed the causes of degraded conditions to develop meaningful assessments of plant management performance and corrective action program effectiveness. Developed written reports to document technical issues and NRC performance assessments.

United States Navy

Nuclear Power Trained Submarine Officer: 1984 - 1989

Responsible for nuclear propulsion plant operations on board nuclear-powered submarine USS Simon Bolivar (SSBN-641). Developed an excellent understanding of design principles and operational characteristics of systems supporting submarine operations and systems associated with naval pressurized water reactors. Utilized principles of system design and operating characteristics to effectively execute the ship's operational mission and ensure safety during maintenance and testing activities. Enforced high standards of safety and workmanship during maintenance and repair periods through frequent inspection.

EDUCATION:

B.S., Marine Engineering, 1984
U. S. Naval Academy, Annapolis, MD

Graduate Studies in Mechanical Engineering, 1992 – 93
University of Maryland, College Park, MD

QUALIFICATIONS AND TRAINING:

Qualified NRC Operations Inspector (PWR), 1998
Qualified Submarine Officer, U. S. Navy, 1989
Qualified Engineering Officer of the Watch/Engineering Duty Officer, U. S. Navy, 1987

Training Courses:

- Westinghouse Technology (full series)
- Combustion Engineering Technology (cross-training series)
- General Electric Technology (short course)
- PRA Technology and Regulatory Perspectives (P-111)
- Perspectives on Reactor Safety (R-800)
- Root Cause/Incident Investigation Workshop (G-205)
- Reactor Inspection and Oversight Program (G-200)
- PRA Basics for Regulatory Applications (P-105)

Inspecting for Performance (G-303)
Fundamentals of Inspection (G-101)

CERTIFICATES AND LICENSES:

Licensed Professional Engineer (Mechanical): Maryland, 1996

ROBERT L. PETTIS, JR., P.E.

Statement of Professional Qualifications

CURRENT POSITION:

Senior Reactor Engineer
Division of Engineering
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Rockville, MD

EDUCATION:

B.S. in Civil Engineering, Northeastern University, 1975
M.S. in Civil Engineering (Structural Major), Northeastern University, 1977

PROFESSIONAL:

- Registered Professional Engineer (Maryland, California, and Massachusetts).
- Former Part-time Faculty Member, California State University (teaching undergraduate civil and structural engineering courses).

SUMMARY:

Over 30 years engineering experience in the commercial nuclear power industry. Significant experience in the following areas:

- Engineering management
- Technical writing
- License renewal reviews and audits
- Nuclear facilities audits, inspections, and design verifications
- Structural engineering and design
- Software quality assurance, verification and validation
- Extended power uprate reviews
- Professional engineer review of ASME Class I component supports

EXPERIENCE:

U.S. Nuclear Regulatory Commission, Staff Engineer, 1984 - Present

Reactor Engineer/Senior Reactor Engineer in the Office of Nuclear Reactor Regulation (NRR). Initially assigned to the Vendor Inspection Branch of NRR, where I was responsible for leading multi-discipline engineering team inspections at nuclear vendor, NSSS, and licensee facilities. Inspection areas included quality assurance compliance to 10 C.F.R. Part 50, Appendix B, and 10 C.F.R. Part 21; licensee procurement and dedication; inspections in support of allegations; and regional initiated inspection requests. For the past several years, my responsibilities primarily included leading on-site audits of licensee scoping and screening programs in support of license renewal activities; extended power uprate reviews of licensee power ascension and testing programs; and required presentations before ACRS.

As part of my responsibilities, I was also involved with the large transient testing issue in the NRC staff's review of the General Electric (GE) Licensing Topical Reports (LTRs), including review of the Constant Pressure Power Uprate (CPPU) LTR report, and I prepared a section of the staff's SE for these submittals. Additional EPU experience was also gained from previous reviews of EPU applications performed prior to the staff's development of a new Standard Review Plan (SRP). I co-authored SRP Section 14.2.1,"Generic Guidelines For Extended Power Uprate Testing Programs," which provides staff guidance on evaluating a licensee's EPU application in relation to the original startup testing performed at the plant under review.

These reviews were performed in accordance with the staff-approved GE LTR NEDC-32424P-A, "General Guidelines for General Electric (GE) Boiling Water Reactor (BWR) Extended Power Uprate," (known as "ELTR-1"). Section 5.11.9 of ELTR-1, "Power Uprate Testing," was the first document to establish the guidelines for large transient testing for GE BWRs. I was also the primary staff presenter to the ACRS for the large transient testing issue associated with the Clinton nuclear power plant's EPU, which utilized the GE CPPU approach, and I presented the staff's draft SE results for the large transient testing issue to the ACRS, at the VYNPS EPU public meeting held in Vermont in November 2005.

ITT Grinnell Corporation, Regional Engineering Manager, 1979 - 1984

Regional Engineering Manager for the Engineered Piping Products Group of ITT Grinnell, located in Huntington Beach, California. Responsible for initial establishment and location of the office, budget, lease negotiating, staffing, and training of over 20 engineers engaged in the preparation of structural pipe support designs for nuclear facilities. Reported to the Vice President of Operations located in headquarters in Providence, RI.

Stone & Webster Engineering Corporation, 1972 - 1977

Performed various assignments within the civil and structural engineering departments of the Boston Design Division while on co-op from Northeastern University and later full-time.

Responsibilities included technical drafting, project management, and preparation of engineering calculations for numerous nuclear power plants designed by Stone & Webster.

GEORGE THOMAS
REACTOR ENGINEER (NUCLEAR)
BWR SYSTEMS BRANCH
DIVISION OF SAFETY SYSTEMS
OFFICE OF NUCLEAR REACTOR REGULATION

GENERAL BACKGROUND

I have a total of 37 years of nuclear power plant experience related to boiling water reactors (BWRs), of which five years are in reactor operations. My experience has included a broad range of functions related to the design, engineering, testing, operations, and evaluation of nuclear plant systems. I performed construction tests, pre-operational tests and normal operations of the plant while working as an operator at Tarapur, a BWR built by General Electric (GE) and Bechtel in India (1967-1972). I was engaged in the design and engineering of reactor and component systems for a BWR while working with Stone & Webster Engineering Corp (1975-1980). While employed at United Engineers & Constructors (1973-1975), I wrote test procedures and system descriptions for a BWR.

NRC EXPERIENCE

Since 1980, I have served as a senior reviewer in the area of reactor systems for Boiling Water Reactors, in the BWR Systems Branch of the Division of Safety Systems, Office of Nuclear Reactor Regulation (NRR), U.S. Nuclear Regulatory Commission (NRC), in Rockville, MD. This involves reviews and evaluations of operating reactor licensing actions including power uprates, license renewals, and the Advanced Boiling Water Reactor (ABWR) and Economical Simplified Boiling Water Reactor (ESBWR) reviews for design certification. I also perform evaluations of multi-plant licensing actions and generic issues in the BWR Systems Branch`s area of responsibility. As part of my duties, I provide expert technical advice, consultation, and recommendations within the BWR Systems Branch`s area of review responsibility to other NRR branches, NRC offices, and NRC regional offices.

Among my responsibilities at the NRC, I was the lead reviewer for the extended power uprate review of the Clinton nuclear power plant, which the NRC approved in 2002. I also participated in the NRC Staff's power uprate review of the Brunswick nuclear power plant. The scope of my review of the Vermont Yankee power uprate included the functional design of the Control Rod Drive System, the Standby Liquid Control System, and transient and accident analyses.

PREVIOUS EMPLOYMENT

From 1975 to 1980, I was a Systems Engineer in the power division of Stone & Webster Engineering Corp. In that capacity, I performed detailed engineering and design of reactor systems of a BWR. My duties included project interface and coordinating work with the nuclear steam system supplier (NSSS) (GE) and the electric utility company.

From 1973 to 1975, I was employed by United Engineers & Constructors (UE&C), in Philadelphia, PA. Initially, I was a Test & Start-Up Engineer in the UE&C Construction Division. In this capacity, I wrote various procedures and systems descriptions for a BWR. Subsequently, I worked as a staff Engineer on the UE&C Nuclear Technical staff. In that capacity, I was engaged in providing technical expertise and consultation services to all nuclear projects of UE&C.

From 1967 to 1972 I served as a Reactor Operator at the Indian Atomic Energy Commission's first commercial nuclear power station, Tarapur 1 & 2 (a BWR built by Bechtel and GE). There, I participated in construction tests, pre-operational tests and normal operations of the station.

EDUCATION

I received a Bachelor of Science degree in Physics from Kerala (India) University in 1963. I also took graduate and professional courses in Nuclear Engineering at the University of Pennsylvania and the Engineers Club, in Philadelphia, PA, in 1975.

Other educational background and training included the following courses:

Perspectives on Reactor Safety - 2000
GE Nuclear Engineering Course - 1999
PRA Basics for Regulatory Applications - 1998
GE BWR/4 Technology Review - 1997
Power Plant Engineering - 1976
PWR technology course - 1980 (NRC sponsored)
BWR/6 simulator course - 1981(NRC sponsored)
Reactor operators training program (Tarapur Atomic Power Station, India) - 1969