September 6, 2005

Bill Eaton, BWRVIP Chairman
Entergy Operations, Inc.
Echelon One
1340 Echelon Parkway
Jackson, MS 39213-8202

SUBJECT: REVISED REQUEST FOR ADDITIONAL INFORMATION - BWRVIP-108: BWR VESSEL AND INTERNALS PROJECT TECHNICAL BASIS FOR THE REDUCTION OF INSPECTION REQUIREMENTS FOR THE BOILING WATER REACTOR NOZZLE-TO-VESSEL SHELL WELDS AND NOZZLE BLEND RADII

Dear Mr. Eaton:

By letter dated November 25, 2002, you submitted for NRC staff review, Electric Power Research Institute (EPRI) proprietary report, BWRVIP-108, “BWR Vessel and Internals Project Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii.” By letter dated May 17, 2005, the staff issued a request for additional information (RAI) regarding its review of the BWRVIP-108 report. The staff has since determined that a revision to RAI-2 is needed. Therefore, the staff is issuing a revised RAI. However, please note that the only change that was made to the May 17, 2005, RAI is with respect to RAI-2. No changes were made to RAI-1 and RAI-3.

The BWRVIP-108 report presents the technical basis to reduce inspection requirements to boiling water reactor (BWR) pressure vessel (PV) nozzle-to-shell welds and nozzle blend radii. Currently, BWR PV nozzle-to-shell welds and nozzle blend radii are inspected per American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI requirements, which require 100% inspection each 10-year interval. The purpose of the BWRVIP-108 report is to provide the technical basis for reducing the inspection requirements for the nozzle-to-shell welds and nozzle blend radii to 25% of the nozzles every 10 years. The 25% coverage refers to 25% of the nozzles for each nozzle type, e.g., 1 of 4 main steam nozzles would be inspected.

The staff has determined that additional information is needed to complete the review. The staff’s revised RAI regarding the BWRVIP-108 report is attached. It should be noted that these questions were raised after the staff had reviewed the Tennessee Valley Authority (TVA) and BWRVIP’s responses to the staff’s RAI, dated November 15, 2004, regarding TVA’s relief request dated July 25, 2003, on the reduction of inspection frequency for the Browns Ferry, Units 2 and 3 nozzle-to-vessel shell welds and nozzle blend radii.

In order to complete the staff's review of the BWRVIP-108 report in an efficient and effective manner, your complete response to the attached RAI is required no later than six months from the date of this letter. If you cannot provide a complete response within six months, please contact Meena Khanna at (301) 415-2150 to discuss the withdrawal of the BWRVIP-108 report and its future resubmittal when you are prepared to respond to the RAI. In addition, if you have any other questions regarding the attached RAI, please contact Ms. Khanna.

Sincerely,

/(RA by M. Mitchell)/

Matthew A. Mitchell, Chief
Vessels & Internals Integrity and Welding Section
Materials and Chemical Engineering Branch
Division of Engineering
Office of Nuclear Reactor Regulation

Project No. 704

Enclosure: As stated

cc: BWRVIP Service List

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cc:

Tom Mulford, EPRI BWRVIP
Integration Manager
Raj Pathania, EPRI BWRVIP
Mitigation Manager
Ken Wolfe, EPRI BWRVIP
Repair Manager
Larry Steinert, EPRI BWRVIP
Electric Power Research Institute
P.O. Box 10412
3412 Hillview Ave.
Palo Alto, CA 94303

Robert Carter, EPRI BWRVIP
Assessment Manager
EPRI NDE Center
P.O. Box 217097
1300 W. T. Harris Blvd.
Charlotte, NC 28221

George Inch, Technical Chairman
BWRVIP Assessment Committee
Constellation Nuclear
Nine Mile Point Nuclear Station (M/S ESB-1)
348 Lake Road
Lycoming, NY 13093

Denver Atwood, Chairman
BWRVIP Repair Focus Group
Southern Nuclear Operating Co.
Post Office Box 1295
40 Inverness Center Parkway (M/S B031)
Birmingham, AL 35242-4809

Jeff Goldstein, Technical Chairman
BWRVIP Mitigation Committee
Entergy Nuclear NE
440 Hamilton Ave. (M/S K-WPO-11c)
White Plains, NY 10601

Charles J. Wirtz, Chairman
BWRVIP Inspection Focus Group
FirstEnergy Corp.
Perry Nuclear Power Plant (M/S A250)
10 Center Road
Perry, OH 44081

Amir Shahkarami
BWRVIP Executive Oversight Committee Member
Exelon Corporation
4300 Winfield Road
Warrenville, IL 60555-4012

Robin Dyle, Technical Chairman
BWRVIP Integration Committee
Southern Nuclear Operating Co.
42 Inverness Center Parkway (M/S B234)
Birmingham, AL 35242-4809

Al Wrape, Executive Chairman
BWRVIP Assessment Committee
PPL Susquehanna, LLC
2 N. 9th St.
Allentown, PA 18101-1139

H. Lewis Sumner, Executive Chairman
BWRVIP Mitigation Committee
Vice President, Hatch Project
Southern Nuclear Operating Co.
M/S BIN B051, P.O. BOX 1295
40 Inverness Center Parkway
Birmingham, AL 35242-4809

Rick Libra
BWRVIP Executive Oversight Committee Member
DTE Energy
6400 N Dixie Hwy (M/S 280 OBA)
Newport, MI 48166-9726

REVISED REQUEST FOR ADDITIONAL INFORMATION REGARDING BWRVIP-108: BWR VESSEL AND INTERNALS PROJECT TECHNICAL BASIS FOR THE REDUCTION OF INSPECTION REQUIREMENTS FOR THE BOILING WATER REACTOR NOZZLE-TO-VESSEL SHELL WELDS AND NOZZLE BLEND RADII

RAI-1. BWRVIP Response to NRC RAI 2-9b: In your probabilistic fracture mechanics (PFM) analysis, which supports the relief requests for Browns Ferry, Units 2 and 3 to use the reduced inspection frequency for the boiling water reactor (BWR) nozzle-to-vessel shell welds and nozzle blend radii, as recommended in the BWRVIP-108 report, the mean stress corrosion cracking (SCC) initiation time is assumed to be the same as that indicated in the BWRVIP-05 report, “BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations,” which uses an initiation time of five times that of the curve that is applied to non-susceptible materials. The staff’s independent analysis, as documented in the final Safety Evaluation (SE) on the BWRVIP-05 report, dated July 28, 1998, did not consider the reactor pressure vessel (RPV) cladding SCC initiation time (i.e., zero SCC initiation time). Therefore, the staff did not take a position on that issue in 1998. Since an initiation time of five times that of the curve, which is being applied to non-susceptible materials, is judgmental, the staff requests that the BWRVIP assess the impact of using the mean curve for cast austenitic stainless steel in BWR plants, (i.e., one times that of the BWRVIP-05 curve which is being applied to non-susceptible materials) on your PFM results. The staff requests that the BWRVIP continue the simulation until several failures have been observed. The staff would like to evaluate the probability of failure, based on simulated vessel failures, so that the PFM results could be adjusted, should it become necessary.

RAI-2. BWRVIP Response to NRC RAI 2-9e: The response states that the flaw size distribution is based on the pressure vessel research users facility (PVRUF) data, and the number of flaws in the weld is based the Marshall distribution. The flaws in the PVRUF data originate from steelmaking, hot rolling/forging RPV plates, welding the plates together, and heat treating the assembly. The PVRUF data does not include flaws that originate as a result of inservice/operating conditions.

The inservice inspections of the RPV nozzle-to-shell welds and RPV nozzle inner radii provide assurance that no degradation mechanism are diminishing the integrity of the RPV.

RPV NOZZLE INNER RADIUS EXAMINATIONS

Early in 1980, operating conditions generated thermal fatigue cracks in BWR feedwater nozzle inner radii. The ultrasonic testing (UT) employed by the industry for the inner radii, at that time, was ineffective in detecting the thermal fatigue cracks.
The effectiveness of UT examinations of the nozzle inner radii steadily improved.
On June 5, 1998, the staff issued a safety evaluation on the Boiling Water Reactor Owners’ Group report GE-NE-523-A71-0594, “Alternate BWR Feedwater Nozzle Inspection Requirements,” which included demonstrating the detectability of a 1/4-inch notch and based the examination frequency on using techniques qualified to the objectives of Appendix VIII of Section XI of the American Society of Mechanical Engineers (ASME) Code. In December 1995, the ASME approved Code Case N-552,

“Qualification for Nozzle Inside Radius Section from the Outside Surface,” which has performance-based acceptance criteria. ASME Code Case N-552 was endorsed by the NRC in the *Federal Register* (64*FR*183) on September 22, 1999. The reliability and effectiveness of UT examinations of the RPV nozzle inner radii is assured by the performance-based qualification process of Supplement 5 to Appendix VIII of Section XI of the ASME Code or the alternative performance-based qualification process of Code Case N-552. Examinations of the RPV nozzle inner radius to the requirements of Appendix VIII, Supplement 5 became mandatory on November 22, 2002.

The staff is requesting a listing of the separate systems and nominal pipe sizes (the listing is the same as that required by Code Case N-702, “Alternative Requirements for BWR Nozzle Inner Radius and Nozzle-to-Shell Welds“) that apply to the RPV nozzle inner radii examination. For each item on the list, provide the total number of RPV nozzle inner radii (for all 35 BWR units). Also, for each item on the list, provide the total number of UT examinations performed using procedures that demonstrate the reliability of these examinations with respect to detection of cracks of various dimensions and locations, and which were performed by personnel that demonstrated their skills in reliably identifying these cracks. The establishment of UT reliability can be accomplished with statistically analyzed data from blind performance demonstrations, such as, Supplement 5 to Appendix VIII, “spirit” of Supplement 5 to Appendix VIII, and Code Case N-552.

<u>RPV-TO-NOZZLE WELD EXAMINATION</u>

In the late 1970s, the Program for the Inspection of Steel Components (PISC) was organized to study the uncertainties in reliability and effectiveness of UT examinations of thick steel components, such as RPVs. In February 1983, the NRC issued Regulatory Guide 1.150, “Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice examinations,” which adapted selected recommendations from the PISC program. Since 1995, personnel evaluated the applicability of Appendix VIII, Supplements 4 and 6 of ASME Code Section XI. Some licensees submitted relief requests to apply a combination of Supplements 4 and 6 as an alternative examination to the ASME Code-required (Section V) RPV-to-nozzle examinations.
By November 22, 2000, when NRC rulemaking made Supplements 4 and 6 mandatory, a sufficient number of personnel qualified to Supplements 4 and 6 were available to handle the industry’s inservice inspection needs. The major differences between combined Supplement 4 and 6 RPV examinations and RPV-to-nozzle weld examinations are the limitations caused by the proximity of nozzle configurations to the RPV-to-nozzle welds. The ASME Code developed Appendix VIII, Supplement 7 to qualify personnel and procedures for RPV-to-nozzle weld examinations. On November 22, 2002, NRC rulemaking made compliance with Appendix VIII, Supplement 7 examinations mandatory.

The staff is requesting a listing of the separate systems and nominal pipe sizes (the listing is the same as that required by Code Case N-702) that apply to the RPV-to-nozzle weld examinations. For each item on the list, provide the total number of RPV-to-nozzle welds (for all 35 BWR units). Also, for each item on the list, provide the total number of UT examinations performed using procedures that demonstrated reliability of these examinations with respect to detection of cracks of various dimensions and locations, and which were performed by personnel that demonstrated their skills in reliably identifying these cracks. The establishment of UT reliability can be accomplished with statistically analyzed data from blind performance demonstrations, such as, Appendix VIII, Supplement 7 qualified personnel and procedures and staff-approved (via the relief request process) Supplements 4 and 6 qualified personnel and procedures.

Since many of these examinations achieved less than essentially 100% coverage, identify on the list the number of RPV-to-nozzle weld examinations with less than essentially 100% coverage in ranges, i.e., 0 to 25%, 25% to 50%, 50% to 75%, 75% to less than 90%, and 90% and greater.

RAI-3. <u>BWRVIP Response to NRC RAI 2-10:</u> In the RAI, the staff describes a PFM analysis using the PFM parameters that were employed in the staff's independent PFM evaluations related to the BWRVIP-05 review (Interim SE dated August 14, 1997; SE dated July 28, 1998) and the Palisades pressurized thermal shock (PTS) review (SE dated April 12, 1995) as the "worst case" study. The staff's wording of "worst case" did not appropriately characterize the nature of the study that assesses the sensitivity of key PFM parameters. The staff requests that the BWRVIP assess the sensitivity of the following PFM parameters:

C Parameter (b), the number of flaws per nozzle for nozzle blend radius, which is based on the SE for the Palisades PTS review, where a flaw density for plates was assumed to be 1/10 of that of the weld;

C Parameters (c) and (d), the standard deviation for fracture toughness and upper-shelf fracture toughness of 15% of the mean values, which are those used by the staff in both the BWRVIP-05 review and the Palisades PTS review;

C Parameter (e), the SCC growth rate, approximately 10 times your rate, which is based on the staff's BWRVIP-05 review;

C Parameter (f), threshold stress intensity factor of 10 ksi / in, which is based on current industry effort in characterizing another type of SCC.