February 15, 2012

Mr. David Czufin, Chairman
Exelon Generation Company, LLC.
Chairman, BWR Vessel and Internals Project
Electric Power Research Institute
3420 Hillview Avenue
Palo Alto, CA 94304-1395

SUBJECT: ACCEPTANCE FOR REVIEW AND REQUEST FOR ADDITIONAL INFORMATION FOR "BWRVIP-241: BWR VESSEL AND INTERNALS PROJECT, PROBABILISTIC FRACTURE MECHANICS EVALUATION FOR THE BOILING WATER REACTOR NOZZLE-TO-VESSEL SHELL WELDS AND NOZZLE BLEND RADII" (TAC NO. ME6328)

Dear Mr. Czufin:

By letter dated April 26, 2011 (Agencywide Documents Access and Management System under Accession Number ML11119A041), the Boiling Water Reactor Vessel and Internals Project (BWRVIP) submitted the Electric Power Research Institute (EPRI) Technical Report (TR) 1021005, "BWRVIP-241: BWR Vessel and Internals Project, Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii'" to the U.S. Nuclear Regulatory Commission (NRC) staff for review and approval.

The NRC staff has performed an acceptance review of the subject TR and found that the material presented provides the technical information in sufficient detail to enable the NRC staff to complete a detailed technical review. The NRC staff estimates that the review will require approximately 150-200 NRC staff hours including project management time. The estimated review costs were discussed and agreed upon in a telephone conference between Mr. Bob Carter of EPRI and the NRC staff on February 8, 2012.

Upon review of the provided information, the NRC staff determined that additional information is needed to complete the review. During the above conference call, Mr. Carter also agreed to provide your response to the enclosed Request for Additional Information (RAI) questions within six months from the date of this letter.

If you have questions regarding this matter, please contact Andrew Hon at (301) 415-8480 or via e-mail at Andrew.Hon@nrc.gov.

Sincerely,

/RA/

John R. Jolicoeur, Chief
Licensing Processes Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 704

Enclosure:
As stated

If you have questions regarding this matter, please contact Andrew Hon at (301) 415-8480 or via e-mail at Andrew.Hon@nrc.gov.

Sincerely,

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REQUEST FOR ADDITIONAL INFORMATION

BOILING WATER REACTOR VESSEL AND INTERNALS PROJECT

"BWRVIP-241: BWR VESSEL AND INTERNALS PROJECT,

PROBABILISTIC FRACTURE MECHANICS EVALUATION FOR THE BOILING

WATER REACTOR NOZZLE-TO-VESSEL SHELL WELDS AND NOZZLE BLEND RADII"

TAC NO. ME6328

RAI-1

The finite element stress analysis results for recirculation inlet and outlet nozzles are documented in Section 4 of this Technical Report (TR). The NRC staff compared the nozzle stress pattern and magnitude from this TR with those from the BWRVIP-108 TR, "BWR Vessel and Internals Project, Technical Basis for the Reduction of Inspection Requirements for the [BWR] Nozzle-to-Vessel Shell Welds and Nozzle Inner Radii." The NRC staff found that under the heatup transient, the Pilgrim recirculation inlet nozzle (Figure 4-9), the Nine Mile Point, Unit 1, recirculation inlet nozzle (Figure 4-21), the Browns Ferry, Unit 2, recirculation outlet nozzle (Figure 4-33), and the Columbia recirculation outlet nozzle (Figure 4-45), all showed high compressive stresses at the inside surface when the steady state is approached. Page 4-3 of the TR provided explanation: "[t]his is due to the difference in thermal coefficient of expansion between the stainless steel clad and the low alloy steel nozzle material." However, the nozzle stresses reported in BWRVIP-108 for recirculation inlet and outlet nozzles showed no or insignificant compressive stresses at the inside surface under the same heatup transient. Please explain the discrepancies for inside surface stresses between these two reports, or confirm that they are caused by the BWRVIP-108 report's not considering clad in the nozzle stress analyses.

In addition, the trend of the through-thickness stress distributions at the nozzle blend radius region and the nozzle-to-shell weld region due to pressure for the Pilgrim recirculation inlet nozzle (Figure 4-8 of the TR; decreasing from the inside surface) is opposite to the trend for the Nine Mile Point, Unit 1, recirculation inlet nozzle (Figure 4-19 of the TR; increasing from the inside surface). Please provide insight or clarification to demonstrate that the opposite trend is not due to finite element modeling errors.

RAI-2

For the Browns Ferry, Unit 2, recirculation outlet nozzle under the loss of feedwater pump transient, the maximum through-thickness hoop stress shape of the nozzle blend radius region (rapidly increasing and then rapidly decreasing) occurred sooner, at 6,710 seconds, than the minimum through-thickness hoop stress shape (rapidly increasing and then slowly increasing), at the 19,336 second (Figure 4-34(a) of the TR). This is opposite to the corresponding hoop stress distribution for the Columbia recirculation outlet nozzle under the loss of feedwater pump transient (Figure 4-47(a) of the TR), where the maximum through-thickness hoop stress shape (rapidly increasing and then rapidly decreasing) occurred later, at the 4,560 sec, than the minimum through-thickness hoop stress shape at the 904 sec. Please provide insight or clarification to demonstrate that the opposite trend is not due to finite element modeling errors.

RAI-3

Figure 4-10 of the TR illustrated the axial and hoop stress distributions across the reactor pressure vessel (RPV) thickness along the nozzle-to-shell weld region, caused by sudden pump start of cold recirculation loop. The axial and hoop stress distributions at t = 2.136 seconds show significant compressive stresses close to inside surface of the nozzle. Please point out the time corresponding to t = 2.136 seconds in the temperature-time plot of the sudden pump start of cold recirculation loop transient shown in Figure 4-5 (b). It appears that at t = 2.136 seconds, the RPV and nozzle inside surfaces are still under the effect of cooldown from 522 °F to 130 °F and are not likely to produce compressive stresses. Please explain this observation.

RAI-4

Figure 4-21 of the TR illustrated the hoop and axial stress distribution across the RPV thickness along Path 1 (nozzle-to-shell weld) and Path 2 (nozzle blend radius) of the recirculation inlet nozzle (see Figure 4-18) of a plant under the heatup transient. Please provide the direction of the axial stress for both paths and confirm that the direction of the hoop stress is perpendicular to the paper of Figure 4-18. Provide an explanation for the sharp change of slope of the hoop stress distribution at about 0.5 inch from the inside surface in Figure 4-21(a). Please discuss whether this explanation can be applied to other nozzles showing similar stress distributions.

RAI-5

Regarding the probabilistic fraction mechanic (PFM) methodology, Section 5.1 of the TR states, “A modified version, VIPER-NOZ [21] was developed to include the evaluation capability for nozzles.” This suggests that the supplemental PFM analyses reported in this report were performed using the VIPER-NOZ Code. Please identify the part(s) of the VIPER-NOZ Code methodology which goes beyond what were accepted in the safety evaluation (SE) for the BWRVIP-108 report and, therefore, needs NRC staff review. Please also justify the inputs (except for plant-specific geometries and thermal loading) which go beyond what were accepted in the SE for the BWRVIP-108 report.

RAI-6

For fatigue crack growth (FCG) analyses, it appears that the approach in the TR is more conservative than that in the BWRVIP-108 report for each of the four nozzles, as indicated in a statement of Section 5.3.1, “For conservatism, all thermal cycles, except the heatup transient are lumped as the sudden pump start on cold recirculation for FCG analysis.” Similar statements can be found under Sections 5.3.2 to 5.3.4. Please confirm that the approach in the TR is more conservative than that in the BWRVIP-108 report. Please also quantify this conservatism, or confirm that the additional conservatism in the TR is insignificant.

RAI-7

Regarding the fracture mechanics (FM) models used in the analyses, Section 5.4 of the TR states that, “For the nozzle blend radius region, a nozzle blend radius corner fracture mechanics model [26] was used…. For the nozzle-to-vessel shell weld, the following crack models [27] were used....” Reference 26 is a private communication from P. M. Besuner to P. C. Riccardella. Reference 27 is a paper by C. B. Buchalet and W. H. Bamford. These references did not appear in the BWRVIP-108 report. To reduce the NRC staff’s review time, please cite any NRC SE on submittals using the FM models in these two references, or provide them for NRC staff review.

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