UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 29, 2008

Mr. Mark B. Bezilla
Site Vice President
FirstEnergy Nuclear Operating Company
Perry Nuclear Power Plant
Mail Stop A-PY-A290
P.O. Box 97, 10 Center Road
Perry, OH 44081-0097

SUBJECT: PERRY NUCLEAR POWER PLANT, UNIT NO. 1 - REQUEST FOR RELIEF RELATED TO INSERVICE INSPECTION RELIEF REQUEST IR-054 (TAC NO. MD8458)

Dear Mr. Bezilla:

By letter to the Nuclear Regulatory Commission (NRC) dated March 31, 2008 (Agencywide Documents Access and Management System Accession No. ML080990625), FirstEnergy Nuclear Operating Company, submitted relief request (RR) IR-054, proposing an alternative to certain requirements of the American Society of Mechanical Engineers, Boiler and Pressure Vessel Code (ASME Code), Section XI inspection requirements regarding examination of certain reactor pressure vessel nozzle-to-vessel welds and nozzle inner radii at the Perry Nuclear Power Plant, Unit No. 1. Instead, an alternative in accordance with ASME Code Case N-702, "Alternative Requirements for Boiling Water Reactor (BWR) Nozzle Inner Radius and Nozzle-to-Shell Welds," was proposed. RR IR-054 is for the second 10-year inservice inspection interval.

The NRC staff has completed its review of RR IR-054 and the details of the NRC staff's review are set forth in the enclosed safety evaluation. Accordingly, RR IR-054 is authorized pursuant to Title 10 of the *Code of Federal Regulations* Section 50.55a(a)(3)(i) based on the NRC staff's determination that the alternative provides an acceptable level of quality and safety. The use of ASME Code Case N-702 is authorized until such time as the ASME Code Case N-702 is published in a future version of Regulatory Guide (RG) 1.147 and incorporated by reference in

10 CFR 50.55a(b). At that time, if the licensee intends to continue implementing this ASME Code Case N-702, it must follow all provisions of ASME Code Case N-702 with conditions as specified in RG 1.147 and limitations as specified in Paragraph 50.55a(b)(4), (b)(5), and (b)(6), if any.

Sincerely,

Joel S Wiebe, for

Russell Gibbs, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-440

Enclosure:
Safety Evaluation

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO RELIEF REQUEST NUMBER IR-054

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION CORP.

OHIO EDISON COMPANY

PERRY NUCLEAR POWER PLANT, UNIT NO. 1

DOCKET NO. 50-440

1.0 INTRODUCTION

By letter to the Nuclear Regulatory Commission (NRC, the Commission) dated March 31, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML080990625), FirstEnergy Nuclear Operating Company (FENOC, the licensee), the licensee for Perry Nuclear Power Plant, Unit No. 1 (PNPP or Perry), submitted a request for relief from American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," regarding examination of reactor pressure vessel (RPV) nozzle-to-vessel welds and nozzle inner radii at PNPP. Instead, the licensee proposed to use an alternative in accordance with ASME Code Case N-702, "Alternative Requirements for Boiling Water Reactor (BWR) Nozzle Inner Radius and Nozzle-to-Shell Welds". The technical basis for ASME Code Case N-702 was documented in an Electric Power Research Institute (EPRI) report for the Boiling Water Reactor Vessel and Internals Project (BWRVIP), "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 Inner Radii," which was approved by the NRC in a safety evaluation (SE) dated December 19, 2007 (ADAMS Accession No. ML073600374). This alternative will be discussed in Section 2.0 below.

The December 19, 2007, SE for the BWRVIP-108 report specified plant-specific requirements which must be met for applicants proposing to use this alternative. This submittal intended to demonstrate that the relevant PNPP RPV nozzle-to-vessel welds and their inner radii meet these plant-specific requirements so that the relief request could be granted.

2.0 REGULATORY EVALUATION

Inservice inspection (ISI) of the ASME Code Class 1, 2, and 3 components is performed in accordance with Section XI of the ASME Code and applicable addenda as required by Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Section 50.55a(g), except where specific relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).

Section 50.55a(a)(3) of 10 CFR states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if: (i) the proposed alternatives would provide an acceptable level of quality and safety or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. 10 CFR 50.55a(g)(4) further states that ASME Code Class 1, 2, and 3 components (including supports) must meet the requirements, except design and access provisions and preservice examination requirements, set forth in the ASME Code, Section XI to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The applicable ISI Code of Record for the second 10-year ISI interval for PNPP is the 1989 Edition of ASME Code, Section XI.

For all RPV nozzle-to-vessel shell welds and nozzle inner radii, ASME Code, Section XI requires 100 percent inspection during each 10-year ISI interval. However, ASME Code Case N-702 proposes an alternative which reduces the inspection of RPV nozzle-to-vessel shell welds and nozzle inner radius areas from 100 percent to 25 percent of the nozzles for each nozzle type during each 10-year interval. As mentioned earlier, the NRC has approved the BWRVIP-108 report, the underlying technical basis document for ASME Code Case N-702. The December 19, 2007, SE regarding the BWRVIP-108 report specified the following plant-specific requirements to be satisfied by applicants using ASME Code Case N-702:

> However, each licensee should demonstrate the plant-specific applicability of the BWRVIP-108 report to their units in the relief request by showing that all the following factors are less than 1.15:
>
> (1) the temperature factor defined as (RPV heat up and cooldown rate)/(100 °F/hour),
>
> For the recirculation inlet nozzle,
>
> (2) the RPV pressure stress factor defined as [(RPV pressure)x(RPV inner radius)/(RPV thickness)]/[1000 psi x 110 inch/5.69 inch],
>
> (3) the nozzle pressure stress factor defined as (pressure/1000 psi)x{[(nozzle outer radius)^2 + (nozzle inner radius)^2]/ [(nozzle outer radius)^2 - (nozzle inner radius)^2]}/{[(13.988 inch)^2 + (6.875 inch)^2]/ [(13.988 inch)^2 - (6.875 inch)^2]},
>
> For the recirculation outlet nozzle,
>
> (4) the RPV pressure stress factor defined as [(RPV pressure)x(RPV inner radius)/(RPV thickness)]/[1000 psi x 113.2 inch/7.0 inch], and

(5) the nozzle pressure stress factor defined as (pressure/1000 psi)x{[(nozzle outer radius)^2 + (nozzle inner radius)^2]/ [(nozzle outer radius)^2 - (nozzle inner radius)^2]}/{[(22.31 inch)^2 + (12.78 inch)^2]/ [(22.31 inch)^2 - (12.78 inch)^2]}.

This plant-specific information was required by the NRC staff to ensure that the probabilistic fracture mechanics (PFM) analysis documented in the BWRVIP-108 report applies to the RPV of the applicant's plant.

3.0 TECHNICAL EVALUATION

3.1 Licensee Evaluation

ASME Code Requirement for which Relief is Requested

The licensee requested relief from the following requirements of ASME Code, Section XI, 1989 Edition:

> [ASME Code] Class 1 nozzle-to-vessel weld and nozzle inner radii examination requirements are given in Subsection IWB, Table IWB-2500-1, "Examination Category B-D Full Penetration Welds of Nozzles in Vessels - Inspection Program B," Item Numbers B3.90 ["Nozzle-to-Vessel Welds"] and B3.100 ["Nozzle Inside Radius Section,"] respectively. The method of examination is volumetric. For the extent of examination, all nozzles with full penetration welds to the vessel shell (or head) and integrally cast nozzles must be examined each interval.

Component(s) for which Relief is Requested

Code Class: 1

Component Numbers: N1A, N1B, N2A, N2B, N2C, N2D, N2E, N2F, N2G, N2H, N2J, N2K, N7, N8, N9A, and N9B Nozzles

Examination Category: B-D

Item Number: B3.90 and B3.100

Licensee's Proposed Alternative to the ASME Code

> Pursuant to 10 CFR 50.55a(a)(3)(i), relief is requested from performing the required examinations on 100% of the identified nozzle assemblies. Alternatively, in accordance with Code Case N-702, a minimum of 25% of the nozzle inner radii and nozzle-to-shell welds, including at least one nozzle from each system and nominal pipe size, would be performed. For each of the identified nozzle assemblies, both the inner radius and the nozzle-to-shell weld would be examined. The following nozzle assemblies would be selected for examination: one of the two 22" recirculation outlet nozzle assemblies; three of the ten 12" recirculation inlet nozzle assemblies;
> one of the two 6" head spray nozzle assemblies; and one of the two 4" jet pump instrumentation nozzle assemblies

Licensee's Bases for Alternative

EPRI Technical Report 1003557, "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," provides the basis for [ASME] Code Case N-702. The evaluation found that failure probabilities due to a Low Temperature Overpressure event at the nozzle blend radius region and nozzle-to-vessel shell weld are very low (i.e., $<1 \times 10^{-6}$ for 40 years) with or without inservice inspection.

The report concludes that inspection of 25 percent of each nozzle type is technically justified.

On December 19, 2007, the NRC issued a Safety Evaluation Report (SER) approving the use of [the] BWRVIP-108 [report] as a basis for using [ASME] Code Case N-702. Within Section 5 of the SER, it states that each licensee should demonstrate the plant-specific applicability of the BWRVIP-108 report to their units in the relief request by meeting the criteria discussed in Section 5 of the SER.

Criterion 1: the maximum RPV heatup/cooldown rate is less than 115° F/hour.

In accordance with Technical Specification (TS) 3.4.11, Reactor Coolant System heatup and cooldown rates are ≤ 100° F in any one hour period.

Criterion 2: for recirculation inlet nozzles, $(pr/t)/C_{RPV} < 1.15$

$[(1045 \times 119) \div 7.19] \div 19332 < 1.15$
The Perry result is 0.89 < 1.15

Criterion 3: for recirculation inlet nozzles, $[p(r_o^2 + r_i^2)/(r_o^2 - r_i^2)]/C_{NOZZLE} < 1.15$

$[1045 \times (11.125^2 + 5.813^2) \div (11.125^2 - 5.813^2)] \div 1637 < 1.15$
The Perry result is 1.12 < 1.15

Criterion 4: for recirculation outlet nozzles, $(pr/t)/C_{RPV} < 1.15$

$[(1045 \times 119) \div 7.19] \div 16171 < 1.15$
The Perry result is 1.07 < 1.15

Criterion 5: for recirculation outlet nozzles, $[p(r_o^2 + r_i^2)/(r_o^2 - r_i^2)]/C_{NOZZLE} < 1.15$

$[1045 \times (17.594^2 + 10^2) (17.594^2 - 10^2)] \div 1977 < 1.15$
The Perry result is 1.03 < 1.15
The results of the above equations demonstrate the applicability of the BWRVIP-108 report to the Perry plant by showing the criteria within Section 5 of the NRC SER is met. Therefore, the basis for using [ASME] Code Case N-702 is demonstrated for the Perry plant.

3.2 Staff Evaluation

Criteria for Applying the BWRVIP-108 Report

The December 19, 2007, SE on the BWRVIP-108 report specified five plant-specific criteria that licensees must meet in order to demonstrate that the BWRVIP-108 report results apply to their plants. The five criteria are related to the driving force of the PFM analyses for the recirculation inlet and outlet nozzles. It was stated in the December 19, 2007, SE that the nozzle material fracture toughness-related (RT_{NDT}) values used in the PFM analyses were based on data from the entire fleet of BWR RPVs. Therefore, the BWRVIP-108 report PFM analyses are bounding with respect to fracture resistance, and only the driving force of the underlying PFM analyses needs to be evaluated. It was also stated in the December 19, 2007, SE, that except for the RPV heatup/cooldown rate, the plant-specific criteria are for the recirculation inlet and outlet nozzles only because the probabilities of failure, P(F|E)s, for other nozzles are an order of magnitude lower.

The licensee stated that Criterion 1 is satisfied because PNPP TS 3.4.11 requires that the RPV heatup/cooldown rate be less than 100° F per hour. For the remaining four criteria, the licensee provided, in the submittal, FENOC's plant-specific data and its evaluation of the driving force factors, or ratios, against the criteria established in the December 19, 2007, SE. The licensee's calculated results showed that remaining four criteria are satisfied. However, information provided by the licensee shows that in November 2007, PNPP had an event with the heatup/cooldown rates exceeding 115° F per hour with three other previous occurrences in the past 10 years occurring in 2001. The December 19, 2007, SE states, "These PFM results show that the total probability of failure for the recirculation inlet nozzle radii is 1.19E-7 for the same LTOP [(low temperature overpressure protection)] event used in [the] BWRVIP-05 [report, "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations,"] and 1.98E-6 for the normal operation." This indicated that the normal operation condition, which is defined in the ASME Code, Section XI as a condition that includes heatups and cooldowns, is limiting. Hence, it is necessary to determine whether the PNPP experienced normal RPV heatups and cooldowns exceeding 115° F per hour. When the NRC staff established Criterion 1 regarding plant-specific heatups and cooldowns in the December 19, 2007, SE for the BWRVIP-108 report, it was not the NRC staff's intention to consider the heatups and cooldowns associated with typical transients. Transients are bounded by the limiting LTOP event used in the PFM analyses for the non-normal condition (i.e., the transient condition). This serves as a clarification of Criterion 1 regarding the RPV heatup and cooldown limit of 115° F per hour specified in the December 19, 2007, SE.

On July 31, 2008, the NRC staff issued a request for additional information (RAI) regarding PNPP plant operation data in recent years which indicated heatups and cooldowns exceeding 115° F per hour (ADAMS Accession No. ML081980628). The staff intended to use this information to assess the impact of these heatups and cooldowns exceeding 115° F per hour on the total probability of failure for the recirculation inlet nozzle radii.

The licensee provided a response to the NRC staff's RAI in a letter dated September 17, 2008 (ADAMS Accession No. ML082680091). The response tabulated events that have occurred over the past 10 years that have exceeded a heatup/cooldown rate of 115° F per hour and concluded that for these events, "reactor pressure vessel temperature rates measured from bulk saturation temperature remained within the 100° F/hr limit." Additional information regarding the November 28, 2007, scram was also provided by the licensee as an example to address the RPV integrity issue due to this kind of event. The NRC staff has reviewed documents related to this November 28, 2007, scram: Event Notification (EN-43808) dated November 29, 2007, Preliminary Notification (PNO-III-07-015) dated November 28, 2007, the Special Inspection Team (SIT) Report dated January 25, 2008, and the closure memorandum for NRC review of the Issue for Resolution (IFR) 2008-006 dated May 21, 2008. The NRC staff concluded from this review that the November 28, 2007, scram at PNPP was a typical transient, which was far from being a normal heatup or cooldown condition with an excessive rate. Other events also fit the description of being transients. As a result, the first criterion regarding the maximum RPV heatup/cooldown rate, which is intended to address normal operating conditions, is satisfied. For the remaining four criteria, the NRC staff verified the licensee's calculations and confirmed that the remaining four criteria are also satisfied.

Evaluation of the Proposed ISI

The licensee states that for each of the identified nozzle assemblies, both the inner radius and the nozzle-to-shell weld would be examined. The licensee will select the following nozzles for examination: one of the two 22 inches recirculation outlet nozzle assemblies, three of the ten 12 inches recirculation inlet nozzle assemblies, one of the two 6 inches head spray nozzle assemblies, and one of the two 4 inches jet pump instrumentation nozzle assemblies.

The licensee notes that ASME Code Case N-702 proposes that VT-1 visual examination may be used in lieu of volumetric examination for the inner radii (Item B3.100); however, for this request they will not be utilizing this aspect of ASME Code Case N-702. For the selected recirculation inlet and jet pump instrumentation nozzles, volumetric examinations (per ASME Code) will be performed as their nozzle inner radii are not fully accessible from the inside of the vessel for VT-1 examination. For the 22 inches recirculation outlet and the 6 inches head spray nozzles, the licensee will be using ASME Code Case N-648-1, "Alternative Requirements for Inner Radius Examinations of Class 1 Reactor Vessel Nozzles," in accordance with the conditions placed upon the use of that Code Case by Regulatory Guide (RG) 1.147, Revision 15, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," which allows VT-1 visual examination for nozzle inner radii. The condition placed on the use of this Code Case is that "in place of UT examination, licensees may perform a visual examination with enhanced magnification that has a resolution sensitivity to detect a 1-mil width wire or crack..." Based on the above information, the staff finds the licensee's proposed alternative examination acceptable as it provides reasonable assurance of verifying structural integrity of the nozzle's inner radii.

4.0 CONCLUSION

The NRC staff has reviewed the submittal and finds that the PNPP RPV meets all five plant-specific criteria specified in the December 19, 2007, SE on the BWRVIP-108 report, which provides technical bases for use of ASME Code Case N-702. Accordingly, RR IR-054 is authorized pursuant to 10 CFR 50.55a(a)(3)(i) based on the NRC staff's determination that the alternative provides an acceptable level of quality and safety. The use of ASME Code Case N-702 is authorized until such time as the ASME Code Case N-702 is published in a future version of RG 1.147 and incorporated by reference in 10 CFR 50.55a(b). At that time, if the licensee intends to continue implementing this ASME Code Case N-702, it must follow all provisions of ASME Code Case N-702 with conditions as specified in RG 1.147 and limitations as specified in Paragraph 50.55a(b)(4), (b)(5), and (b)(6), if any.

Consequently, pursuant to 10 CFR 50.55a(a)(3)(i), the alternative is authorized through the end of the second 10-year ISI interval from the requirements of Table IWB-2500-1 (Inspection Program B) of ASME Code, Section XI, pertaining to inspection of the RPV nozzle-to-vessel shell welds and inner radii for nozzles specified in the Enclosure 1 of the submittal because an acceptable level of quality and safety can be maintained.

All other requirements of the ASME Code, Sections III and XI, for which relief has not been specifically requested and approved remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

Principal Contributors: S. Sheng, NRR
C. Nove, NRR

Date: December 29, 2008

10 CFR 50.55a(b). At that time, if the licensee intends to continue implementing this ASME Code Case N-702, it must follow all provisions of ASME Code Case N-702 with conditions as specified in RG 1.147 and limitations as specified in Paragraph 50.55a(b)(4), (b)(5), and (b)(6), if any.

Sincerely,

/RA/ by JWiebe for

Russell Gibbs, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-440

Enclosure:
Safety Evaluation

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