May 29, 2003

EA-03-025

Mr. Lew Myers
Chief Operating Officer
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, OH 43449-9760

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR A RED FINDING (NRC INSPECTION REPORT 50-346/2003-16) - DAVIS-BESSE CONTROL ROD DRIVE MECHANISM PENETRATION CRACKING AND REACTOR PRESSURE VESSEL HEAD DEGRADATION

Dear Mr. Myers:

The purpose of this letter is to provide you with the Nuclear Regulatory Commission's (NRC's) final significance determination for the performance deficiency that resulted in the control rod drive mechanism penetration cracking and reactor pressure vessel head degradation discovered in February and March 2002. Based on the discovery, the NRC performed an Augmented Inspection, the results of which were documented in Inspection Report 50-346/02-03, issued on May 3, 2002. On October 2, 2002, the NRC issued the Augmented Inspection Followup report documenting ten apparent violations associated with the reactor pressure vessel head degradation. These apparent violations were classified as unresolved items because the significance had not yet been determined.

The performance deficiency associated with the control rod drive penetration cracking and reactor pressure vessel head degradation was documented in the NRC's letter to you dated February 25, 2003. The performance deficiency was FirstEnergy Nuclear Operating Company's (FENOC's) failure to properly implement the boric acid corrosion control and corrective action programs, which allowed reactor coolant system pressure boundary leakage to occur undetected for a prolonged period of time, resulting in reactor pressure vessel head degradation and control rod drive nozzle circumferential cracking. The significance of the performance deficiency was assessed using the Significance Determination Process (SDP) as described in the February 25, 2003, letter.

The performance deficiency resulted in an increase in the risk of reactor core damage through a loss of coolant accident caused by either a rupture in the exposed cladding in the reactor pressure vessel head cavity or a control rod drive mechanism nozzle ejection due to a circumferential crack. The result of the NRC's preliminary significance analysis of the reactor pressure vessel head cavity indicated significance in the Red range (change in core damage frequency $\geq 10^{-4}$ per reactor-year). The result of NRC's significance analysis of the as-found circumferential crack and potential for crack growth indicated significance in the Yellow to Red range (change in core damage frequency in the range of low 10^{-5} to low 10^{-4} per reactor-year). Consequently, the NRC preliminarily determined that the performance deficiency resulting in the reactor pressure vessel head degradation and control rod drive mechanism nozzle cracking had high safety significance in the Red range, an issue of high safety significance that will result in increased NRC inspection and other NRC action.

In addition to describing the NRC's preliminary conclusions concerning the performance deficiency, the NRC's letter of February 25, 2003, provided FENOC an opportunity to request a Regulatory Conference. In lieu of a Regulatory Conference, FENOC submitted a written response dated April 24, 2003, which provided FENOC's additional perspective. FENOC acknowledged the performance deficiency and did not contest the Red finding or the safety significance determination with respect to the scientific evidence presented, the root cause(s), and the risk assessment determinations. FENOC, however, requested the NRC consider making two clarifications to the SDP and Enforcement Review Panel Worksheet attached to NRC's letter of February 25, 2003. The responses to those requested clarifications are contained in the enclosure to this letter. Because the Worksheet was prepared before and reflects the deliberations of the Review Panel regarding these matters, the NRC has concluded that the SDP and Enforcement Review Panel Worksheet will not be reissued.

FENOC's response of April 24, 2003, provided no new information to change the NRC staff's preliminary conclusion that the performance deficiency resulting in the reactor pressure vessel head degradation and control rod drive mechanism nozzle cracking had high safety significance in the Red range. Therefore, the NRC has concluded that the significance of the performance deficiency discussed in the NRC's February 25, 2003, letter is a finding appropriately characterized as Red. You have 30 calendar days from the date of this letter to appeal the staff's final determination of significance for the identified Red finding. Such appeals will be considered to have merit only if they meet the criteria given in NRC Inspection Manual Chapter 0609, Attachment 2.

The safety significance is one of the inputs into the final characterization and resolution of the apparent violations described in the Augmented Inspection Followup Report dated October 2, 2002. The NRC's investigation into the cause of those apparent violations, which were referred to the Office of Investigations, is ongoing. The results of that investigation will be factored into the final enforcement deliberations. As a result, no Notice of Violation is attached at this time. The number and nature of those violations could change as a result of further NRC review.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response, if you chose to respond, will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the Public without redactions. The NRC also includes significant enforcement actions on its Web site at www.nrc.gov; select "What We Do" and "Enforcement," then "Significant Enforcement Actions."

For administrative purposes, this letter is issued as a separate NRC Inspection Report, No. 50-346/03-16. Accordingly, Unresolved Items 50-346/2002-08-01 through -10 are closed.

Apparent Violations 50-346/2003-16-01 through -10, corresponding to each of the Unresolved Items, are open.

Should you have any questions regarding this letter, please contact John A. Grobe, Chairman of the Davis-Besse Oversight Panel, at 630-829-9637.

Sincerely,

/RA by J. Caldwell acting for/

J. E. Dyer
Regional Administrator

Enclosure: As stated

Docket No. 50-346
License No. NPF-3

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ENCLOSURE

RESPONSE TO SPECIFIC COMMENTS IN FENOC'S APRIL 24, 2003, LETTER

1. FENOC Request

"In Section C.1.a, page 2, paragraph two of the SDP and Enforcement Review Panel Worksheet, under the Screening Logic, Results and Assumptions for Phase I, it is stated that "The resulting circumferential cracking and cavity represent a significant loss of the design basis barrier integrity and could be reasonably viewed as a precursor to a significant event." Although FENOC understands that the reference is to the potential that existed for through-wall circumferential cracking to develop if the condition was left uncorrected, in its current context it could be misinterpreted that the cavity resulted from leakage associated with a circumferential crack. As stated in the FENOC root cause and later in the SDP Worksheet (See "Analysis of CRDM [Control Rod Drive Mechanism] Circumferential Cracking,") the circumferential crack identified through inspection at Davis-Besse was associated with CRDM nozzle 2, and this crack was in its initial stages of formation and was not through-wall. The cavity in the RPV [Reactor Pressure Vessel] head was associated with axial cracking in nozzle 3."

NRC Reply

The referenced Section of the Worksheet states, "In accordance with Manual Chapter (MC) 0612, the inspectors determined that the issue was more than minor in safety significance because if left uncorrected, the circumferential cracking and boric acid corrosion would become a more significant safety ***concern*** (emphasis added). The resulting circumferential cracking and cavity represent a significant loss of the design basis barrier integrity and could be reasonably viewed as a precursor to a significant event."

The NRC agrees that the use of the singular "concern" in reference to two, separable reactor pressure vessel degradation effects is potentially ambiguous. The referenced Section could more accurately state:

> In accordance with Manual Chapter (MC) 0612, the inspectors determined that the issue was more than minor in safety significance because if left uncorrected, the resulting circumferential cracking and boric acid corrosion each would become more significant safety concerns. The resulting circumferential cracking and cavity represent a significant loss of the design basis barrier integrity and could each be reasonably viewed as a precursor to a significant event.

However, because the Worksheet was prepared before and reflects the deliberations of the Review Panel regarding these matters, the NRC has concluded that the Worksheet will not be reissued.

2. FENOC Request

"In Section C.1.c, page 5, paragraph two of the SDP and Enforcement Review Panel Worksheet, under "Corrosion rates not known," a possible corrosion rate of 7 inches per year is provided as a reasonable consideration for the last stages of cavity growth for the RPV head. Attachment A, *Risk Assessment and Insights in Support of Phase 3 Risk Significance Determination* . . . , page 8, states that mechanisms for cavity growth cannot be substantiated by data. It is also mentioned that a mechanism could exist which would limit the corrosion process, thus reducing the rate and ultimate size of the cavity. Given that physical evidence provides indications of corrosion products for several cycles over the event time line, a corrosion rate of 7 inches per year, if it occurred, would have been short lived. The extent of boron concentration in the system, amount of boron deposition on the head, and other factors would determine the future rate of corrosion. Seven inches per year, applied as a projected future corrosion rate, would certainly appear to be a bounding assumption and not a realistic estimate of average corrosion rates."

<u>NRC Reply</u>

The referenced Section of the Worksheet states, "Based primarily on the observed levels of boric acid particles in the containment atmosphere, the licensee's root cause analysis report speculates that the cavity found in the RPV head grew at an average rate of 2-inches/year over the 4-year period of the last two operating cycles. The available evidence to support this is certainly not conclusive, and other interpretations are also reasonable. Corrosion rates for aqueous boric acid solutions in a variety of physical situations are provided in the Electric Power Research Institute [EPRI] Boric Acid Corrosion Guidebook. The closest situations covered by the Guidebook appear to be the tests where an aqueous solution of boric acid flowed across a low-alloy steel surface that was heated to 600 degrees fahrenheit, which resulted in corrosion rates as high as 7-inches/year. It seems reasonable to consider the possibility that the last stages of cavity growth on the Davis-Besse RPV head may have experienced a 7-inch/year corrosion rate. It is not known at this time which cases or what intermediate corrosion rate value is more likely."

FENOC correctly observes that the NRC staff assessment uses 7 inches per year as an upper bound for the corrosion rate. The staff performed analyses using both upper and lower bound values because not enough is currently known about the corrosion process to establish a most probable value for the conditions that existed on the reactor vessel head. The 7 inch per year value was selected because it is the rate demonstrated by two EPRI experiments that have conditions similar to those on the Davis-Besse reactor pressure vessel head. The staff knows of no basis for FENOC's assertion that "a corrosion rate of 7 inches per year, if it occurred, would have been short lived." The EPRI experimental data does not support a conclusion that the corrosion rate would be short lived. FENOC's Root Cause Analysis Report proposes 4 inches per year as an upper bound for the corrosion rate at the end of the process. However, at the end of the corrosion period, the unidentified leak rate was on the order of 0.2 gpm, and therefore could have increased by as much as a factor of five before exceeding the limit in Technical Specifications. If the maximum corrosion rate or maximum cavity size is limited by the rate that reactor coolant leaks into the cavity, then corrosion rates and cavity sizes substantially greater than found appear to be possible. On the basis of the available information, the staff confirms that 7 inches per year is a reasonable value to use for the upper bound analysis.