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

September 22, 2011

Barry S. Allen, Vice President
Davis-Besse Nuclear Power Station
FirstEnergy Nuclear Operating Company
5501 North State Route 2
Oak Harbor, OH 43449

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE DAVIS-BESSE NUCLEAR POWER STATION (TAC NO. ME4640)

Dear Mr. Allen:

By letter dated August 27, 2010, FirstEnergy Nuclear Operating Company submitted an application pursuant to Title 10 of the *Code of Federal Regulations* Part 54 for renewal of Operating License NPF-3 for the Davis-Besse Nuclear Power Station. The staff of the U.S. Nuclear Regulatory Commission (NRC or the staff) is reviewing this application in accordance with the guidance in NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants." During its review, the staff has identified areas where additional information is needed to complete the review. The staff's requests for additional information are included in the enclosure. Further requests for additional information may be issued in the future.

Items in the enclosure were discussed with Cliff Custer, of your staff, and a mutually agreeable date for the response is within 30 days from the date of this letter. If you have any questions, please contact me by telephone at 301-415-2946 or by e-mail at Samuel.CuadradoDeJesus@nrc.gov.

Sincerely,

Samuel Cuadrado-De Jesús, Project Manager
Projects Branch 1
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosure:
As stated

cc w/encl: Listserv

DAVIS-BESSE NUCLEAR POWER STATION
LICENSE RENEWAL APPLICATION
REQUEST FOR ADDITIONAL INFORMATION

RAI 3.1.2.2.16-1

Background:

By its letter dated August 17, 2011, the applicant addressed its review results on cracking due to primary water stress corrosion cracking (PWSCC) of steam generator nickel alloy tube-to-tubesheet welds in response to the discussion held in a teleconference call dated July 13, 2011.

In the letter, the applicant stated that upon further review after the conference call with the U.S. Nuclear Regulatory Commission (NRC), it determined that the tube-to-tubesheet welds (Alloy 600 welds) for its steam generators do not have a license renewal intended function and therefore, are not subject to an aging management review. The applicant also stated that the steam generators are Babcock & Wilcox Model 177-FA, once-through design and the tubes and the tubesheets of the steam generators form the pressure boundary between the fluid in the secondary system and the reactor coolant system. The applicant further stated that as provided in Updated Safety Analysis Report (USAR) Section 5.5.2.3, the tubes are expanded (to a partial depth) into the tubesheet and the tubes are seal welded to the tubesheet near the tube ends. In addition, the applicant stated that the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, Division 1, 1995 Edition with 1996 Addenda, IWA-9000 defines a seal weld as a nonstructural weld intended to prevent leakage, where the strength is provided by a separate means. The applicant stated that the "separate means" in this case is the tube-to-tubesheet expansion joint which forms the pressure boundary and that the tube-to-tubesheet welds are seal welds and therefore, are not part of the pressure boundary.

Issue:

The applicant stated the tube-to-tubesheet welds (Alloy 600 welds) for its steam generators do not have a license renewal intended function and therefore, are not subject to an aging management review. The applicant also stated that the tube-to-tubesheet welds are seal welds and therefore, are not part of the pressure boundary. However, the staff noted that the reactor coolant pressure boundary should provide structural and leak-tight integrity. Furthermore, the applicant's statement that the tube-to-tubesheet welds are intended to prevent leakage indicates that these welds perform the intended function of the reactor coolant pressure boundary. Therefore, the staff found a need to confirm whether or not the design analysis of the applicant's once-through steam generators, which was used to establish the current licensing basis (CLB), concludes the following: the interference fits between the tubes and the tubesheets are sufficient to ensure the structural and leak-tight integrity of the tube-to-tubesheet joints, without a need for crediting the tube-to-tubesheet welds.

Request:

1) Confirm whether or not the design analysis, which was used to establish the CLB, concludes that the interference fits are sufficient to ensure the structural and leak-tight integrity of the tube-to-tubesheet joints, without a need for crediting the tube-to-tubesheet welds.

   If the design analysis concludes that the interference fits are sufficient to ensure the structural and leak-tight integrity, provide the technical basis of the conclusion and list the reference(s) addressing the technical basis.

2) If the design analysis, which was used to establish the CLB, credits the tube-to-tubesheet welds for ensuring the structural and leak-tight integrity of the tube-to-tubesheet joints, describe how cracking due to PWSCC will be managed for the steam generator tube-to-tubesheet welds.

RAI 2.3.3.18-4

Background:

In its response to RAI 2.3.3.18-3 dated August 17, 2011, the applicant provided the following information:

1) The letdown coolers performed acceptably from initial startup in 1978 until 1991, when plant personnel detected contamination in the component cooling water (CCW) system, and replaced both letdown coolers in 1993. Then, in 2009, plant personnel identified a small, active reactor coolant leak, and again replaced both letdown coolers in 2010.

2) A failure analysis had not been performed on the leaking letdown coolers to determine the specific leak location or to verify the failure mechanism because of high radiation dose rates associated with that effort.

SRP-LR Section A.1.2.3.4, "Detection of Aging Effects," states that nuclear power plants are licensed using the principles of redundancy, and diversity, and that degraded components reduce the reliability of the systems, challenge safety systems, and contribute to plant risk. The SRP-LR continues by stating that the effects of aging on a component should be managed to ensure its availability to perform its intended function(s) as designed when called upon, and notes that a program based solely on detecting component failure should not be considered as an effective aging management program for license renewal.

Issue:

Based on the information provided in this recent response, as well as the information provided in response to RAI 2.3.3.18-2 for the same issue, the staff did not consider that the applicant has provided sufficient bases to justify the replacement frequency of every seventh refueling outage (approximately 14 years) for the letdown coolers in the makeup and purification system.

The bases for the staff's position are as follows:

a) The applicant established the replacement frequency based on a qualified life, which was empirically derived using two plant-specific data points of 13 and 16 years, after identifying reactor coolant leakage into the component cooling water system.

b) The applicant has not determined the flaw location, performed flaw sizing, or verified flaw characteristics to allow prediction of flaw stability or growth rate. Without having this information, operation of the letdown cooler with ongoing leakage is risking a failure, which would challenge the pressure relief capability of the component cooling water system and the isolation function of the valves in the makeup and purification system.

c) While past operating experience (although limited) may have shown that the flaw was stable for some period of time, the replacement frequency determination did not appear to consider normal operational pressure transients that the letdown coolers would be expected to experience.

d) The letdown cooler replacement frequency appears to be based on overall calendar time and not actual operational time, considering both refueling and extended outages.

Request:

Provide a letdown cooler replacement frequency that includes adequate margin to initiation of tube leakage and provide the basis for the margin, or propose an aging management program that will adequately manage these components that are within the scope of license renewal.

RAI 3.3.2.2.10.4-1

Background:

SRP-LR Table 3.3-1, item 26 states that loss of material due to pitting and crevice corrosion could occur for copper alloy piping, piping components, and piping elements exposed to lubricating oil. The SRP-LR recommends GALL AMP XI.M39, "Lubricating Oil Analysis," to manage the aging effect and further evaluation of a program to verify the effectiveness of the Lubricating Oil Analysis Program, such as XI.M32, "One-Time Inspection," because control of contaminants within the lubricating oil may not have always been adequate to preclude corrosion.

In LRA Tables 3.3.2-14, 3.3.2-18, 3.3.2-30, 3.4.2-1, and 3.4.2-4, the applicant referenced LRA Table 3.3.1, item 3.3.1-26 and generic note I for copper alloy components exposed to lubricating oil and stated that the components have no aging effects requiring management. For these items, the applicant further cited plant-specific notes which state that the components are made of copper alloy with less than 15 percent zinc and are not in contact with a more cathodic metal; therefore, the components have no aging effects requiring management.

Issue:

It is unclear to the staff why the applicant claims that components of copper alloy with less than 15 percent zinc exposed to lubricating oil have no aging effects requiring management. The staff noted that components of copper alloy with less than 15 percent zinc are less susceptible to loss of material than other copper alloys, but that the presence of contaminants (e.g., water) in lubricating oil can create an environment conducive to loss of material, regardless of whether or not the component is in contact with a more cathodic metal.

Request:

Explain why components of copper alloy with less than 15 percent zinc exposed to lubricating oil have no aging effects requiring management or provide an appropriate AMP to manage loss of material.

September 22, 2011

Barry S. Allen, Vice President
Davis-Besse Nuclear Power Station
FirstEnergy Nuclear Operating Company
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Sincerely,

/RA/

Samuel Cuadrado-De Jesús, Project Manager
Projects Branch 1
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosure:
As stated

cc w/encl: Listserv

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DATE	09/14/2011	09/20/2011	09/21/2011	09/22/2011

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Letter to Barry S. Allen from Samuel Cuadrado-De Jesús dated September 22, 2011

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