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

November 8, 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 LICENSE RENEWAL APPLICATION (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 3.1.2.2.16-2

Background:
By letter dated October 21, 2011, the applicant responded to request for additional information 3.1.2.2.16-1, which addressed a need for the aging management of cracking due to primary water stress corrosion cracking (PWSCC) of the steam generator tube-to-tubesheet welds. In its response, the applicant stated that cracking due to PWSCC will be managed for the steam generator tube-to-tubesheet welds (Alloy 600) by a combination of the Pressurized Water Reactor Water Chemistry Program and the Steam Generator Tube Integrity Program. The applicant also stated that the Steam Generator Tube Integrity Program will be enhanced to include enhanced visual (EVT-1 or equivalent) examinations to monitor for cracking of the steam generator tube-to-tubesheet welds. The applicant further indicated that welds included in the inspection sample will be scheduled for examination in each 10-year period that occurs during the period of extended operation and unacceptable inspection findings will be evaluated by the Corrective Action Program using criteria in accordance with Section XI of the American Society of Mechanical Engineers (ASME) Code.

In addition, the applicant indicated that a review of Davis-Besse operating experience has not identified any instances of cracking of the steam generator tube-to-tubesheet welds (Alloy 600): therefore, the weld inspection sample size will include 20 percent of the subject weld population or a maximum of 25, whichever is less. The applicant stated that in this case the maximum of 25 applies since the weld population for the two steam generators is greater than 60,000. The applicant also indicated that if the steam generators are replaced in the future with a design such that the tube-to-tubesheet welds are fabricated of Alloy 690-TT material, the examinations will no longer be required.

Issue:
In its review, the staff found a need to clarify whether the "Alloy 690 TT material," which refers to a potential material for future steam generator welds, means Alloy 690 TT tubes with Alloy 690 type weld material (e.g., Alloy 52). The staff also noted that it is not clear whether Section XI of the ASME Code has acceptance criteria for these steam generator tube-to-tubesheet welds. In addition, the staff found a need to further clarify whether the EVT-1 inspection is capable of detecting cracking in the tube-to-tubesheet weld. The staff also requests that the applicant discuss the extent, to which the routine steam generator tube inspections, using bobbin coil or rotating coil examinations, can detect cracking of the tube-to-tubesheet welds.

The staff also found a need for clarification of why a sample size of 25 is adequate to monitor for the cracking of the steam generator tube-to-tubesheet welds, in view of the following considerations: (1) potential variabilities exist in the weld chemistry, environment and stresses in the approximately 60,000 welds, (2) Alloy 600 is susceptible to PWSCC, (3) the applicant's

ENCLOSURE

steam generator tubes (Alloy 600) have experienced cracking due to PWSCC, indicating that the degradation mechanism (PWSCC) exists for the steam generator tubes, and (4) the applicant's program has not implemented any inspection intended to detect cracking in the tube-to-tubesheet welds.

Request:
The applicant indicated that examinations are no longer required if the steam generators are replaced in the future with a design such that the tube-to-tubesheet welds are fabricated with Alloy 690 TT material. Please, provide information to clarify whether the "Alloy 690 TT material" means Alloy 690 TT tubes with Alloy 690 type tubesheet cladding (e.g., Alloy 52). If not, discuss why inspections are not necessary to manage cracking due to PWSCC of the replacement steam generator welds.

1. It is not clear that Section XI of the ASME Code has acceptance criteria for these steam generator tube-to-tubesheet welds. Please, discuss what acceptance criteria will be used to evaluate the indications found in the inspections.

2. Provide information to demonstrate the EVT-1 inspection is capable of detecting cracking in the tube to tubesheet welds. In addition, discuss the extent, to which the routine steam generator tube inspections, using bobbin coil or rotating coil examinations, can detect cracking of the tube-to-tubesheet welds.

3. Provide justification as to why a sample size of only 25 is adequate to monitor for the cracking of the steam generator tube-to-tubesheet welds in view of the following considerations: (1) potential variabilities exist in the weld chemistry, environment and stresses in the approximately 60,000 welds, (2) Alloy 600 tubes are susceptible to PWSCC, (3) the applicant's Alloy 600 tubes have experienced cracking due to PWSCC, indicating that the degradation mechanism (PWSCC) exists for the steam generator tubes, and (4) the applicant's program has not implemented any inspection intended to detect cracking in the tube-to-tubesheet welds.

RAI 3.3.2.14-2

Background:
License renewal application (LRA) Table 3.3.2-14, "Aging Management Review Results – Fire Protection," item "Heat Exchanger (tubes) – Fire water storage tank heat exchanger (DB-E52)," originally proposed a one-time inspection to manage the reduction in heat transfer of stainless steel tubes. The Generic Aging Lessons Learned Report (NUREG-1801) states that stainless steel components exposed to steam are susceptible to loss of material and stress corrosion cracking; however, the applicant has not identified these aging effects for this component.

By letter dated July 27, 2011, the staff issued request for additional information (RAI) 3.3.2.14-1 requesting that the applicant justify why loss of material and stress corrosion cracking are not applicable aging effects for the fire water storage tank heat exchanger tubes exposed to steam.

In its response dated August 26, 2011, the applicant stated that the only license renewal function for the heat exchanger is reduction of heat transfer and the only aging mechanism that is identified as causing the aging effect of reduction of heat transfer is the aging mechanism of fouling. Loss of material and cracking would ultimately affect the pressure boundary function of the tubes. The applicant also stated that if the heat exchanger tubes should leak, fire water would not leak from the tubes; rather, the higher pressure (i.e., approximately 50 psig) steam from the auxiliary steam system on the external surfaces of the tubes would pass through the tubes and mix with fire water (approximately 25 psig), thereby continuing to add heat to the water. Fire water storage tank level would increase due to water entering the system, but level in the tank could be controlled (i.e., feed-and-bleed) to prevent the tank from overflowing onto the ground. A breach of the heat exchanger tubes would result in continued heat transfer to fire water, and would not prevent the fire water system from performing its functions.

A teleconference was held on September 13, 2011, to further discuss this issue and determine, with a heat exchanger tube failure, whether the fire water storage tank's design could contain a water/steam environment. The applicant stated that the heat exchanger was not subject to license renewal scope based on the Fire Safety Hazard Analysis. The applicant was asked to fully document the basis for this statement.

In a follow-up response dated October 7, 2011, the applicant revised the LRA to delete the fire water storage tank heat exchanger (DB-E52) and fire water storage tank recirculation pump casing (DB-P114). Also License Renewal Boundary Drawing LR-M016A, "Station Fire Protection System," was revised to remove highlighting of the piping and components associated with the fire water storage tank heat exchanger (DB-E52) and fire water storage tank recirculation pump 1-1. The applicant also stated that the fire water storage tank heat exchanger and recirculation pump are not within the scope of license renewal since the subject components do not satisfy the scoping criteria of 10 CFR 54.4(a)(1), (a)(2), or (a)(3). The heat exchanger and the recirculation pump are used to establish initial conditions associated with event assumptions, and perform no fire protection functions. Hence it is the monitoring of the fire water storage tank that is credited with ensuring the appropriate initial conditions and therefore, the heat exchanger and recirculation pump are not in the scope of License Renewal for the Fire Protection regulated event.

However, it is the staff's position that these components are required to maintain temperature in the fire water tank above 35 °F. The Fire Hazard Analysis Report (FHAR), Section 8.1.2, Fire Suppression Water System, states that "...the temperature of the contained water supply is greater than 35 °F every 24 hours during October through March" which is verified using surveillance. These components should not be excluded from the fire water system on the basis that they are not required to function to suppress a fire; rather they should be included to support the tank's primary function of maintaining a useable inventory of water at the appropriate temperature to avoid freezing.

A second teleconference was held on November 1, 2011, to discuss that the deletion of these components was not consistent with the current licensing basis (CLB).

Issue:

It is not clear to the staff how the removal of these fire protection system components is consistent with the FHAR associated with the original Davis-Besse fire protection SERs and the plant's CLB.

The staff lacks sufficient information to understand the basis of the applicant's proposal that these components are not included within scope per 10 CFR 54.4(a)(3). The staff believes that these fire protection SSCs are required for compliance to 10 CFR 50.48 and are subject to an AMR as shown in 10 CFR 54.21.

The revised LRA does not appear to demonstrate that the aging effects associated with the fire protection system are adequately managed, so that there is reasonable assurance that the system components will perform their intended functions in accordance with the CLB, during the period of extended operation as required by 10 CFR 54.4(a)(3).

Requests:
Justify how the fire water storage tank will be maintained greater than 35 °F at all times without the heat exchanger or provide an appropriate aging management program to manage aging for the original component and their subcomponents inclusive of all applicable aging effects.

If components are excluded and other methods are used for the tank's primary temperature function, then describe the procedure steps that would be used to maintain the fire water storage tank level and temperature.

For example, with the heat exchanger tube failure a "feed and bleed" procedure to prevent tank overflow would be required, or without a heat exchanger for the tank an operational procedure would be needed to create recirculation in the tank and provide flow/heating to the tank and keep temperature greater than 35 °F.

When describing these procedures, please document any steps that would require operator action. Please also include a complete list of the SSCs that are part of the procedure including their material, environment, and aging effect that would require age management to operate and support the tank's primary temperature function. Include the aging management program that will be used and list the associated aging management review (AMR) line items documenting their management and associated inspection methods and parameters to be monitored.

Specific to a feed and bleed procedure, please identify the tank temperature upper supply limit to prevent a potential overheating of the tank and net positive suction head issue on the downstream pumps. Please also identify the volume and temperature of feed required to control/maintain the tank's temperature adequately. Also include any isolation steps that would be required to prevent loss of inventory from the tank to the immediate surroundings with a loss of steam pressure to the heat exchanger.

The applicant's most recent response includes the recirculation pump as part of the proposed omitted license renewal scope. Please include any additional AMR line items related to that proposed deletion such as piping components and elements that then would no longer be age managed in the LRA.

Please document the FHAR sections that would support removal of these components while retaining the primary function of adequate fire water supply temperature and maintaining consistency with the plant's CLB.

November 8, 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 LICENSE RENEWAL APPLICATION (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,
/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

DISTRIBUTION:
See next page

ADAMS Accession No.: ML11306A141 *concurrence via e-mail

OFFICE	LA:DLR/RPB1*	PM:DLR/RPB1	BC:DLR/RPB1	PM:DLR/RPB1
NAME	SFigueroa	SCuadrado	DMorey	SCuadrado
DATE	11/7/2011	11/8/2011	11/7/2011	11/8/2011

OFFICIAL RECORD COPY

Letter to Barry S. Allen from Samuel Cuadrado-De Jesús dated November 8, 2011

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

DISTRIBUTION:

HARD COPY:
DLR R/F

E-MAIL:
PUBLIC
RidsNrrDlrResource
RidsNrrDlrRpb1 Resource
RidsNrrDlrRpb2 Resource
RidsNrrDlrRer1 Resource
RidsNrrDlrRer2 Resource
RidsNrrDlrRerb Resource
RidsNrrDlrRpob Resource
RidsNrrDeEvib Resource
RidsNrrDeEpnb Resource
RidsNrrDeEsgb Resource
RidsNrrDeEmcb Resource
RidsNrrDeEeeb Resource
RidsNrrDraAfpb Resource
RidsNrrDraApla Resource
RidsNrrDssSrxb Resource
RidsNrrDssSbpb Resource
RidsNrrDssScvb Resource
RidsOgcMailCenter Resource

PCooper
SCuadrado
DWrona
DMorey
M. Mahoney
EMiller
ICouret, OPA
TRiley, OCA
BHarris, OGC
VMitlyng, RII
JCameron, RIII