FirstEnergy

Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, Ohio 43449-9760

Guy G. Campbell
Vice President - Nuclear

419-321-8588
Fax: 419-321-8337

Docket Number 50-346

License Number NPF-3

Serial Number 2737

November 15, 2001

United States Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555-0001

Subject: Supplemental Information Regarding License Amendment Application to Revise Technical Specification (TS) 3/4.4.5 Steam Generator Tube Repair Roll Requirements (License Amendment Request No. 01-0004; TAC No. MB2107)

Ladies and Gentlemen:

On May 22, 2001, the FirstEnergy Nuclear Operating Company (FENOC) submitted an application for an amendment to the Davis-Besse Nuclear Power Station (DBNPS), Unit Number 1, Operating License Number NPF-3, Appendix A Technical Specifications, regarding Technical Specification (TS) 3/4.4.5, Steam Generators. The proposed amendment (DBNPS letter Serial Number 2705) would revise the once-through steam generator tube repair roll requirements.

On October 11, 2001, during a conference call with the NRC Staff, FENOC was requested to provide a written response to the issues discussed. Enclosure 1 provides the response to this request for additional information and to follow up information requested by the NRC staff on October 16, 2001.

This information does not affect the conclusions stated in the previously submitted license amendment application that there is no adverse impact on nuclear safety and that the proposed amendment involves no significant hazards consideration.

A001

Should you have any questions or require additional information, please contact Mr. David H. Lockwood, Manager - Regulatory Affairs, at (419) 321-8450.

Very truly yours,

MAR/s

Enclosures

cc: J. E. Dyer, Regional Administrator, NRC Region III
S. P. Sands, NRC/NRR Project Manager
D. J. Shipley, Executive Director, Ohio Emergency Management Agency, State of Ohio (NRC Liaison)
D. S. Simpkins, NRC Region III, DB-1 Resident Inspector
Utility Radiological Safety Board

Docket Number 50-346
License Number NPF-3
Serial Number 2737
Enclosure 1

SUPPLEMENTAL INFORMATION IN SUPPORT OF
APPLICATION FOR AMENDMENT
TO
FACILITY OPERATING LICENSE NPF-3
DAVIS-BESSE NUCLEAR POWER STATION
UNIT NUMBER 1

Attached is supplemental information for Davis-Besse Nuclear Power Station (DBNPS), Unit Number 1 Facility Operating License Number NPF-3, License Amendment Request Number 01-0004 (DBNPS Serial Number 2705, dated May 22, 2001).

I, Guy G. Campbell, state that (1) I am Vice President - Nuclear of the FirstEnergy Nuclear Operating Company, (2) I am duly authorized to execute and file this certification on behalf of the Toledo Edison Company and The Cleveland Electric Illuminating Company, and (3) the statements set forth herein are true and correct to the best of my knowledge, information and belief.

By: ______________________________
Guy G. Campbell, Vice President - Nuclear

Affirmed and subscribed before me this 15th day of November, 2001.

Nora L. Flood
____
Notary Public, State of Ohio - Nora L. Flood
My commission expires September 4, 2002.

Response to NRC Request for Additional Information
License Amendment Request 01-0004, TAC Number MB2107:
Steam Generator Tube Repair Roll Requirements
Davis-Besse Nuclear Power Station

NRC Question 1

Discuss degradation in the tube roll joints in the upper and lower tubesheets or reference submitted documents that contain this information.

Response to Question 1

Upper Tubesheet

During the Davis-Besse Nuclear Power Station (DBNPS) tenth refueling outage (10RFO) eddy current inspection in 1996, a single axial indication was detected in the upper roll transition of tube 58-119 in once-through steam generator (OTSG) #2. This tube was pulled and the roll transition region was examined in the laboratory. Inner Diameter (ID) initiated primary water stress corrosion cracking (PWSCC) was confirmed by this laboratory examination. The largest crack was 0.092 inches in length with a maximum depth of 78% through-wall (TW). Four other smaller axial PWSCC cracks were observed which were below the eddy current testing (ECT) detection threshold ranging from 7% to 43%TW and 0.014 to 0.05 inches long. Additionally, very shallow (3% TW) inner diameter intergranular attack (IGA) was observed in a circumferential band approximately 0.06 inches wide around the tube section examined.

Micro-hardness, residual stress, and cold work data from tube 58-119 was compared to that of re-rolled mock-up specimens with and without post-roll stress relief heat treatments. This comparison indicated the tube 58-119 was rolled in to the tubesheet without a post-roll stress relief heat treatment. The lack of a post-roll stress relief treatment leads to increased susceptibility to PWSCC relative to almost all other roll transitions at the DBNPS which are stress relieved. Based on fabrication records, there are currently six other roll transitions at the DBNPS that have been identified as being non-stress relieved.

During 1998 in 11RFO, as identified in Information Notice 98-27, the DBNPS observed axial tube end ID initiated cracking in the heat affected zone of the tube to upper tubesheet weld using rotating coil eddy current. Five tubes with these flaws were discovered in 1998. Four of the tubes with tube end cracking were repaired with re-roll

and the fifth tube plugged due to a volumetric defect. An additional eight indications of this type damage were discovered in 2000 during 12RFO. During 12RFO all tubes with these indications were removed from service by plugging, including those re-rolled in 11RFO. For the four tubes that were re-rolled in 11RFO, their tube end indications were again inspected with eddy current during 12RFO. The 12RFO data was compared to the eddy current data obtained during 11RFO and showed no discernable change in the flaws during one cycle of operation. The 12RFO inspection of the repair rolls installed in 11RFO did not identify any degradation following one cycle of service.

To date, no circumferential crack-like indications have been identified in the DBNPS OTSG upper tubesheet roll transitions or tube ends.

Lower Tubesheet

To date, there have been no indications of degradation in the rolled joints of the lower tubesheet. Although rotating probe eddy-current inspections of the rolled joints have not been performed, an analysis has been prepared to determine when indications might appear. The analysis is based on the time (duration, EFPY) at which degradation was first observed in the upper tubesheet and the differences in temperature between the upper and lower tubesheets. Crack growth models are directly affected by temperature and show that the lower the temperature, the less susceptible the material is to PWSCC. Therefore, it is possible to predict the initiation of degradation in the lower tubesheet by adjusting time to observed degradation in the upper tubesheet by the difference in normal operating temperature between the upper and lower tubesheets.

If the assumption is made that PWSCC defects in the lower tubesheet (cold leg) are subject to the same non-destructive examination (NDE) detection limitations as in the hot leg (upper tubesheet), the estimated time for initiation of PWSCC in the DBNPS OTSG cold legs is at approximately 60 EFPY. Based on the analysis, PWSCC in the lower tubesheet tube-to-tubesheet joint should not be a concern in the operating lifetimes of OTSGs.

NRC Question 2

The NRC staff has approved a similar reroll repair for Oconee Units 1, 2, and 3 and Arkansas Nuclear One, Unit 1 (ANO-1). During the review of the Oconee units and ANO-1 amendment requests, the NRC staff requested additional information from Duke Energy Corporation and Entergy regarding their respective reroll repair method. The NRC staff found that the responses from Duke Energy and Entergy are acceptable. FirstEnergy needs to confirm that its roll repair method and requirements will be consistent with the responses from Duke Energy and Entergy. In addition, FirstEnergy

needs to provide technical basis for any deviation from the responses of Duke Energy or Entergy.

Entergy submitted the relevant documents on September 28, 2000 and February 19, 2001. Duke Energy submitted the relevant documents on September 12, October 4, October 26, November 10, and December 8, 2001.

In addition, the NRC's Safety Evaluation Report for ANO-1 with respect to TAC No. MB2107 for steam generator tube repair roll requirements describes regulatory commitments which the DBNPS should review and consider since the NRC's review of Topical Report BAW-2374 may not be completed in time for DBNPS's needs.

Response to Question 2

References

1) Letter, Duke to USNRC, Response to NRC Questions on License Amendment Request for Technical Specification 5.5.10.e.6 and Topical Report BAW-2303P, Revision 4, dated October 26, 2000

2) Letter, Duke to USNRC, Response to NRC Questions on License Amendment Request for Technical Specification 5.5.10.e.6 and Topical Report BAW-2303P, Revision 4, dated November 10, 2000

3) Letter, Duke to USNRC, Supplemental Information Regarding License Amendment Request for Technical Specification 5.5.10.e.6 and Topical Report BAW-2303P, Revision 4, dated December 8, 2000

4) Letter, Entergy to USNRC, ANO-1 OTSG Reroll Amendment Supplemental Information, Docket No. 50-313, License No. DPR-51, dated February 19, 2001

5) Letter, USNRC to Entergy, ANO-1 - Issuance of Amendment re: the Use of the Reroll Repair Process for Steam Generator Tubes (TAC No. MB0097), dated March 28, 2001.

The DBNPS has reviewed the listed references for applicability. With the exception of those items clarified herein, the responses to the questions contained in the references are applicable to the DBNPS. The September 28, 2000 submittal for Entergy and the September 12, 2001 submittal for Duke Energy were the initial license submittals and do not contain responses to requests for additional information (RAIs). The October 4, 2000 Duke Energy submittal was providing the non-proprietary version of the repair roll

topical report and also does not contain any responses to RAIs. Therefore, these submittals have not been reviewed for applicability to DBNPS.

1) Recognizing that the NRC Staff review is still ongoing with regards to the long term acceptability of eliminating large break loss of coolant accidents (LBLOCAs) from the design basis of B&W steam generators, but with anticipation of being able to apply Topical Report BAW-2374, "Risk-Informed Assessment of Once-Through Steam Generator Tube Thermal Loads Due to Breaks in Reactor Coolant System Upper Hot Leg Large Bore Piping," Revision 1, to the DBNPS Once Through Steam Generators in the future, the DBNPS will commit to the additional interim reporting conditions as listed below if NRC review and approval of Topical Report BAW-2374, Revision 1, is not completed prior to approval of the subject license amendment application. These reporting requirements will be maintained as regulatory commitments similar to that done for ANO-1.

The additional interim reporting requirements are as follows:

1. Following each inservice inspection of steam generator tubes but prior to returning the DBNPS steam generators to service, the DBNPS will verbally notify the NRC of the following:

   a. Indication of circumferential cracking inboard of the roll repair.

   b. Indication of circumferential cracking in the original roll or heat affected zone adjacent to the tube-to-tubesheet seal weld if no re-roll is present.

   c. Determination of the best-estimate total leakage that would result from an analysis of the limiting LBLOCA based on circumferential cracking in the original tube-to-tubesheet rolls, tube-to-tubesheet re-roll repairs, and heat affected zones of seal welds as found during each inspection.

2. Demonstrate that the primary-to-secondary leakage following a LBLOCA, as described in Appendix A to Topical Report BAW-2374, Revision 1, is acceptable, based on the as-found condition of the steam generators. This is required to demonstrate that adequate margin and defense-in-depth are maintained. For the purpose of this evaluation, acceptable means a best estimate of the leakage expected in the event of a LBLOCA that would not result in a significant increase of radionuclide release (e.g., in excess of 10 CFR 100 limits). A summary of this evaluation shall be provided to the NRC within 3 months following completion of steam generator tube inservice inspection.

The DBNPS also recognizes that further NRC review of Topical Report BAW-2374, may necessitate modification to this commitment and may involve additional action on the DBNPS's part to comply with the final NRC conclusion on the topical report.

In addition, the DBNPS has performed a review to assure that its plant-specific Emergency Operating Procedures (EOPs) are consistent with the descriptions in Topical Report BAW-2374 in regard to the key operator actions for mitigation of the accident sequence of concern. The key operator actions are the transfer of the Emergency Core Cooling System (ECCS) suction from the borated water storage tank to the containment sump, and the isolation of the secondary system to minimize any primary-to-secondary leakage. The DBNPS has also confirmed that its EOPs are consistent with the Babcock &Wilcox Owners Group's November 27, 2000, letter with respect to compliance with 10 CFR 50.46.

2) In Reference 1, the licensee (Duke) states that they will limit the application of repair rolls to two repair rolls per tube, not to exceed 50 lbs compressive load change. The DBNPS has elected to evaluate each additional application of repair roll on a case-by-case basis as provided in Section 9.0 item (f) and Section 2.1 of Topical Report BAW-2303P (i.e., 50 lbs will not be a maximum compressive load applicable as a standard maximum per tube). This evaluation will ensure that the compressive loads due to the number of installed rerolls in a tube when combined with the maximum compressive load at that tube location from the limiting transient (plant heatup) will not exceed the compressive loading design limit for the tubes with a margin of 50 lbs. This approach provides the DBNPS additional flexibility to preserve inservice tubes while managing compressive loads resulting from the repair roll process, as was also discussed under Reference 4 and approved for use at ANO-1. It should be noted that the low level heatup which is identified as producing the most limiting compressive tube loads is not performed at the DBNPS, and therefore there is additional compressive tube load margin.

3) In the ANO-1 submittal under Reference 4, there is discussion on observed repair roll degradation. The DBNPS performed inspections of installed repair rolls that were in service for one cycle (21 EFPM) and observed no degradation. All inservice repair rolls will continue to be inspected for degradation during each future inspection of the original OTSGs as presently required by Technical Specification Surveillance Requirement 4.4.5.9.

Docket Number 50-346
License Number NPF-3
Serial Number 2737
Enclosure 2

COMMITMENT LIST

THE FOLLOWING LIST IDENTIFIES THOSE ACTIONS COMMITTED TO BY THE DAVIS-BESSE NUCLEAR POWER STATION (DBNPS) IN THIS DOCUMENT. ANY OTHER ACTIONS DISCUSSED IN THE SUBMITTAL REPRESENT INTENDED OR PLANNED ACTIONS BY THE DBNPS. THEY ARE DESCRIBED ONLY FOR INFORMATION AND ARE NOT REGULATORY COMMITMENTS. PLEASE NOTIFY THE MANAGER – REGULATORY AFFAIRS (419-321-8450) AT THE DBNPS OF ANY QUESTIONS REGARDING THIS DOCUMENT OR ANY ASSOCIATED REGULATORY COMMITMENTS.

COMMITMENTS	DUE DATE
1. Pending resolution of the LBLOCA issue, the DBNPS will verbally notify the NRC of the following: a. Indication of circumferential cracking inboard of the roll repair. b. Indication of circumferential cracking in the original roll or heat affected zone adjacent to the tube-to-tubesheet seal weld if no re-roll is present. c. Determination of the best-estimate total leakage that would result from an analysis of the limiting LBLOCA based on circumferential cracking in the original tube-to-tubesheet rolls, tube-to-tubesheet re-roll repairs, and heat affected zones of seal welds as found during each inspection.	Following each inservice inspection of steam generator tubes but prior to returning the DBNPS steam generators to service
2. Pending resolution of the LBLOCA issue, the DBNPS will demonstrate that the primary-to-secondary leakage following a LBLOCA, as described in Appendix A to Topical Report BAW-2374, Revision 1, is acceptable, based on the as-found condition of the steam generators. For the purpose of this evaluation, acceptable means a best estimate of the leakage expected in the event of a LBLOCA that would not result in a significant increase of radionuclide release (e.g., in excess of 10 CFR 100 limits). A summary of this evaluation shall be provided to the NRC.	Within 3 months following completion of steam generator tube inservice inspection