Commonwealth Edison
1400 Opus Place
Downers Grove, Illinois 60515

December 17, 1993

Dr. Thomas E. Murley, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Document Control Desk

Subject: Response to the Request for Additional Information Regarding the Byron/Braidwood Steam Generator Tube Sleeving Proposed License Amendment

Byron Station Units 1 and 2
(NPF-37/66; NRC Docket Nos. 50-454/455)

Braidwood Station Units 1 and 2
(NPF-72/77; NRC Docket Nos. 50-456/457)

References: See Attachment A

Dear Dr. Murley:

R. R. Assa's letter dated December 2, 1993 (Reference 1) transmitted a request for additional information related to the Byron/Braidwood proposed license amendment regarding steam generator tube sleeving (Reference 2). The enclosed attachments provide CECo's response to this request for additional information.

Attachment B addresses the questions pertaining to Topical Report BAW-2045PA, Revision 1, "Recirculating Steam Generators Kinetic Sleeve Qualification for 3/4 Inch Tubes". Attachment B also contains a copy of J. H. Taylor's letter to H. F. Conrad dated December 10, 1993, accompanying affidavit, and B&W Nuclear Technologies (BWNT) Document #51-1228682-00, "Evaluation of BWNT's Kinetic Sleeving Process". This report has been classified as proprietary by BWNT. Accordingly, it is respectfully requested that the subject document be withheld from public disclosure in accordance with 10 CFR Section 2.790 of the Commission's regulations.

Attachment C addresses the questions pertaining to WCAP-13698, Revision 1, "Laser Welded Sleeves for 3/4 Inch Diameter Tube Feedring-Type and Westinghouse Preheater Steam Generators", dated May 1993 (Proprietary). Attachment C also contains a Westinghouse authorization letter (CAW-93-554), accompanying affidavit, Proprietary Information Notice, and Copyright Notice.

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Please note that the information presented in Attachment C contains information proprietary to Westinghouse Electric Corporation and is supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.790 of the Commission's regulations. Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.790 of the Commission's regulations.

Correspondence with respect to the proprietary aspects of the items in Attachment C or the supporting Westinghouse Affidavit should reference CAW-93-554 and should be addressed to N. J. Liparulo, Manager of Nuclear Safety & Regulatory Activities, Westinghouse Electric Corporation, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

To the best of my knowledge and belief, the statements contained in this document are true and correct. In some respects these statements are not based on my personal knowledge, but on information furnished by other CECo employees, contractor employees, and/or consultants. Such information has been reviewed in accordance with company practice, and I believe it to be reliable.

Please address any comments or questions regarding this matter to this office.

Respectfully,

Joseph A. Bauer

Joseph A. Bauer
Nuclear Licensing Administrator

JAB/gp

Attachments

cc: R. R. Assa, Braidwood Project Manager - NRR
H. Peterson, SRI - Byron
S. G. Dupont, SRI - Braidwood
B. Clayton, Branch Chief - Region III
Office of Nuclear Facility Safety - IDNS

ATTACHMENT A

REFERENCES

1. R. R. Assa Letter to D. L. Farrar dated December 2, 1993 transmitting a Request for Additional Information Regarding the Byron/Braidwood Steam Generator Tube Sleeving Proposed License Amendment (TAC NOS. M87229, M87230, M87227, and M87228)

2. J. A. Bauer Letter to T. E. Murley dated August 13, 1993 transmitting a Byron/Braidwood Proposed License Amendment Regarding Steam Generator Tube Sleeving Methodology

3. WCAP-13698 Revision 1, "Laser Welded Sleeves For 3/4 Inch Diameter Tube Feedring-Type And Westinghouse Preheater Steam Generators"

4. Topical Report BAW-2045PA Revision 1, "Recirculating Steam Generators Kinetic Sleeve Qualification For 3/4 Inch Tubes"

5. B&W Nuclear Technologies (BWNT) Document #51-1228682-00, "Evaluation of BWNT's Kinetic Sleeving Process"

ATTACHMENT B

1. RESPONSE TO QUESTIONS PERTAINING TO TOPICAL REPORT BAW-2045PA, REVISION 1, "RECIRCULATING STEAM GENERATORS KINETIC SLEEVE QUALIFICATION FOR 3/4 INCH TUBES"

2. BWNT LETTER FROM J. H. TAYLOR TO H. F. CONRAD TRANSMITTING BWNT DOCUMENT #51-1228682-00, "EVALUATION OF BWNT'S KINETIC SLEEVING PROCESS"

3. BWNT AFFIDAVIT

4. BWNT DOCUMENT #51-1228682-00, "EVALUATION OF BWNT'S KINETIC SLEEVING PROCESS"

RESPONSE TO QUESTIONS PERTAINING TO TOPICAL REPORT BAW-2045PA, REVISION 1, "RECIRCULATING STEAM GENERATORS KINETIC SLEEVE QUALIFICATION FOR 3/4 INCH TUBES"

NRC QUESTION

1. The report covers Westinghouse Model "D" steam generators (SG); however, the bounding conditions in the Westinghouse report (WCAP-13698, Revision 1, page 3-22) are different than Babcock and Wilcox's (B&W), (page 4-4). Do the conditions stated in the report bound the conditions at Byron and Braidwood.

   RESPONSE: The B&W and Westinghouse reports, noted above, specify the same primary and secondary design pressures: 2485 psig (2500 psia) - primary and 1285 psig (1300 psia) - secondary. These values are consistent with the Byron and Braidwood (B/B) UFSAR. Other parameter values specified on the subject pages for both the Westinghouse and B&W reports, conservatively bound the assumed parameter values as noted in B/B UFSAR Table 5.4-3, "Steam Generator Design Data", Table 15.6-2, "Input Parameters Used in the ECCS Analysis", and Table 15.0-3, "Nominal Values of Pertinent Plant Parameters Utilized in the Accident Analysis" (see attached tables).

   As noted by the question, Westinghouse and B&W list some dissimilar parameters in the design section of the referenced pages. Westinghouse has used more conservative primary to secondary parameters for it's calculations. These assumed parameters (such as 235 psig secondary pressure during design primary pressure conditions and 465 psig primary pressure during design secondary pressure conditions) are included merely to illustrate the origin of the conservative primary to secondary (2250 psig) and secondary to primary (820 psig) differential pressures.

   In summary, CECo's review confirmed that both the Westinghouse and B&W reports are consistent with or bound the applicable parameter values specified in the Byron and Braidwood UFSAR.

2. There is a change in the postweld heat treatment (PWHT) which may require re-qualification. This issue must be addressed.

   RESPONSE: Babcock and Wilcox is providing information to address this issue in three separate documents:

   a. BWNT Document #51-1228682-00, "Evaluation of BWNT's Kinetic Sleeving Process" was previously transmitted to the NRC by J. H. Taylor's letter to H. F. Conrad dated December 10, 1993. This report provides the results of the root cause evaluation of the McGuire Unit 1 1993 tube failure event as well as guidelines for evaluating in-service sleeves. A copy of this document is enclosed for your convenience.

b. BWNT will provide CECo and the NRC with a report that will establish correlations between post weld heat treatment temperatures and tube micro-hardness. It is CECo's understanding that this report will use the McGuire evaluation and independent test results to develop relative tube material corrosion susceptibility thresholds at lower stress relief temperatures. Using the independent test information, B&W will also develop stress relief criteria that is independent of material makeup. In conclusion, the report will arrive at specific recommendations regarding post weld heat treatment for the Byron and Braidwood steam generators, based on tube heat data provided by CECo. BWNT has indicated that this report will be published on or about December 24, 1993.

c. BWNT will then revise the report (from item b above) to incorporate x-ray diffraction information. X-ray diffraction is an indicator of residual stresses and is used to make determinations related to the quality of the stress relief process. The revision to this report will be issued on or about January 15, 1994. In addition, B&W will provide confirmatory corrosion test data later in the summer of 1994.

TABLE 5.4-3

STEAM GENERATOR DESIGN DATA

Design pressure, reactor coolant side, psig	2485
Design pressure, steam side, psig	1185
Design temperature, reactor coolant side, °F	650
Design temperature, steam side, °F	600
Total heat transfer surface area, ft^2	48,300
Maximum moisture carryover, wt percent	0.25
Overall height, ft-in	67-8
Number of U-tubes	4578
U-tube nominal diameter, in.	.750
Tube wall nominal thickness, in.	.043
Number of manways	4
Inside diameter of manways, in.	16
Number of inspection ports	4
Design fouling factor	0.00005
Preheat section	0.00010

TABLE 15.6-2

INPUT PARAMETERS USED IN THE ECCS ANALYSIS

Licensed core power[a], (MWt)	3411
Peak linear power, includes 102% factor (kW/ft)	12.88
Total peaking factor, F_Q^T	2.32
Axial peaking factor, F_Z	1.4968
Power shape	
Large break	Chopped cosine
Small break	See Figure 15.6-48
Fuel assembly array	Optimized 17x17
Accumulator water volume, nominal (ft^3/accumulator)	950
Accumulator tank volume, nominal (ft^3/accumulator)	1350
Accumulator gas pressure, minimum (psia)	600
Safety injection pumped flow	See Figures 15.6-21 and 15.6-47
Containment parameters	See Sec. 6.2
Initial loop flow (lb/sec)	9792
Vessel inlet temperature (^{o}F)	555.4
Vessel outlet temperature (^{o}F)	616.9
Average Reactor coolant pressure (psia)	2280
Steam pressure (psia)	990
Steam generator tube plugging level (%)	0

(a) Two percent is added to this power to account for calorimetric error.

TABLE 15.0-3

Nominal Values of Pertinent Plant Parameters
Utilized In The Accident Analyses*

Thermal output of NSSS (MWt)[a]	3425
Core inlet temperature (^oF)	559.3
Vessel average temperature (^oF)	588.4
Reactor Coolant System pressure (psia)	2250
Reactor coolant flow per loop (gpm)	97,600
Total Reactor Coolant flow (10^6 lb/hr)	145.1
Steam flow from NSSS (10^6 lb/hr)	15.13
Steam pressure at steam generator outlet (psia)	990
Maximum steam moisture content (%)	0.25
Assumed feedwater temperature at steam generator inlet (^oF)	440
Average core heat flux (Btu/hr-ft^2)	197,200

*For accident analyses using the Improved Thermal Design Procedure

[a]See Table 15.0-2