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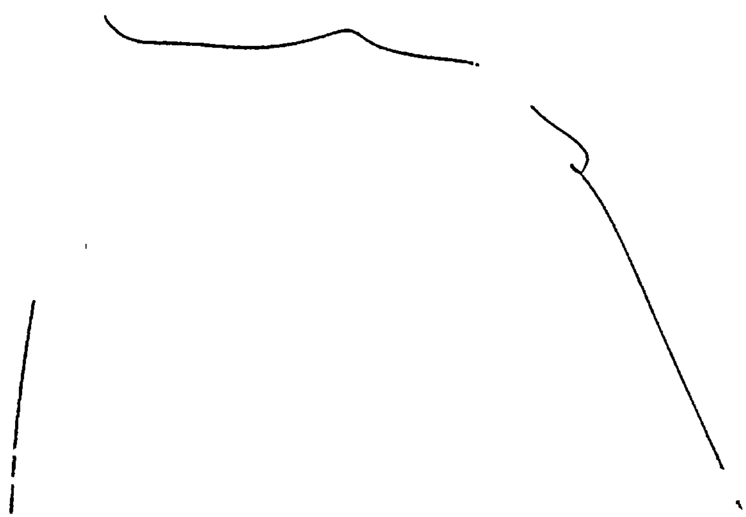

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AMERICAN ELECTRIC POWER

AEP:NRC:1129A

Donald C. Cook Nuclear Plant Unit 1
Docket No. 50-315
License No. DPR-58
ADDITIONAL INFORMATION FOR TECHNICAL SPECIFICATIONS CHANGE
TO ALLOW SLEEVING THE STEAM GENERATOR TUBES

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

Attn: T. E. Murley

October 9, 1990

Dear Dr. Murley

The purpose of this letter is to provide the additional information that was requested by your staff concerning our license amendment request to obtain authorization to use steam generator sleeves to repair defective steam generator tubes. Specific questions concerning the Combustion Engineering report were asked on September 5 and 10. Attachment 1 to this letter contains our responses to these questions as requested by your staff on September 21, 1990. Attachment 2 contains a proprietary Combustion Engineering Report TR-MCC-153 and an accompanying affidavit. It is requested that the information which is proprietary to Combustion Engineering, Incorporated be withheld from public disclosure in accordance with 10 CFR Section 2.790 of the Commission's regulations.

This document has been prepared following Corporate procedures that incorporate a reasonable set of controls to ensure its accuracy and completeness prior to signature by the undersigned.

Sincerely,

M. P. Alexich
Vice President

MPA/eh

A001
1/1

Attachments

cc: D. H. Williams, Jr.
A. A. Blind - Bridgman
J. R. Padgett
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ATTACHMENT 1 TO AEP:NRC:1129A

ADDITIONAL INFORMATION FOR TECHNICAL SPECIFICATION CHANGE

TO ALLOW SLEEVING THE STEAM GENERATOR TUBES

(RESPONSES TO NRC QUESTIONS)

QUESTION NO. 1

Unless it can be demonstrated otherwise, the NRC staff position is that a post weld heat treatment should be utilized to optimize corrosion resistance (Section 4.3). Either commit to a post weld heat treatment or provide adequate justification for not doing so.

RESPONSE NO. 1

As detailed in Section 9.0 of CEN-313-P, C-E has installed approximately 2950 welded sleeves in five operating units since 1984. With the exception of approximately 15 sleeves at Prairie Island Unit 1 in 1990, all of these sleeves were installed using essentially the same process without post weld heat treatment (PWHT). Of these 2950 sleeves, about 175 have seen more than four years, and over 700 more than three years, of service. With the replacement of the Ringhals Unit 2 steam generators in 1989, over 1800 sleeves remain in service at the present time. Other than the two sleeves described in Section 9.2, no leaks or unacceptable indications have been reported from any of the sleeved tubes.

At the time the Ringhals Unit 2 steam generators were removed, six sleeved tubes installed in 1985 and 1986 were removed for examination by a consortium of European utilities at the Belgium laboratory, Laborelec. Results of this evaluation, including metallographic examination, indicated no evidence of either primary or secondary side corrosion associated with the sleeve or the tube in the vicinity of the sleeve to tube joint.

This experience along with corrosion tests described in Section 6.2 has indicated that the sleeve to tube joint could be installed without post weld heat treatment where the service life was expected to be short and/or where PWSCC had not been a factor in the failure of steam generator tubes. However, in order to apply the sleeve to units where primary water stress corrosion cracking (PWSCC) had been a problem C-E has performed accelerated testing to quantify the effects of sleeves installed with and without PWHT.

Because of the Iconel 690 sleeve's superior resistance to PWSCC, the area of the sleeve to tube joint requiring evaluation is the region of the original Inconel 600 tube immediately above the joint. The decision whether or not to

heat treat the weld is based primarily on whether a reduction in residual stress in this region will be of benefit. This determination is based on the original tube's susceptibility to PWSCC and the stresses imposed on this area by the sleeve installation process and operating conditions.

Towards this end, C-E conducted accelerated stress corrosion cracking tests of sleeves installed in highly susceptible tubing with and without post weld heat treatment. As with any accelerated test a method was required to correlate the results with actual operating conditions. Attachment 2 (Report TR-MCC-153) describes these tests and explains the rational upon which this correlation was made. At the time these tests were being conducted, C-E was working closely with a utility which had experienced PWSCC after less than two EFPYs of operation. The approach used in the report was not only reviewed and approved by the utility but also by its independent consultant who has conducted numerous studies on PWSCC. It is this same approach being used here to evaluate the use of PWHT as an option at Cook Nuclear Plant.

The following welded sleeve/tube joint life assessment for installed sleeves is based on a comparison of the stresses associated with the sleeve joint and those associated with the original tube roll transition and its life. By ratioing the function of total stress in the roll transition with that in the tube immediately above the upper weld joint, an assessment of sleeve life can be made. The life of the roll transition is defined as the accumulated time of operation at the last shutdown at which no primary water stress corrosion cracking (PWSCC) was found. At Cook Nuclear Plant, the last shutdown prior to the first installation of plugs due to roll transition cracking in 1987 had accounted for 7.8 EFPYs of operation.

The time for primary water stress corrosion cracking to occur has been empirically determined to be a function of the following relationship (Ref. 1):

$$L = A\,S^{n}\,e^{-Q/RT}$$ where:

L = time to cracking (life)
A = constant
S = total stress (applied + residual)
n = empirically determined exponent
(-4.0 to -5.4, best value = -4.3) (Ref. 2)
Q = empirically determined activation energy for PWSCC
(a function of tube metallurgical condition)
R = gas constant
T = temperature

For the purposes of this assessment, residual stresses in the roll transition have been conservatively taken as 67 Ksi (other estimates have ranged as high as 100 Ksi) (Ref. 3). Section 2.0 of Attachment 2 also addresses residual stress measurements. Based on work in 3/4" tubes, it has been determined that the maximum residual stress at the weld joint is 65 Ksi, oriented in the axial direction (Ref. 4). For the purpose of this analysis it is assumed that the residual stress in the sleeve and the roll transition are the same at 65 Ksi.

The applied stress will be dependent on the specific steam generator; however, in general the applied axial stress on the I.D. of the tube at the weld joint is compressive whereas the stress in the roll transition is tensile. The compressive stress in the weld joint is due to the thermal load imposed on the tube by the higher temperature and coefficient of thermal expansion of the Inconel 690 sleeve. One case in which the operating temperature was higher than the normal operatng temperature for Cook Nuclear Plant resulted in an applied stress of -9.1 Ksi in the tube at the joint. Reference 3 gives the operating stress in 7/8" X .050" wall roll transition tube as +11 Ksi. Using the somewhat more conservative values of -5 Ksi for compressive stress in the weld heat-affected zone and 8 Ksi for the roll transition applied stress, the total stress in the roll transition (S_{roll}) and the sleeve weld joint (S_{slv}) are calculated to be:

$S_{roll} = 8 + 67 = 75$ Ksi and

$S_{slv} = -5 + 65 = 60$ Ksi

Therefore the life, L, of the as-welded sleeve joint under conditions of PWSCC can be calculated as follows:

$$L_{sleeve} = L_{tube} \times (60/75)^{-4.3} = 7.8 \times (2.6) = 20.3 \text{ (EFP) years}$$

This analysis does not take into account the additional life expected from the reduction in the hot leg temperature, initiated in 1989, from the original 599.6°F to 582°F. The lower primary side temperature could be expected to increase the life by approximately 65% (Ref. 5).

Based on our recent field experience, it is estimated that the inclusion of the PWHT step will conservatively increase sleeving time by 25%. Section 3.3 of Attachment 2 addresses the time and temperature parameters. Due to the associated added manrem exposure and cost, it is our opinion that it is more prudent to assess the need for PWHT on a case-by-case basis than to include it as a normal part of the operation. While we agree that plants with highly susceptible tubing that are operated at relatively high temperatures should employ PWHT as part of a sleeve installation process, plants that are not in that category, such as Cook Nuclear Plant, should be able to make the decision whether to incur the additional exposure and expense based on their assessment of their particular operational considerations. Moreover, the decision not to post weld heat treat at the time of sleeve installation does not preclude the application of this process during subsequent outages should considerations such as life extension deem it appropriate.

When considering the need to heat treat sleeves at Cook Nuclear Plant Unit 1, C-E evaluated each of the critical factors. The unit had seen 7.8 EFPYs of operation before any indications of PWSCC were noted. The plant operating temperatures are in a range associated with long term initiation of PWSCC. Based on this input and the analytical methods included in the test report, C-E projected a 20 to 33 EFPY life of sleeved tubes without heat treatment. This is sufficient to complete the planned life of the plant without plant life extension. As noted above, if plant life extension is later desired, the sleeved tubes could be heat treated at that time to increase their projected life.

QUESTION NO. 2

Please provide the most current data from the pure water stress corrosion cracking tests described under Section 6.2.3.

RESPONSE NO. 2

The pure water stress corrosion cracking test was initiated in the early stages of C-E's sleeve development program. In an attempt to accelerate the exposure, the specimens were exposed to higher than normal temperature and stress. At the time the test was initiated the influence of hydrogen on PWSCC had not been identified and was not included in the test design. Subsequently hydrogen was identified as a factor in correlating laboratory tests to field experience. As a result, although the test provides some degree of acceleration, it is not as originally expected. The test has been shut down and its continuation is being reviewed.

As stated in Section 6.2.3, no specimen leaks have been detected. Due to the nature of the encapsulated specimens this is determined only by monitoring the pressure on the primary side of the assembly. When the test is terminated, destructive examinations will be performed to determine whether any incipient attack has occurred.

In reviewing Section 6.2.3, it was discovered that the number of hours of actual run time was in error. Actual run time to date is 35,328 hours.

QUESTION NO. 3

Section 4.4 states that the steam generator plug material is Inconel 606 (Sic). If current material designation is Inconel 690, permission to use code case N-474-1 must be obtained from the NRC.

RESPONSE NO. 3

The material for the welded plug used in the sleeve is SFA 5.14 ERNiCr-3. The reference to Inconel 606 is an outdated, nonstandard designation.

QUESTION NO. 4

Describe your program to maintain awareness of the state of the art with regard to nondestructive testing of steam generator tubes and provide a commitment to utilize improved inspection methods as they become commercially available.

RESPONSE NO. 4

C-E maintains cognizance of technological advances in the industry and will, as described in Section 5.0, use the latest NDE technology once it has been laboratory verified to provide improved sleeve examination.

It has always been Indiana Michigan Power Company's policy to maintain an awareness of the "state-of-the-art" with regard to nondestructive testing of steam generator tubes. This is accomplished through the working relationships with our NDE contractors, original equipment suppliers, other utilities and EPRI. As techniques which would improve our inspection program are developed, verified, and commercially available, they will be utilized.

QUESTION NO. 5

Section 4 discusses coordinated phosphate chemistry. Provide a discussion of the basis for the use of Inconel 690 in current AVT secondary water chemistry and in primary water side service.

RESPONSE NO. 5

C-E considers that the resistance to primary side stress corrosion cracking (PWSCC) also provides an assessment of performance in normal secondary side AVT chemistry.

Section 4.2.2 provides some discussion on the resistance of Inconel 690 to pure water stress corrosion cracking. Additionally, further work, as reported in References 6 and 7, indicates that Inconel 690 is far superior to Inconel 600 in the primary water environment. Tests conducted on highly stressed Inconel 690 and Inconel 600 indicate at least a five fold increase in the life of Inconel 690. In environments such as that of Cook Nuclear Plant, this would represent much more than 32 EFPYs of operation. In addition, Inconel 690 has been identified as the alloy of choice for replacement steam generators at Cook Nuclear Plant Unit 2, Indian Point Unit 3, and Ringhals Unit 2.

QUESTION NO. 6

It is stated in Section 9.3 that sleeves that do not pass the ultrasonic and visual inspection for leak tightness have been accepted by licensees as "potentially leak-limiting". If this is your position, please provide the basis for this position.

RESPONSE NO. 6

As described in Sections 7.3.1 and 7.3.2 both axial pull and fatigue tests were performed on upper weld joints that had been welded for only 330 degrees of the circumference. These tests indicated that the incomplete weld had no measurable affect on the mechanical properties of the joint. For this reason, the use of a sleeve with an indication of lack of fusion for 30 degrees or less is not considered a safety concern. Due to the conservatism of the ultrasonic inspection technique, it is possible to have a weld which does not exhibit 360 degrees of fusion that will not leak or will leak at an acceptably low rate. In the particular case in which a small number of sleeves were accepted as leak limiting, utility personnel chose to accept those with small areas (<30 degrees) of ultrasonic indications after review of the UT data in anticipation that little if any leakage would occur.

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2. F. W. Pement, G. Economy, and R. J. Jacko, "Tubesheet Expansion Improvements," EPRI NP-5547, December 1987, p. 1-3.

3. Data from 1987 EPRI Workshop on PWSCC.

4. TR-MCM-153, "Summary Report Combustion Engineering Steam Generator Tube Sleeve Residual Stress Evaluation," April 1989.

5. Stein, A. A., "Prediction of the Stress Corrosion Cracking Life of Ally 600 Steam Generator Tubing in Primary Water," Paper No. 240, CORROSION 86, NACE, Houston, TX, March 1986.

6. Santarini G., et.al., "Alloy 690: Recent Corrosion Results," EPRI Alloy 690 Workshop, New Orleans, LA, April 1989.

7. Aspden, R. G., Grand, T. F., and Harrod, D. L., "Corrosion Performance of Ally 690," EPRI Alloy 690 Workshop, New Orleans, LA, April 1989.

ATTACHMENT 2 TO AEP:NRC:1129A

COMBUSTION ENGINEERING REPORT TR-MCC-153