OPPD
Omaha Public Power District
444 South 16th Street Mall
Omaha, Nebraska 68102-2247

July 28, 2000
LIC-00-0062

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-137
Washington, DC 20555-0001

Reference: Docket No. 50-285

SUBJECT: Application for Amendment of Facility Operating License No. DPR-40

Omaha Public Power District (OPPD) is submitting this "Application for Amendment of Facility Operating License" to permit the use of Leak Tight Sleeves, developed by Combustion Engineering, Inc. (CE), at Fort Calhoun Station (FCS).

OPPD proposes to amend the applicable sections in the Fort Calhoun Station Unit No. 1 Technical Specifications to allow installation of tube sleeves as an alternative to plugging to repair defective steam generator tubes. Currently, FCS Technical Specifications only allow defective tubes to be plugged and removed from service. The proposed amendment will revise the applicable Technical Specifications to permit the use of Leak Tight Sleeves, developed by Combustion Engineering, Inc. (CE), to be used at FCS. CE provides two types of Leak Tight Sleeves. The first type of repair sleeve spans the expansion transition zone of the tube at the top of the tubesheet. The second type of repair sleeve spans the degraded areas at a support location or in a free span section. The CE process has been in use since 1984 and has been implemented more than 24 times for the installation of over 4,200 sleeves.

Attachment A contains a markup reflecting the proposed changes to Sections 2.1.4, 3.1, 3.17 and Table 3-13, Technical Specifications of the Facility Operating License. Attachment B provides the "Discussion, Justification and No Significant Hazards Consideration." The detailed report on the specific qualifications of the repair sleeve for FCS application is contained in the proprietary CE report CEN-630-P, Revision 02, "Repair of ¾" O.D. Steam Generator Tubes Using Leak Tight Sleeves," Attachment D. The proprietary affidavit for Attachment D appears in Attachment C. The non-proprietary version of this report is included as Attachment E.

OPPD wishes to have sleeving as an option to repair defective steam generator tubes at FCS during the 2001 refueling outage, scheduled to begin in March 2001. Therefore, OPPD requests the Nuclear Regulatory Commission review and approve the proposed amendment on or before February 1, 2001. OPPD respectfully requests 30 days to implement the proposed specifications following NRC approval. If you have additional questions, or require further information, please contact me or members of my staff.

Sincerely,

W. G. Gates

W. G. Gates
Vice President

DLS/dls

Attachments

c: E. W. Merschoff, NRC Regional Administrator, Region IV
L. R. Wharton, NRC Project Manager
W. C. Walker, NRC Senior Resident Inspector
B. E. Casari, Director - Environmental Health Division,
State of Nebraska
Winston & Strawn

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
Omaha Public Power District	)	Docket No. 50-285
(Fort Calhoun Station	)	
Unit No. 1)	)	

APPLICATION FOR AMENDMENT
OF
FACILITY OPERATING LICENSE

Pursuant to Section 50.90 of the regulations of the U. S. Nuclear Regulatory Commission ("the Commission"), Omaha Public Power District, holder of Facility Operating License No. DPR-40, herewith requests that the Bases of Technical Specifications 2.1.4, 3.1, and 3.17 and Technical Specifications set forth in Section 3.17 and Table 3-13 of the Facility Operating License be amended to allow installation of tube sleeves as an alternative to plugging to repair defective steam generator tubes. The proposed amendment will revise the applicable Technical Specifications to permit the use of Leak Tight Sleeves, developed by Combustion Engineering, Inc. (CE), to be used at FCS. CE provides two types of Leak Tight Sleeves. The first type of repair sleeve spans the expansion transition zone of the tube at the top of the tubesheet. The second type of repair sleeve spans the degraded areas at a support location or in a free span section.

The proposed changes to the Technical Specifications are provided in Attachment A of this Application. A Discussion, Justification, and No Significant Hazards Consideration Analysis, which demonstrates the proposed changes do not involve significant hazards considerations, is appended in Attachment B. The proposed changes to Appendix A, Technical Specifications of the Facility Operating License, would not authorize any change in the types or any increase in the amounts of effluents or any change in the authorized power level of the facility.

WHEREFORE, Applicant respectfully requests that Appendix A of the Facility Operating License be amended hereto as Attachment A.

A copy of this Application, including its attachments, has been submitted to the Director - Nebraska State Division of Environmental Health, as required by 10 CFR 50.91.

OMAHA PUBLIC POWER DISTRICT

By R. L. Gary Gates
Vice President

Subscribed and sworn to before me this 28th day of July, 2000

Kristie E. Thompson
Notary Public

GENERAL NOTARY-State of Nebraska
KRISTIE E. THOMPSON
My Comm. Exp. May 1, 2004

U.S Nuclear Regulatory Commission
LIC-00-0062
Enclosure

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
Omaha Public Power District	)	Docket No. 50-285
(Fort Calhoun Station	)	
Unit No. 1)	)	

AFFIDAVIT

W. G. Gates, being duly sworn, hereby deposes and says that he is the Vice President in charge of all nuclear activities of the Omaha Public Power District; that he is duly authorized to sign and file with the Nuclear Regulatory Commission the attached information concerning the Application for Amendment of the Facility Operating License dated July 28, 2000, regarding the change to allow installation of tube sleeves as an alternative to plugging to repair defective steam generator tubes; that he is familiar with the content thereof; and that the matters set forth therein are true and correct to the best of his knowledge, information, and belief.

W. G. Gates

W. G. Gates
Vice President

STATE OF NEBRASKA)
) ss
COUNTY OF DOUGLAS)

Subscribed and sworn to me, a Notary Public in and for the State of Nebraska on this

28th day of July, 2000

Theresa Petersen
Notary Public

GENERAL NOTARY-State of Nebraska
THERESA PETERSEN
My Comm. Exp. Feb. 17, 200

LIC-00-0062
Attachment A
Requested Changes of Technical Specifications Set Forth in Appendix A of the
Facility Operating License
No. DPR-40

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (Continued)
2.1.4 Reactor Coolant System Leakage Limits (Continued)

Limiting primary to secondary leakage is important to ensure steam generator tube integrity. The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 1 gallon per minute, total). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 1 gallon per minute can readily be detected by radiation monitors. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged or repaired.

References

(1) USAR, Section 11.2.3

(2) USAR, Page G.16-1

3.0 SURVEILLANCE REQUIREMENTS

3.1 Instrumentation and Control (Continued)

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

The minimum calibration frequencies of once-per-day (heat balance adjustment only) for the power range safety channels, and once each refueling shutdown for the process system channels, are considered adequate.

The minimum testing frequency for those instrument channels connected to the Reactor Protective System and Engineered Safety Features is based on ABB/CE probabilistic risk analyses and the accumulation of specific operating history. The quarterly frequency for the channel functional tests for these systems is based on the analyses presented in the NRC approved topical report CEN-327-A, "RPS/ESFAS Extended Test Interval Evaluation," as supplemented, and OPPD's Engineering Analysis EA-FC-93-064, "RPS/ESF Functional Test Drift Analysis."

The low temperature setpoint power operated relief valve (PORV) CHANNEL FUNCTIONAL TEST verifies operability of the actuation circuitry using the installed test switches. PORV actuation could depressurize the reactor coolant system and is not required.

Calculation of the Reactor Coolant System (RCS) total flow rate by performance of a precision calorimetric heat balance once every 18 months verifies that the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate (Table 3-3, Item 15, Reactor Coolant Flow).

The frequency of 18 months reflects the importance of verifying flow after a refueling outage when the core has been altered, Steam Generator tubes plugged or repaired, or other activities, which may have caused an alteration of flow resistance.

This requirement is modified by a footnote that requires the surveillance to be performed within 24 hours after ≥95% reactor thermal power (RTP) following power escalation from a refueling outage. The footnote is necessary to allow measurement of the flow rate at normal operating conditions at power in MODE 1.

3.0 **SURVEILLANCE REQUIREMENTS**
3.17 Steam Generator Tubes

Applicability

Applies to in-service surveillance of steam generator tubes.

Objective

To ensure the integrity of the steam generator tubes.

Specifications

Each steam generator shall be demonstrated OPERABLE by performance of the following in-service inspection program.

(1) Steam Generator Sample Selection and Inspection Methods

The in-service inspection shall be performed on each steam generator on a rotating schedule. Under some circumstances, the operating conditions in one steam generator may be found to be more severe than those in the second steam generator. Under such circumstance, the sample sequence shall be modified to inspect the steam generator with the most severe conditions.

(2) Steam Generator Tube Sample Selection and Inspection

The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 3-13. The in-service inspection of steam generator tubes shall be performed according to Specification 3.17(4)(i), "Tube Inspection," and at the frequencies specified in Specification 3.17(3). The inspected tubes shall be verified acceptable per the acceptance criteria of Specification 3.17(4). When applying the exceptions of (i), (ii) and (iii) below, previous degradation, imperfections, or defects in the area of the tube repaired by sleeving are not considered an area requiring reinspection or inspection of adjacent tubes. The tubes selected for each in-service inspection shall include at least 3% of the total tubes in the steam generators and the tubes selected for these inspections shall be selected on a random basis, except:

(i) If the tube is recorded as a degraded tube, then an adjacent tube shall be inspected.

(ii) The first sample inspection during each in-service inspection of each steam generator shall include all non-plugged tubes that previously had detectable wall penetrations (>20%) and shall also include tubes in those areas where experience has indicated potential problems.

(iii) The second and third sample inspections, if required, may be less than an entire tube length inspection provided the inspection concentrates on those areas of the tube

3.0 **<u>SURVEILLANCE REQUIREMENTS</u>**
3.17 Steam Generator Tubes (Continued)

Defect means an imperfection of such severity that it exceeds the plugging or repair limit. ~~A tube containing a defect is defective.~~

Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging, or repaired by sleeving in the affected area because it may become unserviceable prior to the next inspection. Plugging or repair limit ~~and~~ is equal to 40% of the nominal tube wall thickness for the original tube wall. Sleeved tubes shall be plugged upon detection of unacceptable degradation in the pressure boundary region of the sleeve.

Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break.

Tube or Tubing means that portion of the tube or sleeve which forms the primary system to the secondary system pressure boundary.

Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

Tube Repair refers to a process that re-establishes tube serviceability. Acceptable tube repairs will be performed using the Combustion Engineering, Inc. Leak Tight Sleeve as described in a Combustion Engineering, Inc. technical report currently approved by the NRC.

Tube repair includes the removal of plugs that were previously installed as a corrective or preventive measure for the purpose of sleeving the tube. A tube inspection as defined herein is required prior to returning previously plugged tubes to service.

(ii) The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or repair all tubes exceeding the plugging or repair limit and all tubes containing through-wall cracks) required by Table 3-13.

3.0 **SURVEILLANCE REQUIREMENTS**
3.17 Steam Generator Tubes (Continued)

(5) Reporting Requirements

(i) Following each in-service inspection of steam generator tubes, the number of tubes plugged or repaired in each steam generator shall be reported to the Commission within 30 days.

(ii) The complete results of the steam generator tube in-service inspection shall be reported to the Commission within 6 months following completion of the inspection. This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall thickness penetration for each imperfection.

3. Identification of tubes plugged.

4. Identification of tubes repaired by sleeving.

(iii) Results of steam generator tube inspections which fall into Category C-3 and require prompt notification of the Commission shall be reported pursuant to Section 5.6 of the Technical Specifications prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

TABLE 3-13

STEAM GENERATOR TUBE INSPECTION

<table>
<tr><th colspan="3">1st Sample Inspection</th><th colspan="2">2nd Sample Inspection</th><th colspan="2">3rd Sample Inspection</th></tr>
<tr><th>Sample Size</th><th>Result</th><th>Action Required</th><th>Result</th><th>Action Required</th><th>Result</th><th>Action Required</th></tr>
<tr><td rowspan="8">A minimum of 300 tubes per S.G.</td><td>C-1</td><td>None</td><td>N/A</td><td>N/A</td><td>N/A</td><td>N/A</td></tr>
<tr><td rowspan="4">C-2</td><td rowspan="4">Plug or repair defective tubes and inspect additional 600 tubes in this S.G.</td><td>C-1</td><td>None</td><td>N/A</td><td>N/A</td></tr>
<tr><td rowspan="2">C-2</td><td rowspan="2">Plug or repair defective tubes and inspect additional 1200 tubes in this S.G.</td><td>C-1</td><td>None</td></tr>
<tr><td>C-2</td><td>Plug or repair defective tubes</td></tr>
<tr><td>C-3</td><td>Perform action for C-3 result of first sample</td><td>C-3</td><td>Perform action for C-3 result of first sample</td></tr>
<tr><td rowspan="3">C-3</td><td rowspan="3">Inspect all tubes in this S.G., plug or repair defective tubes and inspect 600 tubes in other S.G.

Prompt notification to NRC pursuant to specification 5.6</td><td>The second S.G. is C-1</td><td>None</td><td>N/A</td><td>N/A</td></tr>
<tr><td>The second S.G. is C-2</td><td>Perform action for C-2 result of second sample</td><td>N/A</td><td>N/A</td></tr>
<tr><td>The second S.G. is C-3</td><td>Inspect all tubes in the second S.G. and plug or repair defective tubes. Prompt notification to NRC pursuant to specification 5.6</td><td>N/A</td><td>N/A</td></tr>
</table>

N/A - Not applicable

3.0 **SURVEILLANCE REQUIREMENTS**
3.17 Steam Generator Tubes (Continued)

Basis

The surveillance requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for in-service inspection of the steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1, dated July 1975. In-service inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or in-service conditions that lead to corrosion.

In-service inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled in-service steam generator tube examinations. Plugging or repair will be required for all tubes with imperfections exceeding the plugging or repair limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Defective tubes may be repaired by a Combustion Engineering, Inc. Leak Tight Sleeve. The technical bases for sleeving repair are described in the NRC approved Combustion Engineering, Inc. technical report for Repair of 3/4" O.D. Steam Generator Tubes Using Leak Tight Sleeves.

Whenever the results of any steam generator tubing in-service inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Section 5.6 of the Technical Specifications prior to the resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

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Attachment B
Discussion, Justification and No Significant Hazards Consideration

DISCUSSION, JUSTIFICATION AND NO SIGNIFICANT HAZARDS CONSIDERATION

DISCUSSION

Background

Pressurized water reactor (PWR) steam generators have experienced tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, stress corrosion cracking, and crevice corrosion, along with other phenomena such as denting and vibration wear. Tubes that experience excessive degradation reduce the integrity of the primary-to-secondary pressure boundary. Eddy current examination is used to measure the extent of tube degradation. When the reduction in tube wall thickness reaches a calculated value commonly known as the plugging criteria, the tube is considered defective and corrective action is taken.

Currently, the corrective action taken at many PWRs, including FCS, is to remove the degraded tube from service by installing plugs at both ends of the tube. The installation of steam generator tube plugs removes the heat transfer surface of the plugged tube from service and leads to a reduction in the primary coolant flow available for core cooling.

An alternative to plugging tubes is to repair defective steam generator tubes using Combustion Engineering, Inc. (CE) Leak Tight Sleeves. Sleeving is a steam generator tube repair method where a length of tubing (sleeve) having an outer diameter slightly smaller than the inside of the steam generator tube is installed inside the parent tube spanning the degraded region. Installation of steam generator sleeves does not greatly affect the heat transfer capability or the primary coolant flow rate through the tube being sleeved; therefore, a large number of sleeves can be installed without significantly affecting the operation of the reactor coolant system. The sleeve spans the degraded section of the tube and maintains the structural integrity of the steam generator tube under normal and accident conditions and limits or prevents primary-to-secondary leakage through the sleeved section of the tube should the degradation progress through-wall. This repair method has been approved for use at several other U.S. Nuclear Power Plants, including Calvert Cliffs, Palo Verde, and Prairie Island.

Description of Amendment Request

The Omaha Public Power District proposes to revise the Bases of Technical Specifications 2.1.4, 3.1, and 3.17 and Technical Specifications set forth in Section 3.17 and Table 3-13 for the Fort Calhoun Nuclear Station. This revision will allow installation of tube sleeves as an alternative to plugging to repair defective steam generator tubes. This revision will allow removal of plugs that were previously installed as a corrective or preventive measure for the purpose of sleeving the tube. A tube inspection will be required prior to returning previously plugged tubes to service.

Description of Amendment Request (cont.)

Tube inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg. The proposed amendment will permit FCS to use Leak Tight Sleeves developed by Combustion Engineering, Inc. (CE).

There are two major types of CE Leak Tight Sleeves for steam generator tube repair. Attachment D, CE Report CEN-626-P, "Repair of ¾" O.D. Steam Generator Tubes Using Leak Tight Sleeves," dated June 1997, describes in detail the design and testing of these sleeves for FCS applicability. The analysis was performed for CE designed steam generators with ¾ inch outer diameter, 0.048-inch wall, Alloy 600 tubes. Attachment D provides a detailed description of the design, installation, and testing associated with the CE Leak Tight Sleeves. The sleeve material is thermally treated Alloy 690. The first type of sleeve spans the parent steam generator tube at the top of the tube sheet. This sleeve is welded to the tube near the upper end of the sleeve and is hard rolled into the tube within the steam generator tube sheet. A shorter sleeve of the same design is used to span defective areas of a steam generator tube, which exists just above the tube sheet. The second type of sleeve spans degraded areas of the steam generator tube at a tube support plate or in a free span section of tube. This leak tight sleeve is welded to the steam generator tube near each end of the sleeve. The steam generator tube with the installed welded and/or hard rolled sleeve meets the structural requirements of tubes that are not degraded.

Currently, FCS Technical Specifications only allow defective tubes to be plugged and removed from service. The Technical Specifications marked-up in Attachment A provide the details of the proposed changes.

JUSTIFICATION

OPPD has determined that operation with the proposed amendment would not result in any significant change in the types, or significant increases in the amounts, of any effluents that may be released offsite, nor would it result in any significant increase in individual or cumulative occupational radiation exposure.

These proposed changes to the Technical Specifications and our determination of no significant hazards have been reviewed by our Plant Review Committee and Safety Acceptance and Review Committee. They have concluded that implementation of these changes will not result in an undue risk to the health and safety of the public.

The principal accident associated with this proposed change is the steam generator tube rupture (SGTR) accident. The consequences associated with the SGTR event are discussed in Fort Calhoun Station's Updated Safety Analysis Report Section 14.14, "Steam Generator Tube Rupture." The SGTR event is a breach of the barrier between the reactor coolant system and the main steam system. The integrity of this barrier is significant from the standpoint of radiological safety in that a leaking steam generator

JUSTIFICATION (cont.)

tube allows the transfer of reactor coolant into the main steam system. In the event of a SGTR, radioactivity contained in the reactor coolant mixes with water in the shell side of the affected steam generator. This radioactivity is transported by steam to the turbine and then to the condenser, or directly to the condenser via the turbine bypass valves, or directly to the atmosphere via the atmospheric dump valves, main steam safety valves, or the auxiliary feedwater pump turbine exhaust. Non-condensable radioactive gases in the condenser are removed by the condenser evacuation system and discharged to the plant vent. The use of CE Leak Tight Sleeves will allow the repair of degraded steam generator tubes such that the function and integrity of the tube is maintained; therefore, the SGTR accident is not affected.

The consequences of a hypothetical failure of a CE Leak Tight Sleeve and/or associated steam generator tube would be bounded by the current SGTR analysis described above. Due to the slight reduction in diameter caused by the sleeve wall thickness, primary coolant release rates would be slightly less than assumed for the SGTR analysis (depending on break location), and therefore, would result in lower total primary fluid mass release to the secondary system. A main steam line break (MSLB) or feed line break (FLB) will not cause a SGTR since the sleeves are analyzed for a maximum accident differential pressure greater than that predicted in the Fort Calhoun Station safety analysis. The impact of sleeving on steam generator performance, heat transfer, and flow restriction is minimal and insignificant compared to plugging.

The proposed technical specification change to allow the use of CE Leak Tight Sleeves does not adversely impact any other previously evaluated design basis accident. The structural analyses of the sleeves demonstrate that their design meets all applicable American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) criteria for the steam generator pressure, temperature, and flow design conditions, and establishes the minimum reactor coolant pressure boundary wall thickness requirements. As described in detail in Attachment D, the results of the analyses and testing, as well as plant operating experience, demonstrate that CE Leak Tight Sleeves are an acceptable means of maintaining tube integrity. Sleeved tube plugging limit criteria are established using the guidance of Regulatory Guide (RG) 1.121, "Basis for Plugging Degraded PWR Steam Generator Tubes." Furthermore, per Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes" recommendations, the sleeved tube can be monitored through periodic inspections with present eddy current techniques. These measures ensure that installation of sleeves spanning degraded areas of the tube will restore the tube to a condition consistent with its original design basis.

The material selected for both types of sleeves is thermally treated Alloy 690 due to its corrosion resistance properties. Historically, thermally treated Alloy 690 has been used successfully for steam generator tubes, tube plugs, and tube sleeves. ABB Combustion

JUSTIFICATION (cont.)

Engineering (ABB/CE) conducted a number of bench and autoclave tests to evaluate the corrosion resistance of the welded sleeve joint. Of particular interest is the effect of the mechanical expansion/weld residual stresses and the condition of the weld and weld heat affected zone. Tests have been performed on welded joints with and without a post-weld heat treatment. There was no detectable indication of sleeve or joint corrosion or aggravated tube corrosion. The specific details of the corrosion performance of the thermally treated Alloy 690 material are contained in Section 6 of Attachment D.

The sleeve dimensions, materials, and joints were designed to the applicable ASME Code. An extensive analysis and test program was undertaken to prove the adequacy of both the welded and welded-hard rolled sleeve. This program determined the effect of normal operating and postulated accident conditions on the sleeve-tube assembly, as well as the adequacy of the assembly to perform its intended function. The proposed sleeving provides for a substitution in kind for a portion of a steam generator tube.

Installation of CE Leak Tight Sleeves has no significant effect on the configuration of the plant and does not affect the way in which the plant is operated. Design criteria were established prior to performing the analysis and test program which, if met, would prove that both sleeve types are acceptable repair techniques. These criteria conformed to the stress limits and margins of safety of Section III of the ASME Code. The safety factors of 3 for normal operating conditions and 1.5 for accident conditions were applied. Based upon the results of the analytical and test programs described in Attachment D, the two sleeve types fulfill their intended function as leak tight structural members and meet or exceed the established design criteria.

Evaluation of the sleeved tubes indicates no detrimental effects on the sleeve-tube assembly resulting from reactor coolant system flow, coolant chemistries, or thermal and pressure conditions. The sleeves are designed to be leak tight and therefore have no impact on steam generator leakage limits. Structural analyses of the sleeve-tube assembly, using the demonstrated margins of safety, have established its integrity under normal and accident conditions. The structural analyses performed are applicable to shorter sleeves installed at the top of the tubesheet and the tube support sleeves, which may be installed at FCS. The detailed analyses for the different sleeve types and lengths are included in Section 8 of Attachment D.

Welding development has been performed on clean tubing, dirty tubing which has been taken from pot boiler tests, and contaminated tubing taken from a steam generator. CE installed their first welded sleeves in a demonstration program at Ringhals Unit 2 in May 1984. CE's sleeving history is shown in Table 2-1 of Attachment D. Since 1985, no sleeve that has been accepted based on nondestructive examination (NDE) has been removed from service due to degradation.

JUSTIFICATION (cont.)

Mechanical tests using ASME Code stress allowables were performed on mockup steam generator tubes containing sleeves to provide qualified test data describing the basic properties of the completed assemblies. These tests determined axial load, collapse, burst, and thermal cycling capability. A minimum of three tests of each type was performed. The demonstrated load capacity of the assemblies provided an adequate safety factor for normal operating and postulated accident conditions. The load capability of the upper and lower sleeve joints is sufficient to withstand thermally induced stresses in the weld resulting from the temperature differential between the sleeve and the tube, and pressure-induced stresses resulting from normal operating and postulated accident conditions. The burst and collapse pressures of the sleeve provide a large safety factor over limiting pressure differential. Mechanical testing revealed that the installed sleeve would withstand the cyclical loading resulting from power changes in the plant and other transients.

The effects of sleeve installation on steam generator heat removal capability and reactor coolant system flow rate are discussed in Attachment D. Heat removal capability and reactor coolant system flow rate were considered for installation of one to three sleeves in a steam generator tube. After sleeves are installed, an ultrasonic and eddy current examination is performed. The ultrasonic examination is used to confirm fusion of sleeve to the tube after welding. The eddy current examination serves as baseline to determine if there is sleeve degradation in later operating years. The steam generator tube will be plugged if the sleeve installation is not successful or if there is unacceptable degradation of a sleeve or sleeved steam generator tube. Standard steam generator tube plugs may be used to remove a sleeved tube from service.

Based on past usage and extensive testing, the CE Leak Tight Sleeves provide satisfactory repair of defective steam generator tubes. Design criteria were established based on the requirements of ASME Code and Regulatory Guide 1.121. Qualified nondestructive examination will be used to perform necessary repair sleeve and tube inspections for defect detection and to verify proper installation of the repair sleeve.

NO SIGNIFICANT HAZARDS CONSIDERATION

The proposed change has been evaluated against the standards in 10CFR50.92, "Issuance of Amendment," and has been determined to not involve a significant hazards consideration. In support of this determination, a discussion of each of the significant safety hazards consideration factors with respect to the proposed license amendment is provided.

1. *The proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.*

   The CE Leak Tight Sleeves are designed using the applicable American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code and, therefore, meet the design objectives of the original steam generator tubing. The applicable design criteria for the sleeves conform to the stress limits and margins of safety of Section III of the ASME code. Mechanical testing has shown that the structural strength of repair sleeves under normal, upset, and faulted conditions provides margin to the acceptance limits. These acceptance limits bound the most limiting (three times normal operating pressure differential) burst margin recommended by Regulatory Guide 1.121. Burst testing of sleeved tubes has demonstrated that no unacceptable levels of primary-to-secondary leakage are expected during any plant condition.

   Evaluation of the repaired steam generator tubes indicates no detrimental effects on the sleeve or sleeve-tube assembly from reactor coolant system flow, primary or secondary coolant chemistries, thermal conditions or transients, or pressure conditions as may be experienced at Fort Calhoun Station. Corrosion testing of sleeve-tube assemblies indicates no evidence of sleeve or tube corrosion considered detrimental under anticipated service conditions.

   The installation of the proposed sleeves is controlled via the sleeving vendor's proprietary processes and equipment. The CE process has been in use since 1984 and has been implemented more than 24 times for the installation of over 4,200 sleeves. The FCS steam generator design was reviewed and found to be compatible with the installation processes and equipment.

   The implementation of the proposed amendment has no significant effect on either the configuration of the plant or the manner in which it is operated. The consequences of a hypothetical failure of the sleeved tube is bounded by the current steam generator tube rupture analysis described in Fort Calhoun Station's USAR, Section 14.14. Due to the slight reduction in diameter caused by the sleeve wall thickness, primary coolant release rates would be slightly less than assumed for the steam generator tube rupture analysis, depending on the break location, and therefore, would result in lower total primary fluid mass release to the secondary system. A main steam line break or feed line break will not cause a SGTR since the sleeves are analyzed for a maximum accident differential pressure greater than that predicted in the Fort Calhoun Station safety analysis.

In conclusion, based on the discussion above, these changes will not significantly increase the probability or consequences of an accident previously evaluated.

2. *The proposed amendment would not create the possibility of a new or different kind of accident from any other accident previously evaluated.*

   As discussed above, the CE Leak Tight Sleeves are designed using the applicable ASME Code as guidance; therefore, they meet the objectives of the original steam generator tubing. As a result, the functions of the steam generators will not be significantly affected by the installation of the proposed sleeves. The proposed repair sleeves do not interact with any other plant systems. Any accident as a result of potential tube or sleeve degradation in the repaired portion of the tube is bounded by the existing tube rupture accident analysis. The continued integrity of the installed sleeve is periodically verified by the Technical Specification requirements.

   The implementation of the proposed amendment has no significant effect on either the configuration of the plant or the manner in which it is operated. Therefore, Omaha Public Power District concludes that this proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. *The proposed amendment would not involve a significant reduction in a margin of safety.*

   The repair of degraded steam generator tubes with CE Leak Tight Sleeves restores the structural integrity of the degraded tube under normal operating and postulated accident conditions. The design safety factors utilized for the repair sleeves are consistent with the safety factors in the ASME Code used in the original steam generator design. The portions of the installed sleeve assembly that represents the reactor coolant pressure boundary can be monitored for the initiation and progression of sleeve/tube wall degradation. Use of the previously identified design criteria and design verification testing assures that the margin of safety is not significantly different from the original steam generator tubes. Therefore, OPPD concludes that the proposed change does not involve a significant reduction in a margin of safety.

Based on the above considerations, OPPD concludes that the proposed amendment to FCS Technical Specifications does not involve a significant hazards consideration as defined by 10 CFR 50.92 and that the proposed amendment will not result in a condition which significantly alters the impact of the station on the environment. Thus the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22 (c) (9), and pursuant to the 10 CFR 51.22 (b), no environment assessment need be prepared.

LIC-00-0062
ATTACHMENT C
COMBUSTION ENGINEERING, INC. PROPRIETARY AFFIDAVIT FOR
ATTACHMENT (5), PURSUANT TO 10 CFR 2.790

I, Ian C. Rickard, depose and say that I am the Director, Nuclear Licensing, of C-E Nuclear Power LLC (CENP), duly authorized to make this affidavit, and have reviewed or caused to have reviewed the information which is identified as proprietary and described below.

I am submitting this affidavit in conjunction with the application by Omaha Public Power District and in conformance with the provisions of 10 CFR 2.790 of the Commission's regulations for withholding this information. I have personal knowledge of the criteria and procedures utilized by CENP in designating information as a trade secret, privileged, or as confidential commercial or financial information.

The information for which proprietary treatment is sought, and which document has been appropriately designated as proprietary, is contained in the following:

- *CEN-630-P, Revision 02, "Repair of 3/4" O.D. Steam Generator Tubes using Leaktight Sleeves, Final Report," dated June 1997*

Pursuant to the provisions of Section 2.790(b)(4) of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information included in the document listed above should be withheld from public disclosure.

1. The information sought to be withheld from public disclosure is owned and has been held in confidence by CENP. It consists of information concerning the design, development, and installation process for a welded sleeve used in the repair of 3/4 inch outside diameter steam generator tubes.
2. The information consists of test data or other similar data concerning a process, method or component, the application of which results in substantial competitive advantage to CENP.
3. The information is of a type customarily held in confidence by CENP and not customarily disclosed to the public.
4. The information is being transmitted to the Commission in confidence under the provisions of 10 CFR 2.790 with the understanding that it is to be received in confidence by the Commission.
5. The information, to the best of my knowledge and belief, is not available in public sources, and any disclosure to third parties has been made pursuant to regulatory provisions or proprietary agreements that provide for maintenance of the information in confidence.
6. Public disclosure of the information is likely to cause substantial harm to the competitive position of CENP because:
   a. A similar product is manufactured and sold by major competitors of CENP.
   b. Development of this information by CENP required millions of dollars and thousands of manhours of effort; a competitor would have to undergo similar expense in generating equivalent information. In order to acquire such information, a competitor would also require considerable time and inconvenience to develop the design and installation process for a welded sleeve for repairing 3/4 inch O.D. steam generator tubes.

c. The information consists of technical data and details concerning the design, development, and installation process for a welded sleeve for repairing 3/4 inch O.D. steam generator tubes, the application of which provides CENP a competitive economic advantage. The availability of such information to competitors would enable them to modify their products to better compete with CENP, take marketing or other actions to improve their product's position or impair the position of CENP's product, and avoid developing similar data and analyses in support of their processes, methods or apparatus.

d. In pricing CENP's products and services, significant research, development, engineering, analytical, manufacturing, licensing, quality assurance and other costs and expenses must be included. The ability of CENP's competitors to utilize such information without similar expenditure of resources may enable them to sell at prices reflecting significantly lower costs.

e. Use of the information by competitors in the international marketplace would increase their ability to market steam generator repair systems and services by reducing the costs associated with their technology development. In addition, disclosure would have an adverse economic impact on CENP's potential for obtaining or maintaining foreign licenses.

Sworn to before me this
6th day of July, 2000

I. C. Rickard

Ian C. Rickard
Director, Nuclear Licensing

Catherine P. McCarthy
Notary Public

My commission expires: 1/31/03