October 1, 1996

Mr. Donald Schnell
Senior Vice President - Nuclear
Union Electric Company
Post Office Box 149
St. Louis, Missouri 63166

SUBJECT: AMENDMENT NO. 116 TO FACILITY OPERATING LICENSE NO. NPF-30 - CALLAWAY PLANT, UNIT 1 (TAC NO. M95203)

Dear Mr. Schnell:

The Commission has issued the enclosed Amendment No. 116 to Facility Operating License No. NPF-30 for the Callaway Plant, Unit 1. The amendment consists of changes to the Technical Specifications (TS) in response to your application dated April 12, 1996, as supplemented by letters dated August 2, 1996, August 19, 1996, and September 5, 1996.

The amendment revises TS 3/4.4 and the associated Bases to address the installation of laser welded tube sleeves in the Callaway Plant steam generators.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Original Signed By

Kristine M. Thomas, Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-483

Enclosures: 1. Amendment No. 116 to NPF-30
2. Safety Evaluation

cc w/encls: See next page

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WJohnson, RIV	LHurley, RIV
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JStrosnider, EMBC	
TLH3 (SE)	

DOCUMENT NAME: CAL95203.AMD

OFC	PDIV-2/LA	EMCB:BC	OGC	PDIV-2/PM
NAME	EPeyton	Strosnider	Holler	KThomas:ye
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 1, 1996

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Post Office Box 149
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Office of Nuclear Reactor Regulation

Docket No. 50-483

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2. Safety Evaluation

cc w/encls: See next page

cc w/encls:
Professional Nuclear
Consulting, Inc.
19041 Raines Drive
Derwood, Maryland 20855

Gerald Charnoff, Esq.
Thomas A. Baxter, Esq.
Shaw, Pittman, Potts & Trowbridge
2300 N. Street, N.W.
Washington, D.C. 20037

Mr. H. D. Bono
Supervising Engineer,
Site Licensing
Union Electric Company
Post Office Box 620
Fulton, Missouri 65251

U.S. Nuclear Regulatory Commission
Resident Inspector Office
8201 NRC Road
Steedman, Missouri 65077-1302

Mr. G. L. Randolph, Vice President
Nuclear Operations
Union Electric Company
P.O. Box 620
Fulton, Missouri 65251

Manager - Electric Department
Missouri Public Service Commission
301 W. High
Post Office Box 360
Jefferson City, Missouri 65102

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
Harris Tower & Pavilion
611 Ryan Plaza Drive, Suite 400
Arlington, Texas 76011-8064

Mr. Ronald A. Kucera, Deputy Director
Department of Natural Resources
P.O. Box 176
Jefferson City, Missouri 65102

Mr. Neil S. Carns
President and Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
P.O. Box 411
Burlington, Kansas 66839

Mr. Dan I. Bolef, President
Kay Drey, Representative
Board of Directors Coalition
for the Environment
6267 Delmar Boulevard
University City, Missouri 63130

Mr. Lee Fritz
Presiding Commissioner
Callaway County Court House
10 East Fifth Street
Fulton, Missouri 65151

Mr. Alan C. Passwater, Manager
Licensing and Fuels
Union Electric Company
Post Office Box 149
St. Louis, Missouri 63166

Mr. J. V. Laux, Manager
Quality Assurance
Union Electric Company
Post Office Box 620
Fulton, Missouri 65251

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

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 116
License No. NPF-30

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Union Electric Company (UE, the licensee) dated April 12, 1996, as supplemented by letters dated August 2, 1996, August 19, 1996, and September 5, 1996, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. NPF-30 is hereby amended to read as follows:

(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 116 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective as of its date of issuance and will implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Kristine M. Thomas

Kristine M. Thomas, Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: October 1, 1996

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable steam generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 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 Tables 4.4-2 and 4.4-3. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. When applying the exceptions of 4.4.5.2.a through 4.4.5.2.c, previous defects or imperfections in the area repaired by sleeving are not considered an area requiring inspection. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas;

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

1) All nonplugged tubes that previously had detectable wall penetrations (greater than 20%),

2) Tubes in those areas where experience has indicated potential problems, and

3) A tube inspection (pursuant to Specification 4.4.5.4a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples (if required by Tables 4.4-2 or 4.4-3) during each inservice inspection may be subjected to a partial tube inspection provided:

1) The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and

2) The inspections include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months;

b. If the results of the inservice inspection of a steam generator conducted in accordance with Tables 4.4-2 and 4.4-3 at 40-month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3a.; the interval may then be extended to a maximum of once per 40 months; and

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Tables 4.4-2 and 4.4-3 during the shutdown subsequent to any of the following conditions:

   1) Reactor-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2, or

   2) A seismic occurrence greater than the Operating Basis Earthquake, or

   3) A loss-of-coolant accident requiring actuation of the Engineered Safety Features, or

   4) A main steam line or feedwater line break.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.4 Acceptance Criteria

a. As used in this specification:

1) Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections;

2) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube;

3) Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation;

4) % Degradation means the percentage of the tube wall thickness affected or removed by degradation;

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

6) 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 and is equal to 40% of the nominal tube wall thickness. The plugging limit for laser welded sleeves is equal to 39% of the nominal sleeve wall thickness;

7) 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 as specified in Specification 4.4.5.3c., above;

8) 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. For a tube repaired by sleeving, the tube inspection shall include the sleeved portion of the tube;

SURVEILLANCE REQUIREMENTS (Continued)

9) Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections; and

10) Tube Repair refers to a process that reestablishes tube serviceability. Acceptable tube repairs will be performed by the following processes:

a) Laser welded sleeving as described in Westinghouse Technical Report WCAP-14596-P, "Laser Welded Elevated Tube Sheet Sleeves for Westinghouse Model F Steam Generators." March 1996 (W Proprietary)

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or repair by sleeving all tubes exceeding the plugging or repair limit and all tubes containing through-wall cracks) required by Tables 4.4-2 and 4.4-3.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged or repaired in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2;

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:

1) Number and extent of tubes and sleeves inspected,

2) Location and percent of wall-thickness penetration for each indication of an imperfection, and

3) Identification of tubes plugged or repaired.

c. Results of steam generator tube inspections, which fall into Category C-3, shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No			Yes		
No. of Steam Generators per Unit	Two	Three	Four	Two	Three	Four
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspections	One[1]			One[1]	One[2]	One[3]

TABLE NOTATIONS

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

2. The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.

3. Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N.A.	N.A.	N.A.	N.A.
	C-2	Plug or repair defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N.A.	N.A.
			C-2	Plug or repair defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug or repair defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N.A.	N.A.
	C-3	Inspect all tubes in this S. G., plug or repair defective tubes and inspect 2S tubes in each other S.G. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	All other S.G.s are C-1	None	N.A.	N.A.
			Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N.A.	N.A.
			Additional S. G. is C-3	Inspect all tubes in each S.G. and plug or repair defective tubes. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	N.A.	N.A.

$S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

TABLE 4.4-3

STEAM GENERATOR REPAIRED TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required
A minimum of 20% of repaired tubes (1) (2)	C-1	None	N.A.	N.A.
	C-2	Plug defective repaired tubes and inspect 100% of the repaired tubes in this S.G.	C-1	None
			C-2	Plug defective repaired tubes
			C-3	Perform action for C-3 result of first sample
	C-3	Inspect all repaired tubes in this S.G., plug defective tubes and inspect 20% of the repaired tubes in each other S.G.	All other S.G.s are C-1	None
		Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of first sample
			Additional S.G. is C-3	Inspect all repaired tubes in each S.G. and plug defective tubes. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50

(1) Each repair method is considered a separate population for determination of scope expansion.
(2) The inpection of repaired tubes may be performed on tubes from 1 to 4 steam generators based on outage plans.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System Leakage Detection Systems shall be OPERABLE:

a. The Containment Atmosphere Particulate Radioactivity Monitoring System,

b. The Containment Normal Sump Level Measurement System, and

c. Either the Containment Air Cooler Condensate Flow Rate or the Containment Atmosphere Gaseous Radioactivity Monitoring System.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only two of the above required Leakage Detection Systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed for gaseous and particulate radioactivity or a gamma isotopic analysis of the containment atmosphere is performed using the Post Accident Sampling System at least once per 24 hours when the required Gaseous or Particulate Radioactivity Monitoring System is inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection Systems shall be demonstrated OPERABLE by:

a. Containment Atmosphere Gaseous and Particulate Monitoring System-performance of CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies specified in Table 4.3-3,

b. Containment Normal Sump Level Measurement System-performance of CHANNEL CALIBRATION at least once per 18 months, and

c. Containment Air Cooler Condensate Flow Monitoring System-performance of CHANNEL CALIBRATION at least once per 18 months.

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 600 gpd total reactor-to-secondary leakage through all steam generators not isolated from the Reactor Coolant System and 150 gallons per day through any one steam generator,

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. 8 gpm per RC pump CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig, and

f. The leakage from each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be limited to 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm, at a Reactor Coolant System pressure of 2235 ± 20 psig.*

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, reduce the leakage rate to within limits within 4 hours, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours with an RCS pressure of less than 600 psig.

---

*Test pressures less than 2235 psig but greater than 150 psig are allowed. Observed leakage shall be adjusted for the actual test pressure up to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one-half power.

3/4.4.5 STEAM GENERATORS

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 inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice 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 inservice conditions that lead to corrosion. Inservice 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.

Unscheduled inservice inspections are performed on each steam generator following: (1) reactor to secondary tube leaks; (2) a seismic occurrence greater than the Operating Basis Earthquake; and (3) a loss-of-coolant accident requiring actuation of the Engineered Safety Features, which for this Specification is defined to be a break greater than that equivalent to the severance of a 1" inside diameter pipe, or, for a main steamline or feedline, a break greater than that equivalent to a steam generator safety valve failing open; to ensure that steam generator tubes retain sufficient integrity for continued operation. Transients less severe than these do not require inspections because the resulting stresses are well within the stress criteria established by Regulatory Guide 1.121, which unplugged steam generator tubes must be capable of withstanding.

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 Reactor Coolant System and the Secondary Coolant System (reactor-to-secondary leakage = 150 gallons per day per steam generator). Cracks having a reactor-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 reactor-to-secondary leakage of 150 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located, plugged or repaired.

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 inservice steam generator tube examinations. Plugging or repair will be required for all tubes with imperfections exceeding the plugging or repair limit. 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. Results from WCAP-10043 have been used to establish plugging limit.

STEAM GENERATORS (Continued)

The plugging or repair limit for the pressure boundary portion of laser welded sleeves is determined to be 39% through-wall (by NDE). The laser welded sleeve repair limit applicable to the pressure boundary portion of the sleeve is established in WCAP-14596. Appropriate NDE techniques are also discussed in WCAP-14596.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission pursuant to Specification 6.9.2 prior to 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.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These Detection Systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

The total steam generator tube leakage limit of 600 gpd for all steam generators not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. The 600 gpd limit is conservative compared to the assumptions used in the analysis of these accidents. The 150 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the Leakage Detection Systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow from the reactor coolant pump seals exceeds 8 gpm per RC pump at a nominal RCS pressure of 2235 psig. This limitation ensures adequate performance of the RC pump seals.

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 116 TO FACILITY OPERATING LICENSE NO. NPF-30

DOCKET NO. 50-483

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
3/4 4-11	3/4 4-11
3/4 4-12	3/4 4-12
3/4 4-13	3/4 4-13
3/4 4-14	3/4 4-14
3/4 4-15	3/4 4-15
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 116 TO FACILITY OPERATING LICENSE NO. NPF-30

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

1.0 INTRODUCTION

By letter dated April 12, 1996, as supplemented by letters dated August 2, 1996, August 19, 1996, and September 5, 1996, Union Electric Company (UE), requested changes to the Technical Specifications (Appendix A to Facility Operating License No. NPF-30) for the Callaway Plant, Unit 1. The proposed amendment would revise Technical Specification (TS) 3/4.4 and the associated Bases to address the installation of laser welded tube sleeves designed by Westinghouse Electric Corporation (Westinghouse) in the Callaway Plant steam generators (SG).

The August 2, 1996, August 19, 1996, and September 5, 1996, supplemental letters provided only clarifying information and did not change the original no significant hazards consideration determination published in the Federal Register on May 8, 1996 (61 FR 20857).

The sleeve type is an elevated tubesheet sleeve. It is designed to repair tubes with degradation in the expansion transition zone at the top of the tubesheet. The sleeve is inserted inside the tube and held in position by hydraulically expanding the ends of the sleeve. This hydraulic expansion also brings the sleeve ends into contact with the parent tube in preparation for subsequent welding or rolling. The structural attachment of the sleeve to the tube is accomplished by means of two different joint types: a rolled joint (mechanically expanded) in the (lower) tubesheet end and an autogenous laser weld at the (upper) freespan end. The material of construction for the sleeves is a nickel-iron-chromium alloy, alloy 690, a Code-approved material (ASME SB-163), incorporated in ASME Code Case N-20.

Extensive analyses and testing were performed on the Westinghouse sleeve and sleeve/tube joints to demonstrate that Regulatory and Code design criteria were satisfied under normal operating and postulated accident conditions. The details of the sleeve qualifications are discussed in report WCAP-14596, "Laser Welded Elevated Tubesheet Sleeves for Westinghouse Model F Steam Generators," dated March 1996 (proprietary). This generic sleeving report presents the technical bases supporting the licensing of laser welded sleeves for use in 11/16-inch diameter SG tubes such as those at Callaway. (The bounding assumptions in generic WCAP-14596 were verified by the licensee staff to bound all of the appropriate Callaway Plant conditions).

The staff previously reviewed closely similar Westinghouse documents supporting requests for changes to the TS at other plants. The bulk of the technical and regulatory issues for the present request are identical to those reviewed in previous safety evaluations (SEs) concerning the use of Westinghouse laser welded sleeves. This SE discusses only those issues warranting revision, amplification, or inclusion based upon current experience. A summary of the principle technical issues regarding the design and use of Westinghouse laser welded sleeves follows. Details of the prior staff evaluation of Westinghouse sleeves may be found in SEs for Calvert Cliffs Nuclear Power Plant Units 1 and 2, Docket Nos. 50-317 and 50-318, dated March 22, 1996; DC Cook Nuclear Power Plant Unit 1, Docket No. 50-315, dated January 4, 1996; Maine Yankee Nuclear Power Plant, Docket No. 50-309, dated May 22, 1995; and Joseph M. Farley Nuclear Power Plant, Units 1 and 2, Docket Nos. 50-348 and 50-364, dated October 22, 1990. These evaluations are germane to the proposed Callaway license amendment.

2.0 BACKGROUND

A sleeve is a tube slightly smaller in diameter than an SG tube that is inserted into an SG tube to bridge a degraded or susceptible section of tube. The length of a sleeve is selected according to the individual installation circumstance. Generally, they vary in length between one and three feet. The sleeve becomes the pressure boundary and thereby restores the structural integrity of a degraded or potentially degraded portion of the original SG tube.

Prior to the development of sleeve technology, a defective SG tube was removed from service by plugging. However, this reduces the heat transfer area. The reduction in heat transfer (or other thermal-hydraulic operating parameters) can be tolerated up to a point before other system consequences of the reduced SG performance became limiting. Beyond this limit, a utility has to make operational changes resulting in reduced electrical generating capacity of the affected unit.

Because sleeves have minimal effect upon the thermal-hydraulics of an SG, their use is essentially unrestricted. This means a licensee may restore degraded sections of SG tubes to like new condition without experiencing a serious penalty with regard to unit generating capacity. This has led to increased use of sleeves versus plugs where practical. Recently, some foreign and domestic plants have installed sleeves in previously unprecedented numbers, up to nearly 100 percent of the SG tubes on a single unit.

About 29,000 Westinghouse laser welded sleeves have been installed in foreign and domestic plants since 1988. Over eight years of operating experience with Westinghouse sleeves has shown the technology to be highly reliable. No operationally induced degradation or leakage has occurred in any Westinghouse laser welded sleeves.

3.0 SUMMARY OF PREVIOUS REVIEWS

Previous staff evaluations of Westinghouse sleeves addressed the technical adequacy of the sleeves in the principal areas of pressure retaining component design: structural requirements, material of construction, welding, nondestructive examination, and sleeve plugging limit. The staff found the analyses and tests submitted to address these areas of component design and inspection to be acceptable and are summarized below:

3.1 Structural Requirements

The function of sleeves is to restore the structural integrity of the tube pressure boundary. Consequently, structural analyses were performed for a variety of loadings including design pressure, operating transients, and other parameters selected to envelope loads imposed during normal operating, upset, and accident conditions. Stress analyses of sleeved tube assemblies were performed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III. These analyses, along with the results of qualification testing and previous plant operating experience, were cited to demonstrate the sleeved tube assembly is capable of restoring steam generator tube structural integrity.

3.2 Material of Construction

The sleeves are fabricated from thermally treated alloy 690, a Code approved material (ASME SB-163) covered by ASME Code Case N-20. The staff found the use of alloy 690 is an improvement over the alloy 600 material used in the original SG tubing. Corrosion tests conducted under Electric Power Research Institute (EPRI) sponsorship confirmed test results regarding the improved corrosion resistance of alloy 690 over that of alloy 600. Accelerated stress corrosion tests in caustic and aqueous chloride solutions also indicated alloy 690 resists general corrosion in aggressive environments. Isothermal tests in high purity water have shown that, at normal stress levels, alloy 690 has high resistance to intergranular stress corrosion cracking (IGSCC) in extended high temperature exposure. The NRC concluded, as a result of these laboratory corrosion tests, that alloy 690 is acceptable for use in nuclear power plants. The NRC endorsed the use of Code Case N-20 in Regulatory Guide 1.85, "Materials Code Case Acceptability, ASME Section III, Division 1." The NRC staff has approved use of alloy 690 tubing in replacement steam generators as well as sleeving applications.

3.3 Welding and Post Weld Heat Treatment

Automatic autogenous laser welding is employed to join the sleeve to the parent tube in the freespan regions. The application of this process to the W sleeve design was specifically qualified and demonstrated during laboratory tests employing full scale sleeve/tube mockups. Qualification of the welding procedures and welding equipment operators was performed in accordance with the requirements of the ASME Code, Section IX.

Accelerated corrosion tests have confirmed that a postweld heat treatment (PWHT) significantly improves the IGSCC resistance of the alloy 600 parent tube material in the weld zone. A PWHT reduces the residual stresses resulting from welding. Residual stresses from forming operations (such as bending, welding, etc.) are known to be a principal contributor to IGSCC in alloy 600. Performance of a PWHT greatly reduces the residual stresses from welding thereby enhancing the IGSCC resistance of the alloy 600 portion of the weld zone. The alloy 690 sleeve material is highly resistant to IGSCC either with or without PWHT. All laser welded joints will be heat treated in accordance with the Westinghouse generic sleeving report and the NRR staff position.

The rolled joint used to join the sleeve to the tube within the tubesheet effectively isolates the alloy 600 of the parent tube from the environment and thus is not susceptible to IGSCC. Stress relief of these joints is unwarranted.

3.4 Nondestructive Examination

The baseline nondestructive examination of sleeved tubes is conducted using ultrasonic testing (UT) and eddy-current testing (ECT). UT is performed after welding to confirm the laser welds are consistent with critical process dimensions and are of acceptable weld quality. Westinghouse presented data on a UT system demonstrating post weld examinations of the sleeve/tube assembly will be adequate. Standards which included undersized welds were used in the qualification of the UT technique. The results of the qualification tests demonstrate the system can confirm there is a continuous metallurgical bond between the sleeve and tube and that the weld size (width) meets minimum acceptable dimensions.

ECT is then used to establish baseline inspection data for every installed sleeve/tube. In performing the inspection, the licensee will use Electric Power Research Institute (EPRI) "PWR Steam Generator Tube Examination Guidelines" Appendix G qualified personnel and Appendix H qualified ECT techniques. For future sleeve/tube inspections, the licensee committed to following Revision 4 of the EPRI guidelines. Future revisions of the guidelines will be evaluated and adopted as appropriate. The licensee also modified the TS to incorporate sleeve/tube inspection scope and expansion criteria.

3.5 Tube and Sleeve Plugging/Repair Limits

The tube and sleeve minimum acceptable wall thickness is determined using the criteria of Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," and ASME Code Section III allowable stress values and pressure stress equations. According to RG 1.121 criteria, an allowance for nondestructive evaluation (NDE) uncertainty and postulated operational growth of tube or sleeve wall degradation must be accounted for when using NDE to determine plugging or repair limits. Therefore, a conservative tube or sleeve wall thickness allowance for postulated degradation growth and eddy current uncertainty of 20 percent through wall per cycle were assumed for the purpose of determining the tube or sleeve plugging/repair limit (tubes may be plugged or, if feasible, sleeved, and sleeves would be plugged only).

In accordance with the guidance of NUREG-1431, "Standard Technical Specifications for Westinghouse Plants," the licensee is reducing the tube plugging/repair limit from 48 percent to 40 percent through wall of nominal tube wall thickness (including the 20 percent NDE uncertainty). The sleeve plugging limit, calculated based upon the most limiting of normal, upset, or faulted conditions for the Callaway Plant steam generators, was determined to be 39 percent of the sleeve nominal wall thickness, including the 20 percent NDE uncertainty. Plugging or repair of tubes and plugging of sleeves when degradation reaches the plugging/repair limit provides assurance the minimum acceptable wall thickness will not be violated during the next subsequent cycle of operation.

4.0 DISCUSSION

Experience with all types of SG tube sleeves has led to several areas of concern outside the scope of basic sleeve design and qualification discussed above. These include instances of cracking in sleeved SG tubes, service life predictions for sleeved tubes, application of PWHT and the effect of tube lockup, and primary-to-secondary leakage limits.

4.1 Cracking in Sleeved SG Tubes

Recent experiences at two U.S. plants indicated the freespan joint of a sleeved alloy 600 steam generator tube may be susceptible to IGSCC. The affected joints are of the mechanically expanded type. These employ a hydraulic expansion followed by a hard roll in the center of the hydraulically expanded region. The hard roll forms the structural joint and leak limiting seal. Inner diameter initiated cracks have been detected in the alloy 600 parent tube material at the lower hard roll transition and lower hydraulic transition of freespan joints. The cracks were detected after four to seven years of service. Since a number of sleeved tubes with this joint type have operated up to 14 years in one of the affected units, it is clear that not all such sleeved tubes are likely to develop cracks after a given service interval.

Accelerated corrosion tests of laser welded sleeve joints have shown the hydraulic transition to have little or no susceptibility to IGSCC. Service times exceeding eight years have been achieved for sleeved tubes with laser welded joints at U.S. plants. No instances of service induced IGSCC have occurred in any of these joints. The staff is monitoring these developments for potential impact on welded sleeve installations.

4.2 Service Life Predictions for Sleeved SG Tubes

The staff position on sleeving considers the method unable to assure an unlimited service life for a repaired tube. The conservative view is sleeving creates new locations in the parent tube which may be susceptible to IGSCC after new incubation times are expended. Incubation times are not quantified.

They are observed to vary between individual steam generators and the various tubes within, based upon prior experiences with U-bend and roll transition cracking.

This staff position that sleeving has limited service life is due to the circumstances of the sleeving processes. Sleeve installation methods can enhance one or two of the conditions necessary for IGSCC. The primary contributor is the residual stress resulting from the various joining methods. Secondarily, the local environment of the tube may be altered as a result of the formation of a wetted crevice between the tube and sleeve. Remediation of these contributors would benefit sleeved tube life. Of the two, stress relieving may be the most beneficial given the underlying causes of IGSCC and present sleeve designs. As discussed earlier, the sleeve installation procedure includes a PWHT of the weld joints to increase the resistance to IGSCC.

4.3 PWHT and Tube Lockup

Recent field experience with the installation of welded sleeves with PWHT indicated SG tubes may be constrained ("tube lockup") in their tube supports. The result of such tube locking is distortion of the tube (bowing or bulging) during the PWHT. After the heat treatment is completed, the bow or bulge remains. Measurements of the bowing and bulging have shown them to be of negligible values. These distortions have been analyzed and found to be immaterial to the examination, operation, and safety of the sleeved tubes.

Along with the observed distortion (bowing or bulging) is a residual stress remaining after the heat treatment is completed. Strain gage measurements of this residual stress have shown it to be moderate compared to that resulting from welding without subsequent PWHT. This issue was the subject of additional testing and analysis related to the use of laser welded sleeves at the Maine Yankee facility during a sleeve installation project. Based upon the finding that many tubes are fixed in the tube supports, Westinghouse modified their sleeve installation procedure on the assumption that all tubes are locked. The modified installation procedure thereby minimizes the residual stress of PWHT regardless of tube condition.

4.4 Primary-to-Secondary Leakage Limits

While a laser weld should be inherently leak-tight, the lower (rolled) joint of a tubesheet sleeve may not be leak tight. Westinghouse analyzed the effects of an abnormal lower joint seal. The analysis shows that even under extreme postulated conditions, it will have satisfactory leakage integrity.

The licensee is adopting a change to its TS incorporating a 150 gpd per SG leakage limit and a 600 gpd total primary-to-secondary leakage through all SGs not isolated from the reactor coolant system. This is consistent with the staff's position regarding primary-to-secondary leakage limits for SGs with sleeved tubes.

4.5 Results of Review

Based on the preceding analysis, the NRC staff concludes the repair of SG tubes at Callaway Plant using laser welded tubesheet sleeves designed and installed by Westinghouse in accordance with the methods of WCAP-14596 is acceptable.

Additionally, for the reasons discussed in 3.0 and 4.0 above, the staff finds acceptable these additional changes to the plant TS:

a. The associated sleeve wall depth based plugging limit value of 39 percent (of nominal sleeve wall thickness) and inspection requirements. (See Section 3.5)

b. Reduction of the tube plugging/repair limit from 48 percent to 40 percent through wall (of the nominal tube wall thickness) to be consistent with NUREG-1431, "Standard Technical Specifications for Westinghouse Plants." (See Section 3.5)

c. The reduction of the primary to secondary normal operational leakage limit from 500 to 150 gpd per steam generator in accordance with staff position for units with TS amendments to allow tube repair by sleeving. (See Section 4.4)

d. The addition of sleeve/tube inspection scope and criteria for expansion of sleeve/tube inspections. (See Section 3.4)

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Missouri State official was notified of the proposed issuance of the amendment. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (61 FR 20857). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Geoff Hornseth
Stephanie Coffin

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