December 14 1995

Mr. Ross P. Barkhurst
Vice President Operations
Entergy Operations, Inc.
P. O. Box B
Killona, LA 70066

SUBJECT: ISSUANCE OF AMENDMENT NO. 117 TO FACILITY OPERATING LICENSE NPF-38 - WATERFORD STEAM ELECTRIC STATION, UNIT 3 (TAC NO. M88397)

Dear Mr. Barkhurst:

The Commission has issued the enclosed Amendment No. 117 to Facility Operating License No. NPF-38 for the Waterford Steam Electric Station, Unit 3 (WAT-3). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated December 6, 1993, as supplemented by letters dated May 12, August 9, and September 18, 1995.

The amendment changes the Appendix A TSs to allow installation of steam generator tube repair sleeves at WAT-3. The sleeves are designed and manufactured by Combustion Engineering Incorporated. Based on our evaluation, we find that the proposed sleeving can be accomplished to produce acceptable sleeved tubes with respect to metallurgical properties, corrosion resistance, inservice inspection, and structural integrity. Therefore, we have concluded that the proposed sleeving method for steam generator tubes is an acceptable alternative to plugging.

The staff notes that no accelerated corrosion tests have been performed on locked tubes with heat treated welds. Such tests could better support joint life estimates.

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

Sincerely,
ORIGINAL SIGNED BY:
Chandu P. Patel, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-382

Enclosures: 1. Amendment No. 117 to NPF-38
2. Safety Evaluation

cc w/encls: See next page

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

December 14, 1995

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Entergy Operations, Inc.
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Project Directorate IV-1
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Docket No. 50-382

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cc w/encls: See next page

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Entergy Operations, Inc.

Waterford 3

cc:

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

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-382

WATERFORD STEAM ELECTRIC STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 117
License No. NPF-38

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

   A. The application for amendment by Entergy Operations, Inc. (the licensee) dated December 6, 1993, as supplemented by letters dated May 12, August 9, and September 18, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and 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-38 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 117, 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 license amendment is effective as of its date of issuance to be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Chandu P. Patel

Chandu P. Patel, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: December 14, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 117

TO FACILITY OPERATING LICENSE NO. NPF-38

DOCKET NO. 50-382

Replace the following pages of the Appendix A Technical Specifications with the attached 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 PAGES	INSERT PAGES
3/4 4-11	3/4 4-11
3/4 4-13	3/4 4-13
3/4 4-14	3/4 4-14
3/4 4-16	3/4 4-16
B 3/4 4-2	B 3/4 4-2

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

2. Tubes in those areas where experience has indicated potential problems.

3. A tube inspection (pursuant to Specification 4.4.4.4a.9.) 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 Table 4.4-2) 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.

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)

4.4.4.3 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 calender 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 following service under AVT conditions, 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 Table 4.4-2 at 40-month intervals fall into 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 4.4.4.3a.; the interval may then be extended to a maximum of once per 40 months.

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

    1. Primary-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.5.2.

    2. A seismic occurrence greater than the Operating Basis Earthquake.

    3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

    4. A main steam line or main feedwater line break.

4.4.4.4 Acceptance Criteria

a. As used in this Specification

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

2. 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.

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

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

5. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

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

7. 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 because it may become unserviceable prior to the next inspection and is equal to 40% of the nominal tube wall thickness.

8. 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 4.4.4.3c., above.

9. 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.

10. 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 was performed prior to field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

b. 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 4.4-2. Defective tubes may be repaired in accordance with CENS Report CEN-605-P, "Waterford 3 Steam Generator Tube Repair Using Leak Tight Sleeves," Revision 00-P, dated December 1992.

4.4.4.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged or sleeved 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 completion of the inspection. This Special Report shall include:

1. Number and extent of tubes inspected.

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

3. Identification of tubes plugged or sleeved.

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

The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 6% of the tubes 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.

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	Acion Required
A minimum of S Tubes per S. G.	C-1	None	N. A.	N. A.	N. A.	N. A.
	C-2	Plug or sleeve defective tubes and inspect additional 2S tubes in this S. G.	C-1	None	N. A.	N. A.
			C-2	Plug or sleeve defective tubes and inspect additional 4S tubes in this S. G.	C-1	None
					C-2	Plug or sleeve 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 sleeve defective tubes and inspect 2S tubes in each other S. G. Notification to NRC pursuant to §50.72(b)(2) to 10CFR 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	N. A.	N. A.
			Additional S. G. is C-3	Inspect all tubes in each S. G. and plug or sleeve defective tubes. Notification to NRC pursuant to §50.72(b)(2) of 10CFR Part 50	N. A.	N. A.

$S = \frac{6\%}{n}$ Where n is the number of steam generators inspected during an inspection

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR above 1.20 during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or shutdown cooling train provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops or trains (either shutdown cooling or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single shutdown cooling train provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two shutdown cooling trains be OPERABLE.

The operation of one reactor coolant pump or one shutdown cooling (low pressure safety injection) pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reductions will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a reactor coolant pump in MODES 4 and 5, with one or more RCS cold legs less than or equal to 285°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 100°F above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 4.6×10^5 lbs per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety

STEAM GENERATORS (Continued)

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.

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 = 0.5 gpm per steam generator). 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 0.5 gpm 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 leakage tubes will be located and 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 sleeving will be required for all tubes with imperfections exceeding the plugging or repair limit as defined in Surveillance Requirement 4.4.4.4. Defective tubes may be repaired by sleeving in accordance with CENS Report CEN-605-P, "Waterford 3 Steam Generator Tube Repair Using Leak Tight Sleeves," Revision 00-P, dated December 1992. 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. Sleeved tubes will be included in the periodic tube inspections for the inservice inspection program.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.1 prior 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 117 TO

FACILITY OPERATING LICENSE NO. NPF-38

ENTERGY OPERATIONS, INC.

WATERFORD STEAM ELECTRIC STATION, UNIT 3

DOCKET NO. 50-382

1.0 INTRODUCTION

By application dated December 6, 1993, as supplemented by letters dated May 12, August 9, and September 18, 1995, Entergy Operations, Inc. (the licensee), submitted a request for changes to the Waterford Steam Electric Station, Unit 3 (Waterford 3), Technical Specifications (TSs). The requested changes would allow installation of steam generator tube repair sleeves at Waterford 3. The proposal was for use of two types of leak tight sleeves designed by Combustion Engineering, Inc. (CE).

The two sleeve types are a tube sheet sleeve and a tube support plate sleeve. The tube sheet sleeve is installed by means of two different joint types: a rolled joint in the tube sheet end and an autogenous gas-tungsten arc weld (GTAW) at the free span end. The tube support sleeves are welded (autogenous GTAW) to the SG tube in the free span near each end of the sleeve. The material of construction for the sleeves is nickel alloy 690, a Code approved material (ASME SB-163), incorporated in ASME Code Case N-20.

Extensive analysis and testing were performed on the CE sleeves and sleeve-to-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 CEN-605-P, Revision 00-P "Entergy Operations, Inc. Waterford 3 Steam Generator Tube Repair Using Leak Tight Sleeves", dated December 1992 (proprietary) and report CEN-625-P "Verification of the ABB CENO Steam Generator Tube Sleeve Installation Process and Operating Performance," dated September 1995 (proprietary).

The staff has previously reviewed closely similar CE documents supporting requests for changes to the TSs 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 CE leak tight sleeves. This SE will discuss only those issues that warrant revision, amplification, or inclusion based upon current experience. A summary of the principal technical issues regarding the design and use of CE leak tight sleeves follows. Details of the prior staff evaluation of CE sleeves may be found in SE for Kewaunee Nuclear Power Plant, Docket No. 50-305, dated April 10, 1992, Arkansas Nuclear One, Unit No. 2, Docket No. 50-368, dated January 26, 1993, and Maine Yankee Atomic Power Station, Docket No. 50-309, dated April 14, 1995. These evaluations apply as well to the proposed Waterford license amendment.

The May 12, August 9, and September 18, 1995, letters provided additional information that did not change the initial proposed no significant hazards consideration determination.

2.0 SUMMARY OF PREVIOUS REVIEWS

Previous staff evaluations of CE sleeves addressed the technical adequacy of the sleeves in the 4 principal areas of pressure retaining component design: structural requirements, material of construction, welding, and non-destructive examination. The staff found the analyses and tests that were submitted to address these areas of component design to be acceptable.

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 that the sleeved tube assembly is capable of restoring steam generator tube structural integrity.

The material of construction of the sleeves is nickel Alloy 690, a Code approved material (ASME SB-163), covered by ASME Code Case N-20. The staff has found that the use of Alloy 690 thermally treated (TT) sleeves is an improvement over the Alloy 600 material used in the original steam generator tubing. Corrosion tests conducted under Electric Power Research Institute (EPRI) sponsorship confirm test results regarding the improved corrosion resistance of Alloy 690 TT over that of Alloy 600. Accelerated stress corrosion tests in caustic and chloride aqueous solutions have also indicated that Alloy 690 TT resists general corrosion in aggressive environments. Isothermal tests in high purity water have shown that, at normal stress levels, Alloy 690 TT has high resistance to intergranular stress corrosion cracking in extended high temperature exposure. The NRC has concluded as a result of these laboratory corrosion tests, that Alloy 690 is acceptable to NRC as meeting the guidelines in Regulatory Guide 1.85 (Rev. 24, July 1986). The NRC staff has approved use of Alloy 690 TT tubing in replacement steam generators as well as sleeving applications.

The welding process employed to join the sleeve to the parent tube is automatic autogenous GTAW (gas-tungsten arc welding). The application of this process to the CE sleeve design was specifically qualified and demonstrated during laboratory tests employing full scale sleeve/tube mock-ups.

Qualification of the welding procedures and welding equipment operators was performed in accordance with the requirements of the ASME Code, Section IX (Welding).

The sleeve assemblies can be inspected by nondestructive techniques in accordance with the recommendations of Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes." Nondestructive examination of sleeved tubes is conducted in two primary ways. Initial weld acceptance is performed using ultrasonic testing (UT). This NDE method is appropriate for detecting the types of weld flaws that may occur during the installation process. Service induced flaws are best detected using eddy current testing (ECT). Both techniques were developed and qualified in accordance with the ASME Code requirements and the recommendations of Regulatory Guide 1.83.

3.0 DISCUSSION AND EVALUATION

Recent experiences at two U.S. plants have indicated that the free span joint of a sleeved alloy 600 steam generator tube may be susceptible to stress corrosion cracking (SCC). These instances of SCC in sleeved tubes are limited to a joint type different from that proposed for Waterford. 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. Cracks have been detected in the alloy 600 parent tube material at the lower hard roll transition and lower hydraulic transition of free span joints. The cracks were detected after 4 to 7 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.

This experience with rolled joints is not suggestive of similar difficulties with welded joints. Service times exceeding 10 years have been achieved for sleeved tubes with GTAW joints at U.S. plants. No instances of service induced SCC have occurred in any of these joints.

The staff position on sleeving considers the method unable to assure an unlimited service life for a repaired tube. The conservative view is that sleeving creates new locations in the parent tube which may be susceptible to SCC 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 SCC. 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 SCC and present sleeve designs.

In recognition of the desirability for stress relief, the licensee submitted a proposal for stress relieving the welded sleeve joint(s) to increase the resistance to SCC. The method, which is proprietary, uses a post weld heat treatment (PWHT) developed by CE, which is applied to the weld and associated heat affected zone (HAZ) of free span joints. This PWHT is designed to provide optimum stress relief of the alloy 600 parent tube. The rolled joints performed within the tube sheet effectively isolate the alloy 600 from the environment and thus are not susceptible to SCC. Stress relief of these joints is unwarranted.

EPRI conducted a study of in-situ stress relief methods and performed demonstrations. When this study was performed, the tests and proposed applications were for cold worked, as opposed to welded, alloy 600. Although the forming practices vary, cold worked versus welded, the concepts and application of the EPRI study are equally applicable to welded alloy 600, since the SCC behavior is the same. The results were documented in EPRI report NP-4364-LD, published December 1985. The most essential parameters for performing such a heat treatment were outlined and verified by feasibility and demonstration tests. Verification of the heat treatment methods included tests of comparative corrosion resistance, microstructure, and residual stress measurements. The licensee's submittal and supporting documents draw heavily upon the methods and recommendations of EPRI report NP-4364-LD.

3.1 Qualification of PWHT Method

For process qualification, CE constructed mock-ups of a steam generator tube bundle. The test program was conducted to verify the acceptable performance of the annealing system in meeting the established EPRI heat treating guidelines. Typically, the mock-ups consisted of several tubes held in a frame to simulate a tube sheet and a tube support plate. Tests were conducted to establish or verify the different parameters involved with the heat treating operation. These tests included:

1. positioning (elevation) of the heat source with respect to the weld joint,

2. effect of heater radial position and temperatures achieved at the weld joint due to variations in tube/sleeve diameter,

3. emissivity (black body) effects from different oxides and the tube bundle geometry,

4. relative corrosion resistance for as-welded versus heat treated joints,

5. metallurgical examination of the heat treated joint, and,

6. effect of constraint on the tube during PWHT due to tube "locking" in adjacent tube support plates (TSPs).

Heater positioning with respect to the weld joint was accomplished using previously developed tooling with a known repeatability and accuracy. With this tooling, the effect of small positioning variations of the heater were quantified by thermocouple measurements. A tolerance band for heater positioning was established that was compatible with the accuracy of the positioning device. For the positional accuracy of the tooling, the heater can deliver the correct heat treating temperature at the weld and HAZ.

The EPRI program noted that the radial position of a heater within a sleeve could have a significant effect upon the achieved temperature around the circumference. Such variations, if they exist, must fall within the recommended temperature range to achieve an acceptable stress relief. CE designed a heater that minimized radial positioning problems, and consequent temperature variations, by its having a close fit to the sleeve inside diameter. Instrumented mock-ups verified that the range of achieved temperatures was well within the EPRI recommended temperature range.

Tube emissivity affects the temperature, which can be achieved for a given input wattage to the heater. Different tube/sleeve combinations were tested using tubing aged in an autoclave to produce the range of scale thicknesses and types observed in steam generators. Using the tube array mock-ups, CE established that the heater design was able to maintain the EPRI recommended range of PWHT temperatures for these varying emissivities.

In the EPRI study, accelerated corrosion tests were performed to measure the performance of cold worked versus heat treated material. It is well established that reduction of stress is key to avoiding or minimizing the occurrence of SCC, but the correlation between two different stress levels and the consequent material service life is difficult to predict. Additionally, direct measurement of residual stresses is difficult and uncertain. To avoid these uncertainties, accelerated corrosion tests were selected to verify the benefit of the heat treatment. This test method had the benefit of more directly relating the results of the heat treatment to the desired outcome of greater resistance to SCC.

Identical samples of welded joints were produced. One group was given a PWHT. Both groups were tested simultaneously in environments known to produce SCC in alloy 600. In all cases, the stress relieved samples showed superior resistance to SCC, in agreement with theory and the EPRI results. Thus, the technique developed by CE was verified to be beneficial in reducing weld joint susceptibility to SCC.

The combination of welding followed by PWHT poses the potential for undesirable alteration of the microstructure if the temperatures and times of exposure are excessive. Two possible deleterious effects would be sensitization or grain growth. EPRI established guidance on time and temperature for heat treating. The welding process CE employed was a low heat input process. Due to the low heat input, it was expected that the effects of welding would be minimal. Metallurgical examination of welded sleeve joints verified that adverse microstructural alteration of the materials was absent.

Next, the microstructures of welded and heat treated joints were examined. Joint samples were produced with a variety of PWHT times and temperatures. CE demonstrated that adherence to the EPRI guidelines precluded undesirable microstructural changes. Thus, the heat treatment was shown to be beneficial in reducing residual stresses without inducing undesirable microstructural changes. Additionally, for field application, the heat treatment control system is programmed to automatically limit PWHT temperature and time and to stop the process in the event of an instrument malfunction.

Recent field experience with the installation of welded sleeves with PWHT has indicated that SG tubes may be constrained ("locked") in their tube support plates, even though the TSPs are of designs which are less susceptible to this effect. 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 tube.

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).

3.2 Service Life of Sleeved Tube

Data and discussions were presented which attempted to use the accelerated corrosion test results to predict service life of the sleeve joint. During the development of the PWHT process, CE devised a service life prediction model that was based upon the accelerated corrosion test data. CE welded sleeves, without PWHT, have achieved 10 years of service without service induced problems. The accelerated corrosion tests demonstrated the PWHT joints to be superior in SCC resistance to the as-welded joints. Using these test data in the predictive model, CE estimated the stress relieved sleeve joints would last longer than the remaining licensed life of the plant. The staff has concluded that accelerated corrosion tests should provide a good qualitative assessment of relative service life for various sleeving processes. However, quantitative estimates of service life do not have a high degree of reliability. Periodic inspection as discussed in Section 2.3 and primary-to-secondary leakage monitoring will identify any premature degradation that may occur in the sleeved joints.

3.3 Inspection of Sleeve Joints

For compliance with the Code and Regulatory requirements for initial and periodic examinations under the Inservice Inspection program, the sleeve assemblies can be inspected by eddy current techniques in accordance with the recommendations of Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes."

Inspection of new sleeves is accomplished using two methods. Weld inspection is performed using ultrasonic tests (UT), with examination and acceptance criteria developed by CE and demonstrated with laboratory tests. After production welds are UT accepted, a base line multi-frequency eddy current test is performed of each new sleeve for comparison at future examinations.

For future examinations, the licensee has committed to employ an EPRI recommended, Appendix H qualified, eddy current examination method for sleeve inspections. This includes the use of Plus Point or CECCO probes, which are the present state-of-the-art.

The staff notes that other installations have performed on-site verification of the welding and UT techniques. This assures that plant specific conditions do not affect optimum performance of the welding and inspection methods prior to committing to a large production effort, should it be necessary.

3.4 Heat Treatment of Hydraulic Expansion Transition

Experience at other installations has shown that the hydraulic expansion transition of rolled joint sleeves may be susceptible to SCC. The staff has been monitoring these developments for potential impact on welded sleeve installations. Presently, accelerated corrosion tests of as-welded versus welded and heat treated joints indicate the hydraulic transition to have little or no susceptibility to SCC. The staff notes that the stress levels and crevice conditions at the facilities where cracks have been noted differ substantially from the joint design proposed for Waterford. The current staff position, based on currently available information, is that heat treatment of the hydraulic step is neither required nor prohibited.

3.5 Changes in the Technical Specifications

The licensee has proposed to revise TS 3/4.4.4, Steam Generators, to incorporate the tube sleeving as an acceptable alternative for repairing the degraded steam generator tubes. As discussed in this evaluation the staff finds the tube sleeving methods acceptable. Therefore, the changes in the TS are acceptable. In addition, there are some minor changes that are of an editorial nature, and they therefore are acceptable.

4.0 SUMMARY OF FINDINGS

Based upon the review and evaluation of the information and data presented in the aforementioned CE proprietary reports (CEN-605-P and CEN-625-P), it is concluded that the request by the licensee for a proposed amendment to the Facility License to modify the TS to permit repair of steam generator tubes by installation of sleeves using the CE methodology with PWHT of the welded joints as referenced in the amended TS, is acceptable. Further, the licensee has committed to employ proven and industry accepted eddy current examination techniques for sleeve inspections. The staff understands this commitment to include employment of subsequent advances in eddy current techniques as they become commercially available and industry accepted.

The staff concludes that the proposed sleeving repairs can be accomplished to produce a sleeved tube of acceptable metallurgical properties, strength, mechanical stability, leak tightness and corrosion resistance. We also find that the pre-service integrity of the sleeved tubes can be assured by implementing the proposed sleeve installation examinations.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Louisiana 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 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 (59 FR 2868). 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.

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