May 20, 1997

Mr. C. Randy Hutchinson
Vice President, Operations ANO
Entergy Operations, Inc.
1448 S. R. 333
Russellville, AR 72801

SUBJECT: ISSUANCE OF AMENDMENT NO. 184 TO FACILITY OPERATING LICENSE NO. NPF-6 - ARKANSAS NUCLEAR ONE, UNIT NO. 2 (TAC NO. M97292)

Dear Mr. Hutchinson:

The Commission has issued the enclosed Amendment No. 184 to Facility Operating License No. NPF-6 for the Arkansas Nuclear One, Unit No. 2 (ANO-2). This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated November 26, 1996, as supplemented by letter dated February 12, 1997.

The amendment changes the allowable primary-to-secondary leak rate and in the Surveillance Requirements section of the TSs it changes the acceptance criteria for steam generator tubes. The amendment changes the reference that is included in the tube acceptance criteria from Combustion Engineering topical report CEN-601-P Revision 01-P to CEN-630-P, Revision 01.

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:

George Kalman, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-368

Enclosures: 1. Amendment No.184 to NPF-6
2. Safety Evaluation

cc w/encls: See next page

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

May 20, 1997

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Vice President, Operations ANO
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Mr. C. Randy Hutchinson
Entergy Operations, Inc.

Arkansas Nuclear One, Unit 2

cc:

Executive Vice President
& Chief Operating Officer
Entergy Operations, Inc.
P. O. Box 31995
Jackson, MS 39286-199

Director, Division of Radiation
Control and Emergency Management
Arkansas Department of Health
4815 West Markham Street, Slot 30
Little Rock, AR 72205-3867

Winston & Strawn
1400 L Street, N.W.
Washington, DC 20005-3502

Manager, Rockville Nuclear Licensing
Framatone Technologies
1700 Rockville Pike, Suite 525
Rockville, MD 20852

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P. O. Box 310
London, AR 72847

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-8064

County Judge of Pope County
Pope County Courthouse
Russellville, AR 72801

Vice President, Operations Support
Entergy Operations, Inc.
P. O. Box 31995
Jackson, MS 39286-1995

Wise, Carter, Child & Caraway
P. O. Box 651
Jackson, MS 39205

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

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-368

ARKANSAS NUCLEAR ONE, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 184
License No. NPF-6

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

   A. The application for amendment by Entergy Operations, Inc. (the licensee) dated November 26, 1996, as supplemented by letter dated February 12, 1997, 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 license 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-6 is hereby amended to read as follows:

   2. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 184, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective within 30 days of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

George Kalman

George Kalman, Senior 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: May 20, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 184

FACILITY OPERATING LICENSE NO. NPF-6

DOCKET NO. 50-368

Revise 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 area of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE PAGES	INSERT PAGES
3/4 4-9	3/4 4-9
3/4 4-10	3/4 4-10
3/4 4-14	3/4 4-14
B 3/4 4-2	B 3/4 4-2
B 3/4 4-3	B 3/4 4-3
B 3/4 4-4	B 3/4 4-4

4.4.5.4 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 ≥20% of 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. The plugging or repair limit is equal to 40% of the nominal parent tube and sleeve wall thickness for sleeves installed in accordance with B&W Topical Report BAW-2045-PA-00 as supplemented by the information provided in B&W Report 51-1212539-00, "BWNS Kinetic Sleeve Design - Application to ANO Unit 2". The plugging limit is equal to 29% of the nominal sleeve wall thickness within the sleeve pressure boundary for sleeves installed in accordance with CENO Report CEN-630-P, "Repair of 3/4" O.D. Steam Generator Tubes Using Leak Tight Sleeves," Revision 01, dated November 1996.

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 4.4.5.3.c, 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 shall be performed after the 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:

1) B&W Topical Report BAW-2045PA-00 as supplemented by the information provided in B&W Report 51-1212539-00, "BWNS Kinetic Sleeve Design-Application to ANO Unit 2".

2) CENO Report CEN-630-P, "Repair of 3/4" O.D. Steam Generator Tubes Using Leak Tight Sleeves," Revision 01, dated November 1996. The post weld heat treatment described in CEN-630-P shall be performed.

4.4.5.5 Reports

a. Following each inservice inspection of steam generator tubes the number of tubes plugged or sleeved in each steam generator shall be reported to the Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be reported on an annual basis for the period in which the inspection was completed. This 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 pursuant to Specification 6.9.2 as denoted by Table 4.4-2. Notification of the Commission will be made prior to resumption of plant operation. The written Special Report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

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. A containment atmosphere particulate radioactivity monitoring system,

b. The containment sump level monitoring system, and

c. A 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 at least once per 24 hours when the required gaseous and/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 particulate and gaseous monitoring systems-performance of CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table 4.3-3.

b. Containment sump level monitoring system-performance of CHANNEL CALIBRATION at least once per 18 months.

REACTOR COOLANT S—TEM 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. 300 gallons per day total primary-to-secondary leakage through both steam generators and 150 gallons per day through any one steam generator,

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. Leakage as specified in Table 3.4.6-1 for those Reactor Coolant System Pressure Isolation Valves identified in Table 3.4.6.1.

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, 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, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two valves* in each high pressure line having a non-functional valve and be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

---

* These valves may include check valves for which the leakage rate has been verified, manual valves or automatic valves. Manual and automatic valves shall be tagged as closed to preclude inadvertent valve opening.

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 the limits specified by Specification 3.2.4 during all normal operations and anticipated transients.

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 MODES 4 and 5, a single reactor coolant loop or shutdown cooling loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two shutdown cooling loops to be OPERABLE.

The operation of one Reactor Coolant Pump or one shutdown cooling 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.

3/4.4.2 and 3/4.4.3 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 420,000 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 valves are OPERABLE, an operating shutdown cooling loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the Reactor Coolant System pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., no credit is taken for a direct reactor trip on the loss of turbine) and also assuming no operation of the steam dump valves.

Demonstration of the safety valves' lift setting will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves against water relief. The steam bubble functions to relieve RCS pressure during all design transients.

The requirement that 150 KW of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss-of-offsite power condition to maintain natural circulation at HOT STANDBY.

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.

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 = 150 gallons per day 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 150 gallons per day per steam generator can readily be detected by radiation monitors on the secondary system. 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.

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 tubes examinations. Plugging or sleeving will be required for all tubes with imperfections exceeding the plugging or repair limit as defined in Surveillance Requirement 4.4.5.4.a. Defective tubes may be repaired by sleeving in accordance with the B&W Topical Report BAW-2045PA-00 as supplemented by the information provided in B&W Report 51-1212539-00, "BWNS Kinetic Sleeve Design-Application to ANO Unit 2" or CENO Report CEN-630-P, "Repair of 3/4" O.D. Steam Generator Tubes Using Leak Tight Sleeves," Revision 01, dated November 1996. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the tube wall thickness. For sleeved tubes, the adequacy of the system that is used for periodic inservice inspection will be validated. Additionally, upgraded testing methods will be evaluated and appropriately implemented as better methods are developed and validated for commercial use.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3 certain results will be reported in a Special Report to the Commission pursuant to Specification 6.9.2 as denoted by Table 4.2-2. Notification of the Commission will be made 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 REACTOR COOLANT SYSTEM LEAKAGE

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 10 GPM IDENTIFIED LEAKAGE limitation provides allowances 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 Surveillance Requirements for RCS Pressure Isolation Valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS Pressure Isolation Valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

The total steam generator tube leakage limit of 300 gallons per day for all steam generators ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 150 gallon per day 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.

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.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduce the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2-hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 184 TO

FACILITY OPERATING LICENSE NO. NPF-6

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT NO. 2

DOCKET NO. 50-368

1.0 INTRODUCTION

By letter dated November 26, 1996, and as supplemented by letter dated February 12, 1997, Entergy Operations, Inc. (the licensee), submitted a request to change the Technical Specifications (TS) for Arkansas Nuclear One, Unit 2 (ANO-2). The proposed changes incorporate references to a new Combustion Engineering (CE) topical report describing steam generator (SG) tube sleeves, delete references to the previous CE topical report, change the plugging limit for a CE sleeve to 29% of the nominal sleeve wall thickness, require post weld heat treatment (PWHT) of sleeve welds, and reduce the allowable primary to secondary leakage through any one steam generator to 150 gallons per day (gpd). The information provided in the letter dated February 12, 1997, provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

By letter dated January 26, 1993, the NRC staff issued license amendment number 142 for ANO-2 allowing the repair of steam generator tubes using CE designed welded sleeves as described in the CE topical report CEN-601-P, Revision 01-P, "ANO-2 Steam Generator Tube Repair Using Leak Tight Sleeves," dated July 1992. The requested TS changes will reference a new generic topical report, CEN-630-P, Revision 01, "Repair of 3/4" O.D. Steam Generator Tubes Using Leak Tight Sleeves," dated November 1996. The latter report is not significantly different from CEN-601-P other than it addresses issues identified previously at Prairie Island Unit 1 (PI-1) associated with indications detected in weld joints of CE sleeves resulting from inadequate cleaning. The report also revises the sleeve plugging limit and renders optional the PWHT of the sleeve welds. Because the bulk of the technical and regulatory issues for the present request are identical to those reviewed in the previous Safety Evaluation (SE) described in our January 26, 1993 letter, this SE discusses only those issues warranting revision, amplification or inclusion based on current experience.

Details of prior staff evaluations of CE sleeves may also be found in the SEs for Waterford Steam Electric Station, Unit 3, Docket No. 50-382, dated December 14, 1995; Byron Nuclear Power Station, Units 1 and 2 and Braidwood

Nuclear Power Station, Units 1 and 2, Docket Nos. 50-454, 50-455, 50-456, and 50-457, dated April 12, 1996; and Zion Nuclear Power Station, Units 1 and 2, Docket Nos. 50-295 and 50-304, dated October 29, 1996. These evaluations apply to the proposed ANO-2 license amendment.

2.0 BACKGROUND

The two proposed CE sleeve types are an expansion transition zone (ETZ) sleeve and a tube support (TS) sleeve. An ETZ sleeve is designed to restore the portion of a tube in the vicinity of the top of the SG tubesheet. A TS sleeve can be used to span a support plate elevation or be used on a freespan section of tube. The sleeve material is a nickel-iron-chromium alloy, alloy 690, a Code approved material (ASME SB-163), incorporated in ASME Code Case N-20.

The CE sleeves are installed using gas tungsten arc welding to join the sleeve to the parent tube at the upper (free span) end of the ETZ sleeve and at both ends of a TS sleeve. The lower ETZ sleeve tube joint is hard-rolled into the tubesheet below the expansion zone. The centerline of the welds form the pressure boundary transition between the sleeve and the tube. The weld joint is the subject of the modifications to the installation processes described in the new topical report.

During the Spring 1996, refueling outage at PI-1, roughly 60 upper weld joints in CE sleeved tubes had eddy current testing (ET) indications. Discovery of most of the indications was the result of the licensee employing a new, more sensitive ET probe for its periodic inspection of SG tubes. Tube/sleeve assemblies were removed from the SGs for metallurgical examination and root cause determination. It was found that the ET indications were due to entrapped oxides and/or weld suckback within the sleeve to tube weld. The cause of these weld defects was traced to a previously revised tube cleaning procedure.

As a result of the metallurgical examination, the tube cleaning procedure was revised and revised post cleaning visual inspections (VT) were adopted. The initial weld acceptance inspection, an ultrasonic test (UT), was revised to give greater sensitivity. As an added measure, the initial baseline ET, normally used only as reference for later periodic reinspection, was modified to supplement the UT as part of the initial weld acceptance inspection. All of these refinements to the sleeving procedure were confirmed using a large number of laboratory samples and field mock-ups. These modifications were incorporated into a new generic topical report, CEN-630-P, referenced above and are discussed in more detail in the following section.

3.0 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 in previous SEs. These include weld preparation, weld acceptance inspections, sleeve plugging limits, service life predictions for sleeved SG tubes, and primary-to-secondary leakage limits.

3.1 Weld Preparation

Prior to performing any weld, the surface of the metal(s) to be welded must be cleaned. For sleeve installation, the inner diameter of the parent tube at the desired weld location must be cleaned of service induced oxides. For the CE sleeving process, this is accomplished using motorized wire brushes.

Based upon the metallurgical findings, CE revised the cleaning method to ensure optimum removal of service induced oxides. The revised cleaning procedure entailed some equipment changes. More significantly, from a quality assurance standpoint, a 100 percent VT of the cleaning process was instituted. After the wire brush cleaning step, every tube is given a VT using a remote fiber optic camera system to confirm that adequate surface cleaning has been accomplished. CE advises the 100 percent VT is an interim step until enough field experience is gained to consider adoption of a statistical sampling plan in the future.

3.2 Weld Acceptance Inspections

For compliance with the Code and regulatory requirements, initial and periodic examinations of steam generator tubes and sleeves are performed. Sleeve welds were historically accepted based on VT and UT examinations. ET was used for an initial baseline inspection for comparison with later required periodic inspections. The reason for the different types of nondestructive examinations (NDE) being used for initial acceptance versus periodic reinspection is due to the differences between potential flaws from initial installation defects and service induced degradation. The different NDE techniques have normally been better suited for the respective types of anticipated flaws.

The PI-1 event suggested that the current initial acceptance examinations (VT and UT) may not be sufficient in every circumstance. As a result, the weld acceptance NDE was modified to include:

* 100 percent UT with an enhanced digitized amplitude system
* 100 percent ET using the Plus Point probe

The PI-1 event indicated that cleaning the parent tube prior to welding is a critical step in forming a defect-free sleeve to tube weld. Thus the new CE topical report requires a 100 percent VT of the parent tube after cleaning.

The original UT procedure was based upon the absence of a mid-wall reflection. In that procedure, the sleeve outside diameter wall reflection was readily apparent beyond the fusion zone of the weld, thus signifying lack of fusion with the parent tube. When fusion existed, the mid-wall reflection (mid-wall of the fused sleeve and tube combination) would not appear since no interface would exist. The PI-1 event led CE to discover that lack of fusion caused by axially oriented oxide inclusion from a poorly cleaned weld would not be detected since the oxides did not cause a large sound reflection.
In the enhanced UT procedure, the back wall signal from the outside of the parent tube is also monitored for presence in the fused area. Additionally,

the back wall signal strength is examined for excessive attenuation. Attenuation beyond the normal amount can be interpreted, along with other signal artifacts, as either a weld that is too narrow or one with inclusions or patches of unfused material. The modified UT procedure was extensively tested on laboratory produced welds containing a variety of inclusion/lack of fusion defects. Samples were destructively examined and the metallurgical sections compared with the UT results. Comparison of results demonstrated the revised UT procedure was highly reliable, and that no significant defects could remain undetected by the enhanced UT procedure.

ET with the plus point probe is now part of the sleeve weld acceptance criteria. The PI-1 event led CE to discover that weld suckback and circumferentially oriented oxide inclusions from a poorly cleaned weld would not be detected by UT. CE has shown the plus point probe reliably detects the various process-induced weld defects including blowholes, weld suckback and circumferentially oriented oxide inclusions. CE has also shown the ET can reliably locate the position of the defect with respect to the weld centerline which is considered the pressure boundary. ET indications located above the weld centerline that meet UT requirements can be left in service. Any ET indication found below the weld centerline requires the tube to be plugged.

For future sleeve inspections, the licensee will follow EPRI guidelines for determination of inspection scope and expansion criteria. The licensee will use Electric Power Research Institute (EPRI) "PWR Steam Generator Tube Examination Guidelines" Appendix G qualified personnel and Appendix H qualified ET techniques.

3.3 Sleeve Plugging Limits

The 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 NDE uncertainty and postulated operational growth of tube wall degradation within the sleeve must be accounted for when using NDE to determine sleeve plugging limits. Therefore, a conservative tube wall combined allowance for postulated degradation growth and eddy current uncertainty of 20% throughwall per cycle was assumed for the purpose of determining the sleeve plugging limit. The sleeve plugging limit, which was calculated based on the most limiting of normal, upset, or faulted conditions for 3/4-inch outside diameter steam generator tubes in CE designed generators, was determined to be 49% of the sleeve nominal wall thickness based on ASME Code minimum material properties in accordance with staff positions. Removal of tubes and/or sleeves from service when degradation reaches a plugging limit of 29% provides assurance the minimum acceptable wall thickness will not be violated during the next subsequent cycle of operation.

3.4 Post Weld Heat Treatment

Accelerated corrosion tests confirm a PWHT significantly improves the intergranular stress corrosion cracking resistance of the alloy 600 parent tube material in the weld zone. The licensee committed to performing a PWHT of the welded joints in accordance with the CE topical report and NRR staff position. This commitment is reflected in the TS.

3.5 Primary to Secondary Leakage Limits

With respect to the staff position regarding primary to secondary leakage limits, the licensee proposes to change its TS adopting a 150 gpd per SG leakage limit.

3.6 Technical Specification Changes

The staff finds acceptable the following proposed changes to the plant TS 4.4.5.4 and 3.4.6.2 and associated bases.

1. The definition of the sleeve plugging limit is modified to incorporate a revised plugging limit of 29%.

2. The CE sleeve installation reference document is changed to indicate CE sleeves will be installed as described in CE report CEN-630-P.

3. A PWHT of the sleeve welds as described in CE report CEN-630-P shall be performed.

4. The allowable primary to secondary leakage through any one SG is reduced to 150 gallons per day.

4.0 <u>TECHNICAL CONCLUSIONS</u>

The staff concludes the proposed sleeving repairs, as described in the new CE sleeve topical report, can be accomplished to produce sleeved tubes of acceptable metallurgical properties, structural integrity, leak tightness and corrosion resistance. The staff also finds acceptable the proposed preservice and future inspection methods for examining the tube/sleeve assemblies.

The NRC staff concludes the repair of SG tubes using welded sleeves designed by CE is acceptable, as implemented through appropriate TS changes that 1) modify the sleeve plugging limit, 2) reference the updated CE topical report, 3) perform a PWHT of the welded joints, and 4) modify the TS requirements to incorporate a primary to secondary leakage limit of 150 gpd maximum per SG.

5.0 <u>STATE CONSULTATION</u>

In accordance with the Commission's regulations, the Arkansas 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 (61 FR 64376). 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 Contributors: S. M. Coffin
G. P. Hornseth

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