Entergy

Entergy Operations, Inc.
1448 S.R. 333
Russellville, AR 72801
Tel 501-858-4888

Craig Anderson
Vice President
Operations ANO

June 29, 2000

2CAN060013

U. S. Nuclear Regulatory Commission
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Subject: Arkansas Nuclear One - Unit 2
Docket No. 50-368
License No. NPF-6
Response to Request for Additional Information Regarding the August 18, 1999, Steam Generator Inspection Requirements License Amendment Request

Gentlemen:

In a letter dated August 18, 1999 (2CAN089905), Entergy Operations, Inc. submitted a license amendment request for Arkansas Nuclear One, Unit 2 (ANO-2) regarding modification of the surveillance requirements for the steam generators due to the planned replacement of the steam generators in the fall of this year. In a letter from the Nuclear Regulatory Commission (NRC) dated May 9, 2000, the staff asked four questions in regard to the August 18, 1999 letter.

The ANO staff's responses to the four questions are provided in Attachment 1 to this letter. As a result of the staff's review, minor revisions were required to three of the technical specification pages submitted in the August 18, 1999, license amendment request. These three pages are included in Attachment 2. Additionally, for completeness, all of technical specification pages from the August 18, 1999, submittal are included in Attachment 2; however, only those revisions required as a result of NRC review are discussed in Attachment 1. The discussions associated with the remainder of the proposed technical specification changes are contained in the August 18, 1999, letter and are not repeated in this submittal. Attachment 3 contains a non-proprietary version of WCAP 15406, "Regulatory Guide 1.121 Analysis for Arkansas Nuclear One Unit 2 Replacement Steam Generators." The proprietary version of WCAP 15406, including the basis for classifying the information as proprietary, will be submitted in a separate letter.

A001

The proposed changes have been evaluated in accordance with 10CFR50.91(a)(1) using criteria in 10CFR50.92(c) and it has been determined that the no significant hazards considerations contained in our August 18, 1999, submittal remain bounding.

Should you have any questions or comments, please contact me.

Very truly yours,

CGA/dwb
Attachment/enclosures

To the best of my knowledge and belief, the statements contained in this submittal are true.

SUBSCRIBED AND SWORN TO before me, a Notary Public in and for Pope **County and the State of Arkansas, this** 29 **day of** June**, 2000.**

Notary Public
My Commission Expires 12/15/2007

"OFFICIAL SEAL"
Andrea Pierce
Notary Public, State of Arkansas
County of Pope
My Commission Exp. 12/15/2007

cc: Mr. Ellis W. Merschoff
Regional Administrator
U. S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-8064

NRC Senior Resident Inspector
Arkansas Nuclear One
P.O. Box 310
London, AR 72847

Mr. Thomas W. Alexion
NRR Project Manager Region IV/ANO-2
U. S. Nuclear Regulatory Commission
NRR Mail Stop 04-D-03
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

Mr. David D. Snellings
Director, Division of Radiation
Control and Emergency Management
Arkansas Department of Health
4815 West Markham Street
Little Rock, AR 72205

ATTACHMENT

TO

2CAN060013

PROPOSED TECHNICAL SPECIFICATION

AND

RESPECTIVE SAFETY ANALYSES

IN THE MATTER OF AMENDING

LICENSE NO. NPF-6

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT 2

DOCKET NO. 50-368

ANO Responses to NRC Staff Questions Regarding the Steam Generator Inspection Requirements License Amendment Request

NRC Question #1

In the August 18, letter, the licensee stated that an analysis per Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," will be performed to establish the minimum acceptable wall thickness and the unacceptable defects for the replacement steam generator tubes. Although the licensee indicated which analytical codes and the loading combinations that would be used, the licensee had not completed the analysis, and thus, did not provide a summary of the analysis as recommended by RG 1.121. Therefore, the staff requests that the licensee provide the summary of the analysis to support its proposed minimum wall thickness and plugging criteria.

ANO Response

The Regulatory Guide 1.121 analysis has been completed using the standard Westinghouse methodology and confirms a 40% plugging limit. The controlling structural limit is determined to retain margins of safety against burst consistent with the safety factor margins implicit in the stress limit criteria of the ASME Code, Section III, as referenced in 10 CFR 50.55a, for all service level loadings. Satisfaction of these criteria means that all tubes have been determined to retain the required margin against gross failure or burst under normal plant operating conditions. In addition, the tubes have been determined to retain a margin of safety against gross failure or burst consistent with the margin of safety determined by the stress limits in NB-3225 of Section III of the ASME Code under postulated accidents concurrent with a safe shutdown earthquake. This analysis is summarized in WCAP 15406, "Regulatory Guide 1.121 Analysis for Arkansas Nuclear One Unit 2 Replacement Steam Generators" with a defined structural limit of 57.5% allowable defect depth.

An evaluation of potential defects in replacement steam generator tubing was performed which determined the expected mechanism would be anti vibration bar (AVB) wear. Based upon experience with similar Westinghouse replacement steam generator designs, and a tube wear evaluation specifically for ANO-2 replacement steam generators, an allowance of 5% growth between inspections is established to conservatively bound expected AVB wear. Eddy current sizing uncertainty for wear has been quantified for the bobbin coil probe, which is the inspection probe ANO-2 will be using for general examinations. The technique has been qualified for sizing of wear in accordance with Appendix H of the Electric Power Research Institute (EPRI) Report TR-107569, Vol. 1, Rev. 5, "PWR Steam Generator Examination Guidelines." The sizing root mean square error for the bobbin probe is 4.9% through-wall. Therefore, a 15% allowance for NDE uncertainty and growth between inspections is conservative and the current technical specification plugging limit of 40% is bounded. A non-proprietary version of WCAP 15406 is included in Enclosure 2.

NRC Question #2

The licensee stated that it will be using techniques for flaw detection and sizing as outlined in the EPRI report TR-107569, Vol. 1, Rev. 5, "PWR Steam Generator Examination Guidelines." Surveillance Requirement (SR) 4.4.5.a.10 defines the preservice inspection as an inspection of the full length of each tube in the steam generator performed by eddy current techniques. Since the current SRs define tube inspection as an inspection from the hot-leg tube end, around the U-bend, to the top cold-leg support, provide a discussion of the scope and techniques for the first inservice inspection and whether the licensee will be following the EPRI guideline recommending a 100 percent full-length (end-to-end) inspection and an inspection of all plugs, if any.

ANO Response

Consistent with the EPRI guidelines, a 100% bobbin preservice examination from tube end (cold leg side) to tube end (hot leg side) will be performed onsite prior to installation of the replacement steam generators. The first inservice tube inspection will be performed following the first cycle of operation in accordance with the EPRI guidelines. The SR 4.4.5.4.a.9. <u>Tube Inspection</u> definition has been revised accordingly (see Enclosure 1, page 3/4 4-9). The definition, as currently written and approved, allows for less than a 100% inspection of the tube for the preservice and first inservice inspections.

Regarding the plug inspection, there is only one tube that will potentially be plugged at initial startup due to a failure of the mandrel during the hydraulic expansion process. It will be repaired using a welded shop plug or a mechanical I-690 rolled plug. Plugs will receive a visual examination during the next refueling outage (2R15) and will be inspected in future outages in accordance with EPRI Report TR-107569, Vol. 1.

NRC Question #3

In SR 4.4.5.3.a the licensee proposed that wear degradation not be included in the criteria for determining the tube inspection condition result as applied to allowing an extension of the surveillance interval. Since wear is the most probable degradation mechanism to be experienced in the early inspections, the staff requests that this proposed change be deleted.

ANO Response

We concur with the NRC's request. We withdraw the proposed change to exclude wear degradation from SR 4.4.5.3.a. The proposed added phrase "with the exception of wear" will be removed from the SR (see Enclosure 1, page 3/4 4-8). It was always Entergy's position to include wear as one of the degradation modes to be evaluated. The growth rate of identified wear will be taken into consideration for calculating the next operating interval.

The original request to exclude it was based on the fact that wear is easily detected with the bobbin-coil eddy current technique, which is approved through the EPRI guidelines as a qualified sizing technique. Additionally, the growth rate associated with wear is very low (on the order of 2% through-wall per 18 months) and is predictable.

NRC Question #4

The licensee has proposed to delete wastage as one of the mechanisms in its definition of degradation in SR 4.4.5.4.a.3. Although it is less likely to occur, it remains as a type of degradation that should be included. The staff requests that this proposed change be deleted.

ANO Response

We concur with the NRC's request. We withdraw the proposed change to delete the reference to wastage in the technical specification. The reference to wastage will remain in the definition for degradation in SR 4.4.5.4.a.3 and in the first sentence of the third paragraph of Bases Section 3/4.4.5 (see Enclosure 1, pages 3/4 4-9 and B 3/4 4-3, respectively). The original request to delete it was because wastage of tubing material is only associated with phosphate chemistry control on the secondary side of the generator and phosphate chemistry control has not been utilized by the industry in many years. During normal monitoring of the tubing by bobbin testing, wastage would be detected and repaired.

ENCLOSURE 1

PROPOSED TECHNICAL SPECIFICATION CHANGES

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 generator(s) to OPERABLE status prior to increasing Tavg 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.

NOTE: The requirements for inservice inspection do not apply during the steam generator replacement outage (2R14).

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 Table 4.4-2. 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. 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 (>20%).

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

3. A tube inspection (pursuant to Specification 4.4.5.4.a.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 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 result of each sample inspection shall be classified into one to 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 (>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 following the steam generator replacement outage. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If the first inspection following the preservice inspection results 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 4.4.5.3.a; 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 tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2.

   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 feedwater line break.

4.4.5.4 Acceptance Criteria

a. As used in this Specification

1. Tubing or Tube means that portion of the tube 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 limit. A tube containing a defect is defective.

7. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging because it may become unserviceable prior to the next inspection. The plugging limit 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 4.4.5.3.c, above.

9. Tube Inspection means an inspection of the steam generator tube from tube end (cold leg side) to tube end (hot leg side).

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 hydrostatic test and prior to 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 all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.5.5 Reports

a. Following each inservice inspection of steam generator tubes the number of tubes plugged 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 within 12 months following the completion of the inservice inspection. 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.

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 (i.e., prior to entering Mode 4). 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.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

<table>
<tr><th colspan="3">1ST SAMPLE INSPECTION</th><th colspan="2">2ND SAMPLE INSPECTION</th><th colspan="2">3RD SAMPLE INSPECTION</th></tr>
<tr><th>Sample Size</th><th>Result</th><th>Action Required</th><th>Result</th><th>Action Required</th><th>Result</th><th>Action Required</th></tr>
<tr><td rowspan="9">A minimum of S Tubes per S.G.</td><td>C-1</td><td>None</td><td>N/A</td><td>N/A</td><td>N/A</td><td>N/A</td></tr>
<tr><td rowspan="5">C-2</td><td rowspan="5">Plug defective tubes and inspect additional 2S tubes in this S.G.</td><td>C-1</td><td>None</td><td>N/A</td><td>N/A</td></tr>
<tr><td rowspan="3">C-2</td><td rowspan="3">Plug defective tubes and inspect additional 4S tubes in this S.G.</td><td>C-1</td><td>None</td></tr>
<tr><td>C-2</td><td>Plug defective tubes</td></tr>
<tr><td>C-3</td><td>Perform action for C-3 result of first sample</td></tr>
<tr><td>C-3</td><td>Perform action for C-3 result of first sample</td><td>N/A</td><td>N/A</td></tr>
<tr><td rowspan="3">C-3</td><td rowspan="3">Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in the other S.G.

Special Report to NRC per Specification 6.9.2</td><td>Other S.G. is C-1</td><td>None</td><td>N/A</td><td>N/A</td></tr>
<tr><td>Other S.G. is C-2</td><td>Perform action for C-2 result of second sample</td><td>N/A</td><td>N/A</td></tr>
<tr><td>Other S.G. is C-3</td><td>Inspect all tubes in the other S.G. and plug defective tubes.

Special Report to NRC per Spec. 6.9.2</td><td>N/A</td><td>N/A</td></tr>
</table>

$S = 3 \frac{2}{n} \%$ Where n is the number of steam generators inspected during an inspection.

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.

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 will be required for all tubes with imperfections exceeding the plugging limit as defined in Surveillance Requirement 4.4.5.4.a. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that could affect tube wall integrity. 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.

ENCLOSURE 2

WCAP 15406, "Regulatory Guide 1.121 Analysis for Arkansas Nuclear One Unit 2 Replacement Steam Generators" (non-proprietary version)