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U. S. Nuclear Regulatory Commission
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Subject: Arkansas Nuclear One - Unit 2
Docket No. 50-368
License No. NPF-6
Proposed Technical Specification Change Request Concerning Steam Generator Inspection Requirements for the Replacement Steam Generators

Gentlemen:

Attached for your review and approval is a proposed Arkansas Nuclear One, Unit 2 (ANO-2) Technical Specification amendment associated with the Replacement Steam Generator (RSG) Project that modifies the surveillance requirements for the steam generators. In order to facilitate the replacement of the original steam generators, the requirements for tube repair (sleeving) will be removed and the requirements for preservice and inservice inspection of the tubes will be clarified. Changes to the inspection interval and reporting requirements are also requested.

Entergy is aware of efforts underway to develop revised technical specifications regarding steam generator tube inspections; however, these efforts have not yet culminated in mutually agreeable language between the industry and the NRC. The ANO-2 proposed technical specifications will continue to utilize the general guidance of the existing standard technical specifications language of NUREG-0212, Rev. 4, "Standard Technical Specifications Combustion Engineering Plants."

The proposed change has been evaluated in accordance with 10CFR50.91(a)(1) using criteria in 10CFR50.92(c) and it has been determined that this change involves no significant hazards considerations. The bases for these determinations are included in the attached submittal.

Entergy Operations requests that the effective date for implementation of this change be prior to restart from the 2R14 refueling outage which is currently scheduled to begin on September 15, 2000. Although this request is neither exigent nor emergency, your timely review is requested.

A001

Very truly yours,

C. Randy Hutchinson

CRH/dwb
Attachments

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 Johnson **County and the State of Arkansas, this** 18 **day of** August **, 1999.**

Juana M. Tapp
Notary Public
My Commission Expires 8-11-2000

OFFICIAL SEAL
JUANA M. TAPP
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My Commission Expires: 11 - 8 - 2000

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ATTACHMENT

TO

2CAN089905

PROPOSED TECHNICAL SPECIFICATION

AND

RESPECTIVE SAFETY ANALYSES

IN THE MATTER OF AMENDING

LICENSE NO. NPF-6

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT TWO

DOCKET NO. 50-368

DESCRIPTION OF PROPOSED CHANGES

To accommodate replacement of the original steam generators at Arkansas Nuclear One - Unit 2 (ANO-2), the requirements for tube repair currently specified in ANO-2 Technical Specifications (TS) Surveillance Requirements (SR) 4.4.5.0 and the associated Bases for Specification 3/4.4.5 need to be revised. The resulting ANO-2 requirements for the replacement steam generators will be consistent with the requirements in the standard technical specifications (i.e., NUREG-0212, Revision 4) that do not include provisions for repair (sleeving) of the tubes. The requirements for preservice and inservice inspection of tubes will also be clarified. The specific changes are as follows:

- Technical Specification 4.4.5.0 will be revised to add a "Note" exempting the replacement steam generators from inservice inspection requirements during the replacement steam generator outage.

- Technical Specification 4.4.5.2 will be revised to delete the reference to Table 4.4-3 regarding steam generator tube sleeve inspection and the requirement that tubes selected for inservice inspection include at least 20% of each type of installed sleeves.

- Technical Specification 4.4.5.3 will be revised to extend the inspection interval to a maximum of once per 40 months provided the inspection results from the first inspection following the preservice inspection fall into the C-1 category.

- Technical Specifications 4.4.5.2.c, 4.4.5.3.b, and 4.4.5.3.c will be revised to delete the reference to Table 4.4-3.

- Technical Specifications 4.4.5.4.a.1, 4.4.5.4.a.6, and 4.4.5.4.a.7 will be revised to delete the references to sleeving, repair limit, and the B&W and CENO repair reports. Technical Specification 4.4.5.4.a.3 will be revised to delete the word "wastage."

- Technical Specification 4.4.5.4.a.10 will be revised to delete the requirement that the baseline preservice eddy current inspection be performed after the field hydrostatic test.

- Technical Specification 4.4.5.4.b will be revised to delete the references to sleeving, repair limit, Table 4.4-3, and the B&W and CENO repair reports.

- Technical Specification 4.4.5.5.a and 4.4.5.5.b will be revised to delete the requirement that inservice inspection reports include the number of sleeved tubes. Section 4.4.5.5.b is also revised to be consistent with the guidance for reporting frequency contained in NEI 97-06, Revision 0, "Steam Generator Program Guidance."

- Technical Specification 4.4.5.5.c will be revised to delete the reference to Table 4.4-3. Section 4.4.5.5.c. is also revised to clarify what is meant by "prior to resumption of plant operation."

- Technical Specification Table 4.4-2 will be revised to delete words "non-repaired" from the title and "or sleeve" from the Action Required columns.

- Technical Specification Table 4.4-3 will be deleted.

- Bases for Technical Specification 3/4.4.5 will be revised to remove the references to sleeving, repair and the B&W and CENO repair reports. Additionally, the term "wastage-type" defects and reference to reliably detecting 20% tube degradation will be removed.

BACKGROUND

Prior to 1991, the original steam generators at ANO-2 only experienced mechanical wear at the batwing tube support straps due to cross flow in the upper tube bundle region. In 1991, axially oriented stress corrosion cracking at the eggcrate tube support plates was identified. The cracking was initiated on the outside portion of the tubing on the secondary side of the original steam generators due to corrosion product transport and buildup from the balance-of-plant piping and components.

In 1992, circumferential cracking was first discovered in the portion of tubing just above where the tube is expanded into the tubesheet. This expansion transition area is a region of higher residual stress where the tube decreases in diameter from the expansion down to the nominal diameter of the tubing (3/4").

Because of the large number of defective tubes, sleeves were installed in the tubes at the top-of-tubesheet area to remove the defective area from the pressure boundary and reduce lost margin to the tube plugging limit (maximum number of tubes that could be plugged).

Since 1992, ANO-2 experienced axial cracking in freespan regions of the tubing between the tube support plates and areas of the tube where denting has occurred due to a buildup of secondary side corrosion products reducing the diameter of the tubing.

The original steam generators at ANO-2 will be replaced during the refueling outage that follows fuel cycle 14. ANO-2 will subsequently operate with replacement steam generators starting with fuel cycle 15.

The U-tubes in the ANO-2 steam generators provide for the transfer of heat between the primary and secondary coolants and are part of the primary pressure boundary. The tubes are periodically inspected for indication of degradation. If significant degradation is found, the tube may be removed from service by installing a plug at each end. Plugging of a tube results in a slight reduction of the available heat transfer area and the flow area for the primary coolant. If a large number of tubes are plugged, safety analyses may need to be revised and a power reduction may be required. To address the adverse effects of extensive tube plugging, repair of tubes using a sleeve has been developed. The sleeve spans the degraded area and a connection between the tube and sleeve is made above and below the degraded area.

The sleeving repair design is approved on a sleeve and tube design specific basis. Since there are several differences in tubes in the replacement steam generators from those in the original steam generators, the referenced sleeve designs and associated reports are no longer applicable and are to be removed from the ANO-2 Technical Specifications.

The replacement steam generators use thermally treated Alloy 690 tubes and other design enhancements. Because of the improved materials and manufacturing processes, the tubing in the replacement steam generators is not expected to experience the amount of degradation that has been seen in the original steam generators. This expectation is supported by operating experience with other plants that have replacement steam generators built with the same materials and manufacturing processes. Therefore, a new sleeving design appropriate to the new tube diameter is not expected to be required.

The ASME Code, Section III hydrostatic test for the replacement steam generators is completed in the manufacturing shop prior to shipment. The pressure testing in an operating plant is limited by the rules of ASME Code Section XI. The pressure testing following a repair in an operating plant is a function of the operating pressure. Preservice inspection of the tubes at the ANO-2 site will provide verification that the tubes were not adversely affected by the maximum pressure that they experienced during hydrostatic testing.

DISCUSSION OF CHANGE

For the purpose of evaluation, the requested changes to the ANO-2 Technical Specifications for tube inspection can be grouped into the following six areas:

1. The first area of change is to add a "Note" to exempt the replacement steam generators from inservice inspection requirements during the steam generator replacement outage. Since the replacement steam generators will be subjected to a preservice inspection at the ANO-2 site prior to installation, there is no need to perform inservice inspection upon installation.

2. The second area of change is to remove the references to repair of the original steam generator tubes by sleeving since they are not applicable to the tubes in the replacement steam generators. These changes modify the ANO-2 Technical Specifications to be consistent with the standard technical specification (i.e., NUREG-0212, Revision 0) for tube inspection. The current revised standard technical specifications (NUREG-1432, Rev. 1) do not propose revised testing technical specifications at this time. The tubing diameter and thickness in the replacement steam generators are different from the original steam generators. Therefore, the sleeving designs described in the referenced reports are inappropriate. The replacement steam generators use metallurgy and fabrication technology that has proven to be very resistant to corrosion related degradation. A sleeving design for the replacement steam generators is not expected to be required. Tubes in the replacement steam generators will require plugging for imperfections exceeding the 40% plugging limit. The plugging limits will consist solely of the through-

wall criteria and will require that tubes be plugged when degradation meets or exceeds the plugging limit of 40% of the nominal wall thickness. ASME Section XI, 1992 edition, provides the depth of an allowable outside diameter (OD) flaw for tubes in service. The replacement steam generators have tubing fabricated from SB-163 Alloy UNS N06690 (Alloy 690) inspected preservice per the requirements of Section III, 1992 edition, NB-2250. For such tubing, ASME Section XI IWB-3521.1 defines the depth of allowable indications: for tubing having a radius to thickness (r/t) ratio of less than 8.7, the depth of an allowable OD flaw shall not exceed 40% of the tube wall thickness. The replacement steam generator tubing has an r/t ratio less than 8.70 using either the inside diameter or the OD. Using the OD is conservative:

$$R/t = (OD/2) / (t) = (0.688"/2) / (0.040") = 8.60$$

In addition, analysis per Regulatory Guide 1.121 will be performed to establish the required minimum wall thickness for the replacement steam generators. The Regulatory Guide 1.121 analysis will be performed using Westinghouse methodology and will include assessment of the loads that result from postulated accident conditions, including a safe shutdown earthquake (SSE) in combination with a loss of coolant accident (LOCA) break, a steam line break (SLB) or a feedwater line break (FLB).

For the LOCA plus SSE analysis of dynamic loadings, the most recent version of the ABB-CE computer code CEFLASH-4A will be used to generate the dynamic response (i.e., pressure time histories) due to a LOCA pipe break. The Westinghouse WECAN/Plus computer code will be used for the structural analysis, including the development and assessment of the loads resulting from the SSE in combination with the dynamic response due to a LOCA pipe break. These computer codes are consistent with those used in the pipe break analysis for the original steam generators. CEFLASH-4A is the same computer code that was used for the original steam generators. WECAN/Plus is a Westinghouse developed proprietary structural analysis computer code that is functionally equivalent to the industry developed ANSYS computer code that was used for the original steam generators. WECAN/Plus is the workstation version of the WECAN (Westinghouse Electric Computer Analysis) computer code that has been used extensively by Westinghouse in the design analyses of NSSS seismic Category I components designated as ASME Code Class 1, 2 or 3. The WECAN and WECAN/Plus computer codes have been used by Westinghouse in previous replacement steam generator Regulatory Guide 1.121 analysis. The use of WECAN/Plus in the ANO-2 Regulatory Guide 1.121 analysis will be a first time application to ANO-2.

For the Westinghouse designed tube bundle, dynamic load effects on the tubes resulting from SLB/FLB loads do not result in significant tube stresses relative to the determination of the tube minimum wall thickness. During the postulated SLB/FLB accidents, the predominant primary tube stresses result from the primary to secondary side through-wall pressure gradient. The peak differential pressures for these events are obtained from the results of transient blowdown analyses that are based on a full double-ended rupture of the main steam line/main feedwater line. In both cases, the secondary side of the faulted

steam generator blows down to the ambient pressure. The SLB/FLB dynamic loads on the Westinghouse designed tube bundle are sufficiently small that they are not limiting, and therefore, do not require detailed analysis.

3. The third change is to revise the inspection interval to extend it to a maximum of once per 40 months provided the inspection results from the first inspection following the preservice inspection fall into the C-1 category. Significant industry knowledge has been gained from monitoring the performance of steam generators that have been replaced. Alloy 690 tubing material has proven to be superior to Alloy 600 in regard to corrosion resistance. Plants that have utilized Alloy 690 tubing in their replacement steam generators have not experienced corrosion-induced degradation (some plants have as many as 10 years operating experience without any detectable tube degradation). Also, tubing that has been thermally treated has shown little, if any, corrosion-induced degradation. The primary failure mechanism expected with this type of tube is mechanical wear due to flow-induced vibration.

   Due to the design and construction of the replacement steam generators, and based on the operating experience of the Alloy 690 material, it is acceptable to extend the inspection interval to a maximum of once per 40 months if: 1) the inspection results from the first inspection following the preservice inspection fall into the C-1 category, and 2) after one cycle of operation no corrosion-induced mechanisms are identified and wear is found to be minimal such that growth would not exceed 40% through-wall for the next inspection interval. The basis for excluding wear is that the mechanism is easily detected with the bobbin-coil eddy current technique, which is approved through the Electric Power Research Institute (EPRI) Guidelines as a qualified sizing technique. Typically, the growth rate associated with wear is very low (on the order of 2% through-wall per 18 months) and is predictable. The operational assessment performed following each inservice inspection will address the detection capabilities and growth rates associated with the wear that is identified. Based on the length of the next operating interval, the evaluation would determine if the growth rate of the identified wear would exceed the repair limit. If the repair limit is predicted to be exceeded, the tubes would be removed from service or the operating interval would be decreased.

   The inspection interval may continue to be extended to a maximum of once per 40 months thereafter if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred. Based on today's stringent chemistry controls, which are based on the latest version of the EPRI Guidelines, minimal damage would be expected due to corrosion-induced degradation for Alloy 690 material.

4. The fourth area of change is to revise the preservice inspection requirements to accommodate the difference between the original steam generators and the replacement steam generators regarding when the ASME Code Section III hydrostatic test and the preservice eddy current inspection of the tubes are performed. For the original steam generators, the hydrostatic test was done as part of the reactor coolant system hydrostatic

test. The preservice eddy current inspection for the original steam generators was performed after the field hydrostatic test to find indications of adverse effects due to the hydrostatic test. There was no separate hydrostatic test for the original steam generators. For the replacement steam generators, the hydrostatic test will be performed at the manufacturing facility. The preservice eddy current inspection for the replacement steam generators will be performed at the plant site following the hydrostatic test, but prior to installation, in order to shorten the duration of the replacement outage. It will confirm that there are no adverse effects due to the hydrostatic test and will establish a baseline condition of the tubes for future inspections. The preservice eddy current inspection will be performed under the direction of Entergy to assure that the equipment and techniques used for the preservice inspection are the same as those expected to be used during subsequent inservice inspections, consistent with the requirements of Specification 4.4.5.4.a.10. Following installation of the replacement steam generators, an ASME Code Section XI post-repair leakage test will be performed. Since the post-repair leakage test for an operating plant will be at a much lower pressure than the hydrostatic test, it will not change the baseline condition of the tubes prior to operation.

5. The fifth area of change is to revise section 4.4.5.5.b to be consistent with the guidance for reporting frequency contained in NEI 97-06, Revision 0, "Steam Generator Program Guidance." Rather than continue reporting the complete results of the inservice inspections on an annual basis, the wording would be revised to report the results within 12 months following the completion of the inservice inspection. Currently, reports are submitted each year prior to March 1st. For inservice inspections completed late in the calendar year, little time is available to complete the analysis work, prepare and process the report for submittal to the NRC. The change to the NEI guidance would ease the administrative burden for preparation and submittal of these reports. The change in this reporting frequency would not affect the requirements for submittal of the C-3 Report which is required if the inservice inspection should reveal more than 10% of the total tubes inspected to be degraded or if more than 1% of the inspected tubes are determined to be defective.

   Additionally, the wording in section 4.4.5.5.c. is revised to clarify what is meant by "prior to resumption of plant operation." In the past this phrase has resulted in confusion. The intent is to make notification to the NRC prior to entering Mode 4 when the steam generators are required to be operable.

6. The sixth area of change is to revise the bases to delete reference to tube repair and "wastage-type" defects. Wastage is an historical term that relates to tube degradation associated with phosphate chemistry control which is no longer utilized at ANO-2. Additionally, the bases refer to reliably detecting 20% tube degradation. ANO uses the qualified or equivalent techniques for flaw detection as outlined in EPRI TR-107569, Vol. 1, Rev. 5, "PWR Steam Generator Examination Guidelines." This ensures that the minimum detectability of the technique used is capable of finding those flaws (with high probability) that would challenge structural integrity. The 20% value in the bases was a conservative, arbitrary percentage that was based on the ability to detect wastage-type

(volumetric) flaws. The detection of linear-type flaws is more difficult and more challenging. While 20% tube degradation of both types can be detected, the probability of detection is lower for linear-type flaws. The probability for flaw detection is described in the EPRI document.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

Entergy Operations, Inc. is proposing that the Arkansas Nuclear One Unit 2 (ANO-2) Operating License be amended to modify the surveillance requirements for the steam generators. An evaluation of the proposed change has been performed in accordance with 10CFR50.91(a)(1) regarding no significant hazards considerations using the standards in 10CFR50.92(c). A discussion of these standards as they relate to this amendment request follows:

Criterion 1 - Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The accidents of interest are a tube rupture, loss of coolant accident (LOCA) in combination with a safe shutdown earthquake and a steam line break in combination with a safe shutdown earthquake. A reduction in tube integrity could increase the possibility of a tube rupture accident and increase the consequences of a steam line break or LOCA. The tubing in the replacement steam generators is designed and evaluated consistent with the margins of safety specified in the ASME Code, Section III. The program for periodic inservice inspection provides sufficient time to take proper and timely corrective action if tube degradation is present. The ASME Section XI basis for the 40% through wall plugging limit is applicable to the replacement steam generators just as it was to the original steam generators. As a result there is no reduction in tube integrity for the replacement steam generators.

Addition of a "Note" to clarify that inservice inspection is not required during the steam generator replacement outage is an administrative change that provides clarification regarding inservice inspection requirements. The change in reporting requirements is also an administrative change. The requirements for inservice inspection or the plugging limit for the tubes are not altered by these administrative changes. Additionally, changes were made to the bases to remove potentially misleading information. Bases changes are considered to be administrative in nature.

Elimination of the repair option and the associated references to repair of the original steam generator tubes is an administrative adjustment since the sleeve design is not applicable to the replacement steam generators. The elimination of the repair option does not alter the requirements for inservice inspection or reduce the plugging limit for the tubes.

The proposed change to extend the inspection interval to a maximum of once per 40 months is acceptable based on the use of the superior Alloy 690 tubing material. Significant industry knowledge has been gained from monitoring the performance of steam generators that have

been replaced. Alloy 690 tubing material has proven to be superior to Alloy 600 in regard to corrosion resistance. Plants that have utilized Alloy 690 tubing in their replacement steam generators have not experienced corrosion-induced degradation.

A preservice eddy current inspection will be performed onsite prior to installation of the replacement steam generators. The orientation of the replacement steam generators during the eddy current exam will not impact the results. The hydrostatic test required by the ASME Code Section III for the replacement steam generators is to be performed in the manufacturing facility and not as part of a reactor coolant system hydrostatic test. The post-repair leakage test required by the ASME Code, Section XI for an operating plant is performed at a much lower pressure. No evolutions subsequent to the replacement steam generator hydrostatic test are expected to occur that will change the condition of the tubes prior to operation. This change does not alter the requirement to perform a preservice inspection. As a result, an inservice inspection is not required during the steam generator replacement outage.

The requested ANO-2 Technical Specification changes do not alter the requirements for tube integrity, tube inspection, or tube plugging limit. Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

Criterion 2 - Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The proposed changes do not affect the design or function of any other safety-related component. There is no mechanism to create a new or different kind of accident for the replacement steam generators by eliminating repair criteria or by clarifying the applicable preservice and inservice inspection requirements because a baseline of tube conditions is established and plugging limits are maintained to ensure that defective tubes are removed from service. A change in inspection frequency has a negligible impact on the pre-accident state of the reactor core or post accident confinement of radionuclides within the containment building. Changing the inspection frequency creates no new failure modes or accident initiators/precursors.

The requested ANO-2 Technical Specification changes do not alter the requirements for tube integrity, tube inspection or tube plugging limit. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3 - Does Not Involve a Significant Reduction in the Margin of Safety.

The tubing in the replacement steam generators is designed and evaluated consistent with the margins of safety specified in the ASME Code, Section III. The program for periodic inservice inspection provides sufficient time to take proper and timely corrective action to preserve the design margin if tube degradation is present.

Due to the superior Alloy 690 tubing material and the significant amount of industry knowledge and operating history with this improved tubing material, extending the inspection interval to a maximum of once per 40 months will still allow the integrity of the steam generator tubing to be ensured. The steam generator inspection program is not intended to provide an accident mitigation or assessment function; therefore, this change results in a neutral impact to the margin of safety.

Based upon the reasoning presented above and the previous discussion of the amendment request, Entergy Operations has determined that the requested change does not involve a significant hazards consideration.

ENVIRONMENTAL IMPACT EVALUATION

10 CFR 51.22(c) provides criteria for and identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration, (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released off-site, or (3) result in a significant increase in individual or cumulative occupational radiation exposure. Entergy Operations, Inc. has reviewed this license amendment and has determined that it 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 proposed license amendment. The basis for this determination is as follows:

1. The proposed license amendment does not involve a significant hazards consideration as described previously in the evaluation.

2. As discussed in the significant hazards evaluation, this change does not result in a significant change or significant increase in the radiological doses for any Design Basis Accident. The proposed license amendment does not result in a significant change in the types or a significant increase in the amounts of any effluents that may be released off-site.

3. The proposed license amendment does not result in a significant increase to the individual or cumulative occupational radiation exposure. This change simply revises the surveillance requirements of the steam generator tube inspection program and results in no physical changes to plant equipment or methodologies that could result in an increase in occupational radiation exposure.