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

October 10, 2017

ANO Site Vice President
Arkansas Nuclear One
Entergy Operations, Inc.
1448 S.R. 333
Russellville, AR 72802

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT 1 - ISSUANCE OF AMENDMENT RE: REVISION OF STEAM GENERATOR TECHNICAL SPECIFICATIONS TO REFLECT ADOPTION OF TECHNICAL SPECIFICATIONS TASK FORCE TRAVELER TSTF-510, REVISION 2 (CAC MF9653; EPID L-2017-LLA-0221)

Dear Sir or Madam:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 258 to Renewed Facility Operating License No. DPR-51 for Arkansas Nuclear One, Unit 1 (ANO-1). The amendment consists of changes to the technical specifications (TSs) in response to your application dated April 24, 2017.

The amendment incorporates the guidance of Technical Specifications Task Force (TSTF) Change Traveler TSTF-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection." The guidance of TSTF-510 revises TS 3.4.17, "Steam Generator (SG) Tube Integrity," TS 5.5.9, "Steam Generator (SG) Program," and TS 5.6.7, "Steam Generator Tube Inspection Report," of the Improved Standard Technical Specifications that are applicable to ANO-1.

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

Sincerely,

Thomas J. Wengert

Thomas J. Wengert, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-313

Enclosures:
1. Amendment No. 258 to DPR-51
2. Safety Evaluation

cc w/encls: Distribution via Listserv

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

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 258
Renewed License No. DPR-51

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

   A. The application for amendment by Entergy Operations, Inc. (the licensee), dated April 24, 2017, 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 Renewed Facility Operating License No. DPR-51 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 258, are hereby incorporated in the renewed license. EOI shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance and shall be implemented within 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert J. Pascarelli, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License No. DPR-51 and Technical Specifications

Date of Issuance: October 10, 2017

ATTACHMENT TO LICENSE AMENDMENT NO. 258

RENEWED FACILITY OPERATING LICENSE NO. DPR-51

ARKANSAS NUCLEAR ONE, UNIT 1

DOCKET NO. 50-313

Replace the following pages of the Renewed Facility Operating License No. DPR-51 and Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Operating License

REMOVE	INSERT
3	3

Technical Specifications

REMOVE	INSERT
3.4.16-1	3.4.16-1
3.4.16-2	3.4.16-2
5.0-12	5.0-12
5.0-13	5.0-13
5.0-14	5.0-14
5.0-15	5.0-15
5.0-23	5.0-23

(5) EOI, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

(6) EOI, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

c. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

EOI is authorized to operate the facility at steady state reactor core power levels not in excess of 2568 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 258, are hereby incorporated in the renewed license. EOI shall operate the facility in accordance with the Technical Specifications.

(3) Safety Analysis Report

The licensee's SAR supplement submitted pursuant to 10 CFR 54.21(d), as revised on March 14, 2001, describes certain future inspection activities to be completed before the period of extended operation. The licensee shall complete these activities no later than May 20, 2014.

(4) Physical Protection

EOI shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans, including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Arkansas Nuclear One Physical Security Plan, Training and Qualifications Plan, and Safeguards Contingency Plan," as submitted on May 4, 2006.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 Steam Generator (SG) Tube Integrity

LCO 3.4.16 SG tube integrity shall be maintained.

<u>AND</u>

All SG tubes satisfying the tube plugging criteria shall be plugged in accordance with the Steam Generator Program.

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

--NOTE--
Separate Condition entry is allowed for each SG tube.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube plugging criteria and not plugged in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.	7 days
	<u>AND</u>	
	A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> SG tube integrity not maintained.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.16.2	Verify that each inspected SG tube that satisfies the tube plugging criteria is plugged in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

   1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary to secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary to secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

   2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm.

   3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

c. Provisions for SG tube plugging criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.

2. After the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, c and d below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 144 effective full power months. This constitutes the first inspection period;

b) During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period;

c) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the third inspection period; and

d) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the fourth and subsequent inspection periods.

3. If crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5.10 Secondary Water Chemistry

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events required to initiate corrective action.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safeguards (ES) ventilation systems filters at the frequencies specified in Regulatory Guide 1.52, Revision 2. The VFTP is applicable to the Penetration Room Ventilation System (PRVS) and the Control Room Emergency Ventilation System (CREVS).

a. Demonstrate that an inplace cold DOP test of the high efficiency particulate (HEPA) filters shows:

1. ≥ 99% DOP removal for the PRVS when tested at the system design flowrate of 1800 scfm ± 10%; and

2. ≥ 99.95% DOP removal for the CREVS when tested in accordance with Regulatory Guide 1.52, Revision 2, at the system design flowrate of 2000 cfm ± 10%.

b. Demonstrate that an inplace halogenated hydrocarbon test of the charcoal adsorbers shows:

1. ≥ 99% halogenated hydrocarbon removal for the PRVS when tested at the system design flowrate of 1800 cfm ± 10%; and

2. ≥ 99.95% halogenated hydrocarbon removal for the CREVS when tested in accordance with Regulatory Guide 1.52, Revision 2, at the system design flowrate of 2000 cfm ± 10%.

c. Demonstrate that a laboratory test of a sample of the charcoal adsorber meets the laboratory testing criteria of ASTM D3803-1989 when tested at 30°C and 95% relative humidity for a methyl iodide penetration of:

1. < 5% for the PRVS;

2. when obtained as described in Regulatory Guide 1.52, Revision 2, for CREVS

i. ≤ 2.5% for 2 inch charcoal adsorber beds; and

ii. ≤ 0.5% for 4 inch charcoal adsorber beds.

d. Demonstrate for the PRVS and CREVS, that the pressure drop across the combined HEPA filters, other filters in the system, and the charcoal adsorbers is < 6 inches of water when tested at the following system design flowrates ± 10%:

PRVS	1800 cfm
CREVS	2000 cfm

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

5.6.6 Reactor Building Inspection Report

Any degradation exceeding the acceptance criteria of the containment structure detected during the tests required by the Containment Inspection Program shall undergo an engineering evaluation within 60 days of the completion of the inspection surveillance. The results of the engineering evaluation shall be reported to the NRC within an additional 30 days of the time the evaluation is completed. The report shall include the cause of the condition that does not meet the acceptance criteria, the applicability of the conditions to the other unit, the acceptability of the concrete containment without repair of the item, whether or not repair or replacement is required and, if required, the extent, method, and completion date of necessary repairs, and the extent, nature, and frequency of additional examinations.

5.6.7 Steam Generator Tube Inspection Reports

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.9, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each degradation mechanism,

f. The number and percentage of tubes plugged to date, and the effective plugging percentage in each SG,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 258 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-51

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT 1

DOCKET NO. 50-313

1.0 INTRODUCTION

By application dated April 24, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17114A398), Entergy Operations, Inc. (the licensee), requested changes to the technical specifications (TSs) for Arkansas Nuclear One, Unit 1 (ANO-1).

The proposed changes would revise the TSs for ANO-1, and would adopt U.S. Nuclear Regulatory Commission (NRC)-approved Technical Specification Task Force (TSTF) Standard Technical Specifications (STSs) Change Traveler TSTF-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection" (ADAMS Accession No. ML110610350). The guidance of TSTF-510 revises TS 3.4.17, "Steam Generator (SG) Tube Integrity"; TS 5.5.9, "Steam Generator (SG) Program"; and TS 5.6.7, "Steam Generator Tube Inspection Report," of NUREG-1430, Revision 4, "Standard Technical Specifications – Babcock and Wilcox Plants" (ADAMS Accession No. ML12100A177), applicable to ANO-1. The specific changes concern SG inspection periods, and address applicable administrative changes and clarifications.

The licensee stated that the license amendment request (LAR) is consistent with the Notice of Availability of TSTF-510, Revision 2, announced in the *Federal Register* on October 27, 2011 (76 FR 66763), as part of the consolidated line item improvement process.

The current STS requirements in the above specifications were established in May 2005 with the NRC staff's approval of TSTF-449, Revision 4, "Steam Generator Tube Integrity" (NRC *Federal Register* Notice of Availability (70 FR 24126)). The TSTF-449 changes to the STS incorporated a new, largely performance-based approach for ensuring the integrity of the SG tubes is maintained. The performance-based requirements were supplemented by prescriptive requirements relating to tube inspections and tube repair limits to ensure that conditions adverse to quality are detected and corrected on a timely basis. As of September 2007, the TSTF-449, Revision 4, changes were adopted in the plant TSs for all pressurized water reactors (PWRs).

The proposed changes in TSTF-510, Revision 2, reflect licensees' early implementation experience with respect to TSTF-449, Revision 4. TSTF-510 characterizes the changes as editorial corrections, changes, and clarifications intended to improve internal consistency,

consistency with implementing industry documents, and usability without changing the intent of the requirements. The proposed changes are an improvement to the existing SG inspection requirements and continue to provide assurance that the licensing basis will be maintained between SG inspections.

2.0 REGULATORY EVALUATION

The SG tubes in PWRs have a number of important safety functions. These tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied upon to maintain primary system pressure and inventory. As part of the RCPB, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system and are relied upon to isolate the radioactive fission products in the primary coolant from the secondary system. In addition, the SG tubes are relied upon to maintain their integrity to be consistent with the containment objectives of preventing uncontrolled fission product release under conditions resulting from core damage during severe accidents.

The regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) establish the requirements with respect to the integrity of SG tubing. Specifically, the General Design Criteria (GDC) in Appendix A to 10 CFR Part 50 state that the RCPB:

- shall have "an extremely low probability of abnormal leakage. . .and gross rupture" (GDC 14, "Reactor pressure coolant boundary"),
- "shall be designed with sufficient margin" (GDC 15, "Reactor coolant system design," and GDC 31, "Fracture prevention of reactor coolant pressure boundary"),
- shall be of "the highest quality standards possible" (GDC 30, "Quality of reactor coolant pressure boundary"), and
- shall be designed to permit "periodic inspection and testing. . .to assess. . .structural and leaktight integrity" (GDC 32, "Inspection of reactor coolant pressure boundary").

The ANO-1 plant was designed and constructed to meet the intent of the Atomic Energy Commission's GDC, as originally proposed in July 1967, and the construction permit was issued prior to the 1971 publication of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50. As such, compliance with the explicit requirements of the GDC is not required, as long as the intent of the GDC are met. Section 1.4 of the ANO-1 Safety Analysis Report lists the manner in which the ANO-1 GDC meet the intent of the GDC in Appendix A of 10 CFR Part 50. The ANO-1 GDC addressing the RCPB are Criterion 14, "Reactor Coolant Pressure Boundary"; Criterion 15, "RCS [Reactor Coolant System] Design"; Criterion 30, "Quality of Reactor Coolant Pressure Boundary"; Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary"; and Criterion 32, "Inspection of Reactor Coolant Pressure Boundary." These ANO-1 GDC are similar to GDC 14, 15, 30, 31, and 32 in Appendix A of 10 CFR Part 50.

The regulations in 10 CFR 50.55a specify that RCPB components must meet the requirements for Class 1 components in Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). Section 50.55a further requires, in part, that throughout the service life of a PWR, ASME Code Class 1 components meet the requirements, except design and access provisions and preservice examination requirements, in Section XI, "Rules for Inservice Inspection (ISI) of Nuclear Power Plant Components," of the ASME Code, to the extent practical. This requirement includes the inspection and repair criteria of Section XI of the ASME Code.

As part of the plant's licensing basis, applicants for PWR licenses are required to analyze the consequences of postulated design-basis accidents such as an SG tube rupture or main steamline break. These analyses consider the primary-to-secondary leakage that may occur during these events and must show that the radiological consequences do not exceed the applicable limits of the 10 CFR Part 100.11, "Determination of exclusion area, low population zone, and population center distance," guidelines for offsite doses (or 10 CFR 50.67, "Accident source term," as appropriate), GDC-19 of Appendix A to 10 CFR Part 50 for control room operator doses (or some fraction thereof as appropriate to the accident), or the NRC-approved licensing basis.

The regulation at 10 CFR 50.36, "Technical specifications," establishes the requirements related to the content of the TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five categories related to station operation:

(1) safety limits, limiting safety system settings, and limiting control settings;
(2) limiting conditions for operation (LCOs);
(3) surveillance requirements (SRs);
(4) design features; and
(5) administrative controls.

For ANO-1, the LCOs (and accompanying Action statements) and SRs in the TSs, relevant to SG tube integrity, are in TS 3.4.13, "RCS Operational Leakage," and TS 3.4.16, "Steam Generator (SG) Tube Integrity." The SRs in the "Steam Generator (SG) Tube Integrity" specification reference the SG Program, which is defined in the Administrative Controls section of the TSs.

The regulation at 10 CFR 50.36(c)(5) defines administrative controls as "the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure the operation of the facility in a safe manner." Programs established by the licensee to operate the facility in a safe manner, including the SG Program, are listed in the Administrative Controls section of the TSs. In the STSs, the SG Program is defined in TS 5.5.9 and the reporting requirements related to implementation of the SG Program are in TS 5.6.7.

TS 5.5.9, for ANO-1, requires that an SG Program be established and implemented to ensure that SG tube integrity is maintained. Tube integrity is maintained by meeting the performance criteria specified in TS 5.5.9.b for structural and leakage integrity, consistent with the plant design and licensing bases. TS 5.5.9.a requires that a condition monitoring assessment be performed during each outage, during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met. TS 5.5.9.d includes provisions regarding the scope, frequency, and methods of SG tube inspections. These provisions require that the inspections be performed with the objective of detecting flaws of any type that (1) may be present along the length of a tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and (2) may satisfy the applicable tube plugging criteria.

The applicable tube repair criteria, specified in TS 5.5.9.c, are that tubes found during ISI to contain flaws with a depth equal to or exceeding 40 percent of the nominal wall thickness shall be plugged.

3.0 TECHNICAL EVALUATION

The changes proposed by the licensee for ANO-1 LCO 3.4.16, TS 5.5.9, and TS 5.6.7 are consistent with the corresponding TS changes described in TSTF-510, Revision 2, including the proposed revised inspection intervals, which are appropriate since ANO-1 has SGs with thermally treated Alloy 690 tubing. Except for the administrative changes and variations discussed below, the proposed changes are consistent with TSTF-510, Revision 2. As a result, the NRC staff's evaluation is focused on these administrative changes and variations, since the other changes were previously evaluated in the model safety evaluation (ADAMS Accession No. ML112101513), which is applicable to ANO-1.

3.1 Administrative Changes and Variations

- The ANO-1 TSs utilize different numbering than the STSs on which TSTF-510, Revision 2, was based. For ANO-1, the "Steam Generator (SG) Tube Integrity" TS is numbered 3.4.16 rather than 3.4.17.

- An NRC letter dated June 17, 2013 (ADAMS Accession No. ML13120A541), clarified that if LARs proposing to implement TSTF-510, Revision 2, corrected an administrative inconsistency in paragraph 5.5.9.d.2 of the Steam Generator (SG) Program, it would not result in removal of submitted LARs from the consolidated line item improvement process. Accordingly, since ANO-1 does not have any approved tube repair methods, this LAR fixes the administrative inconsistency in paragraph 5.5.9.d.2, by replacing "tube repair criteria" with "tube plugging criteria."

- The TSTF-510, Revision 2, changes, require some TS information to be moved from one page to another. Specifically, TS 5.11, "Ventilation Filter Testing Program" is relocated in its entirety to TS Page 5.0-15, and therefore, the change to this page is a result of pagination only.

The differences noted above are administrative and do not affect the applicability of TSTF-510, Revision 2, to the ANO-1 TSs. As a result, the NRC staff finds that the differences between what was previously approved for TSTF-510, Revision 2, and the licensee's proposed TS changes, are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Arkansas State official was notified of the proposed issuance of the amendment on September 11, 2017. The State official had no comments.

5.0 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 published in the *Federal Register* on July 5, 2017 (82 FR 31092). 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.

6.0 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) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: A. Johnson

Date: October 10, 2017

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT 1 - ISSUANCE OF AMENDMENT RE: REVISION OF STEAM GENERATOR TECHNICAL SPECIFICATIONS TO REFLECT ADOPTION OF TECHNICAL SPECIFICATIONS TASK FORCE TRAVELER TSTF-510, REVISION 2 (CAC MF9653; EPID L-2017-LLA-0221) DATED: OCTOBER 10, 2017

DISTRIBUTION:
PUBLIC
LPL4 R/F
RidsACRS_MailCTR Resource
RidsNrrDorlLpl4 Resource
RidsNrrDssStsb Resource
RidsNrrLAPBlechman Resource
RidsNrrDlrRasb Resource
RidsNrrPMANO Resource
RidsRgn4MailCenter Resource
AJohnson, NRR

ADAMS Accession No. ML17235A519; BWI: ML17235A520 *via memorandum

OFFICE	NRR/DORL/LPL4/PM	NRR/DORL/LPL4/LA	NRR/DSS/STSB/BC (A)	NRR/DLR/RCCB/BC*
NAME	TWengert	PBlechman	JWhitman	SBloom
DATE	9/15/17	9/6/17	9/19/17	7/17/2017
OFFICE	OGC / NLO	NRR/DORL/LPL4/BC	NRR/DORL/LPL4/PM	
NAME	JGillespie	RPascarelli	TWengert	
DATE	10/3/17	10/6/17	10/10/17	

OFFICIAL RECORD COPY