October 4, 2000

Mr. Craig G. Anderson
Vice President, Operations ANO
Entergy Operations, Inc.
1448 S. R. 333
Russellville, AR 72801

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: REPLACEMENT STEAM GENERATOR INSPECTION REQUIREMENTS (TAC NO. MA6341)

Dear Mr. Anderson:

The Commission has issued the enclosed Amendment No. 223 to Facility Operating License No. NPF-6 for Arkansas Nuclear One, Unit No. 2. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated August 18, 1999, as supplemented by letters dated June 29, July 19, and August 9, 2000.

The amendment revises TS 4.4.5, "Steam Generators," to note that the requirements for inservice inspection do not apply during the steam generator replacement outage (2R14), to revise the requirement for tube inspection to mean an inspection from tube end (cold leg side) to tube end (hot leg side), to delete inspection requirements associated with steam generator tube sleeving and repair limits, to revise the preservice inspection requirements on when the hydrostatic test and the eddy current inspection of the tubes would be performed, and to revise the reporting frequency of the results of steam generator tube inspections to within 12 months following completion of the inservice inspection. Related changes to the Bases are also made.

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

Sincerely, /RA/
Thomas W. Alexion, Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-368

Enclosures:
1. Amendment No. 223 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

October 4, 2000

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Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

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cc w/encls: See next page

cc:

Executive Vice President
& Chief Operating Officer
Entergy Operations, Inc.
P. O. Box 31995
Jackson, MS 39286-1995

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. 223
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 August 18, 1999, as supplemented by letters dated June 29, July 19, and August 9, 2000, 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. 223, are hereby incorporated in the license. The licensee 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 prior to startup from the 2R14 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

for

Robert A. Gramm, Chief, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: October 4, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 223

FACILITY OPERATING LICENSE NO. NPF-6

DOCKET NO. 50-368

Replace the following pages of the 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.

Remove	Insert
3/4 4-6	3/4 4-6
3/4 4-7	3/4 4-7
3/4 4-8	3/4 4-8
3/4 4-9	3/4 4-9
3/4 4-10	3/4 4-10
3/4 4-12	3/4 4-12
3/4 4-12a	---
B 3/4 4-2	B 3/4 4-2
B 3/4 4-3	B 3/4 4-3

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.

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. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40 month intervals fall into Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3.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

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in the other S.G. Special Report to NRC per Specification 6.9.2	Other S.G. is C-1	None	N/A	N/A
			Other S.G. is C-2	Perform action for C-2 result of second sample	N/A	N/A
			Other S.G. is C-3	Inspect all tubes in the other S.G. and plug defective tubes. Special Report to NRC per Spec. 6.9.2	N/A	N/A

$S = 3 \frac{2}{n}$ % 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 223 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 August 18, 1999, as supplemented by letters dated June 29, July 19, and August 9, 2000, Entergy Operations, Inc. (the licensee), submitted a request for changes to the Arkansas Nuclear One, Unit No. 2 (ANO-2), Technical Specifications (TSs). The requested changes would revise Technical Specification (TS) 4.4.5, "Steam Generators," to account for changes associated with replacement of the original steam generators (SGs). Specifically, the proposed changes would note that the requirements for inservice inspection do not apply during the SG replacement outage (2R14), revise the requirement for tube inspection to mean an inspection from tube end (cold leg side) to tube end (hot leg side), delete inspection requirements associated with SG tube sleeving and repair limits, revise the preservice inspection requirements on when the hydrostatic test and the eddy current inspection of the tubes would be performed, and revise the reporting frequency of the results of SG tube inspections to within 12 months following completion of the inservice inspection (ISI). Related changes to the Bases would also be made.

The application was renoticed to include the June 29, July 19, and August 9, 2000, supplements as indicated in Section 6.0 of this safety evaluation.

2.0 BACKGROUND

The ANO-2 plant design currently uses two Model 2815 SGs manufactured by Combustion Engineering with mill annealed Alloy 600 tubing that is explosively expanded along the full depth of the tube sheet. The licensee is scheduled to replace the ANO-2 SGs during refueling outage 2R14 which is scheduled to begin about September 15, 2000. The replacement SGs are designed and fabricated by Westinghouse. The replacement SGs incorporate a number of design and material changes when compared to the original SGs. With regard to the heat transfer surfaces, there are changes in (1) the number, size (outside diameter and wall thickness), and material selection of the tubing, and (2) tube support design and material selection for the straight leg and U-bend sections.

The current SG tubes are made of Alloy 600 and the tube support plates are made of carbon steel. Use of these materials contributed, in part, to the existing SG tube degradation. As degradation occurred, the licensee requested amendments to the TSs for tube sleeving repair criteria which the Nuclear Regulatory Commission (NRC) subsequently reviewed and approved.

The replacement SGs will use thermally-treated Alloy 690 tube material and stainless steel tube support plates (flat-contact broached trifoil tube hole plates) and anti-vibration bars. The Alloy 690 tubing material is more resistant to stress corrosion cracking than Alloy 600 tubing material. The Alloy 405 stainless steel tube support plates and anti-vibration bars will be more resistant to magnetite formation than carbon steel support plates and minimize tube denting. Licensees that have used these materials in their replacement SGs have reported minimal tube degradation. Thus, the licensee is not requesting approval of tube sleeving repair criteria for the replacement SGs. Additionally, the licensee has requested various modifications to the surveillance requirements (SRs) for the replacement SGs.

3.0 EVALUATION

3.1 ISI During the SG Replacement Outage (2R14) and Definition of Tube Inspection

SR 4.4.5.0 requires that each SG be demonstrated operable, in part, through the performance of the tube ISI program. The licensee proposed the revision of a note under SR 4.4.5.0 that states that the requirements for ISI do not apply during the SG replacement outage (2R14) since the replacement SGs will be subjected to a preservice inspection as required by the TSs.

In a letter, dated June 29, 2000, in response to an NRC request for additional information dated May 9, 2000, the licensee stated that 100% preservice examination will be performed onsite prior to installation of the SGs. It also stated that the first ISI of the SGs will be performed after six effective full power months but within 24 calendar months of initial criticality. Consistent with TS requirements in TS Table 4.4-1, which states that the first ISI may be limited to one SG, the licensee has proposed that the first ISI is a 100 percent inspection of the tubes in both SGs after the initial cycle of SG operation to determine whether any unexpected degradation occurs. Along with the preservice inspection results, the first ISI provides baseline information for comparison with future examination results. Since the TS requirements for a preservice inspection of the replacement SGs will be satisfied, the staff finds the proposed change acceptable.

In the same letter dated June 29, 2000, the licensee responded to the staff regarding, in part, the extent of preservice inspection and the first ISI as being from tube end to tube end. The definition in TS 4.4.5.4.a.9 has been revised according to the Electric Power Research Institute Guidelines as from the hot leg tube end to the cold leg tube end. The staff finds that this change is conservative, and therefore, acceptable.

3.2 SG Tube Sleeving and Repair Limits

The licensee has proposed changing SR 4.4.5.4.a.7, "Plugging or Repair Limit," SR 4.4.5.4.b, and Table 4.4-2 to delete requirements and associated references to (1) repair of the SG tubes by sleeving, and (2) plugging limit of tubes previously repaired by sleeving. Further, the licensee proposed the deletion of Table 4.4-3, "Steam Generator Tube Sleeve Inspection," in its entirety and the deletion of the reference to this table in SRs 4.4.5.2 and 4.4.5.3. Also, SR 4.4.5.2 will be modified to delete the requirement to select a sample of installed sleeves for inspection.

In addition, the licensee has proposed to change the definitions in (1) SR 4.4.5.4.a.1 for "Tubing or Tube" to delete "sleeve" as forming part of the primary to secondary system pressure boundary, and (2) SR 4.4.5.4.a.6 for "Defect" to delete reference to a sleeve repair limit. The reporting of the identification and number of tubes repaired by sleeving in SR 4.4.5.5 would also be deleted.

The current surveillance requirements detail the approved sleeve designs and installation requirements described in the specified Babcock & Wilcox and Combustion Engineering Nuclear Operations reports. The licensee states that the referenced reports are no longer appropriate since the replacement SGs use tubing with a diameter and wall thickness that is different from the original SG tubing. Further, the licensee stated that the replacement SGs use metallurgy and fabrication technology that has proven to be very resistant to corrosion related degradation and a sleeving design is not expected to be required. Thus, the current sleeving requirements are not valid for the replacement SGs. The staff finds deletion of these requirements acceptable.

In a letter dated June 29, 2000, regarding an analysis per Regulatory Guide 1.121, the licensee provided a report entitled "Analysis for Arkansas Nuclear One Unit 2 Replacement Steam Generators." The analysis was performed using the standard Westinghouse methodology. The analysis confirms a defined structural limit of 57.5% allowable defect depth and confirms the 40% plugging limit which will remain in the TSs for the tubing.

3.3 Preservice Inspection Requirements

The licensee proposed to revise the preservice inspection requirements in TS 4.4.5.4.a.10 to accommodate the difference between the original SGs and the replacement SGs regarding when the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code) Section III hydrostatic test and the preservice eddy current inspection of the tubes are performed. The licensee stated that for the original SGs, the hydrostatic test was done in the plant as part of the reactor coolant system hydrostatic test, which was followed by the preservice eddy current inspection. For the replacement SGs, the licensee stated that the hydrostatic test will be performed at the manufacturing facility instead of at Arkansas Nuclear One (ANO). The preservice inspection will be performed at the ANO site prior to the actual installation of the SGs. After the installation of the replacement SGs, the licensee will conduct a field post-repair system leakage test in accordance with ASME Code Section XI.

The staff approved a similar TS change regarding preservice inspection for the North Anna Power Station, Units 1 and 2 replacement SGs in a letter to Virginia Power Corporation on December 4, 1991, and for the Joseph M. Farley Nuclear Plant, Units 1 and 2 in a letter to Southern Nuclear Operating Company on December 29, 1999. Since the only change from the original requirements relates to the performance of the hydrostatic test at the manufacturer's facility instead of the plant, the staff finds the proposed changes to the preservice inspection requirements provide reasonable assurance of component integrity and are acceptable.

3.4 Reporting Frequency of SG Tube Inspection Results

The licensee proposed to change the requirement for submission of the complete results of the SG inspection in TS 4.4.5.5.b from "an annual basis for the period in which the inspection was completed" to "within 12 months following the completion of the inservice inspection." The licensee stated that since the reports are currently submitted each year by March 1, it has little time to complete the analysis and prepare the report for inspections conducted late in the prior year. For this situation, the current requirement creates an administrative burden.

The NRC staff finds that the proposed change may delay the submission of the complete ISI results but does not modify the requirement to submit the report or the content of the report. The licensee is also not modifying the reporting requirement in TS 4.4.5.5.c when the SG inspection results fall into Category C-3. Because the proposed change provides for submission of the report in a reasonable period of time, the staff finds that the proposed change is acceptable.

Additionally, the licensee proposed adding a clarification to TS 4.4.5.5.c that would define "prior to resumption of plant operation" as "prior to entering Mode 4" (Cold Shutdown) for making notifications to the NRC of Category C-3 inspection results. Since the proposed change provides a clearer definition of the requirement and that entering Mode 4 occurs prior to significant plant heatup, the staff finds the proposed change to be administrative in nature and is acceptable.

3.5 Associated Bases Changes

The NRC staff has reviewed the proposed changes to TS Bases Section 3/4.4.5 and finds them consistent with the other changes discussed above. Therefore, the staff finds these changes acceptable.

4.0 EVALUATION SUMMARY

Based on the staff's review and evaluation of ANO Unit 2's proposed TS changes, the staff has determined that the proposed changes are acceptable. Although not discussed in the preceding, each of the proposed changes have been compared with the Improved Standard TSs (ISTS) and have been determined consistent with the ISTSs.

5.0 STATE CONSULTATION

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 SRs. 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 (65 FR 9005, dated February 23, 2000, and 65 FR 51353, dated August 23, 2000). The amendment also modifies recordkeeping or reporting requirements. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (c)(10). 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 Contributor: Z. Bart Fu

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