Mr. Jerry W. Yelverton
Vice President, Operations ANO
Entergy Operations, Inc.
Route 3 Box 137G
Russellville, Arkansas 72801

January , 1995

SUBJECT: ISSUANCE OF AMENDMENT NO. 158 TO FACILITY OPERATING LICENSE NO. NPF-6 - ARKANSAS NUCLEAR ONE, UNIT NO. 2 (TAC NO. M90989)

Dear Mr. Yelverton:

The Commission has issued the enclosed Amendment No.158 to Facility Operating License No. NPF-6 for the Arkansas Nuclear One, Unit No. 2 (ANO-2). This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated November 29, 1994 as supplemented by letters dated December 20 and 21, 1994.

The amendment deletes the requirement to perform the full complement of steam generator surveillances as outlined in the TSs when the steam generators are subjected to special inspections that are in addition to the inspections required by the TSs. This amendment is applicable only to the special steam generator inspection scheduled for January 1995.

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

Sincerely,
ORIGINAL SIGNED BY:
George Kalman, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-368

Enclosures:
1. Amendment No. 158 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

January 5, 1995

Mr. Jerry W. Yelverton
Vice President, Operations ANO
Entergy Operations, Inc.
Route 3 Box 137G
Russellville, Arkansas 72801

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Sincerely,

George Kalman

George Kalman, Senior Project Manager
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Docket No. 50-368

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1. Amendment No. 158 to NPF-6
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cc w/encls: See next page

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

ENTERGY OPERATIONS INC.

DOCKET NO. 50-368

ARKANSAS NUCLEAR ONE, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 158
License No. NPF-6

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

   A. The application for amendment by Entergy Operations, Inc. (the licensee) dated November 29, 1994, as supplemented by letters dated December 20 and 21, 1994, 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.

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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. 158, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

George Kalman

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

Attachment: Changes to the
Technical Specifications

Date of Issuance: January 5, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 158

FACILITY OPERATING LICENSE NO. NPF-6

DOCKET NO. 50-368

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE PAGES	INSERT PAGES
3/4 4-6	3/4 4-6
3/4 4-7	3/4 4-7
3/4 4-12	3/4 4-12
B3/4 4-2	B3/4 4-2

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with a water volume of $\leq$ 910 cubic feet (equivalent to $\leq$ 82% of wide range indicated level) and both pressurizer proportional heater groups shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

(a) With the pressurizer inoperable due to water volume $\geq$910 cubic feet, be in at least HOT SHUTDOWN with the reactor trip breakers open within 12 hours.

(b) With the pressurizer inoperable due to an inoperable emergency power supply to the pressurizer heaters, either restore the inoperable emergency power supply within 72 hours or be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.4.1 The pressurizer water volume shall be determined to be within its limits at least once per 12 hours.

4.4.4.2 The pressurizer proportional heater groups shall be determined to be OPERABLE:

(a) At least once per 12 hours by verifying emergency power is available to the heater groups, and

(b) At least once per 18 months by verifying that the summed power consumption of the two proportional heater groups is $\geq$ 150 KW.

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 T_{avg} 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 surveillance requirements of Specification 3.4.5 do not apply to the special steam generator tube inspection to be performed during the 2P95-1 outage scheduled to begin on January 6, 1995. The scope and expansion criteria for this inspection are specified in correspondence to the NRC submitted under separate cover. The scope and criteria shall be approved by the NRC prior to exiting Mode 5. The results of this inspection shall be reviewed by the Plant Safety Committee prior to resumption of plant operation and reported to the NRC within 30 days of resumption of plant operation.

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 tube inspection provided:

1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

2. The inspections include those portions of the tubes where imperfections were previously found.

The result of each sample inspection shall be classified into one to the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. 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 tubes 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.

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 or sleeve defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug or sleeve defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug or sleeve 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 or sleeve defective tubes and inspect 2S tubes in each other S.G. Special Report to NRC per Specification 6.9.2	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug or sleeve defective tubes. Special Report to NRC per Spec. 6.9.2.	N/A	N/A

$S = 3 \frac{N}{n} \%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR above the limits specified by Specification 3.2.4 during all normal operations and anticipated transients.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

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

The operation of one Reactor Coolant Pump or one shutdown cooling pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reductions will, therefore, be within the capability of operator recognition and control.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 420,000 lbs. per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating shutdown cooling loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

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

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

3/4.4.4 PRESSURIZER

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

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

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 0.5 GPM 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 0.5 GPM per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged or repaired.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 158 TO

FACILITY OPERATING LICENSE NO. NPF-6

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT NO. 2

DOCKET NO. 50-368

1.0 INTRODUCTION

By letter dated November 29, 1994, as supplemented by letters dated December 20 and 21, 1994, Entergy Operations, Inc., the licensee, submitted a request to change the technical specifications for Arkansas Nuclear One, Unit 2 (ANO-2). The requested amendment revises, in part, Technical Specifications 4.4.5.0 and 4.4.5.2.b.3, Table 4.4-2, and Bases 3/4.4.5. The proposed changes to the technical specifications would permit, in part, the licensee to implement an alternate expansion criteria for the steam generator tube inspections to be performed during the 2P95-1 outage in January 1995. The requested amendment also makes several administrative changes to remove inconsistencies and a misspelling introduced in previous amendments to the technical specifications. The proposed inspection scope and expansion criteria for the steam generator tubing are designed to address a specific type of degradation mechanism, that is, circumferential cracking at the expansion-transition region.

The licensee's letters of December 20 and 21, 1994, provided additional clarifying information with respect to its proposed inspection scope and expansion criteria that did not change the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

In March 1992, ANO-2 shut down as a result of a primary-to-secondary leak of approximately 0.25 gallons per minute. During the performance of eddy current examinations of the steam generator tubing during this outage (2F92), the licensee discovered a number of circumferential cracks at the expansion-transition which is located near the top of the tubesheet. The licensee decided to pull several tubes to characterize the root cause of failure. As a result of the pulled tube metallurgical examinations and rotating pancake coil (RPC) inspections at the expansion-transition, the licensee concluded that the plant could be operated safely until the scheduled refueling outage in the Fall of 1992 (approximately four-months of operation).

During the 1992 refueling outage (2R9), the licensee detected additional circumferential indications on a number of tubes at the expansion-transition

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location. The licensee pulled several tubes for metallurgical examination during the outage and also performed an in-situ pressure test of the most degraded tube detected as characterized by the RPC probe. The licensee concluded that all detected circumferential indications were capable of meeting the structural criteria of Regulatory Guide 1.121. The licensee also concluded, based on the distribution of indications detected, that the plant could be safely operated for six-months prior to the next inspection.

During the scheduled plant shutdown in the Spring of 1993 (2P93-1), the licensee examined the expansion-transition location of approximately 6000 tubes with an RPC probe. These examinations were concentrated in a region of the steam generator tube bundle where the majority of circumferential cracking had been detected previously. Based on the inspection findings from this outage and other considerations, the licensee concluded that the plant could be operated safely for the remainder of the cycle, approximately 10-months. The licensee concluded that a 10-month operating cycle was acceptable based, in part, on:

- 100-percent RPC probe inspection in the area of interest (i.e., approximately 70-percent of the total tube population was inspected at the expansion-transition with the RPC probe).
- All the detectable indications were repaired (i.e., plugged or sleeved).
- An assessment that concluded that none of the detected cracks exceeded Regulatory Guide 1.121 acceptance criteria based on RPC data after 6-months of operation.
- An analysis which determined the required through-wall depth necessary to sustain Regulatory Guide 1.121 pressure loads.
- A projection of the total number of tubes expected to be detected at the end of a 10-month operating interval.
- Improved eddy current testing data analysis guidelines and acquisition techniques.
- Enhanced primary-to-secondary leak rate monitoring including the use of Nitrogen-16 monitors.
- Improvements in the secondary water chemistry program.
- Lowering of the primary coolant hot leg temperature.
- Evaluation of operator response to single and multiple steam generator tube ruptures (with a concurrent main steam line break (MSLB) in some instances) during simulator training exercises.

Subsequent to the Spring 1993 outage (2P93-1), the plant was operated for approximately 10-months after which a refueling outage inspection of the steam generator tubing was performed (i.e., 2R10). This outage commenced in the March 1994 time-frame. During 2R10, the licensee inspected 100% of the tubes with an RPC probe at the expansion-transition location. Based on an evaluation of the inspection findings and other considerations, the licensee concluded that the plant could be operated safely for approximately 10-months until a scheduled outage in January 1995 (i.e., 2P95-1).

As a result of these previous inspection findings, the licensee is proposing to modify their technical specifications to permit them to focus their inspection efforts during 2P95-1 in the area of the steam generator where circumferential cracking has historically been observed.

3.0 PROPOSED REVISIONS TO THE TECHNICAL SPECIFICATIONS

The following revisions to the technical specifications are being proposed by the licensee:

1. Correcting a reference in Technical Specification 4.4.5.2.b.3 to specify that the reference for the definition of tube inspection is in Technical Specification 4.4.5.4.a.9.

2. Deleting a one-time exemption included as Footnote 1 to Table 4.4-2.

3. Correcting a misspelled word in Table 4.4-2.

4. Correcting an inconsistency in Bases 3/4.4.5 to allow an unserviceable tube to be either plugged or repaired.

5. Adding a note to Technical Specification 4.4.5.0 which states that:

   a. the surveillance requirements of Specification 3.4.5 do not apply to the special steam generator tube inspection to be performed during the 2P95-1 outage scheduled to begin on January 6, 1995;

   b. the scope and expansion criteria for this inspection are specified in correspondence to the NRC submitted under separate cover;

   c. the scope and criteria shall be approved by the NRC prior to exiting Mode 5; and

   d. the results of this inspection shall be reviewed by the Plant Safety Committee prior to resumption of plant operation and reported to the NRC within 30 days of resumption of plant operation

4.0 EVALUATION

4.1 Minor Administrative Changes

The licensee has proposed to correct several minor administrative errors in the technical specifications. The staff's evaluation of these changes are provided below.

The staff finds that correcting the reference to "tube inspection" in Technical Specification 4.4.5.2.b.3 from 4.4.5.4.a.8 to 4.4.5.4.a.9 is appropriate since the reference to Technical Specification 4.4.5.4.a.8 references a section of the technical specifications pertaining to an unserviceable condition of a tube rather than the inspection of the tube. The proposed technical specification change is acceptable.

The staff finds that deleting footnote 1 of Table 4.4-2 which references a one time exemption to the C-3 inspection requirements as a result of not inspecting two tubes during 2R9 is appropriate since the tubes have subsequently been inspected and the permitted interval of applicability of this footnote has expired. As a result, the proposed technical specification change is acceptable.

The staff finds that changing the spelling of the word "mininum" to "minimum" in Table 4.4-2 is appropriate. As a result, the proposed technical specification change is acceptable.

The staff finds that changing Technical Specification Bases 3/4.4.5 to clarify that a tube may be either plugged or repaired in cases when the plant is shutdown as a result of primary-to-secondary leakage in excess of the technical specification limits is acceptable since tube repair (i.e., sleeving) was approved for ANO-2 as a result of the issuance of Amendment No. 133 on April 22, 1992.

4.2 <u>Inspection Scope and Expansion Criteria</u>

By letter dated November 29, 1994, the licensee proposed to exempt all special steam generator tube inspections (e.g., inspections conducted at frequencies shorter than those specified in the technical specifications, inspections with defined scopes for examinations of a specific area of interest, etc.) from the surveillance requirements of Specification 3.4.5. Based upon subsequent discussions with the NRC, the licensee narrowed the focus of the proposed amendment in a letter dated December 20, 1994. The licensee's revised submittal provides a one-time exemption to the requirements of Specification 3.4.5 for the special steam generator tube inspection to be performed during the 2P95-1 outage scheduled to begin on January 6, 1995. Specifically, the licensee has proposed to add a note to Technical Specification 4.4.5.0 which states that (1) the surveillance requirements of Specification 3.4.5 do not apply to the special steam generator tube inspection to be performed during the 2P95-1 outage scheduled to begin on January 6, 1995, (2) the scope and expansion criteria for this inspection are specified in correspondence to the NRC submitted under separate cover, (3) the scope and criteria shall be approved by the NRC prior to exiting Mode 5, and (4) the results of this inspection shall be reviewed by the Plant Safety Committee prior to resumption of plant operation and reported to the NRC within 30 days of resumption of plant operation.

By letter dated December 21, 1994, the licensee submitted their proposed scope and expansion criteria for the special steam generator tube inspection to be performed during 2P95-1. The scope of the 2P95-1 steam generator tube inspection includes performing RPC inspections of approximately 5000 tubes in both steam generators in the sludge pile region at the hot-leg expansion-transition location. The licensee may also use ultrasonic probes to size selected indications detected with the RPC probe.

The licensee's proposed expansion criteria for the 2P95-1 inspection is as follows:

> If a circumferential crack indication is found on the periphery of the inspection zone in either steam generator, the zone at the location of the flaw will be expanded to bound the detected indication by two tubes. Should more than one circumferential crack indication be found on the periphery of the inspection zone of a steam generator, the periphery of the entire sample area will be expanded by two tubes. The peripheral expansion will be continued until no further indications are found.

As noted above, the inspections to be performed during 2P95-1 are designed to monitor the progression of circumferential cracking at the expansion-transition location on the steam generator tubing which is located near the top of the tubesheet. Historically, circumferential cracking at ANO-2 has primarily been observed at the expansion-transition location in an area of the hot-leg portion of the steam generator commonly referred to as the sludge pile region. Since the proposed inspection scope essentially includes 100% of the tubes in the area of interest, that is the region where circumferential cracking has historically been observed, the staff finds the initial RPC inspection scope and the proposed expansion criteria to be acceptable for 2P95-1. The expansion criteria provides added confidence that if circumferential cracking starts to occur outside the initial inspection boundary that it will be detected.

The staff notes that the licensee will repair all tubes exhibiting circumferential indications that are detected during the 2P95-1 outage. Authorized repair techniques are those delineated in Specification 3.4.5.

Based on the evaluation provided above, the staff finds the proposed changes to the technical specifications to be acceptable. Furthermore, the staff finds the proposed inspection scope and expansion criteria delineated in the licensee's letter dated December 21, 1994, to be appropriate for 2P95-1.

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 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 (59 FR 62416). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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