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November 8, 1984

Docket No. 50-313

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Mr. John M. Griffin, Senior Vice President
of Energy Supply
Arkansas Power and Light Company
P. O. Box 551
Little Rock, Arkansas 72203

Dear Mr. Griffin:

The Commission has issued the enclosed Amendment No. 86 to Facility Operating License No. DPR-51 for Arkansas Nuclear One, Unit No. 1 (ANO-1). This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated August 13, 1984.

The amendment modifies the ANO-1 TS for Steam Generator Surveillance to (1) provide clarity, (2) modify the designation of those areas identified as special areas in the steam generator where imperfections have been previously found and (3) allow the sleeving of ten steam generator tubes as part of a demonstration program.

The material contained in the enclosed Safety Evaluation is considered to be proprietary and therefore is withheld from public disclosure per 10 CFR 2.790. A non-proprietary version of the Safety Evaluation is also enclosed and is being made publicly available.

Notice of Issuance will be included in the Commission's next Monthly Notice.

Sincerely,

"ORIGINAL SIGNED BY
JOHN F. STOLZ"

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 86
2. Safety Evaluation-Proprietary and Non-Proprietary Versions

cc w/non-proprietary
Safety Evaluation:
See next page

No legal objection
McGuire

ORB#4:DL	ORB#4:DL	ORB#4:DL	OELD	AD:OR:DL	MEB:DE
RIngram	GVissing;cf	JFStolz	McGuire	GLainas	RBosnak
10/31/84	10/31/84	11/01/84	11/02/84	11/7/84	11/8/84

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

ARKANSAS POWER & LIGHT COMPANY

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 86
License No. DPR-51

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

   A. The application for amendment by Arkansas Power and Light Company (the licensee) dated August 13, 1984, 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 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. DPR-51 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 86 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 8, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 86

FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
v	v
-	vi
110j	110j
110k	110k
110l	110l
110m	110m
110n	110n
110o	110o
-	110o1
-	110o2

3.5.2-2E	ROD POSITION LIMITS FOR TWO-PUMP OPERATION FROM 0 to 60 EFPD-ANO-1, CYCLE 5	48c4
3.5.2-2F	ROD POSITION LIMITS FOR TWO-PUMP OPERATION FROM 50 to 200 ± 10 EFPD-ANO-1, CYCLE 5	48c5
3.5.2-2G	ROD POSITION LIMITS FOR TWO-PUMP OPERATION FROM 200 ± 10 TO 400 ± 10 EFPD-ANO-1, CYCLE 5	48c6
3.5.2-2H	ROD POSITION LIMITS FOR TWO-PUMP OPERATION FROM 400 ± 10 TO 435 ± 10 EFPD-ANO-1, CYCLE 5	48c7
3.5.2-3A	OPERATIONAL POWER IMBALANCE ENVELOPE FOR OPERATION FROM 0 TO 60 EFPD-ANO-1, CYCLE 5	48d
3.5.2-3B	OPERATIONAL POWER IMBALANCE ENVELOPE FOR OPERATION FROM 50 TO 200 ± 10 EFPD-ANO-1, CYCLE 5	48d1
3.5.2-3C	OPERATIONAL POWER IMBALANCE ENVELOPE FOR OPERATION FROM 200 ± 10 TO 400 ± 10 EFPD-ANO-1, CYCLE 5	48d2
3.5.2-3D	OPERATIONAL POWER IMBALANCE ENVELOPE FOR OPERATION FROM 400 ± 10 TO 435 ± 10 EFPD-ANO-1, CYCLE 5	48d3
3.5.2-4	LOCA LIMITED MAXIMUM ALLOWABLE LINEAR HEAT RATE	48e
3.5.2-4A	ASPR POSITION LIMITS FOR OPERATION FROM 0 to 60 EFPD-ANO-1, CYCLE 5	48f
3.5.2-4B	ASPR POSITION LIMITS FOR OPERATION FROM 50 to 200 ± 10 EFPD-ANO-1, CYCLE 5	48g
3.5.2-4C	APSR POSITION LIMITS FOR OPERATION FROM 200 ± 10 to 400 ± 10 EFPD-ANO-1, CYCLE 5	48h
3.5.2.4C	APSR POSITION LIMITS FOR OPERATION FROM 400 ± 10 to 435 ± 10 EFPD-ANO-1, CYCLE 5	48i
3.5.4-1	INCORE INSTRUMENTATION SPECIFICATION AXIAL IMBALANCE INDICATION	53a
3.5.4-2	INCORE INSTRUMENTATION SPECIFICATION RADIAL FLUX TILT INDICATION	53b
3.5.4-3	INCORE INSTRUMENTATION SPECIFICATION	53c
4.4.2-1	NORMALIZED LIFTOFF FORCE - HOOP TENDONS	85b
4.4.2-2	NORMALIZED LIFTOFF FORCE - DOME TENDONS	85c
4.4.2-3	NORMALIZED LIFTOFF FORCE - VERTICAL TENDONS	85d

4.18.1	UPPER TUBE SHEET VIEW OF SPECIAL GROUPS PER SPECIFICATION 4.18.3.a.3	110o2
6.2-1	MANAGEMENT ORGANIZATION CHART	119
6.2-2	FUNCTIONAL ORGANIZATION FOR PLANT OPERATION	120

4.18 STEAM GENERATOR TUBING SURVEILLANCE

Applicability

Applies to the surveillance of tubing of each steam generator.

Objective

To ensure integrity of the steam generator tubing through a defined inservice surveillance program, and to minimize exposure of personnel to radiation during performance of the surveillance program.

Specification

4.18.1 Baseline Inspection

The first steam generator tubing inspection performed according to Specifications 4.18.2 and 4.18.3.a shall be considered as constituting the baseline condition for subsequent inspections.

4.18.2 Examination Methods

Inservice inspection of steam generator tubing shall include nondestructive examination by eddy-current testing or other equivalent techniques. The inspection equipment shall provide a sensitivity that will detect defects with a penetration of 20 percent or more of the minimum allowable as-manufactured tube wall thickness.

4.18.3 Selection and Testing

The steam generator sample size is specified in Table 4.18.1. The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.18.2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.18.4 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.18.5. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in both steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. The first sample inspection during each inservice inspection (subsequent to the baseline inspection) of each steam generator shall include:

   1. All nonplugged tubes that previously had detectable wall penetrations (>20%), and

   2. At least 50% of the tubes inspected shall be in those areas where experience has indicated potential problems, except where specific groups are inspected per Specification 4.18.3.a.3.

A tube inspection (pursuant to Specification 4.18.5.a.8) 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.

3. Tubes in the following groups may be excluded from the first random sample if all tubes in a group in both steam generators are inspected. The inspection may be concentrated on those portions of the tubes where imperfections were previously found. No credit will be taken for these tubes in meeting minimum sample size requirements. Where only a portion of the tube is inspected, the remainder of the tube will be subjected to the random inspection.

   (1) Group A-1: Tubes within one, two or three rows of the open inspection lane.

   (2) Group A-2: deleted

   (3) Group A-3: Tubes in the wedge-shaped group on either side of the lane region (Group A-1) as defined by Figure 4.18.1.

b. The second and third sample inspections during each inservice inspection as required by Table 4.18.2 may be less than a full tube inspection by concentrating the inspection on those areas of the tube sheet array and on those portions of the tubes where tubes with imperfections were previously found.

The results of each sample inspection shall be classified into one of 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.

NOTES: (1) In all inspections, previously degraded tubes must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.

(2) Where special inspections are performed pursuant to 4.18.3.a.3, defective or degraded tubes found as a result of the inspection shall be included in determining the Inspection Results Category for that special inspection but need not be included in determining the Inspection Results Category for the general steam generator inspection.

4.18.4 Inspection Intervals

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

a. The baseline inspection shall be performed during the first refueling shutdown. Subsequent inservice inspections shall be performed at intervals of not less than 10 nor more than 24 calendar months after the previous inspection. If the results of two consecutive inspections for a given group* of tubes following service under all volatile treatment (AVT) conditions fall 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 for that group may be extended to a maximum of 40 months.

b. If the results of the inservice inspection of a steam generator performed in accordance with Table 4.18.2 at 40-month intervals for a given group* of tubes fall in Category C-3, subsequent inservice inspections shall be performed at intervals of not less than 10 nor more than 20 calendar months after the previous inspection. The increase in inspection frequency shall apply until a subsequent inspection meets the conditions specified in 4.18.4.a and the interval can be extended to 40 months.

c. Additional unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.18.2 during the shutdown subsequent to any of the following conditions:

1. Primary-to-secondary leakage in excess of the limits of Specification 3.10 (inservice inspection not required if leaks originate from tube-to-tubesheet welds),

2. A seismic occurrence greater than the Operating Basis Earthquake,

*A group of tubes means: (a) All tubes inspected pursuant to 4.18.3.a.3, or (b) All tubes in a steam generator less those inspected pursuant to 4.18.3.a.3.

3. A loss-of-coolant accident requiring actuation of the engineered safeguards, or

4. A main steam line or feedwater line break.

4.18.5 Acceptance Criteria

a. As used in this specification:

1. 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.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either the inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections ≧ 20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service because it may become unserviceable prior to the next inspection; it is equal to 40% of the nominal tube wall thickness.

7. 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 Specification 4.18.4.c.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry completely to the point of exit.

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 4.18.2.

4.18.6 Reports

Following each inservice inspection of steam generator tubes, the complete results of the inspection shall be reported to the NRC. This report, to be submitted within 45 days of inspection completion, shall include:

a. Number and extent of tubes inspected;

b. Location and percent of wall-thickness penetration for each indication of an imperfection; and

c. Identification of tubes plugged.

This report shall be in addition to the report of results of steam generator tube inspections which fall into Category C-3 and which require prompt notification of the NRC per Specification 6.12.3.

Bases

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.

TABLE 4.18-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No
No. of Steam Generators per Unit	Two
First Inservice Inspection	Two
Second & Subsequent Inservice Inspections	One[1]

Table Notation:

1 The inservice inspection may be limited to one steam generator on alternating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

TABLE 4.18-2
STEAM GENERATOR TUBE INSPECTION[2, 3]

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.[1]	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 SG	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 other S.G. Prompt notification to NRC pursuant to specification 6.12.3.	Other S.G. is C-1	None	N/A	N/A
			Other S.G. is C-2	Perform action for C-2 results of second sample	N/A	N/A
			Other S.G. is C-3	Inspect all tubes in each S.G. and plug defective tubes. Prompt notification to NRC pursuant to specification 6.12.3, and request NRC approval of remedial action	N/A	N/A

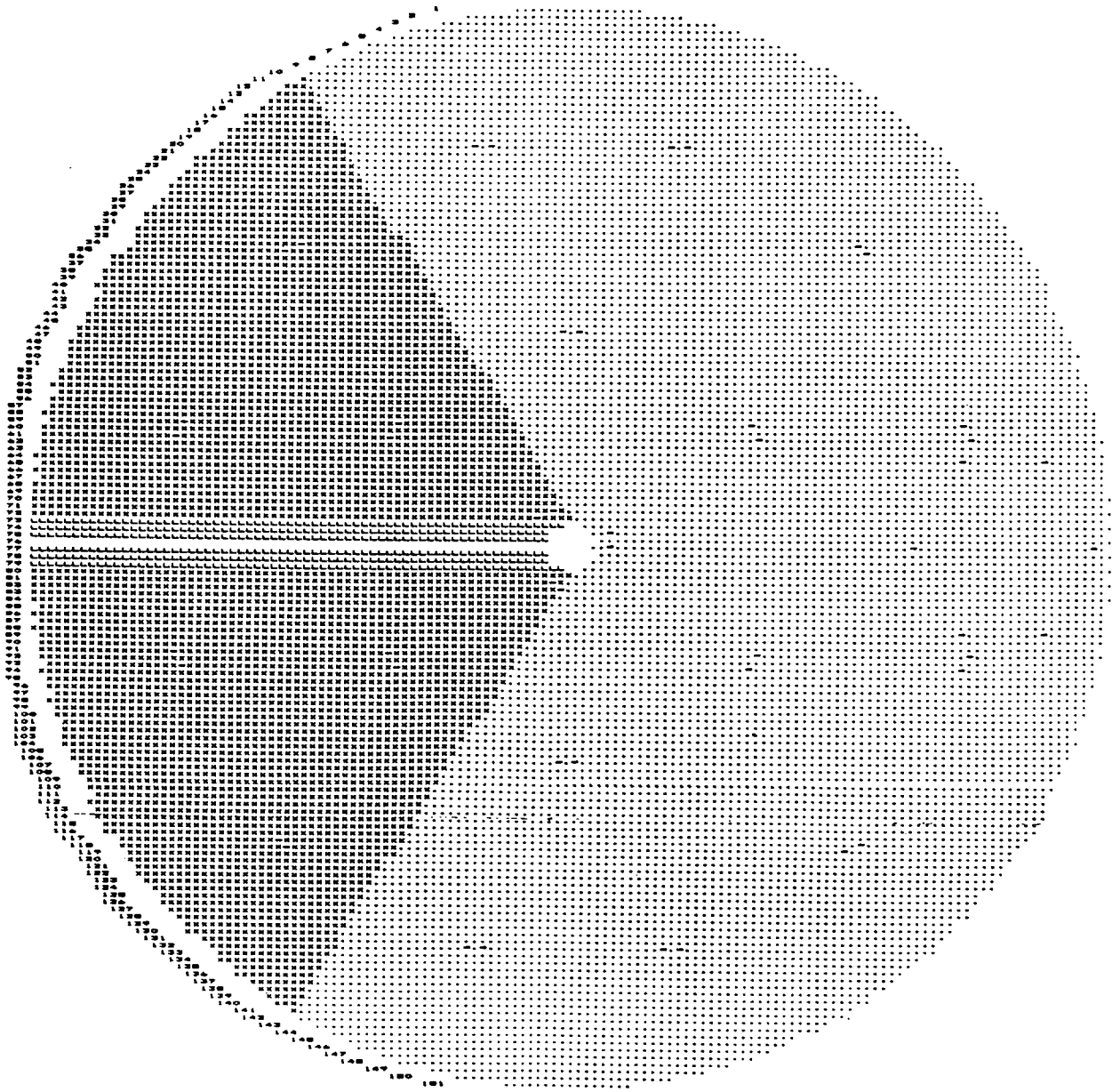
NOTES: [1] $S=3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

[2] For tubes inspected pursuant to 4.18.3.a.3: No action is required for C-1 results. For C-2 results in one or both steam generators plug defective tubes. For C-3 results in one or both steam generators, plug defective tubes and provide prompt notification of NRC pursuant to specification 6.12.3.

[3] As part of a steam generator sleeving qualification program up to 10 demonstration sleeves may be installed in defective tubes in lieu of plugging during the sixth ANO-1 refueling.

FIGURE 4.18.1

Upper Tube Sheet View of Special Groups per Specification 4.18.3.a.3



Plot Character	Description	Tube Count
L	Group A-1: Lane region tube	382
X	Group A-3: Wedge adjacent to lane	4831
-	Support rod location	NA

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 86 TO FACILITY OPERATING LICENSE NO. DPR-51

ARKANSAS POWER AND LIGHT COMPANY

ARKANSAS NUCLEAR ONE, UNIT NO. 1

DOCKET NO. 50-313

1.0 Introduction

By letter dated August 13, 1984, supplemented with supporting information by letter dated October 10, 1984, Arkansas Power and Light Company (AP&L or the licensee) requested amendment to the Technical Specifications (TSs) appended to Facility Operating License DPR-51 for Arkansas Nuclear One, Unit No. 1 (ANO-1). The amendment would revise the TSs for Steam Generator Surveillance to (1) provide clarity, (2) modify the designation of those areas identified as special areas in the steam generator where imperfections have been previously found and (3) allow the sleeving of ten steam generator tubes as part of a demonstration program.

2.0 Background

As a result of degradation in the ANO-1 Once Through Steam Generators (OTSGs) a substantial number of tubes with eddy current indications in excess of the 40% through-wall plugging limit were removed from service prior to the November 1982 outage; 26 tubes were plugged in the A-OTSG and 45 in the B-OTSG.

Since that outage, there have been two additional inspections performed on the steam generators. During the July 1983 outage which was due to a leaking tube, the A-OTSG was partially inspected. There was also a midcycle inspection in which a wedge-shaped area of tubes on either side of the inspection lane was examined from the 15th tube support plate (TSP) on up, in both steam generators. As a result of these two inspections, 75 tubes were plugged in the B-OTSG. This makes a current total of 196 tubes plugged in the A-OTSG and 88 plugged in the B-OTSG.

Of the defective tubes, about 88% were plugged due to degradation in a limited area. This area extends from the midspan of the upper tubesheet (UTS) crevice down to about 1/2 inch below the lower face of the UTS and is also in the wedge-shaped area adjacent to the inspection lane. Examination of two tubes which were pulled from the B-OTSG disclosed outside diameter surface intergranular attack of up to 20% through-wall.

Because of the number of defective tubes being identified, AP&L is initiating a steam generator sleeving qualification program which would provide the basis for a large scale sleeving program. The suspected corrosion mechanism affecting the tubes in the UTS region is corrosion attact by concentrated chemical contaminants carried by moisture in the steam up through the lane region. The lane region is cooler, therefore, more moisture would be in the

steam in the lane region at the UTS region. The contaminants carried by this moisture would be deposited on the tubes in the UTS region. Plugging the lane region tubes increases the area of cooler water flowing up through the lane region and, therefore, the amount of moisture in the steam, thus increasing the amount of contaminants carried by the steam to be deposited on the tubes in the UTS region. This aggravates the degradation of the tubes in the UTS region. Sleeving should improve these conditions by preventing additional loss of heat transfer areas which will limit the spread of degradation. In addition, the sleeved tube has better corrosion resistance than the original tubes.

As part of a long range, large scale sleeving program, AP&L has proposed a demonstration sleeving program to verify the field installation capability of the process. The demonstration program will verify and benchmark the actual field leakage rates against design criteria and laboratory leakage rates and confirm the reliability of tube sleeves under actual operating conditions.

3.0 Discussion and Evaluation

The discussions and evaluations of the OTSG Tube Sleeving Demonstration Program and other proposed TS changes are provided separately below.

3.1 OTSG Tube Sleeving Demonstration Program

A method for extending the service life of a degraded steam generator tube is to install a sleeve inside the original tube to bridge the degraded area, thus permitting the tube to remain in service. Babcock and Wilcox (B&W) has developed and qualified a mechanical tube sleeve that can be installed in degraded tubes of OTSGs. The report BAW-1823P, "Once-Through Steam Generator Mechanical Sleeve Qualification," submitted by AP&L, describes the sleeving methodology, design criteria and qualification testing for the proposed demonstration sleeving of ten sleeves in the ANO-1 steam generators. This section addresses the following aspects of the sleeve/tube qualification program.

1. Sleeve Design
2. Leak Tight Integrity
3. Corrosion Considerations
4. Eddy Current Inspection
5. Pullout Strength of Sleeves
6. Joint Development Considerations
7. Flow-Induced Vibration Effects
8. Effect of Sleeve Installation on Adjacent Sleeves and Tubes
9. Plant Performance Considerations
10. Structural and Functional Integrity Considerations
11. Radiological Considerations
12. Conclusions Regarding OTSG Tube Sleeving Program

3.1.1 Sleeve Design

Tube degradation found in OTSGs was located most frequently within a few inches above or below the secondary face of the UTS. There is some concentration at the 15th TSP and in the 16th tube span (i.e., between the 15th TSP and the UTS), and the remainder seems to be rather randomly distributed at elevations below the 15th TSP. The radial location of

degradation tends to be more frequent toward the outer periphery of the tube bundle, and there is some concentration near the open tube lane.

The qualified design of the sleeve is 30 inches to 80 inches long to span the entire 16th tube span and 15th TSP. The sleeve is made from tubing that has been mill-annealed and heat treated at The tube size is The head clearance over the outermost tubes in the OTSG is only about 13 inches, whereas the sleeve length required to extend 6 inches beyond the UTS secondary face is 30 inches. Therefore, installation requires that the sleeves be prebent to a gentle radius in order to clear the head and then straightened as they are fed into the tubes. This sleeve design can be installed in any tube in the steam generators.

There is about in the tubesheet, and expansions near the free end of the sleeve. The installation method is roller expanding both ends of the sleeve. This gives adequate leak tightness and pull out strength. Roller expansion has been routinely used to seal tubes into tubesheets in all types of heat exchangers.

The amount of expansion is controlled by limiting the torque applied to the expansion tool since there is a rapid increase in torque as the tube is squeezed against the tubesheet and the tubesheet begins to be extruded axially. Roller expansion of the sleeve in the free-span tube without a tubesheet backup is an innovative application.

The sleeve design loading requirements have been established to be equivalent to those of the unsleeved tube such that a sleeved tube could be totally severed without affecting the function of the tube. The sleeve is a structural member that meets all normal, up-set, emergency, and faulted conditions resulting from normal operation and accident transients.

3.1.2 Leak Tight Integrity Considerations

The leakage rate for the sleeved tubes was determined for normal operating and accident conditions. Data indicate that leakage is The resulting overall service factors and cumulative leakage after 40 years simulated service have been provided. Within the range of normal operation the maximum leakage is well within the 2.5 ml/h average

The tightness of a rolled joint is highly dependent upon the yield strengths of the materials used. In this qualification, both tubes and sleeves were Type 600 Inconel per ASME Specification SB-163, which permits a minimum yield

strength of 35,000 psi. All sleeves were made from a single heat of material which had a yield strength of

Results indicate that stronger tubes are also much tighter.

Two samples were tested to assess the effects of test temperature on tube leakage. When the specimens were maintained at 388°F, the mean leakage with no axial load was At the ambient temperature of 67°F, the corresponding mean leakage was Although it was not practical to leak test at the full design temperature, the hot test results Since both sleeve and

In service, the annulus between sleeve and tube may tend to insulate the tube so that the sleeve is hotter

Space limitations require that the sleeves be bent into a gentle arc outside the OTSG and straightened as they are fed into the tubes.

Two sample sleeves, which had been bent and straightened, were expanded into tubes and leak tested to verify that the insertion process does not degrade the quality of the expansion joint. and there

The axial load on a tube during operation is a function of the pressures, the position of the tube in the OTSG, and the tube and shell temperature difference, which in turn is a function of the service transient. For normal operation, the transient that results in the greatest total tube load is a cooldown from 15% power. During this transient, the load on a center tube reaches 649 lb while the load on a peripheral tube reaches 1107 lb. The worst accident condition is a main steam line break (MSLB), which results in a 3140-lb total load on a peripheral tube of high-yield strength, or 2620 lb if the tube has a low-yield strength. For a central tube, the maximum load of 1585 lb is a result of a loss-of-coolant accident (LOCA). It was determined from the data that under maximum operating load, the greatest expected leakage for a peripheral tube would range from 0.32 ml/h for a tube of high-yield strength to 1.43 ml/h for a tube of low-yield strength. Furthermore, it was determined that under maximum accident load, the greatest leakage for a peripheral tube would range

In order to predict the leakage in a sleeved OTSG, it is necessary to make some assumptions regarding the location and yield strength of the sleeved tubes.

Thus, the predicted maximum leakage for a under normal operating loads and If there were 10,000 sleeved tubes in a plant (and all of the tubes leaked), the predicted leakage in normal operation would be Technical Specification plant shutdown limit. For accident conditions, the rate would be

In the event of a complete failure of a rolled joint, such as a full circumferential tube crack at the lower roll transition, the tube and sleeve have been designed to remain engaged under worst-case accident conditions The maximum leakage which could occur in such a failed tube has been calculated as For comparison, the maximum leakage for an unsleeved ruptured tube has been calculated as

Conclusion

Based on a review of the test data as discussed above, we conclude that the maximum leakage under normal operation for a sleeved tube after 40 years simulated service is well within the 2.5 ml/h average leakage objective. Under worst accident conditions, the tube and sleeve will remain engaged even if the rolled sleeve joint is assumed to fail completely.

3.1.3 Corrosion Considerations

Accelerated stress corrosion cracking tests in autoclaves with a 10% sodium hydroxide solution at 550°F and a +190 mV applied potential were performed on sleeved tube mockup specimens to determine whether residual stresses from the sleeving process are sufficent to cause stress corrosion cracking of the sleeve or the OTSG tube.

Sleeve mockups showed

which demonstrates that the test was rigid enough to produce severe cracking in highly strained specimens.

Use of the accelerated caustic corrosion test results to predict failure in service with all-volatile-treated (AVT) water is based upon a correlation developed by the Electric Power Research Institute. Although the data are limited,

These correlations are for well controlled AVT water and may not apply to other water treatments or water with impurities which may be present in steam generator operations. Actual steam generator water chemistries and loading conditions may reduce this prediction

The cracks in

Because of the limited nature of these data, the licensee has committed to continuing corrosion tests during the 10 tube demonstration period. As part of this program, the following test was proposed by AP&L.

Since the outside diameter of the tubes to be sleeved in the ANO-1 OTSGs could possibly contain intergranular attack (IGA) at the elevation of the free-span roll expansion of the sleeve, a corrosion test is proposed to confirm that the expanded sleeve design will not significantly accelerate or propagate the existing IGA in the ANO-1 OTSGs.

A specimen will be fabricated from a portion of a tube pulled from the ANO-1 B-OTSG in January 1983, on which IGA was observed. The specimen will be fabricated using the process developed for field sleeve installation. The specimen will be exposed in an autoclave at approximately 600°F in an environment that contains approximately 6 times the typical feedwater contaminant concentrations. A tensile load of 500 lbs. will be placed on the expanded joint for the duration of the 2000 hour test. Upon completion of this exposure, one of the two expanded joints will be removed for metallurgical examination. The remaining joint will be replaced in the autoclave and wet layup conditions at 150°F will be established for a period of one month. This joint will be removed from the autoclave and both joints will be metallurgically examined for evidence that the existing IGA has or has not progressed under the test conditions.

In addition, AP&L will evaluate the effect of roll expanding a sleeve into existing ANO-1 tubing. Three ANO-1 tube samples will be cut from a previously pulled ANO-1 tube sample obtained from a portion of a tube adjacent to an area known to have IGA present on the outside diameter surface. A sleeve will be double roll expanded into each tube sample to the maximum qualified expansion. The tube samples will then receive a destructive metallurgical examination consisting of SEM, stereo macroscopy, and metallography in the rolled area to determine if the tube expansion caused any progression of the existing IGA. Eddy current examination of each tube before and after rolling will be made in order to accumulate data to be compared with the destructive examination data.

In another test to be conducted, the impact of the sleeving processes relative to residual stress levels will be evaluated using stress corrosion cracking susceptibility. Sensitized Inconel Alloy 600 is known to be susceptible to stress corrosion cracking in sulfur bearing environments if tensile stresses above a threshold value are present. The time to initiate cracks is related to the magnitude of the tensile stress increase. Stress corrosion cracking tests will be performed in polythionic acid using stressed C-rings and actual sleeved joints. Residual stress levels in the original tubing as a result of the sleeving operation will be evaluated by comparing time-to-crack for C-rings stressed to known levels with the performace; i.e., time-to-crack of actual joints. Due to the limited quantity of archive tubing, it will be necessary that the bulk of the time-to-crack vs. stress level data be generated using a surrogate material. The surrogate material would be sensitized to result in the same level of sensitization as observed in the archive material. This surrogate material would also be employed to develop a stress level vs. time-to-crack curve following a simulated braze cycle.

Following evaluation of the results of the testing described above and the experience gained via the demonstration sleeving program, the extent of

additional qualification testing needed to fully support a large scale sleeving program will be determined by AP&L.

3.1.4 Eddy Current Inspection

The issue of eddy current testing of the sleeved tubes was not addressed in the B&W report. It is generally known that the sleeved tubes are more difficult to inspect by eddy current method than the nonsleeved tubes. However, the licensee has later committed to a program to demonstrate the adequacy of eddy current inspection techniques when inspecting sleeved tubes. We will evaluate the efficacy of the licensee's eddy current inspection techniques when such a program is submitted for review.

3.1.5 Pullout Strength of Sleeves

The expected joint strength was determined by measuring the under a range of axial loads, and adding the service-cycled to the as-installed This results in an end-of-life which can be compared to the as-installed curve. The maximum expected for normal operation and accident conditions are read from these curves at the appropriate axial loads. The as-installed,

The data provided by the licensee indicates that

The vibration cycling results in

The cumulative calculated by the licensee represent the expected joint of simulated service.

The strength of a rolled joint is dependent upon the yield strength of the materials used. In this qualification, most of the tubes were made from

All sleeves were made from a

made with sleeves that had been bent and straightened were measured at various axial tube loads. The results show slightly greater and straightened sample, but both are quite compared to specimens that had been

A comparison of the expected to the design requirement has been provided. Under operating loads, the maximum joint Under accident loads, the maximum for a low-yield peripheral tube.

based upon low-yield tubes on the outer periphery of the OTSG undergoing the maximum cooldown rate or maximum hypothetical accident conditions. They also presume that the sleeved tube is totally severed so that the rolled sleeve joint must carry the entire axial

load on the tube. In actual practice, the tube would normally share the load with the sleeve because a severed tube is quite rare. As previously discussed, the mean tube

Thus, it would be a relatively rare occurrence for a sleeve in a low-yield peripheral tube to be subjected to the full maximum operating loads. However, when this happens, the rolled joint is capable of carrying the load without failure.

Although no acceptance criteria were established for the ultimate failure of a rolled joint, most of the test specimens were pulled to failure. These failures

Conclusion

No joints failed at loads below the tube The maximum slippage was A slipping joint could be displaced Our review of the test results indicates that the acceptance criterion for pullout strength of at least has been satisfied. The acceptance criterion of proportionally smaller loads

Since the licensee's acceptance criterion is based on the worst case accident loading, we conclude that the joint strengths are acceptable.

3.1.6 Joint Development Considerations

The tightness of a rolled joint depends upon the amount of roller expansion. In the qualification tests, all sleeves which were rolled at the tubesheet end had a regardless of tube dimensions and strength. The free-span expansions were

The amount of free-span diameter increase depends on

The average tube diameter increase in the free span was In the event that dimensional stack-ups result in a much Three specimens were tested with

The light expansions show a

It is seen that The effect of light expansions on The sensitivity of the joint to

The roller expansion tool

The effect of light expansions on an OTSG can be estimated by applying factors to the expected portion of light expansions.

The mean leakage under maximum accident conditions in a light expanded tube has been determined to be If leak at the maximum leak rate under accident conditions for normal expansions, the predicted leakage for the total plant would be or about Under the worst normal operation loads, the expected mean leakage is practically

Conclusion

The data obtained by the licensee indicate that approximately and average leakage of under maximum accident loads. Based on a review of the data, we conclude that the sleeve installation process will be controllable, and predictable joint quality is likely to be maintained.

3.1.7 Flow-Induced Vibration Effects

The effect which a sleeve has on the vibration characteristics of a tube is not obvious because the sleeve stiffens the tube and tends to increase the system damping. However, the additional mass tends to reduce the natural frequency. The characteristics of the sleeved tube depend upon the

The response plots from the in-air tests were used to determine approximate critical damping ratios These values were rounded off to conservative test damping values, and then more conservative operational damping values for the analysis were obtained

A NASTRAN finite element model was used for the analysis with at the tubesheet face and damping as listed above. Secondary side crossflow velocities of OTSGs with both internal and external auxiliary feedwater headers were evaluated.

Conclusion

Based on a review of the computed results showing the worst case generic fluid-elastic stability margins and the and for the random vibration and vortex shedding responses, we conclude that the fluid-elastic stability margin for the sleeved tube is and the maximum are less for all cases. In addition, the maxium calculated corresponding to the number of design cycles for a 40-year life. Therefore, it is concluded that flow-induced vibration will not be detrimental in any OTSG tube sleeved in the upper span, even if the tube is completely

3.1.8 Effects Of Sleeve Installation On Adjacent Tubes

Specimens were processed to evaluate the sleeve and tube During the tubesheet roll, the sleeve

Based on our review of data, it is concluded

There was some concern that rolling a sleeve in a tube adjacent to a tube which previously had been sleeved may loosen the first expansion, causing increased leakage.

it is conservative when applied to rolled sleeves in the OTSG tubesheet.

Based on a review of the adjacent tube test results, we conlude that adjacent sleeve installations are unaffected by a new sleeve installation.

3.1.9 Plant Performance Considerations

The installation of a significant number of sleeves in the OTSG could reduce the OTSG's thermal performance due to the insulating effect of the sleeve (especially, the annulus between sleeve and tube) and the change in primary flow distribution caused by higher flow resistance. The net result of these effects

The analysis of thermal and hydraulic effects assumed that 5000 80-inch long sleeves were installed in the peripheral tubes of each OTSG. These worst-case assumptions reduce primary flow by

The effect of this reduction in superheat temperature on plant operation is considered to be minimal. The first OTSG put into operation was warranted to produce a minimum of 35°F superheat steam at full power,

The new operating point for the OTSGs, turbines, and feedwater control system would be

Conclusion

Based on a review of the licensee's analysis of the effects of sleeving on plant performance, we conclude that

The thermal/hydraulic effects are therefore considered acceptable.

3.1.10 Tube/Sleeve Structural and Functional Integrity Considerations

The minimum acceptable wall thickness for degraded sleeves was determined in accordance with the allowable stress and pressure limits of ASME Section III and NRC Regulatory Guide 1.121. Primary membrane stress, burst pressure, and fatigue analyses were considered for normal operation, and primary membrane stress, burst pressure, collapse pressure, and primary membrane plus bending stresses were considered for postulated accident conditions. In addition, primary plus thermal stresses were evaluated. The minimum sleeve wall thickness was calculated for these eight different acceptance criteria. For the expected type of defects, the greatest required minimum wall was found to be

70% through-wall defect would require that a sleeve be removed from service. This compared to a 69% defect limit for the OTSG tubes.

The sleeve must be bent and straightened for installation in the outermost OTSG tubes. This results in a slightly elliptical cross section, which was evaluated for buckling pressure. The maximum expected ovality (i.e., difference in extreme ODs at any one cross section) was found to be inch based

on sample dimensions. The critical external pressure depends on the material yield strength.

Under the maximum secondary pressure of 1050 psi with no primary pressure, neither tube nor sleeve would collapse.

annular pressure increase is more likely to blow out the corrosion products which plugged the leak than to collapse the sleeve. Thus, the likelihood of sleeve collapse is very small.

Conclusion

The results of the licensee's analysis indicate that the minimum required sleeve wall for normal and accident conditions is The licensee's analysis is in compliance with the requirements of ASME Code Section III and NRC Regulatory Guide 1.121. We, therefore conclude that sleeve/tube integrity will be maintained by operating within the above specific limits.

3.1.11 Radiological Considerations

We have evaluated the radiation protection measures extablished for the demonstration tube sleeving program at ANO-1 by AP&L, including those features intended to ensure that doses will be maintained as low as is reasonably achievable (ALARA). The bases for our review are the criteria outlined in NUREG-0800 (SRP) and Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable."

The licensee has provided information regarding their tube sleeving program in submittals dated August 13, 1984, and October 10, 1984. Additional related information has been utilized by the NRC staff based on visits to the B&W mockup facilities and discussions with Rochester Gas and Electric (RG&E) and B&W personnel on June 23, 1982, and September 15, 1982, and participation in the Independent Design Review Panel meetings for the Ginna tube sleeving program. The licensee will minimize individual and collective doses through the use of remote automatic systems, the use of mock-up training, ALARA preplanning including the estimation of doses, and application of methods to reduce radiation sources. The licensee's preparations and programs for radiation protection meet our requirements in 10 CFR 20 and are consistent with our guidance in Regulatory Guide 8.8 and are, therefore, acceptable.

3.1.11.1 Radiation Protection

The licensee has projected performing tube sleeving at ANO-1 for approximately 10 steam generator tubes as part of a Steam Generator Sleeving Qualification Program for an anticipated large scale sleeving program at ANO-1. Special surveys will be conducted to establish radiological conditions in the steam generator, and dose estimates will be performed in preparation for the task. Suitable protective clothing and respiratory protection will be worn.

Multiple dosimetry will be provided to monitor personnel exposure. Staytime will be calculated and carefully controlled by Health Physics Technicians. Communications via headset will be maintained with personnel entering the steam generator. Normal and special radiological control procedures have been developed and will be applied for the steam generator entries. Prompt update of exposure through TLD readout will be performed.

3.1.11.2 ALARA Considerations

AP&L's ALARA Committee and vendor representatives will perform an ALARA review of the steam generator tube sleeving program to assure that occupational doses will be maintained as low as is reasonably achievable, consistent with 10 CFR 20.1(c) and Regulatory Guide 8.8.

Prior to beginning the tube sleeving project, AP&L will perform a decontamination by hydrolancing and evaluate shielding techniques which can effectively reduce dose rates in the steam generator channel head. A principle ALARA feature is the performance of most aspects of the tube sleeving process by remote, automatic means. Technicians and supervisory personnel can remain in a trailer outside of the containment and outside of radiation areas, and only one individual may be needed at the steam generator manway to change tooling. Actual entry into a steam generator channel head is needed to install and remove the automatic equipment. Work can be monitored remotely by closed-circuit TV and intercom from the trailer. The remote, automatic equipment can perform inspection and repair functions, through tooling changes at the manway, greatly minimizing dose by minimizing entries into the high radiation fields and minimizing the numbers of personnel necessary in radiation areas to conduct operations.

Planning will include the use of mock-up training and pre-job briefings conducted outside of radiation areas. The vendor has prior experience, including the tube sleeving efforts at Ginna, to factor into development and design. Surveys and manpower/time estimates will be used to estimate the dose for the task as part of the ALARA review. The experience gained as a result of the demonstration program may form the basis for a subsequent major tube sleeving effort at ANO-1 and can serve to identify potential problems as well identify the effectiveness of ALARA and radiation protection measures.

Personnel exposure will be controlled by radiation and contamination surveys, airborne radioactivity surveys, use of protective clothing and respiratory protection, and controlled access.

3.1.11.3 Conclusions

AP&L's radiation protection provisions and efforts to maintain occupational doses ALARA during the steam generator repair work are in accordance with 10 CFR 20 and the guidelines of Regulatory Guide 8.8 and are acceptable.

3.1.12 Conclusions Regarding OTSG Tube Sleeving Demonstration Program

We have reviewed the licensee's submittals and on the basis of our evaluation of the aspects considered above, we have determined that it is acceptable to change the TSs to allow the proposed sleeving of 10 OTSG tubes.

3.2 Other Proposed Technical Specification Changes

The current plant Technical Specifications identify a special group of tubes to be inspected regularly. The existing special group A2 includes tubes having a drilled opening in the 15th TSP. These tubes were included as a special group since tube denting similar to that frequently found in recirculating steam generators was possible in OTSGs. Experience on B&W plants to date has shown that the drilled hole group has not shown any significant tube denting. B&W now recommends only random inspection of the drilled hole group.

Because this special group has not shown significant degradation, AP&L does not consider inspection of 100% of the tubes in this group to be necessary in the future. Not inspecting all the tubes in this group would require that these tubes be part of the makeup of the 50% sample required by the Technical Specifications. To prevent diluting the 1st sample, elimination of the special group was requested. We agree with the licensee's basis for this request and find this change to be acceptable.

In another change request, new special group A-3 is added. The new special group consists of 4,831 tubes adjacent to the lane region. The new group is a large wedge (1/3 of the tube bundle) originating at the center of the bundle fanning out on either side of the lane region. This new special group is selected due to tube degradation noted in recent steam generator tubing inservice inspections and recent leaker outage tube inspections. Sample analyses performed on tubes pulled during recent inspections have shown that IGA is the root cause of tube degradation leading to the plugging of approximately 280 tubes in the OTSGs. This attack has been concentrated in and around the lane region, group A-1, and at the periphery of the tube bundle near the lane. The proposed group A-3 more than adequately bounds the region of degraded tubing found to date.

Prior to the 11/82 outage, 26 tubes had been plugged due to service related defect indications. All 26 tubes were in the A-OTSG (no defects had been recorded in the B-OTSG prior to the 11/82 outage). Twenty-two of these defects were located at the UTS secondary face or within the UTS. All 22 of these defects would have fallen into this new A-3 special group.

The 100% eddy current examination performed during the 11/82 refueling outage identified 83 tubes with pluggable defects in the A-OTSG and 45 tubes with pluggable defects in the B-OTSG. The majority of these defects were located in the UTS (69 tubes with UTS defects in A-OTSG and 34 tubes with UTS defects in B-OTSG).

The UTS defects in both the A&B-OTSGs were located predominantly in a quadrant around the open lane. Defects below the UTS were scattered randomly over the entire tube bundle. Of these defective tubes, 66 in the A-OTSG and 29 in the B-OTSG would have been detected while conducting a special inspection of the new A-3 group from the 15th TSP up. The defects in the remainder of the tubes located within the bounds of the special group were randomly dispersed below the 15th TSP.

The results of the March 1984 mid-cycle outage also support the proposed configuration of the new A-3 group. During the mid-cycle inspection, an examination identical to that proposed by the new special A-3 group was conducted. The defective and degraded indications identified during this inspection (excluding those tubes with previously reportable indication) all

fell well within the bounds of the large wedge. The heaviest concentration of degraded tubes were found within 10 rows of the lane and around the periphery either side of the lane. The majority of the IGA found to date has been located at and above the secondary face of the UTS. The inspection of the proposed new A-3 group from the 15th TSP up would provide adequate monitoring of the IGA problem.

Including this large group in the 1st random sample inspection could be expected to lead to a C-3 classification of the ANO-1 generators in the next inservice inspection. This would require 100% inspection of all generator tubing. The special groups were originally added to the Technical Specification to prevent unnecessary 100% inspection of the entire generator. This same rationale was used in proposing the new special A-3 group.

By enacting the provisions of Specification 4.18.3.a.3 with the new A-3 group, 100% of the tubes in the special group must be inspected as opposed to a random inspection of the same tubes if the provision were not in place. Thus a more thorough examination of this newly identified special group will be performed by its addition as a special group. The inclusion of the new A-3 group will also make the Specification more restrictive since it will require the inspection of additional tubes to satisfy the requirements of the Technical Specification. For the above reasons, we find this Technical Specification change request to be acceptable. In another requested change, wording is added to permit limiting the inspection of "potential problem" areas to those portions of tubes where imperfections have previously been found. Clarification is also provided to specify that when only a portion of a tube is inspected, the remainder of the tube will be subjected to the random inspection per the Technical Specification.

The existing Specification designates a special group of tubes (i.e, areas of the tube bundle) where experience has indicated potential problems exist. These "potential problem" areas were originally identified in Technical Specification Amendment No. 41. They represent areas where AP&L has acquired enough data from inspections to designate them "critical areas unique to the ANO-1 steam generators." Amendment No. 41 permitted the option of inspecting 100% of the tubes in the "critical area" in lieu of including these areas in the 1st random inspection.

Recent inservice inspections at ANO-1 have shown that certain portions of the tubing can also be used to bound "critical areas." Examples of this include from the 15th TSP to the UTS primary face for the lane region (region A-1) and those additional tubes which make up the new A-3 group. Inspection of the tubes in this group below the 15th TSP is unnecessary to detect further degradation due to the IGA mechanism observed at ANO-1. Past inspections at ANO have shown the condition of these tubes to show only normal wear for their age relative to other B&W plant experience. Therefore, AP&L requested that portions of the tube (from the 15th TSP down) need only be subjected to random inspection. This random inspection is assured by an addition to the Technical Specification.

Limiting the inspection to that portion of the tubes where previous imperfections have been found is not unique. The current Technical Specification allows the second and third sample inservice inspections to be concentrated on those areas of the tubesheet array and on those portions of the tubes where previous imperfections have been found.

The change only provides the option to further define a special group for a detailed inspection. Only the specific problem area in the generator is subject to a 100% examination, while the remainder of the generator is subjected to the random inspection, which is the intent of the Specification. For this reason, we find the change request to be acceptable.

The phrase "except where specific groups are inspected per Specification 4.18.3.a.3" is proposed to be added to this section to clarify the relationship between the random selection provision and the special areas. The added statement does not change the intent of the Specification, but rather adds specific wording to make it comply with the Safety Evaluation Report to Technical Specification Amendment No. 41 which establishes the "potential problem" areas (special group). We, therefore, find this change to be acceptable.

Enviromental Consideration

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and a change in surveillance requirements. We have 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this 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 this amendment.

Conclusion

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: November 8, 1984
Principal Contributors: H. Conrad
J. Rajan
R. Serbu
G. Vissing

DMB 016

November 8, 1984

Docket No. 50-313

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Mr. John M. Griffin, Senior Vice President
of Energy Supply
Arkansas Power and Light Company
P. O. Box 551
Little Rock, Arkansas 72203

Dear Mr. Griffin:

The Commission has issued the enclosed Amendment No. 86 to Facility Operating License No. DPR-51 for Arkansas Nuclear One, Unit No. 1 (ANO-1). This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated August 13, 1984.

The amendment modifies the ANO-1 TS for Steam Generator Surveillance to (1) provide clarity, (2) modify the designation of those areas identified as special areas in the steam generator where imperfections have been previously found and (3) allow the sleeving of ten steam generator tubes as part of a demonstration program.

The material contained in the enclosed Safety Evaluation is considered to be proprietary and therefore is withheld from public disclosure per 10 CFR 2.790. A non-proprietary version of the Safety Evaluation is also enclosed and is being made publicly available.

Notice of Issuance will be included in the Commission's next Monthly Notice.

Sincerely,

"ORIGINAL SIGNED BY
JOHN F. STOLZ"

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 86
2. Safety Evaluation-Proprietary and Non-Proprietary Versions

cc w/non-proprietary
Safety Evaluation:
See next page

ORB#4:DL	ORB#4:DL	ORB#4:DL	OELD	AD:OR:DL	MEB:DE
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10/31/84	10/31/84	10/31/84	10/31/84	10/31/84	10/31/84