-20CS MS-016

Docket No. 50-336

DEC 3 0 1983

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Mr. W. G. Counsil, Senior Vice President
Nuclear Engineering and Operations
Northeast Nuclear Energy Company
P. O. Box 270
Hartford, Connecticut 06141-0270

Dear Mr. Counsil:

SUBJECT: STEAM GENERATOR REPAIRS BY SLEEVING

The Commission has issued the enclosed Amendment No. 89 to Facility Operating License No. DPR-65 for Millstone Nuclear Power Station, Unit 2, in response to your application dated June 3, 1983, as supplemented on August 18, 1983 and November 17, 1983.

This amendment changes the plant Technical Specifications to allow repair of degraded steam generator tubes by installing metal sleeves in addition to the current repair method, which is plugging.

Based on our evaluation, we find the proposed sleeving repair method for degraded steam generator tubes to be an acceptable repair alternative to plugging. The proposed sleeving repairs can be accomplished to produce a sleeved tube of acceptable integrity with respect to metallurgical properties, corrosion resistance, leak tightness and inservice inspectability. Your commitment to use state-of-the-art inspection methods and to utilize improved techniques as they are developed will assure continued integrity.

The design verification program was developed to ensure acceptable performance, leak resistance and structural integrity of the sleeved tube under normal operation, accident and transient conditions. Analytical and structural evaluations of the Millstone Unit No. 2 sleeve and tube assembly were performed in accordance with applicable ASME Boiler and Pressure Vessel Code, Section III, 1980 Edition Criteria. Millstone steam generator design and test loading conditions including Loss of Coolant and Main Steamline Break conditions constituted the bases for the analyses. Based on our review the staff concludes that all primary stresses for the sleeved tube assemblies are well within allowable ASME Code stresses. The maximum range of stress intensities complies with the requirements of ASME Code, Subsection NB-3222.2 at all sleeved tube assembly locations. The cumulative fatigue usage factors are below the allowable value of 1.0 specified in the ASME Code.

The mechanical testing program includes the results of the thermal cycling and fatigue testing of both upper and lower joints to demonstrate the joints' structural integrity and leak resistance. Push-out and pull-out strengths of both the upper and lower joints are greater than potential plant loading conditions and therefore the joint strength of mechanical sleeves will ensure tube integrity.

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The staff has also reviewed the ALARA and radiological assessment of the proposed program and finds it acceptable.

The staff finds certain portions of your submittals did not fully respond to our concerns. Therefore, we request your response to certain open or confirmatory items as follows:

1. You have proposed that the Technical Specification plugging limit of 40% be applied to degraded sleeves. Since this plugging limit was intended for degraded tubes, it is necessary to establish that a 40% degraded sleeve is equivalent in strength to a 40% degraded tube. You have not yet established that the two have equivalent bending strengths. We therefore request that you establish this equivalency. In the interim, we have not accepted the proposed plugging limit for sleeves. However, we will reconsider this promptly on receipt of your response on this subject, allowing this to resolve before your next refueling outage.

2. In our Safety Evaluation dated March 16, 1983 relating to a Technical Specification change for steam generator inspections, we noted that you had committed to certain actions which would result in a more accurate determination of the absolute primary to secondary leakage rate. We request that you provide a status report of these actions.

These open or confirmatory items should be provided within 120 days of your receipt of this letter. This request for information affects fewer than 10 respondents; therefore OMB clearance is not required under P.L. 96-511.

A copy of our Safety Evaluation is enclosed. A non-proprietary version of our Safety Evaluation is being made publicly available. The notice of issuance will be included in the Commission's next monthly Federal Register notice.

Sincerely,

Original signed by

Kenneth L. Heitner, Project Manager
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 89 to DPR-65
2. Safety Evaluation-Proprietary and non-proprietary version to licensee

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

*See previous page for concurrences

ORB#3:DL*	ORB#3:DL*	ORB#3:DL	OELD *	AD:OR:DL *
PMKreutzer	KHeitner/pn	JRMiller		GCLainas
12/23/83	12/23/83	12/30/83	12/27/83	12/27/83

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The staff finds certain portions of your submittals did not fully respond to our concerns. Therefore, we request your response to certain confirmatory items as follows:

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12/23/83	12/23/83	12/ /83	12/27/83	12/27/83

Northeast Nuclear Energy Company

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

NORTHEAST NUCLEAR ENERGY COMPANY

THE CONNECTICUT LIGHT AND POWER COMPANY

THE WESTERN MASSACHUSETTS ELECTRIC COMPANY

DOCKET NO. 50-336

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 89
License No. DPR-65

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

   A. The application for amendment by Northeast Nuclear Energy Company, et al. (the licensee) dated June 3, 1983 as supplemented on August 18 and November 17, 1983, 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 (1) 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-65 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 89 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective on the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James R. Miller

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 30, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 89

FACILITY OPERATING LICENSE NO. DPR-65

DOCKET NO. 50-336

Remove and 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. The corresponding overleaf pages are provided to maintain document completeness.

Remove	Insert
3/4 4-7a	3/4 4-7a
3/4 4-7b	3/4 4-7b
3/4 4-7f	3/4 4-7f
B 3/4 4-2a	B 3/4 4-2a

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.1.3 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 4.4-6 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 4.4.5.1.3.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 4.4-6 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.

*For the outage commencing on March 1, 1983 only, the inservice inspection requirements of Category C-1 of Table 4.4-6 are deferred to the 1983 refueling outage.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.1.4 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 inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections $\geq$20% of the nominal wall thickness caused be 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 or sleeve containing a defect is defective.

6. Plugging Limit means the imperfection depth at or beyond which the tube or sleeve shall be repaired because it may become unserviceable prior to the next inspection and is equal to 40% of the nominal tube wall thickness for tubes.*

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 4.4.5.1.3.c, above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U - Bend to the top support of the cold leg.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or sleeve all tubes exceeding the plugging limit and plug all defective sleeves) required by Table 4.4-6.

*The plugging limit for sleeves will be determined prior to next refueling outage.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.1.5 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 included in the Annual Operating Report for the period in which this inspection was completed. 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 or sleeved.

c. Results of steam generator tube inspections which fall into Category C-3 and require prompt notification of the Commission shall be reported pursuant to Specification 6.9.1 prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

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TABLE 4.4-6

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	Repair defective tubes and inspect additional 25 tubes in this S.G.*	C–1	None	N/A	N/A
			C–2	Repair defective tubes and inspect additional 45 tubes in this S.G.*	C–1	None
					C–2	Repair 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., repair defective tubes and inspect 25 tubes in each other S.G.* Prompt notification to NRC pursuant to specification 6.9.1	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 repair defective tubes.* Prompt notification to NRC pursuant to specification 6.9.1.	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

*Repair of defective tubes shall be limited to plugging with the exception of those tubes which may be sleeved. Tubes with defective sleeves shall be plugged.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System leakage detection systems shall be OPERABLE:

a. A containment atmosphere particulate radioactivity monitoring system,

b. The containment sump level monitoring system, and

c. A containment atmosphere gaseous radioactivity monitoring system.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one of the above radioactivity monitoring leakage detection systems inoperable, operations may continue for up to 30 days provided:

1. The other two above required leakage detection systems are OPERABLE, and

2. Appropriate grab samples are obtained and analyzed at least once per 24 hours;

otherwise, be in COLD SHUTDOWN within the next 36 hours.

b. With the containment sump level monitoring system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. Containment atmosphere gaseous and particulate monitoring systems-performance of CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table 4.3-3, and

b. Containment sump level monitoring system-performance of CHANNEL CALIBRATION TEST at least once per 18 months.

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 gallon per minute 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.

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 tube examinations. Plugging or sleeving will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Sleeving repair will be limited to those steam generator tubes with a defect between the tube sheet and the first eggcrate support. Tubes containing sleeves with imperfections exceeding the plugging limit will be plugged. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.1 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 89 TO FACILITY OPERATING LICENSE NO. DPR-65

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

DOCKET NO. 50-336

1.0 Introduction

By letter dated June 3, 1983, Northeast Nuclear Energy Company (licensee) proposed revisions to the Millstone Unit 2 Technical Specifications (TS) which would allow for the repair of defective steam generator tubes by plugging or sleeving. Repair is required when a tube contains an imperfection which exceeds the plugging limit of 40% of the nominal wall thickness.

In addition to the attachments to the June 3, 1983 letter describing the sleeving program, a Westinghouse Electric Company Report, WCAP-10267 (Proprietary), "Millstone Unit 2, Steam Generator Sleeving Report," was also submitted by letter dated August 18, 1983. In response to our request for further information on eddy current inspection techniques, a letter dated November 16, 1983 was provided.

The tubes to be sleeved were selected by the licensee after a review of eddy current test data. Selection was based on the location of the tube in the tubesheet, tooling accessibility, the location and size of the eddy current indication and ALARA considerations.

2.0 Background

Millstone Unit No. 2 is a Combustion Engineering design, two loop pressurized water reactor rated at 2700 megawatts thermal. The two steam generators are of the vertical shell U-tube type with each rated at 5,603,000 lb/hr steam flow at 870 psig. Each steam generator contains 8519 heat transfer tubes. The tube material is Inconel 600 with dimensions of 0.750 inch O.D. and .048 inch nominal wall thickness. The tubes are fully expanded into the tubesheet and seal welded.

Millstone Unit No. 2 began commerical operation in December 1975. All volatile treatment (AVT) secondary water chemistry control has been used since initial operation. Full flow condensate polishing was introduced in November 1977.

Steam generator tube degradation experienced at Millstone has included tube support plate and eggcrate denting and tube pitting. Between November 1977 and December 1980, 361 tubes in steam generator #1 and 439 tubes in steam generator #2 were preventatively plugged because of denting. During the 1981/1982 refueling outage, eddy current examination of heat transfer tubing in both steam generators revealed indications of secondary side tube degradation in the hot and cold legs of both steam generators. Eddy current testing (ECT) characterized the degradation as discrete and small volume defects (confirmed later as pits) located within the tube bundles between the tubesheet secondary face and the lowest eggcrate support. Estimated depth of the indications varied from less than 20 percent through-wall to essentially through-wall.

The sleeves have been designed to span degraded regions of tubes in order to maintain them in service. Degradation due to pitting attack has occurred in both the hot and cold legs of the tube bundle with most located on the cold leg and confined to a region approximately one foot above the tubesheet.

[[redacted]] Sleeve fabrication techniques were verified, and material properties are in conformance with ASME SB-163 and ASME Code Case N379. Installation and inspection processes, parameters, and procedures were developed and tested. Laboratory testing included metallurgical evaluations, corrosion testing and mechanical testing of materials and fabricated tube/sleeve assemblies. Extended corrosion and additional verification tests are continuing for information purposes. Analytical work was performed to verify the structural integrity of the sleeve design and its effects upon the overall nuclear plant system in accordance with applicable ASME Boiler and Pressure Vessel Code and Nuclear Regulatory Commission regulations and guidelines.

A considerable amount of actual field experience in installing sleeves has been obtained on other installations. The process tooling, techniques and sequences are essentially the same as those sleeving programs utilized previously by Westinghouse Electric Company. Although the steam generator is of the Combustion Engineering design, the sleeving technology is fully applicable since the only significant difference is the exact tube dimensions.

At the San Onofre Unit 1, more than 6,400 degraded tubes (including leakers) were sleeved, tested and returned to service using remote installation equipment. After this project, the tooling was redesigned, to incorporate the field experience. The resultant remote sleeving system was adapted to a Westinghouse Model 44 series steam generator and utilized in the sleeving operations at Indian Point 3. Process modifications were employed during the installation of 13 demonstration sleeves in a hands-on mode at the Point Beach Unit 1. To date, more than 12,000 sleeves have been successfully installed utilizing both remote and hands-on tooling. This sleeving system was modified to perform under the field conditions of the Combustion Engineering steam generators at Millstone 2.

3.0 Sleeving Process Description

3.1 Sleeve Design

The sleeving process consisted of installing within the original steam generator tube, a smaller diameter tube to span the degraded section of the parent tube. The Westinghouse sleeve design employed is a [[redacted]] inch long, thin walled, [[redacted]] sleeve joined to the steam generator tube at both the sleeve's upper and lower ends. The sleeve's lower joint is located at the tube inlet end near the tubesheet's primary side surface. The [[redacted]] inch length of the sleeve will ensure the upper joint is located above the segment of each tube which is degraded due to pitting and that it is above the existing sludge pile. The tubesheet is 21.5 inches thick and the sludge pile and pit defect heights range from approximately 0 to 11 inches above the tubesheet as determined by ECT during the last refueling outage.

The sleeves are fabricated from thermally treated bimetallic tubing. [[redacted]] While [[redacted]] will provide greater pitting corrosion resistance in environments representative of Millstone's secondary side conditions, [[redacted]] will provide optimal stress corrosion resistance to primary water and secondary side (caustic) environments known to have resulted in tube degradation in other operating nuclear units' steam generators.

The upper and lower joints are designed to be the structural joints and thus also serve as the leak limiting seals. [[redacted]] The design criteria for these joints are (1) that they are structurally adequate to maintain the steam generator tubing primary-to-secondary pressure boundary under normal and accident conditions, (2) that they are sufficiently leak limiting such that total leakage between the primary and secondary for all sleeved tubes is less than plant technical specification limits during normal operation and postulated accidents, and (3) that they do not impair the pressure retaining capability of the steam generator tube.

Hydrostatic testing of pilot sleeve joints at pressures simulating normal operation and faulted conditions have shown acceptable leak-limiting characteristics of the sleeve assembly. Based on 1500 sleeves per steam generator, the Technical Specification primary to secondary maximum allowable leak rate of 0.5 gallons per minute per steam generator, is equivalent to an allowable leak rate per sleeve of 0.0002 gallons per minute. The maximum leak rate per sleeve under the accident conditions of steamline break and LOCA are [[redacted]] and [[redacted]], respectively, based on [[redacted]] sleeves.

The sleeve design, materials, and joints were designed to the ASME Boiler and Pressure Vessel Code, 1980 Edition including the 1980 Winter addenda. The sleeve design also met the requirements of the original Millstone Unit No. 2 steam generator design specification under normal and design basis accident conditions (Loss of Coolant Accident and Main Steamline Break).

3.2 Bimetallic Material Corrosion Testing

In addition to verifying the fabricability of the bimetallic sleeve, the effects of processing parameters on mechanical properties, microstructure, and corrosion performance were evaluated. [[redacted]] Metallurgical evaluations included hardness measurements, tensile tests, burst tests, free expansion tests, metallography and stress corrosion cracking testing.

[[redacted]]

Corrosion testing of the bimetallic materials included beaker, electrochemical, autoclave immersion, and heat transfer tests. Test environments representative of the Millstone Unit No. 2 secondary side sludge pile as well as aggressive pitting and stress corrosion environments were utilized in the test program. Beaker test results revealed no evidence of pitting [[redacted]] Beaker test results showed that tube strain had no effects upon pit initiation.

[[redacted]]

3.3 Sleeve Installation and Verification

The sleeve installation process consists of tube preparation, sleeve insertion, lower and upper joint fabrication, and joint and sleeve inspections. The sleeve installation process can be performed either automatically or manually. The process tooling, techniques, and sequences are essentially the same as those sleeving programs previously utilized by Westinghouse. The process and tooling differences that exist between Millstone and previous Westinghouse sleeving programs are the result of improvements in automatic tooling, sleeve design, and/or sleeve dimensions. The process differences are primarily with specific process parameters rather than techniques.

The following installation sequence was used to install the sleeves. [[redacted]]

[[redacted]]

In process sampling by eddy current testing or gauging of a percentage of the sleeves was performed to confirm that the equipment and tooling was performing satisfactorily. Undersized diameters were corrected by an additional expansion step to produce the correct diameter. If it was judged necessary to remove a sleeved tube from service, due to incorrect dimensions for example, the lower portion of the sleeve was machined to provide clearance for a plug to be inserted. A system pressure test in accordance with the ASME, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components" was conducted following sleeve installation.

4.0 Analytical Verification

Analytical structural evaluations of the Millstone Unit No. 2 sleeve and tube assembly were performed in accordance with applicable ASME Boiler and Pressure Vessel Code, Section III, 1980 Edition criteria. The assembly was modeled mathematically and evaluated with the Westinghouse computer analyses codes WECAN[1] and WECEVAL[4]. The analytical procedure was essentially identical to that previously completed for the sleeving effort at San Onofre 1, Point Beach and Indian Point 3.

Millstone steam generator design loads (including the faulted condition) and test loading conditions constituted the bases for the analyses. Analyses were performed for both the upper and lower joints (Figures 1 and 2). Both intact and severed steam generator tubes were considered.

Special considerations and additional analytical evaluations included the following.

(1) Allowable Sleeve Degradation per Regulatory Guide 1.121 Criteria
(2) Tube Vibrational Analysis
(3) Evaluation of effects due to prebowing of sleeves
(4) Effect of Tubesheet/Support Plate Interaction.

The sleeve/tube assembly consists of an upper joint, a lower joint and straight sections of the sleeve and tube between the two joints. Two finite element models were developed to represent the hybrid expansion joint (HEJ) of the upper joint and the lower joint areas. In developing the roll region of the model, the estimated bulging at the OD of the sleeve and tube were evaluated in combination with the nominal tube and sleeve wall thickness. However, the slight wall thinning was neglected for the stress levels computed in the roll transition regions. [[redacted]] The tubesheet is low alloy carbon steel, SA 508 [redacted] Inconel cladding on the primary face.

Finite element models were developed for evaluating the sleeve configuration shown in Figures 1 and 2 (Proprietary). Thermal analysis was performed to obtain the temperature distribution needed for the thermal stress evaluation. Since the thermal stress solutions were used for the fatigue calculations, the maximum range of the stress intensities during any of the loading conditions considered were calculated. In previous transient runs done on the tube sleeves, it was observed that the temperature responses closely followed the variations of the boundary temperature. Thus it was unnecessary to perform time history analyses.

The WECAN models were used to determine the stress levels in the tube/sleeve configuration including the rolled transition region for both the temperature and pressure loading conditions. At any given point or section of each model, the program WECEVAL determined the total stress distribution for a given loading condition and categorized that total distribution per the ASME Section NB requirements. The total stress of a given cross section through the thickness was categorized into membrane, linear bending and non-linear components. These categorized stresses were then compared with Subsection NB allowables. In addition, when supplied with a complete transient history at a given location in the model, the program WECEVAL calculated the total cumulative fatigue usage factor per Code Paragraph NB-3216.2. For the fatigue evaluation, the effect of local discontinuities was considered at each location. The ASME Code stress criteria which were satisfied are as follows:

1. Primary General Membrane Stress Intensity: Pm

   (a) Design Condition: $Pm \leq Sm$
   (b) Test Condition: $Pm \leq 0.9\ Sy$
   (c) Abnormal Conditions: Upset $Pm \leq Sm$
   Faulted $Pm \leq 0.7\ Su$ or $2.4\ Sm$ whichever is lower

   where Sm = stress intensity limit as defined in the ASME Section III Code,
   Sy = yield stress at temperature
   and Su = ultimate tensile stress at temperature.

2. Primary Local Membrane Stress Intensity:

   (a) Design Conditions: $P_L \leq 1.5\ Sm$
   (b) Test Conditions: $P_L \leq 1.35\ Sy$
   (c) Abnormal Conditions Upset: $P_L \leq 1.5\ Sm$
   Faulted $P_L \leq 1.5 \times 0.7\ Su$ or $1.5 \times 2.4\ Sm$
   whichever is lower

4.1 Results of the Analyses

(a) Primary pressure stresses:
The maximum primary pressure stresses are summarized in Tables 6.2-11 and 6.2-12 of Reference 5. All primary stresses for the sleeved tube assemblies are well within ASME allowable Code stresses. The minimum stress intensity margin relative to the Code allowable occurs in the sleeve.

(b) Range of the Primary and Secondary Stress Intensities

The maximum range stress intensity values for the sleeved tube assemblies are summarized in Tables 6.2-17a thru 6.2-17d of Reference 5. The requirements of ASME Code Paragraph NB 3222-2 are met at all locations and required no further consideration. For the four sleeved tube configurations analyzed, Table 6.2-17e of Reference 5 provides a summary of the maximum stress intensity range with its location in the component and the available margin to the allowable stress.

(c) Range of Total Stress Intensities

The fatigue analysis considered a design life of 35 years for the sleeved tube assemblies. A stress intensity factor of 5.0 was included for the hoop stress only. The results of the fatigue analysis are provided in Tables 6.2-18 through 6.2-21 of Reference 5. All of the cumulative usage factors are below the allowable value of 1.0 specified in the ASME Code. The tube fatigue was found to be negligible.

4.2 Allowable Sleeve Degradation

The licensee has provided an evaluation of the minimum sleeve wall thickness requirements to sustain normal and accident condition loads in accordance with guidelines of Regulatory Guide (R.G.) 1.121. In this evaluation, the surrounding tube is assumed to be severed between the upper and lower joints of the sleeve.

According to the R.G. 1.121 guidelines, a factor of safety (FS) of 3 is required against failure by bursting under the normal operating pressure differential. On Page 6.3-5 of Reference 5, the licensee has taken exception to this position and recommends that a FS of 2 be used against failure by bursting. However, the approved Technical Specification reflects the staff position requiring a FS of 3. This is acceptable.

The current Millstone Unit No. 2 Technical Specification contains a plugging limit of 40% which satisfies the FS=3 requirement stated earlier and is acceptable for tubes. The 40% plugging limit in the Technical Specifications is for degraded tubes, but is now being proposed by the licensee for degraded sleeves. Before this can be accepted, it is necessary to establish that a 40% degraded sleeve is equivalent in strength to a 40% degraded tube. The licensee has not yet established that the two have equivalent bending strengths. This, therefore, remains an open item. The plugging limit proposed by the licensee will be evaluated when the licensee submits the requested information.

A leak-before-break evaluation for the sleeve based on the leak rate and burst pressure test data obtained on 11/16 in. OD x 0.040 in. wall tubing with various amounts of uniform thinning simulated by machining the tube O.D. was performed. The maximum permissible leak rate is 0.5 gpm at an operating pressure differential of 1365 psi. It was determined for Millstone 2 that the maximum permissible crack length in the sleeve, making an assumption that the total allowable leak rate of 0.5 gpm emanates from a single crack, is less than 0.48 inches. The critical crack length for burst is compiled using the Westinghouse normalized test data. For the Millstone 2 sleeve, the burst crack length based on ΔP of 2285 psi (Main Steam Line Break, MSLB pressure) and the maximum properties from recent test data for the Millstone sleeve samples is 0.51 inch. Since the critical crack length for burst is greater than the permissible crack size for leak, the largest permissible crack will not burst during a MSLB condition.

4.3 Vibrational Effects of Sleeves

The effects on the vibrational response of the tubes due to the installation of sleeves have been examined. Analyses to determine the effects of cross flow, parallel flow, excitation due to coolant pump imbalance and acoustic pressure pulses due to pump operation have been performed. Cross flow and parallel flow vibratory stress effects have been determined to be insignificant for sleeved tubes at Millstone Unit No. 2 due to their location between the tubesheet and first eggcrate.

The installation of sleeves adds stiffness and mass to the tube. Therefore, the tube/sleeve frequencies would be greater than the frequencies of the unsleeved tubes due to mechanical excitation and acoustic pressure pulses. Any increase in tube/sleeve frequencies from those of the tubes would reduce dynamic stresses. Therefore, sleeving would not be detrimental to the vibratory stresses and fatigue life.

4.4 Effects of Pre-Bowing of Sleeves

A slight bow to the sleeves was introduced to facilitate automatic installation. This bow is likely to introduce a small initial residual stress to the sleeve which will have no significant impact on primary stress intensity, primary plus secondary stress intensity range for racheting, or total stress intensity range for fatigue evaluation.

4.5 Effects of Tubesheet/Support Plate Interaction

Since the pressure is normally higher on the primary side of the tubesheet than on the secondary side, the tubesheet becomes concave upward. Under this condition, the tubes protruding from the top of the tubesheet will rotate from the vertical. This rotation depends on the boundary condition for the edges of the tubesheet.

In assessing the case for the Millstone 2 tubesheet, data was taken from the Westinghouse finite element analysis of a similar geometry using a simple axisymmetric model. [[redacted]]. These stresses are not considered large enough to affect the fatigue usage factors.

5.0 Mechanical Testing

As part of the design verification program, the licensee has conducted a full range of qualification testing which included the following.

(a) Leakage resistance testing
(b) Fatigue testing
(c) Thermal cycle testing
(d) Push-out and pull-out strength tests
(d) LOCA and steam line break simulation tests.

Specific test parameters and conditions, e.g., temperature, pressure and loadings, simulated Millstone Unit 2 operating and design conditions.

The design verification program was developed to ensure acceptable performance/leak resistance and structural integrity of the sleeved tube under normal operation, accident, and transient conditions.

Test samples included in the test program were upper joint, lower joint, and fixed-fixed mock-up samples. The fixed-fixed mock-up sample is a partial simulation of the steam generator that allows the testing of both the upper and lower joints in one test sample.

5.1 Test Specimens

[redacted]

[redacted]

[redacted]

5.2 Test Description and Acceptance Criteria

Leak resistance tests

[redacted]

Thermal cycling test

The thermal cycling test simulated the transient temperature conditions that the joint is likely to experience during the plant life. [redacted] Any leakage through the joint was measured in terms of drops of water per minute at room temperature at intervals during the test. Leakage could also be detected during the test by measurement of condensed water vapor.

Pull-out and Push-out Strength Tests

The purpose of these tests was to determine the strength of the upper and lower sleeves joint in tension and compression. A tensile strength greater than [redacted] and a compressive strength greater than [redacted] was considered to be acceptable.

[redacted]

[redacted]

Fatigue Tests

[redacted]

5.3 Test Results

Leak rate test results have been provided for specimens made with sleeves. [redacted] Comments and explanations relative to the leak rates have also been provided.

The thermal cycling tests were successful in all cases [redacted]

[redacted]]

[[redacted]]

[[redacted]] The acceptance criteria stated earlier, for leak resistance, thermal cycling, fatigue and axial load (push-out, pull-out) tests envelope the potential plant loading conditions and are, therefore, acceptable. Compliance with these acceptance criteria demonstrate the structural and leak tight integrity of the tube sleeve design.

6.0 Leak Rate Determination

In our Safety Evalaution dated March 16, 1983 relating to a Technical Specification change regarding steam generators inspections, we noted that the licensee had committed to certain actions which would result in a more accurate determination of the absolute primary to secondary leakage rate. We consider this item to be a confirmatory item and it should be implemented by the licensee within 120 days after returning to power operation.

7.0 Eddy Current Inspection

After sleeve installation, all sleeved tubes were subject to a series of eddy current inspections, some of these inspections were intended as a process control procedure to verify correct installation. Each tube/sleeve assembly also received a baseline eddy current inspection to which all subsequent inservice inspections will be compared.

Eddy current inspections will be periodically carried out on the steam generator tubes in accordance with the technical specifications. The purpose of the inspection is to detect tube degradation that may have occurred during plant operation so that corrective action can be taken to minimize further degradation and to reduce the likelihood of primary-to-secondary leakage.

In the unsleeved portion of the parent tube, conventional bobbin coil inspection techniques were used by the licensee. However, since the diameter of the sleeve is smaller than that of the tube, the fill factor of a probe inserted through the sleeve can result in a decreased capability for detecting tubing degradation. Thus, it was necessary to inspect the unsleeved portion of the tube above the sleeve by inserting a standard size probe over the U-bend from the unsleeved leg of the tube.

The standard inspection procedure involved the use of two circumferentially wound bobbin coils connected in the differential mode and excited in the multi-frequency mode. For the straight length regions of the sleeve/tube assembly, the inspection of the sleeve and tube was consistent with normal tubing inspections. In the regions where the tube/sleeve assembly joint occurs, the detection and sizing of degradation in the vicinity of geometric discontinuities of the sleeved assembly is affected by interference from the geometry. For those regions of the sleeve where the assembly has no geometric transitions, the conventional bobbin coil probe provides acceptable detection and sizing capability. For the regions of geometric transitions, the licensee has chosen a cross-wound coil configuration which significantly reduces the noise signal from the transitions.

The overall inspection procedure involved the use of the cross-wound probe, which significantly reduces the interference of the transitions, coupled with a multi-frequency technique for further reduction of the remaining interference signals. This system reduced the interference from all discontinuities which have 360-degree symmetry, providing improved visibility for discrete discontinuities. [[redacted]]

Another difficult region of the assembly to inspect is the region at the end of the sleeve. Here, for the conventional bobbin inspection, the response from the transition regions is still larger than that of the expansion regions. Thus, the signal-to-noise ratios for this part of the tube/sleeve assembly is about a factor of four less sensitive than that of the expansion regions. [[redacted]] The cross-wound coil also significantly reduces the noise response of the sleeve end.

In general, lower frequencies tend to suppress signals from transition regions relative to signals from degradation at the expense of the ability to quantify the size of the defect. Similarly, the inspection of the tube through the sleeve requires the use of low frequencies to achieve detection with an associated loss in quantification. Therefore, a compromise between detection and quantification must be made.

The licensee has made a commitment that his eddy current testing techniques will incorporate the most recent state-of-the-art technology for inspection of the sleeved assembly and that as improved techniques are developed they will be utilized. Therefore, we find the inspection plans to be acceptable.

8.0 ALARA Considerations

The Northeast Nuclear Energy Company (NNECO) has taken into account ALARA considerations for each of the activities to be involved in the proposed steam generator sleeving program at Millstone Unit No. 2. ALARA activities specifically directed to reduction of occupational radiation doses include decontamination of steam generator channel heads; special shielding to reduce exposure to personnel during channel head and tube sheet operations; a control ventilation system for the channel heads and other surrounding work areas; special remote and semi-remote tools designed for high radiation areas; remote control of the sleeving process; TV and audio surveillances of all platform and channel head operation; and personnel training in full size mock-ups. NNECO has verified that the training program is in accordance with Regulatory Guide 8.27, 8.29, and 8.13 or equivalent. In addition, NNECO and its sleeving contractor, Westinghouse make extensive use of classroom and mock-up training for individuals who perform the sleeving operation. All personnel assigned to the project have received operational experience from recent sleeving operations at Indian Point Unit 3 and were trained at the Westinghouse Training Facility.

Administrative control of personnel exposures is effected by planning of maintenance procedures for the job, in order to minimize the number of personnel used to perform the various tasks involving relatively high doses and dose rates. An internal channel head platform covered by a layer of lead blankets is installed to reduce time in the channel head and personnel exposures. The lead blankets provide shielding from some of the radiation coming out of the channel head bowl, and its cushioning effect provides better traction for workers inside the channel head. Temporary shielding is also used to reduce the general area background radiation at work stations inside containment such as adjacent to the non-regenerative heat exchanger. T.V. surveillance of personnel during tasks is used to identify areas and activities involving high exposures and to initiate suitable dose reducing actions.

NNECO has described provisions for special local ventilation associated with the steam generator sleeving program. Ventilation through the steam generator channel heads are provided by a portable ventilation rig equipped with HEPA and pre-filters. Each steam generator is ventilated through the hot leg manway for cold side work. This maintains a negative pressure in the working manway to prevent airborne radioactivity on the steam generator platform. Each steam generator is ventilated by providing suction and supply via the secondary side manways and flexible ducting. A decontamination tent is used for cleaning tools and materials removed from the steam generator.

The major source of the radiation dose rate inside the steam generator head is a tenacious layer of "oxide" which includes deposited activated corrosion products. In order to remove this deposited activity from the inside of the channel head and thereby reducing dose rates in this region, NNECO decontaminated the M2 steam generator channel heads prior to tube sleeving. NNECO used a tube honing precess. The honing of tubes removes the oxide film from tube surfaces in preparation for installing sleeves and provides decontamination in addition to channel head surface cleaning. NNECO has made use of experience gained in prior channel head decontamination in planning for the proposed tube sleeving activities. Data were available for Point Beach (Unit 1), San Onofre (Unit 1), Turkey Point (Unit 3) and Indian Point (Unit 3). In particular, NNECO considered information on mechanisms used in prior decontaminations, and has provided information relevant to projected occupational radiation exposures.

In a letter dated August 18, 1983, NNECO provided information that the occupational exposure incurred to date for the steam generator channel head decontamination was 103.17 person-rems. NNECO estimated that 19.4 person-rem will be incurred during follow up work directly associated with the decontamination project. NNECO had estimated 122.4 person-rems for the decontamination. Based on field experience from sleeving projects at other plants, NNECO has estimated an average dose of 164.7 person-rems and 185 person-rems for sleeving each of the steam generators. The total collective dose will be 350 person-rems for the M2 sleeving project. This collective dose will include all occupational doses resulting from the sleeving operation including all site and contractor support personnel. NNECO estimated that 90% of the dose is received by technicians (platform and channel head workers).

A breakdown of each task by estimated dose rates, person-hours and person-rems has been provided.

Based on our review of the Millstone Report, we conclude that the projected activities and estimated person-rem doses for this project appear reasonable. NNECO intends to take ALARA considerations into account, and to implement reasonable dose-reducing activities. We conclude that NNECO will be able to maintain individual occupational radiation exposures within the applicable limits of 10 CFR Part 20, and maintain doses ALARA, consistent with the guidelines of Regulatory Guide 8.8. Therefore, the proposed radiation protection aspect of the sleeving program is acceptable.

9.0 Conclusions

As a result of our review of the analytical structural evaluations performed by the licensee, the staff concludes:

(1) All primary stresses for the sleeved tube assemblies are well within allowable ASME Code stresses.

(2) The maximum range of stress intensities complies with the requirements of the ASME Code, Paragraph NB-3222.2, at all sleeved tube assembly locations.

(3) Fatigue analysis results confirmed that all the cumulative usage factors are below the allowable value of 1.0 specified in the ASME Code.

(4) The licensee has proposed that the Technical Specification plugging limit of 40% be applied to degraded sleeves. Since this plugging limit was intended for degraded tubes, it is necessary to establish that a 40% degraded sleeve is equivalent in strength to a 40% degraded tube. The licensee has not yet established that the two have equivalent bending strengths. This, therefore, remains an open item, and was not approved.

(5) Cross flow and parallel flow vibratory stress effects are insignificant for sleeved tubes at Millstone Unit 2 due to their location between the tubesheet and first eggcrate. Sleeving is not considered detrimental to the vibratory stresses and fatigue life.

(6) Thermal cycling and fatigue testing of both upper and lower joints had no adverse effect upon the structural integrity or leak resistance of the joints. Push-out and pull-out strengths of both the upper and lower joints are greater than potential plant loading conditions and therefore the joint strength of mechanical sleeves will ensure tube integrity.

On the basis of the above evaluation, the staff concludes that the analytical verification and mechanical testing portions of the steam generator tube sleeving program proposed by the licensee is acceptable. The issue of allowable tube degradation will be resolved by the staff prior to the time that Regulatory Guide 1.121 will have to be implemented on Millstone 2.

Further, we find that the sleeving repairs can be accomplished to produce a sleeved tube of acceptable integrity with respect to metallurgical properties, corrosion resistance, leak tightness and inservice inspectability. We also find that the licensee's commitment to use state-of-the-art inspection methods and to utilize improved techniques as they are developed, in combination with stringent allowable leak rate requirements, will assure the continued integrity of the steam generator tubes. In addition, the licensee's commitments made during the preceding outage to certain actions to provide a more accurate method of determination of leakage rates is a confirmatory item and should be implemented within 120 days after returing to power operation.

10.0 Technical Specification Changes

The Millstone Unit 2 Technical Specifications have been changed as follows:

Section 4.4.5.1.4.a - The definition of defect has been extended to include a defect in a sleeve. The definition of plugging limit has not been extended to include sleeves at this time.

Section 4.4.5.1.4.b - The definition of OPERABLE has been modified to allow sleeving as an acceptable repair.

Section 4.4.5.1.5 - The reporting requirements have been modified to include sleeves.

Table 4.4.6 - The table has been modified to reflect sleeving as an acceptable repair.

These changes to the Technical Specifications reflect the use of sleeving as an acceptable repair technique and are, therefore, acceptable in accordance with the findings of Section 9.0.

11.0 Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

12.0 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.

Date: December 30, 1983

Principal Contributors:
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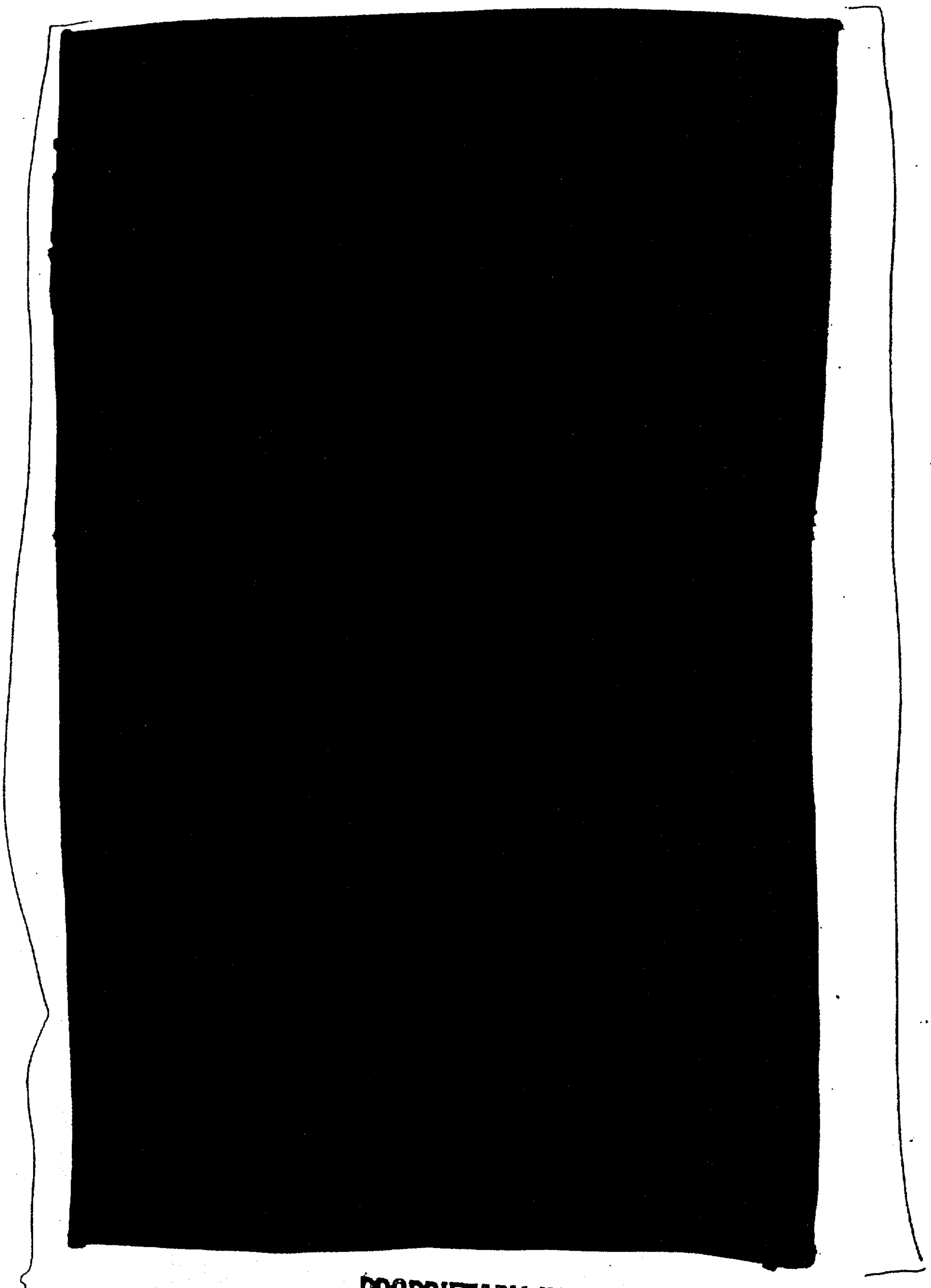
[redacted]

FIGURE 1

PROPRIETARY INFORMATION



PROPRIETARY INFORMATION

FIGURE 2