July 28, 1992

Docket No. 50-346

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Mr. Donald C. Shelton
Vice President, Nuclear - Davis-Besse
Centerior Service Company
c/o Toledo Edison Company
300 Madison Avenue
Toledo, Ohio 43652

Dear Mr. Shelton:

SUBJECT: AMENDMENT NO. 171 TO FACILITY OPERATING LICENSE NO. NPF-3 (TAC NO. M81298)

The Commission has issued Amendment No.171 to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station, Unit No. 1. The amendment revises the Technical Specifications in response to your application dated August 16, 1991, supplemented February 3, 1992.

This amendment revises Technical Specification 3/4.4.5, "Steam Generators", and its bases to allow use of a Babcock & Wilcox steam generator tube sleeving process to effect repairs of defective steam generator tubes.

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

Sincerely,

Original Signed By:
J. B. Hopkins

Jon B. Hopkins, Sr. Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 171 to License No. NPF-3
2. Safety Evaluation

cc w/enclosures:
See next page

change noted on clipped page

LA/PD3-3/DRPW	PM/PD3-3/DRPW	D/PD3-3/DRPW	OGC-WF1
PKreutzer	JHopkins/sw	JHannon	J. Hull ~~DON~~ of SE
7/7/92	7/10/92 7-28-92	7/29/92	7/16/92

JBH

changes made. JBH 7-28-92

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

July 28, 1992

Docket No. 50-346

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Vice President, Nuclear - Davis-Besse
Centerior Service Company
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1. Amendment No. 171 to
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2. Safety Evaluationn

cc w/enclosures:
See next page

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Davis-Besse Nuclear Power Station
Unit No. 1

cc:

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

TOLEDO EDISON COMPANY

CENTERIOR SERVICE COMPANY

AND

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

DOCKET NO. 50-346

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.171
License No. NPF-3

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

   A. The application for amendment by the Toledo Edison Company, Centerior Service Company, and the Cleveland Electric Illuminating Company (the licensees) dated August 16, 1991, supplemented February 3, 1992, 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. NPF-3 is hereby amended to read as follows:

(a) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 171, are hereby incorporated in the license. The Toledo Edison Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented not later than 90 days after issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Jon B. Hopkins

Jon B. Hopkins, Sr. Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of issuance: July 28, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 171

FACILITY OPERATING LICENSE NO. NPF-3

DOCKET NO. 50-346

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

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

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with:

a. A steam bubble,

b. A water level between 45 and 305 inches.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the pressurizer inoperable, be in at least HOT STANDBY with the control rod drive trip breakers open within 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.4 The pressurizer shall be demonstrated OPERABLE by verifying pressurizer level to be within limits at least once per 12 hours.

STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE with a water level between 18 and 348 inches.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one or more steam generators inoperable due to steam generator tube imperfections, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

b. With one or more steam generators inoperable due to the water level being outside the limits, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4-4.1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

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

1. All tubes or tube sleeves that previously had detectable wall penetrations (> 20%) that have not been plugged or repaired by sleeving in the affected area.

2. At least 50% of the tubes inspected shall be in those areas where experience has indicated potential problems.

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.

d. The provisions of Specification 4.0.2 are not applicable.

4.4.5.4 Acceptance Criteria

a. As used in this Specification:

1. Tubing or Tube means that portion of the tube or tube sleeve which forms the primary system to secondary system boundary.

2. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

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

4. Degraded Tube means a tube containing imperfections $\geq$ 20% of the nominal wall thickness caused by degradation that has not been repaired by sleeving in the affected area.

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

6. Defect means an imperfection of such severity that it exceeds the repair limit. A defective tube is a tube containing a defect that has not been repaired by sleeving in the affected area or a sleeved tube that has a defect in the sleeve.

7. Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving in the affected area because it may become unserviceable prior to the next inspection and is equal to 40% of the nominal tube wall thickness. The Babcock and Wilcox process described in Topical Report BAW-2120P will be used for sleeving.

8. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.5.3.c, above.

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

SURVEILLANCE REQUIREMENTS (Continued)

10. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or repair by sleeving in the affected areas all tubes exceeding the repair limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.5.5 Reports

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be submitted on an annual basis in a 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.

4.4.5.6 The steam generator shall be demonstrated OPERABLE by verifying steam generator level to be within limits at least once per 12 hours.

4.4.5.7 When steam generator tube inspection is performed as per Section 4.4.5.2, an additional but totally separate inspection shall be performed on special interest peripheral tubes in the vicinity of the secured internal auxiliary feedwater header. This testing shall only be required on the steam generator selected for inspection, and the test shall require inspection only between

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No			Yes		
No. of Steam Generators per Unit	Two	Three	Four	Two	Three	Four
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspections	One[1]			One[1]	One[2]	One[3]

Table Notation:

1. The inservice inspection may be limited to one steam generator on a rotating 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 circum stances the sample sequence shall be modified to inspect the most severe conditions.

2. The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.

3. Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION (2)

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 or repair by sleeving defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug or repair by sleeving defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug or repair by sleeving defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., plug or repair by sleeving defective tubes and inspect 2S 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 plug or repair by sleeving defective tubes. Prompt notification to NRC pursuant to specification 6.9.1	N/A	N/A

(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 4.4.5.2.b: No action is required for C-1 results. For C-2 results in one or both steam generators plug or repair by sleeving defective tubes. For C-3 results in one or both steam generators, plug or repair by sleeving defective tubes and provide prompt notification of NRC pursuant to Specification 6.6.

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and pilot operated relief valve against water relief.

The low level limit is based on providing enough water volume to prevent a reactor coolant system low pressure condition that would actuate the Reactor Protection System or the Safety Feature Actuation System. The high level limit is based on providing enough steam volume to prevent a pressurizer high level as a result of any transient.

The pilot operated relief valve and steam bubble function to relieve RCS pressure during all design transients. Operation of the pilot operated relief valve minimizes the undesirable opening of the spring-loaded pressurizer code safety valves.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken. A process equivalent to the inspection method described in Topical Report BAW-2120P will be used for inservice inspection of steam generator tube sleeves. This inspection will provide ensurance of RCS integrity.

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 chemistry 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 = 1 GPM). 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 1 GPM can be detected by monitoring the secondary coolant. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged or repaired by sleeving in the affected areas.

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. As described in Topical Report BAW-2120P, degradation as small as 20% through wall can be detected in all areas of a tube sleeve except for the roll expanded areas and the sleeve end, where the limit of detectability is 40% through wall. Tubes with imperfections exceeding the repair limit of 40% of the nominal wall thickness will be plugged or repaired by sleeving the affected areas. Davis-Besse will evaluate, and as appropriate implement, better testing methods which are developed and validated for commercial use so as to enable detection of degradation as small as 20% through wall without exception. Until such time as 20% penetration can be detected in the roll expanded areas and the sleeve end, inspection results will be compared to those obtained during the baseline sleeved tube inspection.

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.

The steam generator water level limits are consistent with the initial assumptions in the FSAR.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to detect and monitor leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

Industry experience has shown that, while a limited amount of leakage is expected from the RCS, the UNIDENTIFIED LEAKAGE portion of this can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The total steam generator tube leakage limit of 1 GPM for all steam generators ensures that the dosage contribution from tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 1 GPM limit is consistent with the assumptions used in the analysis of these accidents.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The CONTROLLED LEAKAGE limit of 10 GPM restricts operation with a total RCS leakage from all RC pump seals in excess of 10 GPM.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 171 TO FACILITY OPERATING LICENSE NO. NPF-3

TOLEDO EDISON COMPANY

CENTERIOR SERVICE COMPANY

AND

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

DOCKET NO. 50-346

1.0 INTRODUCTION

By letter dated August 16, 1991, supplemented February 3, 1992, Toledo Edison Company requested a revision to the Technical Specifications for the Davis-Besse Nuclear Power Station. The proposed change would revise the surveillance requirements of Technical Specification 4.4.5, "Steam Generators," to permit the option of using the Babcock & Wilcox (B&W) mechanical sleeving process for Once-Through Steam Generator (OTSG) tube repair.

The requested Technical Specification change will allow the use of B&W mechanical sleeves for steam generator tube repair as an alternative to plugging degraded tubes. The change references B&W topical report BAW-2120P, "OTSG Mechanical Sleeve Qualification (Alloy 690)," in Technical Specification Section 4.4.5.4. The topical report was originally submitted to the NRC by a letter dated March 26, 1991. The staff approved the topical report as being suitable for referencing in a letter to James H. Taylor of B&W from James E. Richardson of NRR dated August 1, 1991.

2.0 EVALUATION

The B&W mechanical sleeving methodology consists of inserting a tube of smaller diameter (the sleeve) inside the defective original tube, bridging the defect, and forming a new primary-to-secondary pressure boundary. The sleeve is joined to the inside of the original tube wall by mechanically rolled expansion joints at the free-span end of the sleeve and at the tube sheet end and therefore it is of a leak limiting design. The B&W OTSG mechanical sleeve methodology was originally reviewed and approved for use in Arkansas Unit 1 in 1984. The methodology has been used for the installation of over 1400 sleeves in OTSGs. With the exception of the change in material from the previously used Alloy 600 to the superior thermally treated Alloy 690, the proposed sleeve design is identical to the original sleeve design.

B&W topical report BAW-2120P describes in detail the analytical methods used for design and qualification of the B&W sleeve. The topical report also contains the results of the sleeve design verification which included analysis and confirmatory testing to demonstrate the acceptability of the technique.

A sleeve installed in a steam generator tube (1) must maintain the structural integrity of the steam generator tube under normal operating and postulated accident conditions, and (2) must limit or prevent leakage if a through-wall crack in the steam generator tube should develop. B&W performed tests and analyses to demonstrate the capability of the sleeve to perform these functions under normal operating and postulated accident conditions. Design analyses were performed to verify that the sleeve conformed to Section III or XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) and Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes."

The structural adequacy of the sleeve (e.g., the minimum sleeve wall thickness) was evaluated in accordance with Section III for design-basis loads. A vibration analysis was performed to demonstrate the adequacy of the sleeved tube. Fatigue loadings used in qualifying the sleeve joints were established. For expanded attachments that depend on frictional forces for their strength, Section III of the ASME Code does not provide design rules and theoretical stress analysis is inadequate. In such cases, Appendix II to Section III of the ASME Code permits the use of experimental stress analysis to substantiate the critical or governing stresses. The adequacy of the sleeve attachment to withstand cyclic loadings was demonstrated by means of a fatigue test. The fatigue analysis considered bounding values of temperatures and pressures associated with transients, and calculations included load ranges based on conservative assumptions. On the basis of the results of the fatigue tests and analyses, B&W found that the fatigue strength of the sleeve was adequate.

The staff's evaluation of the B&W sleeving process was based on the following considerations: The sleeving of steam generator tubes is a repair technique that is an alternative to removing defective or degraded tubes from service by plugging. Sleeves are designed to span a defective or degraded region of a steam generator tube and to maintain the steam generator tubing primary-to-secondary pressure boundary under normal and accident conditions. A successful sleeving system must provide a corrosion-resistant sleeve material and restore the structural integrity of the sleeved tube.

The tubesheet and free span joints are mechanical seals produced by roll expanding the sleeves into the tube. The structural integrity of the joints was demonstrated by subjecting sleeve/tube specimens to a series of tests representing service conditions which included leak tests, thermal cycling, fatigue, and tensile loading.

BAW-2120P contains the results of the sleeve design verification which included analysis and confirmatory testing to demonstrate the acceptability of the steam generator sleeving technique for defective tubes. The licensee has confirmed that the design and operating conditions (including transient conditions and cycles) specified for the sleeve in the topical report bound the Davis-Besse Nuclear Power Station steam generator design conditions. The topical report includes a technical description of the sleeve design; design verification, including analyses and tests; process qualification; sleeve installation procedures; and nondestructive examinations.

The B&W mechanical sleeve is manufactured from thermally treated Inconel Alloy 690. Worldwide corrosion studies reported in the open literature and work at the B&W laboratories indicate that the corrosion resistance in primary and secondary water of thermally treated Alloy 690 is superior to that of the Alloy 600 that was used for steam generator tubes. The higher chromium content of Alloy 690 is believed to be the major factor in regard to its enhanced corrosion resistance. Alloy 690 is a Code-approved material (ASME SB-163), covered by ASME Code Case N-20-3, for use as tubing in condensers and heat exchangers and is acceptable to the NRC staff under Regulatory Guide 1.85 (Rev. 28, April 1992.)

The NRC staff has approved the use of thermally treated Alloy 690 tubing in replacement steam generators. Corrosion test results presented in BAW-2120P confirm that the sleeving methodology used by B&W does not promote stress corrosion cracking in the sleeve/tube assemblies used in the tests. The staff concurs that thermally treated Alloy 690 used for sleeves is an improvement over the Alloy 600 used in the original steam generator tubing. A calculation to determine the minimum allowable wall thickness in accordance with Regulatory Guide 1.121 guidelines gave an allowable wall degradation limit of 70% of wall thickness, but a conservative plugging limit of 40% has been established based on staff positions concerning the need for an additional allowance for operational degradation and for eddy-current testing uncertainty.

Eddy current techniques are available to perform necessary sleeve/tube inspections for defect detection and to verify proper installation of the sleeve. As described in topical report BAW-2120P, degradation as small as 20% through wall can be detected in all areas of the tube sleeve except for the roll expansion and the sleeve end, where the limit of detectability is 40% through wall. This is the current state-of-the-art capability. The licensee has stated that it will evaluate and implement better testing methods when they are developed and validated for commercial use to enable detection of degradation as small as 20% through wall in the roll expanded area and the sleeve ends. Until better methods are available for inspection of these areas, the licensee will compare inservice inspection results with those obtained during the baseline sleeved tube inspection. This is acceptable to the NRC staff.

On the basis of its review of the analytical results, structural tests, and metallurgical evaluations provided by B&W for its sleeving methodology, the NRC staff has found that (1) the sleeve-to-tube joints have an acceptable leak resistance, (2) the structural strength of the sleeve and tube/sleeve joints under normal and accident conditions and the fatigue strength under transient loads are adequate, and (3) the use of thermally treated Alloy 690 should provide additional assurance against the possibility of stress corrosion cracking.

The above conclusions are based on the NRC staff's previous approval for referencing of the topical report BAW-2120P with associated supplemental information and commitments provided by the licensee. The NRC staff has also concluded that the design and operating conditions for the Davis-Besse steam generators are appropriately bounded by those discussed in the approved topical report. Therefore, the NRC staff has concluded that the issuance of this amendment is acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Ohio State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact has been prepared and published in the Federal Register on June 11, 1992 (57 FR 24832). Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

5.0 CONCLUSION

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

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