Distribution w/encls:

Docket File	GHill (2)
PUBLIC	WBeckner
PD3-3 Reading	ACRS
EAdensam	OGC
GGrant, RIII	RSavio
THarris (TLH3, SE only)	
AKeim	

April 14, 1998

Mr. John K. Wood
Vice President - Nuclear, Davis-Besse
Centerior Service Company
c/o Toledo Edison Company
Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, OH 43449-9760

SUBJECT: AMENDMENT NO. 220 TO FACILITY OPERATING LICENSE NO. NPF-3 - DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1 (TAC NO. MA1003)

Dear Mr. Wood:

The Commission has issued the enclosed Amendment No. 220 to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station, Unit No. 1. The amendment revises the Technical Specifications (TSs) in response to your application dated February 26, 1998, as supplemented March 20, 1998.

This amendment revises Technical Specification (TS) Section 3/4.4.5, "Reactor Coolant System - Steam Generators," TS Section 3/4.4.6.2, "Reactor Coolant System - Operational Leakage," and the associated bases to allow use of the "repair roll" steam generator tube repair process.

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:

Allen G. Hansen, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosures: 1. Amendment No. 220 to License No. NPF-3
2. Safety Evaluation

cc w/encls: See next page

DF01

DOCUMENT NAME: G:\DAVISBES\DBA1003.AMD

To receive a copy of this document, indicate in the box: "C" = Copy without enclosures "E" = Copy with enclosures "N" = No copy

OFFICE	PD33-LA E	PD33-PM E	OGC
NAME	EBarnhill	AHansen	
DATE	4/7/98	4/7/98	4/14/98

4/14/98

OFFICIAL RECORD COPY

9805180444 980414
PDR ADOCK 05000346
P PDR

NRC FILE CENTER COPY

Distribution w/encls:

Docket File	GHill (2)
PUBLIC	WBeckner
PD3-3 Reading	ACRS
EAdensam	OGC
GGrant, RIII	RSavio
THarris (TLH3, SE only)	
AKeim	

April 14, 1998

Mr. John K. Wood
Vice President - Nuclear, Davis-Besse
Centerior Service Company
c/o Toledo Edison Company
Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, OH 43449-9760

SUBJECT: AMENDMENT NO. 220 TO FACILITY OPERATING LICENSE NO. NPF-3 - DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1 (TAC NO. MA1003)

Dear Mr. Wood:

The Commission has issued the enclosed Amendment No. 220 to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station, Unit No. 1. The amendment revises the Technical Specifications (TSs) in response to your application dated February 26, 1998, as supplemented March 20, 1998.

This amendment revises Technical Specification (TS) Section 3/4.4.5, "Reactor Coolant System - Steam Generators," TS Section 3/4.4.6.2, "Reactor Coolant System - Operational Leakage," and the associated bases to allow use of the "repair roll" steam generator tube repair process.

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:

Allen G. Hansen, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosures: 1. Amendment No. 220 to License No. NPF-3
2. Safety Evaluation

cc w/encls: See next page

DOCUMENT NAME: G:\DAVISBES\DBA1003.AMD

To receive a copy of this document, indicate in the box: "C" = Copy without enclosures "E" = Copy with enclosures "N" = No copy

OFFICE	PD33-LA	E	PD33-PM	E	OGC [illegible]
NAME	EBarnhill		AHansen		[illegible]
DATE	4/7/98		4/7/98		4/14/98

4/14/98

OFFICIAL RECORD COPY

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

April 14, 1998

Mr. John K. Wood
Vice President - Nuclear, Davis-Besse
Centerior Service Company
c/o Toledo Edison Company
Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, OH 43449-9760

SUBJECT: AMENDMENT NO. 220 TO FACILITY OPERATING LICENSE NO. NPF-3 - DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1 (TAC NO. MA1003)

Dear Mr. Wood:

The Commission has issued the enclosed Amendment No. 220 to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station, Unit No. 1. The amendment revises the Technical Specifications (TSs) in response to your application dated February 26, 1998, as supplemented March 20, 1998.

This amendment revises Technical Specification (TS) Section 3/4.4.5, "Reactor Coolant System - Steam Generators," TS Section 3/4.4.6.2, "Reactor Coolant System - Operational Leakage," and the associated bases to allow use of the "repair roll" steam generator tube repair process.

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,

Allen G. Hansen, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosures: 1. Amendment No. 220 to License No. NPF-3
2. Safety Evaluation

cc w/encls: See next page

cc:

Mary E. O'Reilly
FirstEnergy
Davis-Besse Nuclear Power Station
5501 North State - Route 2
Oak Harbor, OH 43449-9760

James L. Freels
Manager - Regulatory Affairs
Toledo Edison Company
Davis-Besse Nuclear Power Station
5501 North State - Route 2
Oak Harbor, OH 43449-9760

Gerald Charnoff, Esq.
Shaw, Pittman, Potts
and Trowbridge
2300 N Street, NW.
Washington, DC 20037

Regional Administrator
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, IL 60523-4351

Robert B. Borsum
Babcock & Wilcox
Nuclear Power Generation Division
1700 Rockville Pike, Suite 525
Rockville, MD 20852

Resident Inspector
U.S. Nuclear Regulatory Commission
5503 North State Route 2
Oak Harbor, OH 43449

James H. Lash, Plant Manager
Toledo Edison Company
Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, OH 43449-9760

Robert E. Owen, Chief
Bureau of Radiological Health
Service
Ohio Department of Health
P.O. Box 118
Columbus, OH 43266-0118

James R. Williams, Chief of Staff
Ohio Emergency Management Agency
2855 West Dublin Granville Road
Columbus, OH 43235-2206

Donna Owens, Director
Ohio Department of Commerce
Division of Industrial Compliance
Bureau of Operations & Maintenance
6606 Tussing Road
P.O. Box 4009
Reynoldsburg, OH 43068-9009

Ohio Environmental Protection Agency
DERR--Compliance Unit
ATTN: Zack A. Clayton
P.O. Box 1049
Columbus, OH 43266-0149

State of Ohio
Public Utilities Commission
180 East Broad Street
Columbus, OH 43266-0573

Attorney General
Department of Attorney
30 East Broad Street
Columbus, OH 43216

President, Board of County
Commissioner of Ottawa County
Port Clinton, OH 43252

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

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. 220
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 February 26, 1998, as supplemented by letter dated March 20, 1998, 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:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 220 , 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 120 days after issuance.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Allen G. Hansen, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of issuance: April 14, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 220

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.

Remove	Insert
TS 3/4 4-6b	TS 3/4 4-6b
TS 3/4 4-8	TS 3/4 4-8
TS 3/4 4-9	TS 3/4 4-9
TS 3/4 4-9a	TS 3/4 4-9a
TS 3/4 4-10	TS 3/4 4-10
TS 3/4 4-10a	TS 3/4 4-10a
TS 3/4 4-12	TS 3/4 4-12
TS 3/4 4-15	TS 3/4 4-15
TS 3/4 4-16	TS 3/4 4-16
TS B 3/4 4-2	TS B 3/4 4-2
TS B 3/4 4-3	TS B 3/4 4-3
TS B 3/4 4-3a	TS B 3/4 4-3a
TS B 3/4 4-4	TS B 3/4 4-4

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 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 repair roll or sleeving in the affected area. (Tubes repaired by sleeving or repair roll remain available for random selection).

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

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 4.4.5.2.b, 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.

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

a. Inservice inspections shall be performed at intervals of not less than 12 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 4.4-2 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 4.4.5.3a 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 4.4-2 during the shutdown subsequent to any of the following conditions:

1. Primary-to-secondary tube leaks (not including leaks originating from tube-to tube sheet welds) in excess of the limits of Specification 3.4.6.2.

   If the leak is determined to be from a repair roll joint, rather than selecting a random sample, inspect 100% of the repair roll joints in the affected steam generator. If the results of this inspection fall into the C-3 category, perform additional inspections of the new roll areas in the unaffected steam generator.

*A group of tubes means:
(a) All tubes inspected pursuant to 4.4.5.2.b, or
(b) All tubes in a steam generator less those inspected pursuant to 4.4.5.2.b.

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 repair roll or 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 repair roll or 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 repair roll or 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.

SURVEILLANCE REQUIREMENTS (Continued)

(Continued) 7. The repair roll process will only be used to repair tubes with defects in the upper tubesheet area. The repair roll process will be performed only once per steam generator tube using a 1 inch reroll length. The new roll area must be free of degradation in order for the repair to be considered acceptable. The repair roll process used is described in the Topical Report BAW-2303P, Revision 3.

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. The previously existing tube and tube roll, above the new roll area in the upper tube sheet, can be excluded from future periodic inspection requirements because it is no longer part of the pressure boundary once the repair roll is installed.

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 repair roll or 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, sleeved or repair rolled.

c. Results of steam generator tube inspections which fall into Category C-3 and require notification of the Commission shall be reported prior to resumption of plant operation. 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

SURVEILLANCE REQUIREMENTS (Continued)

the upper tube sheet and the 15th tube support plate. The tubes selected for inspection shall represent the entire circumference of the steam generator and shall total at least 150 peripheral tubes.

4.4.5.8 Visual inspections of the secured internal auxiliary feedwater header, header to shroud attachment welds, and the external header thermal sleeves shall be performed on each steam generator through the auxiliary feedwater injection penetrations.

These inspections shall be performed during the third and fourth refueling outages and at the ten-year ISI.

4.4.5.9 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 tubes that have been repaired by the repair roll process. This inspection shall be performed on 100% of the tubes that have been repaired by the repair roll process. The inspection shall be limited to the repair roll joint and the roll transitions of the repair roll. Defective or degraded tubes found in the repair roll region as a result of the inspection need not be included in determining the Inspection Results Category for the general steam generator inspection.

TABLE 4.4-2
STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G. (1)	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug or repair by repair rolling or 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 repair rolling or sleeving defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug or repair by repair rolling or 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 repair rolling or sleeving defective tubes and inspect 2S tubes in each other S.G. Report to the NRC prior to resumption of plant operation.	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 repair rolling or sleeving defective tubes. Report to the NRC prior to resumption of plant operation.	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.

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 150 GPD primary-to-secondary leakage through the tubes of any one steam generator,

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. 10 GPM CONTROLLED LEAKAGE, and

f. 5 GPM leakage from any Reactor Coolant System Pressure Isolation Valve as specified in Table 3.4-2.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours except as permitted by paragraph c below.

c. In the event that integrity of any pressure isolation valve specified in Table 3.4-2 cannot be demonstrated, POWER OPERATION may continue, provided that at least two valves in each high pressure line having a non-functional valve are in and remain in, the mode corresponding to the isolated condition.(a)

d. The provisions of Section 3.0.4 are not applicable for entry into MODES 3 and 4 for the purpose of testing the isolation valves in Table 3.4-2.

---

[a]Motor operated valves shall be placed in the closed position and power supplies deenergized.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere gaseous or particulate radioactivity at least once per 12 hours.

b. Monitoring the containment sump level and flow indication at least once per 12 hours.

c. Measurement of the CONTROLLED LEAKAGE from the reactor coolant pump seals to the makeup system when the Reactor Coolant System pressure is 2185 ± 20 psig at least once per 31 days.

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours during steady state operation.

e. An evaluation of secondary water radiochemistry for determination of primary to secondary leakage through the steam generators at least once per 72 hours during steady state operations.

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-2 shall be individually demonstrated OPERABLE by verifying leakage testing (or the equivalent) to be within its limit prior to entering MODE 2:

a. After each refueling outage,

b. Whenever the plant has been in COLD SHUTDOWN for 7 days, or more, and if leakage testing has not been performed in the previous 9 months, and

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve.

d. The provisions of Specification 4.0.4 are not applicable for entry into MODES 3 or 4.

4.4.6.2.3 Whenever the integrity of a pressure isolation valve listed in Table 3.4-2 cannot be demonstrated, determine and record the integrity of the high pressure flowpath on a daily basis. Integrity shall be determined by performing either a leakage test of the remaining pressure isolation valve, or a combined leakage test of the remaining pressure isolation valve in a series with the closed motor operated containment isolation valve. In addition, record the position of the closed motor-operated containment isolation valve located in the high pressure piping on a daily basis.

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 = 150 GPD through any one steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 150 GPD 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 repair rolling or 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 repair rolling or 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.

An additional repair method for degraded steam generator tubes consists of rerolling the tubes in the upper tubesheet to create a new roll area and pressure boundary for the tube. The repair roll process will ensure that the area of degradation will not serve as a pressure boundary, thus permitting the tube to remain in service. The degraded area of the tube can be excluded from future periodic inspection requirements because it is no longer part of the pressure boundary once the repair roll is installed in the upper tubesheet.

All tubes which have been repaired using the repair roll process will have the new roll area inspected during the inservice inspection. Defective or degraded tube indications found in the new roll area as a result of the inspection of the repair roll and any indications found in the originally rolled region of the rerolled tube need not be included in determining the Inspection Results Category for the general steam generator inspection.

The repair roll process will be performed only once per steam generator tube using a 1 inch reroll length as described in the Topical Report BAW-2303P, Revision 3. Thus, multiple applications of the rerolling process to any individual tube is not acceptable. The new roll area must be free of degradation in order for the repair to be considered acceptable. After the new roll area is initially deemed acceptable, future degradation in the new roll area will be analyzed to determine if the tube is defective and needs to be removed from service. The rerolling process is described in the Topical Report BAW-2303P, Revision 3.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results shall be reported to the Commission 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 USAR. While in MODE 3, examples of Main Feedwater Pumps that are incapable of supplying feedwater to the Steam Generators are tripped pumps or a manual valve closed in the discharge flowpath. The reactivity requirements to ensure adequate SHUTDOWN MARGIN are provided in plant operating procedures.

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 recommendation 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 that 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The steam generator tube leakage limit of 150 GPD through any one steam generator ensures that the dosage contribution from tube leakage will be limited to a small fraction of 10 CFR Part 100 limits in the event of either a steam generator tube rupture or steam line break. A 1 GPM total primary to secondary leakage limit is 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.

The surveillance requirements for RCS Pressure Isolation Valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS Pressure Isolation Valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 220 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. INTRODUCTION

By letter dated February 26, 1998, as supplemented by letter dated March 20, 1998, the Toledo Edison Company, Centerior Service Company, and the Cleveland Electric Illuminating Company (the licensees) submitted a request for changes to the Davis-Besse Nuclear Power Station, Unit No. 1, Technical Specifications (TSs). The proposed amendment would revise TS Section 3/4.4.5, "Reactor Coolant System - Steam Generators," TS Section 3/4.4.6.2, "Reactor Coolant System - Operational Leakage," and the associated bases.

The purpose of the amendment is to implement alternate repair criteria in the TSs for steam generator tubes that have degraded roll joints inside of the upper tubesheet. The alternate repair criteria would allow new roll joints to be installed below the degraded roll joints in the upper tubesheet. The alternate repair criteria were based on a qualification program documented in Framatome (formerly Babcock and Wilcox) Topical Report, BAW-2303P (proprietary), "OTSG Repair Roll Qualification Report," Revision 3, which was a part of the submittal. This topical report was approved by the NRC staff in the safety evaluation dated November 21, 1997, for Oconee Nuclear Station, Units 1, 2 and 3.

DBNPS has two model 177FA Once-Through Steam Generators (OTSGs) that were manufactured by Babcock and Wilcox. The OTSG tubes were fabricated from Inconel Alloy 600 material and were restrained by roll expansion joints in the upper and lower tubesheets. The original tube-to-tubesheet rolls were expanded by a hard roll process and are about 1-2 inches in axial length extended into the tubesheet. The tubesheet is about 24 inches thick and a tube seal weld is provided at the primary face of the tubesheet to prevent leakage from the primary to secondary systems.

2. BACKGROUND

General Design Criterion (GDC) 14 of Appendix A to 10 CFR Part 50 requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. A significant portion of the reactor coolant pressure boundary is maintained by steam generator tubes which have experienced various levels of degradation. Regulatory Guide (RG) 1.121 provides guidance on an acceptable method for establishing the limiting safe conditions of tube degradation. In addition, the plant TSs require periodic inspections of steam generator tubes. TSs also require those tubes with defects in excess of the repair limits (for example, 40 percent through-wall) be repaired or removed from service.

The joint between the tube and tubesheet is an interference fit constructed by roll expanding the tube into the bore of the tubesheet, followed by a seal weld at the primary face of the tubesheet. The tube-to-tubesheet roll joint provides sufficient strength to maintain adequate structural and pressure boundary integrity.

The industry experience has shown that defects have developed in the tube-to-tubesheet roll joints as a result of various degradation processes. In general, tubes with degraded roll joints are either plugged or repaired by sleeving. However, the NRC has accepted alternate repair criteria allowing repaired tubes with degraded roll joints to remain in service provided that the repaired tubes can maintain adequate structural and leakage integrity under loadings from normal operation, anticipated operational occurrence, and postulated accident conditions.

RG 1.121 recommends that the margin of safety against tube rupture under normal operating conditions should be equal to or greater than 3 at any tube location where defects have been detected. For postulated accidents, RG 1.121 recommends that the margin of safety against tube rupture be consistent with the margin of safety determined by the stress limits specified in NB-3225 of Section III of the Boiler and Pressure Vessel Code of the American Society of Mechanical Engineers (ASME). Normal structural loads imposed on the tube-to-tubesheet roll primarily are derived from the differential pressure between the primary and secondary sides of the tubes. Loadings from a postulated main steam line break event can be significant. In addition, cyclic loading from transients (for example, startup or shutdown) should also be considered in the qualification of the roll joints.

3. EVALUATION

3.1 Qualification Program

On the basis of the qualification program, the licensees established that a 1-inch roll length for the new roll joints would carry all structural loads and minimize potential leakage. The qualification program consisted of (1) preparing the mockup, (2) establishing tube loads for the qualification tests, and (3) performing verification tests and analyses.

The mockup consisted of a perforated metal block inserted with eight steam generator tubes that simulates the tube-to-tubesheet configuration in the field. The tubes were expanded into the mockup tubesheet using an expanding tool that had the same critical dimensions as the tool used in the field.

To determine the strength of the roll joints, the licensees applied loads to the sample tubes to simulate or exceed normal, thermal and pressure cycling transient, and postulated accident conditions. In accordance with RG 1.121, the test pressure applied to the sample tubes exceeded 3 times normal operating pressure and 1.43 times main steam line break pressure. To obtain conservative leakage results, the sample tubes were severed 360 degrees through the tube wall in the roll joints.

In the qualification program, the licensees also considered the impact of tubesheet bowing on the roll joints because the tubesheet bore diameter can change during certain operating conditions. The combined effects of primary to secondary pressure differential and thermal loads may cause the tubesheet to bow in one direction or the other which can lead to tubesheet bore dilation or shrinkage. When the tubesheet bore is dilated, the contact stress between the roll joint and the tubesheet would decrease, reducing the pullout resistance of the roll joint. Considering the bowing effect, the Topical Report, BAW-2303, Revision 3, specified an exclusion zone in the tubesheet where the reroll joint would not be installed.

3.2 Structural and Leakage Integrity

Based on the results of the qualification testing, the licensees determined a roll length of 1 inch is necessary to ensure adequate margins of structural and leakage integrity. With regard to the structural integrity, the licensees demonstrated by their ultimate load testing (testing to simulate accident conditions) that the tube with the new roll would not be pulled out from the upper tubesheet under the worst possible combination of loadings. Also, no motion of the tubes relative to the simulated tubesheet were observed during the thermal and fatigue cycling tests.

With regard to the leakage integrity, the qualification tests showed that if each of the tubes (about 15,500 tubes) in a steam generator was rerolled in the upper tubesheet and had a 100 percent through-wall flaw in the reroll, the total leakage from all flaws would be minimal. As a defense-in-depth measure, the licensees proposed to implement a primary-to-secondary leakage limit of 150 gallons per day (gpd) per steam generator in the plant TSs. The 150 gpd requirement is more limiting and conservative than the current TS limit of 1440 gpd and is consistent with the staff's position regarding alternate tube repair criteria.

3.3 Field Installation and Inspection

The licensees propose to apply a single repair method to install one roll (reroll) in the tubes that have degradation in the original roll region. The repaired roll is typically installed using a manipulator and a tool head, monitored by a control system that tracks the position and monitors the torque of the roll expander. The roll expander is 1-inch long but the actual roll will have a 1/4-inch taper on each end. The torque is automatically controlled during the rerolling and is recalibrated after installation of a certain number of rerolls to ensure the minimum torque is maintained to produce proper fit.

After the installation, as provided in revised TS 4.4.5.4 and TS 4.4.5.9, the licensees will inspect all rerolls using eddy current techniques to ensure proper diametral expansion and that the reroll regions are free of degradation. Any reroll not satisfying the acceptance criteria will be either plugged or repaired with a method other than rerolling. For future inservice inspections, the licensees will inspect all rerolled tubes during steam generator inspection activities. If degradation is found in the reroll region, the affected tube will be plugged or repaired by means other than rerolling because only one reroll per tube is allowed by the proposed amendment (the licensees did not propose a method for multiple rerolls on the same tube in this amendment).

3.4 Proposed Technical Specification Changes

Surveillance Requirement (SR) 4.4.5.2.a will be revised by removing the parenthetical statement:

> (subsequent to the baseline inspection)

SR 4.4.5.3.a will be revised by removing the first sentence which states:

> The baseline inspection shall be performed to coincide with the first scheduled refueling outage but no later than April 30, 1980.

The above statements in SR 4.4.5.2.a and SR 4.4.5.3.a are no longer necessary because the baseline inspection has been performed and continued reference is no longer necessary. Since these changes are administrative they are acceptable.

SR 4.4.5.2.a.1. will be revised as shown:

> All tubes or tube sleeves that previously had detectable wall penetrations (>20%) that have not been plugged or repaired by repair roll or sleeving in the affected area. (Tubes repaired by sleeving or repair roll remain available for random selection.)

The proposed revision to this SR adds "repair roll" as a type of repaired tube that is not required to be included in the tubes selected for the first sample of a steam generator inservice inspection. This revision also adds the statement that tubes with a repair roll will still remain available for inspection on a random basis. The required inspection for repair roll repairs (repair roll region) is addressed in a proposed new SR 4.4.5.9. The new reference to "repair roll" is acceptable based on the evaluation presented below. The parenthetical statement on availability of repaired tubes for random selection is acceptable because it is conservative and provides a restriction consistent with current staff guidance.

The licensees propose to revise SR 4.4.5.3.c.1 to add the following paragraph:

> If the leak is determined to be from a repair roll joint, rather than selecting a random sample, inspect 100% of the repair roll joints in the affected steam generator. If the results of this inspection fall into the C-3 category, perform additional inspections of the new roll areas in the unaffected steam generator.

This proposed revision provides guidance for escalating the inspection scope when the inspection is required by SR 4.4.5.3.c.1 and repair roll joints are identified as the cause of the primary-to-secondary leak. Since this revision is consistent with current staff positions on additional SG inspections, it is acceptable.

The proposed revisions of SRs 4.4.5.4.a.4, 4.4.5.4.a.6, and 4.4.5.4.a.7 are to include the use of "repair roll" as a repair method. These are administrative changes consistent with the addition of "repair roll" as an acceptable technique, and are therefore acceptable.

The following paragraph is proposed to be added to SR 4.4.5.4.a.7:

> The repair roll process will only be used to repair tubes with defects in the upper tubesheet area. The repair roll process will be performed only once per steam generator tube using a 1 inch [sic] reroll length. The new roll area must be free of degradation in order for the repair to be considered acceptable. The repair roll process used is described in the Topical Report BAW-2303P, Revision 3.

This proposed modification defines the applicable usage of the reroll repair process. The reroll repair is limited to the upper tubesheet and to one reroll repair per steam generator tube. The specification also provides a clear definition of length of the reroll and acceptance criteria of rerolled tubes. Additionally, this change identifies the previously NRC-approved Topical Report BAW-2303P, Revision 3, as the topical report that describes the repair roll process to be used. The staff finds this modification acceptable as the process described in the topical report will provide reasonable assurance of steam generator tube integrity.

SR 4.4.5.4.a.9 will be revised to add the following sentence:

> The previously existing tube and tube roll, above the new roll area in the upper tube sheet, can be excluded from future periodic inspection requirements because it is no longer part of the pressure boundary once the repair roll is installed.

This requirement clarifies the scope of tube inspection. Since the reroll regions are the new pressure boundary, the original roll areas need not be inspected. Therefore, the staff finds this specification acceptable.

SR 4.4.5.4.b will be revised as follows:

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

This proposed modification adds "repair roll" as a repair method to repair by plugging and sleeving when declaring the SGs operable. Since this safety evaluation finds acceptable the repair by rolling under the specified conditions, the above revision to SR 4.4.5.4.b is acceptable.

SR 4.4.5.5.b will be revised to include "repair roll" as a repair method, similar to plugging or sleeving, to be reported to the NRC. Since this safety evaluation finds acceptable the repair by rolling under the specified conditions, this revision is acceptable.

Proposed SR 4.4.5.9 will state the following:

> 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 tubes that have been repaired by the repair roll process. This inspection shall be performed on 100% of the tubes that have been repaired by the repair roll process. The inspection shall be limited to the repair roll joint and the roll transitions of the repair roll. Defective or degraded tubes found in the repair roll region as a result of the inspection need not be included in determining the Inspection Results Category for the general steam generator inspection.

This surveillance requirement clarifies the licensees' intent that all reroll regions of the repaired tubes will be inspected during each scheduled inservice SG inspection. The staff finds this requirement acceptable because it provides a comprehensive monitoring of potential degradation in the rerolled regions of the repaired tubes.

Table 4.4-2, "Steam Generator Tube Inspection," will be revised to include "repair rolling" as a repair method. Since this safety evaluation finds acceptable the repair by rolling under the specified conditions, this revision is acceptable.

Limiting Condition of Operation (LCO) 3.4.6.2.c, "Operational Leakage," will be revised to state:

> 150 GPD primary-to-secondary leakage through the tubes of any one steam generator,

This change reduces the present limit of 1440 gpd total primary-to-secondary leakage through the steam generators to 150 gpd primary-to-secondary leakage through the tubes of any one steam generator. The proposed leakage limit of 150 gpd for one steam generator is more limiting and conservative than the current limit. Therefore, this change is acceptable.

SR 4.4.6.2.1.e will be added to state:

> An evaluation of secondary water radiochemistry for determination of primary to secondary leakage through the steam generators at least once per 72 hours during steady state operations.

This new SR will reflect a more accurate method for detecting steam generator primary-to-secondary leakage in accordance with the more restrictive LCO limit. Therefore, it is acceptable.

Bases Section 3/4.4.5, "Steam Generators," will be revised to include references to the repair roll method of repair and to a leakage limitation of 150 gpd through any one steam generator, instead of 1 gallon per minute (1440 gpd) total leakage through the steam generators. The revision will provide reference to the repair roll method of repair and to the steam generator primary-to-secondary leakage limit of 150 gpd through any one steam generator. These revisions are consistent with the proposed TS changes, and are therefore acceptable.

Three paragraphs will be added to Bases section 3/4.4.5, to provide a description of the repair roll process. The descriptions are consistent with the TS changes and provide clarifying information, and are therefore acceptable.

Bases Section 3/4.4.6.2, "Operational Leakage," will be revised to include a reference to the new leakage limitation of 150 gpd through any one steam generator instead of 1 gallon per minute (1440 gpd) total primary-to-secondary leakage limit. Also, the phrase "consistent with the assumptions" used in the third paragraph will be removed. The revised paragraph will state that although the leakage limit is 150 gpd primary-to-secondary leakage through the tubes of any one steam generator, the accident analysis assumed 1 gallon per minute (1440 gpd). The revised sentence will read "A 1 GPM total primary to secondary leakage limit is used in the analysis of these accidents." These revisions are consistent with the approved changes above, and are therefore acceptable.

4. SUMMARY

The licensees have proposed to implement an alternate repair method using reroll to repair tubes having indications in the original roll regions of the upper tubesheet. The technical basis for the proposed reroll method is documented in Topical Report, BAW-2303P, Revision 3.

The staff has determined that (1) the licensees' alternate repair criteria using reroll were established on the basis of the qualification tests that used specimens simulating the actual tube-to-tubesheet joint configuration of the steam generators; (2) the loads for structural and leakage tests were specified and applied in accordance with RG 1.121; and (3) the proposed changes to plant TSs satisfied the regulatory requirements and technical basis.

On the basis of the submitted information, the staff concludes that the proposed reroll repair for degraded roll joints in the steam generators at Davis-Besse is acceptable because the licensees have demonstrated through an acceptable qualification program that the reroll satisfies GDC 14 of Appendix A to 10 CFR Part 50 and RG 1.121.

5. 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.

6. ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes a surveillance requirement. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (63 FR 11460). The supplemental information submitted by the licensees did not affect the proposed findings. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7. 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.

Principal Contributor: A. Keim

Date: April 14, 1998