December 7, 1988

Docket No. 50-346
Serial No. DB-88-008

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Mr. Donald C. Shelton
Vice President, Nuclear
Toledo Edison Company
Edison Plaza - Stop 712
300 Madison Avenue
Toledo, Ohio 43652

Dear Mr. Shelton:

SUBJECT: AMENDMENT NO. 125 TO FACILITY OPERATING LICENSE NO. NPF-3: STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM, TURBINE STOP VALVE CLOSURE TIME (TAC 65070)

The Commission has issued the enclosed Amendment No. 125 to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station, Unit No. 1. This amendment consists of changes to the Appendix A Technical Specifications (TS's) in response to your application dated March 23, 1987 (No. 1354).

This amendment revises the TS's relating to Steam and Feedwater Rupture Control System (SFRCS), Turbine Stop Valve (TSV) response times and the associated Bases Sections.

The amendment, specifically, revises Technical Specification 3.3.2.2, Table 3.3-13, Steam and Feedwater Rupture Control System Response Times, by decreasing the maximum overall closure time allowable for the Turbine Stop Valves (TSV's) from 6 seconds to 1 second. Additionally, the associated Bases Sections 3/4.3.1 and 3/4.3.2 are also revised. These changes were proposed by Toledo Edison Company in accordance with their commitment made in a letter dated June 6, 1987 (No. 1280).

The NRC Incident Investigation Team (IIT) investigating the circumstances surrounding the June 9, 1985, event (the event), concluded that "...neither the SFRCS nor the auxiliary feedwater system meet the single failure criterion with respect to (re)opening an AFW containment isolation valve to feed an intact steam generator." The Team's concern arose after a review of Updated Safety Analysis Report (USAR) Section 15.4.4 (Steam Line Break Analysis) which at that time showed that both SG's would depressurize below the 600 psig SFRCS trip setpoint. This trip would cause both AFW containment isolation valves (AF 599 and AF 608) to close. Following the SFRCS trip and subsequent repressurization of the intact SG above the trip setpoint, a single failure of the AFW containment isolation valve associated with the intact SG would block AFW to that SG.

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Following the event, Toledo Edison Company formed a Decay Heat Removal Task Force to evaluate and recommend improvements to the AFW system and to SFRCS. One of the short term recommendations of this task force was to modify SFRCS logic to prevent isolation of AFW flow to both SG's so that if low-pressure conditions were to be sensed at both SG's, only the first SG with low-pressure would be isolated. This logic modification was incorporated prior to restart of the facility in December 1986.

As part of the Course-of-Action document submitted by Toledo Edison Company following the event, a reanalysis of the MSLB accident was performed. This reanalysis used a 1-second closure time for the TSV's instead of the 6-second closure time used in the then existing USAR analysis. The reanalysis shows that for a TSV closure time of 1 second, the intact SG would remain above the 600 psig SFRCS trip.

The single failure issue and the reanalysis of the MSLB accident were evaluated by the staff and are discussed in the Safety Evaluation Report related to restart, NUREG-1177 (cf. Section 3.3.1.1). In NUREG-1177, the staff requested that Toledo Edison Company propose to change the TS requirement for TSV closure response time from not greater than 6 seconds to not greater than 1 second to be consistent with the MSLB reanalysis.

Toledo Edison Company has asserted in its application that based on past testing, a 1 second response time for the TSV can be met without any modifications. The overall response time is the sum of the response times for the following:

a) Low pressure sensing device	< 0.050 sec
b) SFRCS cabinet logic (Approximate)	0.035
c) TSV closure (Longest measured value)	0.576

Thus, the overall response time for TSV closure when low pressure is detected in the main steam lines is less than 1 second. The associated Bases Sections 3/4.3.1 and 3/4.3.2 are also changed to reflect the factors comprising the overall response time. However, to ensure that the closure requirement will be correctly interpreted, a footnote applicable to the TSV response time was added to Table 3.3-13 after discussion with Toledo Edison Company representatives. This footnote clarifies that the response time of the TSV's is to be the time elapsed from the main steam line low pressure trip until the TSV's are closed fully. The response time is not applicable for any other SFRCS trip which closes the TSV's. Toledo Edison has confirmed that the MSLB accident is the only event in which credit for TSV closure is taken in the USAR.

Based on 1) the above considerations, 2) the prior evaluation by the staff of the reanalysis of the MSLB accident, and 3) considering that the proposed TS modification is more stringent than the current requirement and, therefore, should ensure that any TSV degradation will be detected and corrected, the staff finds the proposed change acceptable.

This amendment involves changes to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or a change to 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 effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration, and there has been no public comment on such finding.

Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement nor environmental assessment need be prepared in connection with the issuance of this amendment.

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, 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.

Notice of issuance will be included in the Commission's next biweekly Federal Register notice.

On December 21, 1987, the NRC issued Inspection Report 50-346/87026(DRP). Item 2 i. of that report indicates that Violation (346/85030-02E (DRP) will remain open pending resolution of two issues, one of which is the issuance of this TS change.

Sincerely,

/s/

Albert W. De Agazio, Sr. Project Manager
Project Directorate III-3
Division of Reactor Projects - III, IV,
V & Special Projects

Enclosures:
Amendment No.125 to
License No. NPF-3

cc: See next page

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Mr. Donald C. Shelton
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Davis-Besse Nuclear Power Station
Unit No. 1

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

TOLEDO EDISON 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. 125
License No. NPF-3

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

   A. The application for amendment by the Toledo Edison Company and The Cleveland Electric Illuminating Company (the licensees) dated March 23, 1987 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:

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(a) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 125 , 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 January 20, 1989.

FOR THE NUCLEAR REGULATORY COMMISSION

Timothy M. Colburn for

John N. Hannon, Director
Project Directorate III-3
Division of Reactor Projects - III, IV, V, & Special Projects

Attachment: Changes to the Technical Specifications

Date of Issuance: December 7, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 125

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 3-29	3/4 3-29
B 3/4 3-1	B 3/4 3-1
--	B 3/4 3-1a

TABLE 3.3-13

STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM RESPONSE TIMES

ACTUATED EQUIPMENT	RESPONSE TIME IN SECONDS
1. Auxiliary Feed Pump	≤ 40
2. Main Steam Isolation Valves*	
a. Main Steam Low Pressure Channels	≤ 6
b. Feedwater/Steam Generator High Differential Pressure Channels	≤ 6.5
3. Main Feedwater Valves	
a. Main Control	≤ 8
b. Startup Control	≤ 13
c. Stop Valve	≤ 16
4. Turbine Stop Valves**	≤ 1

* The response time is to be the time elapsed from the monitored variable exceeding the trip setpoint until the MSIV is fully closed.

** The response time is to be the time elapsed from the main steam line low pressure trip condition until the TSV is fully closed.

TABLE 4.3-11

STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM
INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Instrument Channel			
a. Steam Line Pressure - Low	S	R	M+
b. Steam Generator Level - Low	S	R	M
c. Steam Generator - Feedwater Differential Pressure - High	S	R	M
d. Reactor Coolant Pumps - Loss of	S	R	M
2. Manual Actuation	NA	NA	R

+The surveillance period for Steam Line Pressure-Low Instrument is extended to 2400 hours, September 16, 1982.

3/4.3.1 and 3/4.3.2 REACTOR PROTECTION SYSTEM AND SAFETY SYSTEM INSTRUMENTATION

The OPERABILITY of the RPS, SFAS and SFRCS instrumentation systems ensure that 1) the associated action and/or trip will be initiated when the parameter monitored by each channel or combination thereof exceeds its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available for RPS, SFAS and SFRCS purposes from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundance and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The measurement of response time at the specified frequencies provides assurance that the RPS, SFAS, and SFRCS action function associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

An SFRCS channel consists of 1) the sensing device(s), 2) associated logic and output relays (including Isolation of Main Feedwater Non Essential Valves and Turbine Trip), and 3) power sources.

The SFRCS response time for the turbine stop valve closure is based on the combined response times of main steam line low pressure sensors, logic cabinet delay for main steam line low pressure signals and closure time of the turbine stop valves. This SFRCS response time ensuresthat the auxiliary feedwater to the unaffected steam generator will not be isolated due to a SFRCS low pressure trip during a main steam line break accident.

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3/4.3.1 and 3/4.3.2 REACTOR PROTECTION SYSTEM AND SAFETY SYSTEM INSTRUMENTATION (Continued)

Safety-grade anticipatory reactor trip is initiated by a turbine trip (above 25 percent of RATED THERMAL POWER) or trip of both main feedwater pump turbines. This anticipatory trip will operate in advance of the reactor coolant system high pressure reactor trip to reduce the peak reactor coolant system pressure and thus reduce challenges to the power operated relief valve. This anticipatory reactor trip system was installed to satisfy Item II.K.2.10 of NUREG-0737.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

3/4.3.3.2 INCORE DETECTORS

The OPERABILITY of the incore detectors ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core. See Bases Figures 3-1 and 3-2 for examples of acceptable minimum incore detector arrangements.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event so that the response of those features important to safety may be evaluated. This capability is required to permit comparison of the measured response to that used in the design basis for the facility. This instrumentation is consistent with the recommendations of Regulatory Guide 1.12 "Instrumentation for Earthquakes," April 1974.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public. This instrumentation is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of