June 4, 1991

Docket No. 50-346

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

Dear Mr. Shelton:

SUBJECT: AMENDMENT NO. 157 TO FACILITY OPERATING LICENSE NO. NPF-3 (TAC NO. 79701)

The Commission has issued Amendment No. 157 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 November 30, 1990 as supplemented December 19, 1990.

This amendment revises the main steam low pressure block permit setpoint and the steam pressure setpoint where the block permit is automatically removed.

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,

/S/

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. 157 to License No. NPF-3
2. Safety Evaluation

cc: See next page

LA/PDIII-3/DRPW	PM/PDIII-3/DRPW	D/PDIII-3/DRPW	OGC-WF1
PKreutzer	JHopkins:rc	JHannon	R Bachmann
5/21/91	5/21/91 6-3-91	5/21/91	5/23/91

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Mr. Donald C. Shelton
Toledo Edison Company

Davis-Besse Nuclear Power Station
Unit No. 1

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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. 157
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 November 30, 1990 as supplemented December 19, 1990 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. 157, 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 45 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: June 4, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 157

FACILITY OPERATING LICENSE NO. NPF-3

DOCKET NO. 50-346

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

Remove	Insert
3/4 3-27	3/4 3-27

TABLE 3.3-11 (Continued)

TABLE NOTATION

* May be bypassed when steam pressure is below 750 psig. Bypass shall be automatically removed when the steam pressure exceeds 800 psig.

The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 16 - With the number of OPERABLE Channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required CHANNEL FUNCTIONAL TEST provided the inoperable section of the channel is placed in the tripped condition within 1 hour.

ACTION 17 - With the number of OPERABLE Channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

TABLE 3.3-12

STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNITS	TRIP SETPOINTS	ALLOWABLE VALUES
1. Steam Line Pressure - Low	$\geq$ 591.6 psig	$\geq$ 591.6 psig* $\geq$ 586.6 psig**
2. Steam Generator Level - Low(1)	$\geq$ 16.4"	$\geq$ 15.6"* $\geq$ 12.9"**
3. Steam Generator Feedwater Differential Pressure - High (2)	$\leq$ 197.6 psid	$\leq$ 197.6 psid* $\leq$ 199.6 psid**
4. Reactor Coolant Pumps - Loss of	High $\leq$ 1384.6 amps Low $\geq$ 106.5 amps	$\leq$ 1384.6 amps# $\geq$ 106.5 amps#

(1) Actual water level above the lower steam generator tubesheet.
(2) Where differential pressure is steam generator minus feedwater pressure.
*Allowable Value for CHANNEL FUNCTIONAL TEST
**Allowable Value for CHANNEL CALIBRATION
#Allowable Value for CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 157 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 November 30, 1990, as supplemented December 19, 1990, and May 15, 1991, Toledo Edison proposed an amendment to the Technical Specifications for the Davis-Besse Nuclear Power Station, Unit 1 (DBNPS). The proposed change involves Technical Specifications 3/4.3.2.2, "Instrumentation, Steam and Feedwater Rupture Control System Instrumentation." The supplemental letters simply provided a notarized original and notification that a copy of the amendment application has been provided to the State of Ohio, respectively.

The proposed changes would increase the Steam and Feedwater Rupture Control System (SFRCS) Main Steam (MS) low pressure block permit setpoint from 700 psig to 750 psig and increase the steam pressure setpoint where the block permit is automatically removed from 750 psig to 800 psig. These changes would increase the pressure margin between the SFRCS block permit and the SFRCS MS low pressure trip setpoint of 591.6 psig and minimize the possibility of an inadvertent MS low pressure trip occurring during plant cooldown and heatup operations. These changes would also increase the margin between the automatic block reset and the low pressure trip reset which ensures that the low pressure trips have cleared prior to the automatic reset of the SFRCS during plant startup operations.

The SFRCS is an automatic system designed to detect and mitigate the effects of major upsets in the MS and Main Feedwater (MFW) systems, including MS and MFW line ruptures, Steam Generator (SG) overfeed, and a loss of Reactor Coolant System (RCS) forced circulation cooling. The SFRCS detects these events through sensing and logic channels and mitigates their consequences by automatically positioning valves in the MS, FRW, and Auxiliary Feedwater (AFW) systems with appropriate actuation signals dependent upon the initiating event.

An MS low pressure trip of an Actuation Channel of SFRCS during plant operation would be indicative of a main steam line break (MSLB). The MS low trip instrumentation includes pressure switches for all four SFRCS logic channels on each MS header. An SFRCS Actuation Channel trip would cause the complete isolation of the SG connected to the MS header experiencing the trip signal, the realignment of the affected SG's AFW pump to the opposite SG, and the initiation of AFW to the unaffected SG. It would also close the main steam line isolation valve and selected main feedwater valves on the unaffected SG.

2.0 EVALUATION

By raising the block permit setpoint to 750 psig, the RCS temperature at which the MS low pressure instrumentation can be blocked is increased from approximately 506°F to 514°F. This represents an increase of 8°F in the range of RCS temperatures in Mode 3 where the MS low pressure instrumentation would be unavailable during cooldown operations. The RCS temperatures from 506°F to 514°F represent a range of transient plant operations in Mode 3 and do not represent temperatures in Mode 3 where the RCS would be stabilized for any long periods of time. Using a nominal cooldown rate of 15°F/hr., the raising of the block permit value to 750 psig would increase by approximately 30 minutes the period of time in Mode 3 where the MS low pressure instrumentation would be unavailable during the normal plant cooldown operation. The increased time period over the life of the plant is so short that the probability that an MSLB would occur in this short time period is extremely low.

By raising the automatic reset pressure of the block to 800 psig from 750 psig, the dead band associated with the resetting of the low pressure trip switches will not overlap the automatic reset of the block switches. Based on past surveillance tests of the block permit pressure switches, the switches typically reset within 20-30 psi above the block permit setpoint. Consequently, the 50 psi difference between the block permit setpoint and automatic reset point specified by Technical Specifications is considered to be an appropriate pressure range for the equipment in use.

By raising the automatic reset value to 800 psig, the RCS temperature at which the block automatically resets is increased from approximately 514° to approximately 520°F. This increases by 6°F the range of RCS temperature in Mode 3 where MS low pressure instrumentation would be unavailable during heatup operations. However, it still ensures that the automatic reset occurs before the plant enters Mode 2, since per TS 3.1.1.4, "Reactivity Control Systems, Minimum Temperature for Criticality," the plant is not allowed to go critical until RCS Tavg is greater than or equal to 525°F. Using a nominal heat-up rate of 15°F/hr., this temperature increase of 6°F would increase the time period during heatup where the MS low pressure instrumentation is blocked by approximately 30 minutes. As with the increased time period associated with plant cooldown operations, this time period over the life of the plant is so short that the probability of an MSLB during this 30 minutes is extremely low.

Since normal plant operation is in Modes 1 and 2 where the MS line pressure is typically 870 psig, the raising of the block permit pressure to 750 psig and the automatic reset to 800 psig will have no impact. Use of the block permit in Modes 1 and 2 is not possible due to the large difference in pressure between its setpoint and normal MS operating pressure. Consequently, the protection against MSLBs during power operation provided by the MS low pressure instrumentation is unaffected by the proposed change.

By raising the block permit value to 750 psig and raising the automatic reset setpoint to 800 psig, the time period is increased during cooldown and heatup where the MS low pressure instrumentation is blocked. This increased time period is approximately 30 minutes. As discussed above, this time period is so short that the probability of an MSLB during this interval is extremely low. From its review of the above information, the NRC staff finds that the proposed changes to Technical Specifications 3/4.3.2.2 are 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

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 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 the amendment involves no significant hazards consideration and there has been no public comment on such finding. 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.

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.

Principal Contributor: S. Rhow, ICSB

Date: June 4, 1991