July 25, 1984

Docket No. 50-346

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Mr. Richard P. Crouse
Vice President, Nuclear
Toledo Edison Company
Edison Plaza - Stop 712
300 Madison Avenue
Toledo, Ohio 43652

Dear Mr. Crouse:

SUBJECT: 1. AMENDMENT NO. 73 TO FACILITY OPERATING LICENSE NO. NPF-3, ANTICIPATORY REACTOR TRIP SYSTEM

2. SAFETY EVALUATION RELATED TO RESPONSE TO GENERIC LETTER 82-16

The Commission has issued Amendment No. to Facility Operating License NPF-3 for the Davis-Besse Nuclear Power Station, Unit No. 1. This amendment modifies the Appendix A Technical Specifications in response to item 2 of your application dated November 25, 1981 (No. 757) and as modified by your letter dated February 22, 1983 (No. 913).

This amendment revises the Technical Specifications to add new Sections 3.3.2.3 and 4.3.2.3 and new Tables 3.3-15 and 4.3-15 to reflect the addition of the final Anticipatory Reactor Trip System. The basis for the specifications has also been added.

The enclosed Safety Evaluation supporting this amendment also includes an evaluation of your response, dated November 15, 1982 (No. 873), to NRC Generic Letter 82-16 (September 20, 1982). Our evaluation indicates that certain items related to GL 82-16 are not satisfactorily resolved. Several of these are currently under review and resolution is in process, however, for several other items additional submittals by Toledo Edison Company will be required. The additional submittals were requested by our letter of February 8, 1984.

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DCR 016

The Notice of Issuance will be included in the Commission's Monthly Notice in the Federal Register.

Sincerely,

George W. Rivenbark, Acting Chief
Operating Reactors Branch No. 4
Division of Licensing

Enclosures:
1. Amendment No. 73
2. Safety Evaluation

cc w/enclosures:
See next page

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Toledo Edison Company

cc w/enclosure(s):

Mr. Donald H. Hauser, Esq.
The Cleveland Electric
Illuminating Company
P. O. Box 5000
Cleveland, Ohio 44101

Gerald Charnoff, Esq.
Shaw, Pittman, Potts
and Trowbridge
1800 M Street, N.W.
Washington, D. C. 20036

Paul M. Smart, Esq.
Fuller & Henry
300 Madison Avenue
P. O. Box 2088
Toledo, Ohio 43603

Mr. Robert B. Borsum
Babcock & Wilcox
Nuclear Power Generation Division
7910 Woodmont Avenue, Suite 220
Bethesda, Maryland 20814

President, Board of County
Commissioners of Ottawa County
Port Clinton, Ohio 43452

Attorney General
Department of Attorney General
30 East Broad Street
Columbus, Ohio 43215

Harold Kohn, Staff Scientist
Power Siting Commission
361 East Broad Street
Columbus, Ohio 43216

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
5503 N. State Route 2
Oak Harbor, Ohio 43449

Regional Radiation Representative
EPA Region V
230 South Dearborn Street
Chicago, Illinois 60604

Ohio Department of Health
ATTN: Radiological Health
Program Director
P. O. Box 118
Columbus, Ohio 43216

James W. Harris, Director (Addressee Only)
Division of Power Generation
Ohio Department of Industrial Relations
2323 West 5th Avenue
P. O. Box 825
Columbus, Ohio 43216

Mr. James G. Keppler, Regional Administrator
U. S. Nuclear Regulatory Commission, Region III
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Mr. Robert F. Peters
Manager, Nuclear Licensing
Toledo Edison Company
Edison Plaza
300 Madison Avenue
Toledo, Ohio 43652

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

THE 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. 73
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 November 25, 1981, as modified February 22, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (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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Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 73 , are hereby incorporated in the license. 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.

FOR THE NUCLEAR REGULATORY COMMISSION

George W. Rivenbark

George W. Rivenbark, Acting Chief
Operating Reactors Branch No. 4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 25, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 73

FACILITY OPERATING LICENSE NO. NPF-3

DOCKET NO. 50-346

Relace and add the following pages to the Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

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3/4 3-30b (new)
3/4 3-30c (new)
3/4 3-30d (new)
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ANTICIPATORY REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.3 The Anticipatory Reactor Trip System instrumentation channels of Table 3.3-15 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-15

ACTION: As shown in Table 3.3-15

SURVEILLANCE REQUIREMENTS

4.3.2.3 The Anticipatory Reactor Trip System shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST for the modes and at the frequencies shown in Table 4.3-15.

TABLE 3.3-15

ANTICIPATORY REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Turbine Trip	4	2[a]	3	1[b]	16
2. Trip of Both Main Feed Pump Turbines	4	2	3	1	17
3. Output Logic	4	2	3	1	18

(a) Trip automatically bypassed below 25 percent of RATED THERMAL POWER
(b) Applicable only above 25 percent of RATED THERMAL POWER

TABLE 3.3-15 (CONTINUED)

ACTION STATEMENTS

ACTION 16 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirements, restore the inoperable channel to OPERABLE status within 72 hours or reduce reactor power to less than 25 percent of RATED THERMAL POWER within the next 6 hours.

ACTION 17 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirements, restore the inoperable channel to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours.

ACTION 18 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and POWER OPERATION may proceed provided both of the following conditions are satisfied:

a) The control rod drive trip breaker associated with the inoperable channel is placed in the tripped condition within one hour.

b) The Minimum Channels OPERABLE requirement is met; however, one additional control rod drive trip breaker associated with another channel may be tripped for up to 2 hours for surveillance testing per Specification 4.3.2.3, after reclosing the control rod drive trip breaker opened in a) above.

TABLE 4.3-15

ANTICIPATORY REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE IS REQUIRED
1. Turbine Trip [a]	S	Not Applicable	M	1[b]
2. Main Feed Pump Turbine Trip	S	Not Applicable	M	1
3. Output Logic	Not Applicable	Not Applicable	M	1

(a) Trip automatically bypassed below 25 percent of RATED THERMAL POWER
(b) Applicable only above 25 percent of RATED THERMAL POWER

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 test 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.

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO.73 TO FACILITY OPERATING LICENSE NO. NPF-3, AND

THE TOLEDO EDISON COMPANY

AND

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

DOCKET NO. 50-346

I. Introduction

A. Background Information

Following the accident at Three Mile Island, Unit 2, the staff developed an NRC Action Plan, NUREG-0660, to provide a comprehensive and integrated plan to improve safety at power reactors. Specific NUREG-0660 items, approved by the Commission for implementation at power reactors, were issued as NUREG-0737. NUREG-0737 specified that new Technical Specifications (TSs) would be required for several of the items. Accordingly, on September 20, 1982, the NRC issued Generic Letter 82-16, requesting information on the following NUREG-0737 items:

1. STA Training (I.A.1.1.3).
2. Shift Manning - Overtime Limits (I.A.1.3.1).
3. Short Term Auxiliary Feedwater System (AFWS) Evaluation (II.E.1.1).
4. Safety Grade AFW Initiation and Flow Indication (II.E.1.2).
5. Dedicated Hydrogen Penetrations (II.E.4.1).
6. Containment Pressure Setpoint (II.E.4.2.5).
7. Containment Purge Valves (II.E.4.2.6).
8. Radiation Signal on Purge Valves (II.E.4.2.7).
9. Upgrade B&W AFWS (II.K.2.8).
10. B&W Safety-Grade Anticipatory Reactor Trip (II.K.2.10).
11. B&W Thermal-Mechanical Report (II.K.2.13).
12. Reporting Safety and Relief Valve Failures and Challenges (II.K.3.3).
13. Anticipatory Trip on Turbine Trip (II.K.3.12).

B. Licensee's Response to Generic Letter 82-16

By letter dated November 15, 1982, Toledo Edison Company (TED) responded to NRC Generic Letter No. 82-16, "NUREG-0737 Technical Specifications." The licensee summarized the results of the requested review and concluded that all but two items had been appropriately addressed. For Item 7 (Containment Purge Valve, II.E.4.2.6) it stated that it would review the model TSs when they were issued. For Item 13 (Anticipatory Trip on Turbine

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Trip, II.K.3.12) Toledo Edison stated it would resubmit proposed TSs. (Note - Item 13 is for Westinghouse plants and therefore would not apply to Davis-Besse. However, Item 10, which was inadvertently omitted from the generic letter, is essentially the same item for B&W plants. Therefore, we addressed Toledo Edison's response to Item 13 as essentially being a response to Item 10).

C. Scope of Review

The staff's review consisted of a verification of the status of each of the items and an evaluation of the proposed TSs against the model TSs provided in Generic Letter 82-16 and other guidance related to NUREG-0737. For the Davis-Besse facility, only Item 10 is evaluated in this Safety Evaluation (SE). The remaining items are not being evaluated in this SE because (1) the item has not been completed at the facility, (2) the item has been previously closed out by the staff for the facility, (3) the staff position has not been finalized for the item, or (4) the item does not apply to the Davis-Besse facility. A summary of each of the items is given below.

1. STA Training (I.A.1.1.3)

Our July 2, 1980 letter provided model TSs and TMI lessons learned category "A" ITEMS. Included were TSs that specified the qualifications, training and on-duty requirements for the Shift Technical Advisors (STAs). In a previous submittal of September 16, 1980, the licensee committed to a training program for STAs. Amendment 37 was issued on March 24, 1981 in response to this submittal. By letter of March 29, 1982, the staff advised that its post-implementation review indicated that the licensee's STA training program was acceptable and consistent with all current NRC guidance. However, STA training requirements are now under consideration by the Commission, and no action will be taken to further amend the TSs until guidance is provided by the Commission. Therefore, with respect to GL 82-16 this item is closed.

2. Shift Manning - Overtime Limits (I.A.1.3.1)

In its response, Toledo Edison stated that it has revised administrative procedures to limit overtime and that it considers this item implemented and closed by NRC Inspection Report 81-10 dated June 19, 1982 (the date is actually 1981). That inspection report stated that the administrative procedures were in good agreement with the criteria of pages 3-6 and 3-7 of NUREG-0737. Since then, however, revised criteria have been issued. By letter of February 8, 1982 (Generic Letter 82-02) the Commission issued a policy statement on factors causing fatigue of operating personnel and informing licensees that they would be contacted by separate letter to request that administrative sections of their technical

specifications be revised to adopt the final policy. On February 11, 1982, however, the NRC informed Toledo Edison that based on its review of the Toledo Edison responses to this action item, it found Toledo Edison's policy on overtime limits acceptable since it was more conservative than NUREG-0737 requirements.

By letter of June 15, 1982 (Generic Letter 82-12) the NRC then issued a slightly revised policy statement on factors causing fatigue and requested licensees to review their past actions on this matter to assure they were consistent with the revised policy statement. The revision was only minimally different from the original statement. In a letter dated January 31, 1983, the licensee stated that its procedures had been revised to include the guidelines of Generic Letter 82-12.

Toledo Edison has not incorporated any of this policy into its technical specifications. While it has met or exceeded the intent of the guidelines as evidenced by the NRC letter of February 11, 1982 accepting the commitment which was subsequently included in Administrative Procedures AD 1829, Station Operations; AD 1839.04, STA Administrative Procedure; AD 1844, Maintenance; and AD 1842, Chemistry and Health Physics and closed in Inspection Report 40-346/81-10, it has not met the intent of GL 82-16. Toledo Edison will be requested to submit such a change. This item remains open.

3. Short Term AFWS Evaluation (II.E.1.1)

On August 3, 1982, the NRC requested Toledo Edison to review an NRC status report of the Davis-Besse auxiliary feedwater system reliability evaluation including proposed Technical Specifications. Toledo Edison responded to this letter on September 14, 1982 (Serial No. 857), and disagreed that additional Technical Specifications were needed. This was followed by a telephone conference on September 29 in which the NRC reiterated its position. Generic letter 82-16, which included task action item II.E.1.1, then was sent to all PWR licensees on September 20, 1982. Toledo Edison responded to this letter on November 15, 1982. With regard to this item, Toledo Edison referenced its September 14 response to the NRC request of August 3. The NRC sent another letter on December 1, 1982, disagreeing with Toledo Edison and requesting that it commit to proposed Technical Specifications. Toledo Edison responded on December 15, 1982 and still maintained that the suggested Technical Specifications were not warranted. However on June 15, 1983, it submitted an application to amend its license to include Technical Specifications to verify the AFW flow path following any modifications or repairs to the AFW piping and during the refueling outage. This application currently is being reviewed by the NRC headquarters staff under a separate action (TAC No. 51964); therefore, with respect to GL 82-16 only this item is closed.

4. Safety Grade AFW Initiation and Flow Indication (II.E.1.2)

A safety evaluation was performed by the Instrumentation and Control Systems Branch of NRR on this item. By memorandum dated May 3, 1983, from R. Wayne Houston to G. Lainas, that Branch concluded that "with the addition of redundant flow indication,...we find that the Davis-Besse Nuclear Power Station Unit 1 will meet the requirements of NUREG-0737 - Item II.E.1.2." Amendment 68, issued May 30, 1984, incorporated the requirement for two operable channels of AFW flow indication per steam generator. Therefore, this item is closed.

5. Dedicated Hydrogen Penetrations (II.E.4.1)

Toledo Edison uses the hydrogen dilution system for post accident combustible gas control at the Davis-Besse plant. This was acknowledged by the NRC in Section 6.2.5 of the safety evaluation report (NUREG-0136) following review of the FSAR and determined to be acceptable because it conformed to the guidelines of Regulatory Guide 1.7 and the requirements of General Design Criteria 41, 42, and 43.

In addition, Table 3.6-2 of the Technical Specifications for Davis-Besse lists penetration numbers 51, 67, and 69 for exclusive use of the containment hydrogen dilution and purge systems. Therefore, this item is closed.

6. Containment Pressure Setpoint (II.E.4.2.5)

By letter of January 30, 1981, Toledo Edison submitted information on the containment isolation pressure setpoint. The NRC reviewed that information and by letter of April 14, 1982, informed Toledo Edison that its pressure setpoint met the requirements of II.E.4.2.5; therefore, no Technical Specification change was required. A safety evaluation was included in the letter. This item is therefore closed.

7. Containment Purge Valve (II.E.4.2.6)

Model Technical Specifications on this item were not sent to Toledo Edison. However, by letter of December 3, 1982, Toledo Edison was requested to provide a commitment by December 31, 1982, to seal closed the purge isolation valves when the Davis-Besse plant is in operational modes 1-4 until mechanical stops are installed to prevent opening the valves beyond 55°, and to verify at least every 31 days that the isolation valves are closed. On December 30, 1982 (Serial No. 890) Toledo Edison requested an extension of

time to evaluate the NRC recommended actions, and committed to keep the containment purge and isolation valves closed in operational modes 1-4 until such time that it responded to the December 3 NRC request. By letter of February 16, 1983, the NRC requested Toledo Edison to inform it in writing within 15 days whether or not the purge and vent valves would be maintained closed, and be verified closed at least every 31 days. Toledo Edison responded on March 16, 1983 (Serial No. 920) that the purge and vent valves were closed with the control power off and that the position was being verified once each shift.

The licensee is still evaluating the operability of the large vent and purge valves and will submit a Technical Specification change when the evaluation is complete. Until the evaluation is completed, these purge valves are presently considered to be inoperable; therefore, existing Technical Specification 3.6.3.1.b applies and the affected penetrations must remain isolated by use of at least one deactivated automatic valve secured in the isolation position.

In this interim period, the existing Technical Specification to keep the penetrations isolated and the licensee's commitment to maintain the valves closed and to verify their closure at least every 31 days is considered to meet the requirements of this Item. When the application to amend the Technical Specifications is submitted, the review action will be carried as a separate licensing action; therefore, this item is considered closed.

8. Radiation Signal on Purge Valves (II.E.4.2.7)

In a letter to all PWR licensees on July 2, 1980, the NRC proposed model Technical Specifications including some for containment isolation to satisfy TMI Category "A" action items. Toledo Edison responded on September 16, 1980 (Serial No. 650) with a request for a license amendment to include the model Technical Specifications. On March 24, 1981, the NRC issued Amendment 37 to the Davis-Besse license which included Technical Specifications for closing of the containment purge valve on a high radiation signal. IE Inspection Report 80-29 states, "The licensee has performed the testing required by IE Bulletin 80-06 to determine that the containment purge isolation valve will isolate by the Safety Features Actuation System (SFAS) and that upon resetting of that signal the valve will remain in its safety actuation mode." In its response to Generic Letter 82-16 on this item, Toledo Edison stated that the radiation signal on purge valves is part of the SFAS original design and no additional technical specifications are required.

Based upon the issuance of Amendment 37, and upon the inspection report, Toledo Edison has met the requirements of this item. Therefore, this item is closed.

9. Upgrade B&W AFW System (II.K.2.8)

See response to items 3. and 4. above.

10. B&W Safety Grade Anticipatory Reactor Trip (II.K.2.10)

In a letter dated February 22, 1983, Toledo Edison submitted a revised application for amendment of its Technical Specifications concerning the Anticipatory Reactor Trip System to provide new limiting conditions of operation and surveillance requirements. The evaluation of this proposal is covered in Section II below.

11. B&W Thermal-Mechanical Report (II.K.2.13)

By letter of May 12, 1981, the Babcock and Wilcox Regulatory Response Group submitted a "Letter Report on Reactor Vessel Brittle Fracture Concerns in B&W Operating Plants." This was a generic report which was to be followed by specific reports from B&W licensees. Toledo Edison submitted its report on May 22, 1981, listing design features which make Davis-Besse unique. Based on its evaluation, Toledo Edison concluded that the reactor vessel thermal shock issue was not an immediate safety concern for its facility.

On January 4, 1982, the NRC requested information from B&W owners on this TMI action item. Toledo Edison responded to this letter on March 3, 1982 (Serial No. 790) by stating that since its May 22, 1981 submittal, B&W performed a specific analysis for Davis-Besse on the effects of thermal shock. The conclusion of the analysis was that for at least the first six effective full power years of operation, a 0.023 ft^2 small break loss of coolant accident, with loss of auxiliary feedwater, and no operator action to throttle HPI, would not lead to brittle fracture of the reactor vessel. Therefore, there was no need to establish procedures to instruct operators to throttle HPI and maintain 100°F subcooling margin following a LOCA with no auxiliary feedwater and no training was necessary to familiarize operators with such operations. There was no NRC response to this letter. However, there is nothing further to be done at this time because the issue still is being discussed and will be picked up under the overall issue of thermal shock, which is carried under an individual action item (TAC No. 43428). Therefore, this item is considered closed.

12. Reporting SV and RV Failures and Challenges (II.K.3.3)

On May 7, 1980, the NRC sent a letter to all operating reactor licensees informing them of five additional TMI-2 related requirements including a requirement that Item II.K.3.3 be implemented by January 1, 1981. No model Technical Specifications were included. In its response to this letter dated June 26, 1980 (Serial No. 624), Toledo Edison committed to report challenges of reactor coolant safety and relief valves in its monthly operating reports. Failures of such valves are reported in accordance with Section 6.9.1.8.a. of the Technical Specifications. Following the issuance of Generic Letter 82-16, Toledo Edison responded by reiterating its previous commitment and stating its belief that Technical Specifications were not required. The staff does not agree with this position. If Toledo Edison chooses to report SV and RV challenges in the monthly report, Section 6.9.1.6 of the Technical Specifications should be revised to include this requirement. Toledo Edison will be requested to submit such a change. This item remains open.

13. Anticipatory Trip on Turbine Trip (II.K.3.12)

This item is directed at Westinghouse plants and therefore is not applicable to Davis-Besse. However, Item 10, which was inadvertently omitted from the Generic Letter, is essentially the same item for B&W plants. Therefore, we addressed TECo's response to Item 13 as essentially being a response to Item 10.

II. Safety Evaluation of Licensee's Proposed License Amendment for TMI Action Item II.K.2.10

As a result of the accident at TMI-2, B&W plants were required to install a safety grade anticipatory reactor trip (ARTS) on loss of feedwater and turbine trip. Toledo Edison initially submitted a request to revise its Technical Specifications for this on November 25, 1981 (Serial No. 757). In response to Generic Letter 82-16, Toledo Edison stated it would submit a revised application. This was done on February 22, 1983. The revised application contains proposed Technical Specifications which would impose limiting conditions for operation if the ARTS was inoperable. For turbine trip, if the number of operable ARTS channels is one less than the minimum required, the inoperable channel would have to be restored to an operable condition within 72 hours. If this could not be done, reactor power would have to be reduced to less than 25 percent of rated thermal power within six hours.

For the trip of both main feed pump turbines, if the number of operable ARTS channels is one less than the minimum required, the inoperable channel would have to be restored to an operable condition within 72 hours or the reactor would have to be in at least hot standby within six hours. The ARTS would trip the reactor if the main turbine trips or if both main feed pump turbines are lost to prevent the reactor coolant pressure from increasing to the point where the power operated relief valves would open.

The proposed Technical Specifications are nearly identical to the model Technical Specifications included with Generic Letter 82-16 and meet the intent of TMI action item II.K.2.10. They are therefore acceptable.

III. Environmental Consideration

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. 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 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 or environmental assessment need be prepared in connection with the issuance of this amendment.

IV. Conclusion

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

Dated: July 25, 1984

The following NRC personnel have contributed to this Safety Evaluation Report:

T.N. Tambling
R. W. DeFayette
K. R. Ridgway