January 31, 1990

Docket No. 50-346

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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. 146 TO FACILITY OPERATING LICENSE NO. NPF-3 (TAC NO. 66418)

The Commission has issued Amendment No. 146 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 March 7, 1988.

This amendment removes the requirement in Specification 4.6.1.2.a of Appendix A, Technical Specifications, that the third test of each set of three Type A tests (i.e., the overall integrated containment leak rate tests) be conducted during the same shutdown when the 10-year plant inservice inspection is being conducted. Section 3/4.6.1.2 of the Bases has also been revised to reflect this uncoupling. In a parallel action, we have previously issued an Exemption in our letter dated January 29, 1990, from the requirement regarding the coupling of these two types of tests contained in Section III.D.1(a) of Appendix J to 10 CFR Part 50.

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/

Thomas V. Wambach, Sr. Project Manager
Project Directorate III-3
Division of Reactor Projects - III, IV, V & Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 146 to License No. NPF-3
2. Safety Evaluation

cc: See next page

Office:	LA/PDIII-3	SPE/PDIII-3	PM/PDIII-3	BC/SPLB
Surname:	PKreutzer	MDLynch/mw	TWambach	CMcCracken
Date:	01/10/90	01/10/90	01/10/90	01/11/90

Office:	PDIII-3/PD	OGC
Surname:	JHannon	BMB
Date:	01/11/90	01/23/90

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Mr. Donald C. Shelton
Toledo Edison Company

Davis-Besse Nuclear Power Station
Unit No. 1

cc:
David E. Burke, Esq.
The Cleveland Electric
Illuminating Company
P. O. Box 5000
Cleveland, Ohio 44101

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

Gerald Charnoff, Esq.
Shaw, Pittman, Potts
and Trowbridge
2300 N Street N.W.
Washington, D.C. 20037

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

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

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

Radiological Health Program
Ohio Department of Health
1224 Kinnear Road
Columbus, Ohio 43212

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

Mr. 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

Ohio Environmental Protection Agency
361 East Broad Street
Columbus, Ohio 43266-0558

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

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

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. 146
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 7, 1988 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. 146, 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

John N. Hannon

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

Attachment: Changes to the Technical Specifications

Date of Issuance: January 31, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 146

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 6-2	3/4 6-2
B 3/4 6-1	B 3/4 6-1

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that:

   1. All penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-2 of Specification 3.6.3.1, and

   2. All equipment hatches are closed and sealed.

b. By verifying that each containment air lock is OPERABLE per Specification 3.6.1.3.

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*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that verification of these penetrations being closed need not be performed more often than once per 92 days.

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of $\leq L_a$, 0.50 percent by weight of the containment air per 24 hours at P_a, 38 psig.

b. A combined leakage rate of $\leq 0.60\ L_a$, for all penetrations and valves subject to Type B and C tests, when pressurized to P_a.

c. A combined leakage rate of $\leq 0.015\ L_a$ for all penetrations identified in Table 3.6-1 as secondary containment bypass leakage paths, when pressurized to P_a.

d. A single penetration leakage rate of $\leq 0.15\ L_a$ for the containment purge and exhaust isolation valve special test.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With either (a) the measured overall integrated containment leakage rate exceeding $0.75\ L_a$ (b) with the measured combined leakage rate for all penetrations and valves subject to Type B and C tests exceeding $0.60\ L_a$, or (c) with the combined bypass leakage rate exceeding $0.015\ L_a$, restore the leakage rate(s) to within the limit(s) prior to increasing the Reactor Coolant System temperature above 200°F.

b. With a single containment purge and exhaust isolation valve penetration having leakage rate exceeding $0.15\ L_a$; restore the leakage rate to within limits in 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and provisions of ANSI N45.4 - 1972:

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at P_a, 38 psig, during each 10 year service period.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure of 38 psig, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to $\leq 0.75\ L_a$, during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of 10 CFR Part 50, Appendix J with the following exemption. The third test of each Type A testing set need not be conducted when the plant is shutdown for the 10-year plant inservice inspections. The operational readiness of the vessel is considered proven by the ILRT, and in accordance with license requirements, when completed per the 40 $\pm$ 10 months frequency.

The special test for the containment purge and exhaust isolation valves is intended to detect gross degradation of seals on the valve seats. The special test is performed in addition to the Appendix J requirements.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provide assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the annulus atmosphere of 0.5 psi and 2) the containment peak pressure does not exceed the design pressure of 40 psig during LOCA conditions.

The maximum peak pressure obtained from a LOCA event is 37 psig. The limit of 1 psig for initial positive containment pressure will limit the total pressure to 38 psig which is less than the design pressure and is consistent with the safety analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the accident analysis for a LOCA.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 38 psig in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The limitation on use of the Containment Purge and Exhaust System limits the time this system may be in operation with the reactor coolant system temperature above 200°F. This restriction minimizes the time that a direct open path would exist from the containment atmosphere to the outside atmosphere and consequently reduces the probability that an accident dose would exceed 10 CFR 100 guideline values in the event of a LOCA occurring coincident with purge system operation. The use of this system is therefore restricted to non-routine usage not to exceed 90 hours in any consecutive 365 day period which is equivalent to approximately 1% of the total possible yearly unit operating time.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the containment spray system ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 146 TO FACILITY OPERATING LICENSE NO. NPF-3

TOLEDO EDISON COMPANY

AND

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

DOCKET NO. 50-346

1.0 INTRODUCTION

In its letter dated March 7, 1988, the Toledo Edison Company (the licensee) requested an amendment to the operating license for the Davis-Besse Nuclear Power Station, Unit No. 1, to remove from Specification 4.6.1.2.a of Appendix A, Technical Specifications, a specific surveillance requirement. This requirement presently states that certain primary containment leakage rate tests be conducted during the same shutdown when the 10-year plant inservice inspection is being conducted. The licensee also requested that the Bases in the Davis-Besse Technical Specifications (TSs) be revised to reflect the uncoupling of these two types of tests.

2.0 DISCUSSION

Section III.D.1(a) of Appendix J to 10 CFR Part 50 states in part that "... a set of three Type A tests shall be performed, at approximately equal intervals during each 10-year service period. The third test of each set shall be conducted when the plant is shutdown for the 10-year plant inservice inspections." Appendix J defines Type A tests as those "...intended to measure the primary reactor containment overall integrated leakage rate..." These tests are also identified as integrated leakage rate tests (ILRTs). This Appendix J requirement has been incorporated into the surveillance requirements in Specification 4.6.1.2.a of the Davis-Besse TSs. The 10-year inservice inspection (ISI) is composed of the series of inspections performed every 10 years in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a.

The length of time required to perform the Type A tests (i.e., the ILRTs) necessitates that they be performed during refueling outages. The Appendix J requirement to conduct these ILRTs at approximately equal intervals requires an interval of about 40 months. However, this schedule may not coincide with the plant's refueling outages. Accordingly, the TSs issued with the Davis-Besse operating license allow a variation of 10 months in this interval; i.e., the ILRT test interval can range from 30 to 50 months.

The 10-year ISI is presently scheduled to be performed during the forthcoming refueling outage (i.e., the sixth) starting in February 1990.

The licensee had previously requested, in its letter dated November 20, 1987, an exemption from the Appendix J requirement to couple the third ILRT of the set of three Type A tests every 10 years from the 10-year ISI. This Exemption was issued in the staff's letter dated January , 1990. The intent of the subject amendment request is to revise the Davis-Besse TSs to introduce flexibility in the scheduling of the ILRTs now permitted by the Exemption cited above. Specifically, the requested amendment would permit the next ILRT to be conducted during the seventh refueling outage starting in January 1992 since the last ILRT for the Davis-Besse facility was successfully completed in September 1988 during the fifth refueling outage. The next ILRT could be conducted no later than November 1992. However, the present coupling requirement in the Davis-Besse TSs requires that the next ILRT be conducted during the forthcoming sixth refueling outage starting in February 1990. As discussed in the Exemption cited above, the NRC staff found that this requirement is not necessary to achieve the underlying purpose of the subject rule.

3.0 EVALUATION

The purpose of the periodic ILRTs is to demonstrate that the leakage rate from the primary containment and systems and components penetrating primary containment do not exceed the allowable leakage rate specified in TS 3.6.1.2.a of the Davis-Besse TSs. This demonstration in turn is required to ensure that the calculated offsite radiological doses using the primary containment design basis leakage rate of 0.5 percent by weight of the containment atmosphere under the conditions associated with the design basis accident (DBA) remain valid. These offsite radiological doses for the DBA were originally calculated in accordance with 10 CFR Part 100 and compared to the reference guidelines for determining the suitability of the Davis-Besse site. In summary, the periodic ILRTS are required to support the original site suitability determination for the Davis-Besse plant in accordance with 10 CFR 100.11(a) which requires a "... demonstrable leak rate from the containment..." (Emphasis supplied.) The licensee has successfully performed this required demonstration for the Davis-Besse facility in the three ILRTs performed to date. Additionally, the licensee will continue to perform these Type A tests throughout the plant's lifetime on a schedule consistent with the requirements of Appendix J to 10 CFR Part 50. Furthermore, the licensee's compliance with

Section III.D.1(a) of Appendix J which requires that a set of three ILRTs be performed during each 10-year service period will ensure that the permissible 10-month variation in the surveillance interval will not accumulate indefinitely. On this basis, the NRC staff concludes that the proposed uncoupling of the Type A tests from the 10-year ISI will not affect the required continuing demonstration of the validity of the original calculations performed in compliance with 10 CFR Part 100.

The purpose of the 10-year ISI is to provide assurance of the structural integrity of the Davis-Besse safety-related structures, systems and components in compliance with 10 CFR 50.55a. The licensee will also continue to perform the 10-year inservice inspections in compliance with 10 CFR 50.55a. On this basis, the NRC staff concludes that the proposed uncoupling of the Type A tests from the 10-year ISI will not affect the required continuing demonstration of the structural integrity of the Davis-Besse safety-related structures, systems and components.

Inasmuch as the purposes of the Type A tests and the 10-year ISI are independent of each other and the performance of one does not directly affect the other, there is no safety-related concern associated with the present requirement that they be coupled in the same refueling outage. Furthermore, both tests will continue to be conducted on a schedule consistent with the Commission's regulations. On this basis, the NRC staff finds that the proposed uncoupling of the Type A test from the 10-year ISI is acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change 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 commulative 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.

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, 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.

Principal Contributor: M.D.Lynch

Dated: January 31, 1990