November 27, 1985

Docket No. 50-346

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Mr. Joe Williams, Jr.
Vice President, Nuclear
Toledo Edison Company
Edison Plaza - Stop 712
300 Madison Avenue
Toledo, Ohio 43652

Dear Mr. Williams:

SUBJECT: AMENDMENT NO. 90 TO FACILITY OPERATING LICENSE NO. NPF-3; SPECIAL LEAK TESTING REQUIREMENTS FOR THE CONTAINMENT PURGE VALVES

The Commission has issued the enclosed Amendment No. 90 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 (TSs) in response to your application dated August 18, 1983 (Item 1), as modified November 20, 1984.

The amendment incorporates requirements for special leak testing of containment purge isolation valves. The purpose of the special testing will be to determine if excessive degradation of the valve seats has occurred.

A copy of the Safety Evaluation supporting this amendment is also enclosed. Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Original signed by

George F. Dick, Jr., Project Manager
PWR Project Directorate #6
Division of PWR Licensing-B

Enclosures:
1. Amendment No. 90 to NPF-3
2. Safety Evaluation

cc w/enclosures:
See next page

M 11/27/85

*See previous white for concurrences:

PBD-6	PBD-6	PBD-6	PBD-6	OELD
RIngram *	GDick;cr*	ADeAgazio*	JStolz*	CBarth*
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Sincerely,

George F. Dick, Jr., Project Manager
Operating Reactors Branch #4
Division of Licensing

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

cc w/enclosures:
See next page

ORB#4:DL
ADeAgazio
11/15/85

ORB#4:DL	ORB#4:DL	ORB#4:DL	OELD	AD:OR:DL
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11/13/85	11/14/85	11/18/85	11/20/85	11/ /85

Mr. J. Williams
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

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. 90
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 August 18, 1983, as modified November 20, 1984, 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:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 90 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
PWR Project Directorate #6
Division of PWR Licensing-B

Attachment:
Changes to the Technical Specifications

Date of Issuance: November 27, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 90

FACILITY OPERATING LICENSE NO. NPF-3

DOCKET NO. 50-346

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed 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
3/4 6-3	3/4 6-3
3/4 6-4	3/4 6-4
B 3/4 6-1	B 3/4 6-1

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.

*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 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 $\pm$ 10 month intervals during shutdown at P_a, 38 psig, during each 10 year service period. The third test of each set shall be conducted during the shutdown for the 10 year plant inservice inspection.

SURVEILLANCE REQUIREMENTS (Continued)

b. If any periodic Type A test fails to meet .75 L_a, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet .75 L_a, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet .75 L_a at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

1. Confirms the accuracy of the Type A test by verifying that the difference between supplemental and Type A test data is within 0.25 L_a.

2. Has a duration sufficient to establish accurately the change in leakage between the Type A test and the supplemental test.

3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25 percent of the total measured leakage rate at P_a, 38 psig.

d. Type B and C tests shall be conducted with gas at P_a, 38 psig, at intervals no greater than 24 months except for tests involving:

1. Air locks,

2. Penetrations using continuous leakage monitoring systems, and

3. Valves pressurized with fluid from a seal system.

e. The combined bypass leakage rate shall be determined to be < 0.015 L_a by applicable Type B and C tests at least once every 24 months except for penetrations which are not individually testable; penetrations not individually testable shall be determined to have no detectable leakage when tested with soap bubbles while the containment is pressurized to P_a, 38 psig, during each Type A test.

f. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.

SURVEILLANCE REQUIREMENTS (Continued)

g. Leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a, 41.8 psig, and the seal system capacity is adequate to maintain system pressure for at least 30 days.

h. Type B tests for penetrations employing a continuous leakage monitoring system shall be conducted at P_a, 38 psig, at intervals no greater than once per 3 years.

i. Each time the containment purge and exhaust isolation valves are opened, a special test shall be performed within 72 hours after valve closure or prior to entering mode 4 from mode 5, whichever is later. The special test is conducted by pressurizing the piping section including one valve inside and one valve outside the containment to a pressure greater or equal to 20 psig. The leakage rate per penetration shall not exceed 0.15 L_a.

j. The special test as defined in Surveillance Requirement 4.6.1.2.i shall be performed for the containment purge and isolation valves when the plant has been in any combination of modes 3, 4, 5 or 6 for more than 72 hours provided that the tests required by Surveillance Requirements 4.6.1.2.i or 4.6.1.2.d have not been performed in the previous 6 months.

k. All test leakage rates shall be calculated using observed data converted to absolute values. Error analyses shall be performed to select a balanced integrated leakage measurement system.

l. The provisions of Specification 4.0.2. are not applicable.

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 Appendix "J" of 10 CFR 50.

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 than 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 occuring 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

SUPPORTING AMENDMENT NO. 90 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 AND BACKGROUND

By letter dated August 18, 1983, as modified November 20, 1984, the Toledo Edison Company (licensee) requested that the Davis-Besse Nuclear Power Station, Unit 1 (Davis-Besse) Technical Specifications (TSs) be revised to include a special leakage test requirement for the containment purge and exhaust isolation valve penetrations.

The current TSs require that the containment isolation valves be leak tested once for each operating cycle during refueling. We requested that the valves be tested more frequently to determine if excessive degradation of the valve seats has occurred. Specifically, we recommended that the valves be tested at three-month intervals or after each use of the purge system with a maximum interval not to exceed six months.

2.0 EVALUATION

In the November 20, 1984 submittal, the licensee proposed to perform the special leak testing of the purge valves within 72 hours after each use of the purge system (limited to Modes 5 and 6) or prior to entering Mode 4 from Mode 5, whichever is later. In addition, the special leak testing will be performed after being in Modes 3, 4, 5, or 6 for more than 72 hours following power operation if the special or Type C leak tests have not been performed within the past six months. The licensee contends that implementing a fixed, six-month special test schedule is not feasible because the test requires access to the annulus area which has a radiation level in excess of 1 R/hr during power operation. Without shutting down the reactor, the test could result in high personnel radiation exposures.

Due to the high radiation levels that may be present in the annulus during and shortly after leaving Modes 1 and 2, we have agreed that the licensee may perform the special leak test (1) after each use of the purge system, and (2) when the plant has been in Modes 3, 4, 5, or 6 for more than 72 hours and the valves have not been tested in accordance with Appendix J or the special leak test requirements within the past six months. We find this approach acceptable because containment purge isolation valves are normally sealed closed during Modes 1 through 4, and the outboard valve is within the secondary containment annulus. Therefore, the valve seats will not be subject to degradation in Modes 1 through 4 stemming from system use, will be protected from environmental extremes and will be checked following each use of the purge system.

3.0 ENVIRONMENTAL CONSIDERATION

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

4.0 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: November 27, 1985

The following NRC personnel contributed to this Safety Evaluation: J. Guo