August 23, 1991

Docket No. 50-346

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

Dear Mr. Shelton:

SUBJECT: AMENDMENT NO. 160 TO FACILITY OPERATING LICENSE NO. NPF-3 (TAC NO. 80347)

The Commission has issued Amendment No. 160 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 1, 1991.

This amendment increases the allowed secondary containment bypass leakage rate from 0.015 L_a to 0.03 L_a, relocates the list of secondary containment bypass leakage paths (TS Table 3.6-1) from the TS to the Updated Safety Analysis Report (USAR), and deletes surveillance requirements for types of containment penetrations and containment isolation valves which are not incorporated in the DBNPS design.

A copy of the Safety Evaluation is also enclosed. Notice of issuance will be included in the Commission's next biweekly <u>Federal Register</u> notice.

Sincerely,

Original Signed By:

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

cc: See next page

LA/PDIII-3/	PM/PDIII-3/ JBH	BC/PRPB	BC/SPLB	D/PDIII-3	OGC-WF1
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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. 160
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 March 1, 1991 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. 160, 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: August 23, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 160

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
3/4 6-3	3/4 6-3
3/4 6-4	3/4 6-4
3/4 6-5	3/4 6-5
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 those valves that may be opened under administrative controls per 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 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 $\leq 0.03\ L_a$ for all penetrations that are 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 $0.03\ L_a$, 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.

SURVEILLANCE REQUIREMENTS (Continued)

b. If any periodic Type A test fails to meet 0.75 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 0.75 L_a, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 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 that the rate at which gas is injected into the containment or bled from the containment during the supplemental test is between 0.75 L_a and 1.25 L_a.

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 air locks.

e. The combined bypass leakage rate shall be determined to be < 0.03 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. 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.

h. The special test as defined in Surveillance Requirement 4.6.1.2.g 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 test required by Surveillance Requirement 4.6.1.2.g has not been performed in the previous 6 months.

i. 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.

j. The provisions of Specification 4.0.2 are not applicable.

TABLE 3.6-1

DELETED

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of $\leq 0.002\ L_a$ at P_a, 38 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With an air lock inoperable, restore the air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. *After each opening except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying either no detectable seal leakage when the volume between the door seals is pressurized to 10 psig, or by verifying a seal leakage rate of $\leq 0.0015\ L_a$ when the volume between the door seals is pressurized to P_a, 38 psig, and the air lock door holddowns are installed,

b. At least once per 6 months by conducting an overall air lock leakage test at P_a, 38 psig, and by verifying that the overall air lock leakage rate is within its limit, and

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

*Exemption to Appendix "J" of 10 CFR 50.

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.

USAR 6.2.4 identifies all penetrations that are secondary containment bypass leakage paths.

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. 160 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 March 1, 1991, Toledo Edison (the licensee) proposed changes to the Technical Specifications for the Davis-Besse Nuclear Power Station, Unit 1 (DBNPS). The proposed changes involve Technical Specification (TS) 3/4.6.1.2, "Containment Leakage," and its bases. The proposed changes would increase the allowed secondary containment bypass leakage rate from 0.015 L_a to 0.03 L_a, relocate the list of secondary containment bypass leakage paths (TS Table 3.6-1) from the TS to the Updated Safety Analysis Report (USAR), and delete surveillance requirements for types of containment penetrations and containment isolation valves which are not incorporated in the DBNPS design.

2.0 EVALUATION

The allowable containment leakage is defined in terms of L_a where L_a is the overall integrated leakage rate. L_a is defined as a leakage rate of 0.50% (by weight) of the containment air when the containment is pressurized to 38 psig (P_a) per 24-hour period. P_a is the peak safety analysis accident pressure. The containment leakage is comprised of two components: filtered air and unfiltered air or bypass leakage. Currently the bypass leakage is limited by TS 3.6.1.2c to 0.015 L_a. This TS change proposes increasing the bypass leakage to 0.03 L_a. The consequence of the proposed change is to increase the fraction of the containment volume released which is unfiltered while effectively reducing the fraction released which is filtered resulting in an increase in the radiological consequences.

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The NRC staff performed a confirmatory analysis during the original licensing process using a value of 3 percent for the rate of bypass leakage. That analysis is documented in NUREG-0136, Supplement 1, "Safety Evaluation Report related to operation of Davis-Besse Nuclear Power Station Unit 1," dated April 1977. Based on the analysis, the NRC staff found that the calculated doses met 10 CFR Part 50, Appendix A, General Design Criteria (GDC) 19, "Control Room," and 10 CFR Part 100, "Reactor Site Criteria," and were acceptable. This conclusion is documented in Sections 6.4.1, "Radiation Protection Provisions," and 15.3.1, "Loss of Coolant Accident," of NUREG-0136, Supplement 1.

The licensee benchmarked their existing USAR analysis, and then performed a new analysis using the same assumptions except the bypass leakage was 0.03 L_a instead of 0.015 L_a. The evaluation shows that the increased thyroid doses are in accord with or less than the doses that the NRC staff calculated and found acceptable to meet GDC 19 and 10 CFR Part 100 in NUREG-0136, Supplement 1. Also, the whole body gamma doses were reevaluated by the licensee to determine the impact of increased iodine release on the whole body gamma dose. The licensee found that the increase in whole body gamma dose is negligible.

The NRC staff has reviewed the above information. Based on this review, the NRC staff finds the increase in allowed secondary containment bypass leakage to 0.03 L_a to be acceptable.

The following is a discussion of the administrative changes. TS Table 3.6-1 is being deleted and a reference is being added to the bases to refer to USAR Section 6.2.4 for the list of secondary containment bypass leakage paths. This change is similar to License Amendment 147 which relocated the list of containment isolation valves from the TS to the USAR. Also, in TS 4.6.1.2 references to penetrations using continuous leakage monitoring systems and valves pressurized with fluid from a seal system are being deleted, because DBNPS does not have either of those types of systems. Purely editorial changes are also made to TS 4.6.1.2. The NRC staff has reviewed these proposed changes and finds that they are administrative in nature and 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 (56 FR 24219). 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: Jon B. Hopkins, NRR

Date: August 23, 1991