UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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. 205
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 December 12, 1995, and supplemented by facsimile transmission dated January 26, 1996, 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:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 205 , 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 90 days after issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Linda L. Gundrum

Linda L. Gundrum, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of issuance: February 22, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 205

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.

Remove	Insert
VI	VI
3/4 6-1	3/4 6-1
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 6-9	3/4 6-9
3/4 6-32	3/4 6-32
B 3/4 6-1	B 3/4 6-1
B 3/4 6-2	B 3/4 6-2
B 3/4 6-5	B 3/4 6-5

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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 in compliance with the requirements of Specification 3.6.1.3

c. By performing required visual examinations of the containment vessel and shield building in accordance with 10 CFR 50, Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide 1.163, dated September 1995.

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*Except valves, blind flanges, and deactivated automatic valves which are located inside the Shield Building (including the annulus and 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 $< 0.60\ L_a$, 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 as-left 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 as follows:

a. Perform Type A tests in accordance with 10 CFR 50, Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide 1.163, dated September, 1995.

SURVEILLANCE REQUIREMENTS (Continued)

b. Deleted

c. Deleted

d. Perform Type B and C tests in accordance with 10 CFR 50, Appendix J, Option A, as modified by approved exemptions.

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 in compliance with the requirements of Specification 3/4.6.1.3.

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. Deleted

j. The provisions of Specification 4.0.2 are not applicable.

Page 3/4 6-9 Deleted.
Next page is 3/4 6-10.

Page 3/4 6-32 Deleted.
Next page is 3/4 6-33.

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 and air lock door requirements, 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 as-left 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 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

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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

3/4.6.5.2 SHIELD BUILDING INTEGRITY

SHIELDING BUILDING INTEGRITY ensures that the release of radioactive material from the containment vessel will be restricted to those leakage paths and associated leak rates assumed in the safety analysis. The closure of the airtight doors and blowout panels listed in Table 4.6-1 ensure that the Emergency Ventilation System (EVS) can provide a negative pressure between 0.25 and 1.5 inches Water Gauge within the annulus between the shield building and containment vessel and within the interconnecting mechanical penetration rooms after a loss-of-coolant accident (LOCA). This restriction, in conjunction with the operation of the EVS, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

3/4.6.5.3 SHIELD BUILDING STRUCTURAL INTEGRITY

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