50-346

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

March 21, 1995

Mr. John P. Stetz
Vice President - Nuclear, Davis Besse
Centerior Service Company
c/o Toledo Edison Company
Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, OH 43449

SUBJECT: AMENDMENT NO. 196 TO FACILITY OPERATING LICENSE NO. NPF-3 FOR THE DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1 (TAC NO. M91138)

Dear Mr. Stetz:

The Commission has issued the enclosed Amendment No. 196 to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station, Unit No. 1. The amendment revises the Technical Specifications (TS) in response to your application dated December 6, 1994.

This amendment revises the following TS:

1. The change to Technical Specification 3/4.1.3.2 will delete Surveillance Requirement (SR) 4.1.3.2.2, that presently requires the movement of at least 2% for each Axial Power Shaping Rod not fully withdrawn every 31 days.

2. TS 4.1.3.1.2 for the Movable Control Assemblies "Group Height - Safety and Regulating Rod Groups," will relax testing requirements from at least once every 31 days to every 92 days.

3. TS 4.4.6.2, for "Operational Leakage," relaxes the requirement to leakage test RCS pressure isolation valves prior to MODE 2, whenever the plant has been in COLD SHUTDOWN for 72 hours to whenever the plant has been in COLD SHUTDOWN for 7 days.

4. SR 4.5.2.c.2 for TS 4.5.2, "ECCS Subsystems - Tavg equal to or greater than 280°F," relaxes the inspection requirements for ensuring no debris in containment at the completion of each containment entry to at least once daily for the affected areas.

5. TS 4.6.2.1.d, for the "Containment Spray System," relaxes the SR to perform an air or smoke flow test through the spray header and nozzles from once per 5 years to once per 10 years.

6. TS 4.10.4.2 for "Special Test Exceptions Shutdown Margin" relaxes the SR interval for testing rod insertion capability prior to reducing shutdown margin from 24 hours to 7 days.

DF01

Changes 2 through 6 revise surveillance intervals in accordance with the guidance of Generic Letter 93-05, "Line Item Technical Specifications Improvements to Reduce Surveillance Requirements For Testing During Power Operation," and NUREG-1366, "Improvements to Technical Specifications Surveillance Requirements."

We have reviewed the proposed changes and concluded that both the line-item improvement and the plant specific changes are acceptable.

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,

Original signed by:

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

Docket No. 50-346

Enclosures: 1. Amendment No. 196 to License No. NPF-3
2. Safety Evaluation

cc w/encls: See next page

Distribution w/encls:

Docket File	GHill (2)
PUBLIC	ACRS (4)
PD3-3 Reading	OC/LFDCB
EAdensam	JHannon
OPA	
WAxelson, RIII	

DOCUMENT NAME: G:\DAVISBES\DB91138.AMD

To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy *See previous concurrences

OFFICE	LA:PDIII-3	E	PM:PDIII-3	E	OTSB:BC*		OGC*		PD:PDIII-3*	
NAME	MRushbrook		LGundrum		CGrimes		CWoodhead		LNorrholm	
DATE	3/21/95		3/21/95		02/15/95		02/23/95		03/08/95	

OFFICIAL RECORD COPY

Changes 2 through 6 revise surveillance intervals in accordance with the guidance of Generic Letter 93-05, "Line Item Technical Specifications Improvements to Reduce Surveillance Requirements For Testing During Power Operation," and NUREG-1366, "Improvements to Technical Specifications Surveillance Requirements."

We have reviewed the proposed changes and concluded that both the line-item improvement and the plant specific changes are acceptable.

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,

Linda L. Gundrum

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

Docket No. 50-346

Enclosures: 1. Amendment No. 196 to License No. NPF-3
2. Safety Evaluation

cc w/encls: See next page

cc:
Mary E. O'Reilly
Centerior Energy Corporation
300 Madison Avenue
Toledo, Ohio 43652

Mr. William T. O'Connor, Jr.
Manager - Regulatory Affairs
Toledo Edison Company
Davis-Besse Nuclear Power Plant
5501 North State - Route 2
Oak Harbor, Ohio 43449

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

Regional Administrator
U.S. NRC, Region III
801 Warrenville Road
Lisle, Illinois 60523-4351

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

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

Mr. John K. Wood, Plant Manager
Toledo Edison Company
Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, Ohio 43449

Robert E. Owen, Chief
Bureau of Radiological Health
Service
Ohio Department of Health
P. O. Box 118
Columbus, Ohio 43266-0118

Attorney General
Department of Attorney
General
30 East Broad Street
Columbus, Ohio 43216

Mr. James W. Harris, Director
Division of Power Generation
Ohio Department of Industrial
Regulations
P. O. Box 825
Columbus, Ohio 43216

Ohio Environmental Protection Agency
DERR--Compliance Unit
ATTN: Zack A. Clayton
P. O. Box 1049
Columbus, Ohio 43266-0149

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

Mr. James R. Williams
State Liaison to the NRC
Adjutant General's Department
Office of Emergency Management Agency
2825 West Granville Road
Columbus, Ohio 43235-2712

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

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. 196
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 6, 1994, 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. 196 , 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: March 21, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 196

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
3/4 1-20	3/4 1-20
3/4 1-21	3/4 1-21
3/4 4-16	3/4 4-16
3/4 5-4	3/4 5-4
3/4 6-12	3/4 6-12
3/4 10-4	3/4 10-4

GROUP HEIGHT - SAFETY AND REGULATING ROD GROUPS

LIMITING CONDITION FOR OPERATIONS

ACTION: (Continued)

c) A power distribution map is obtained from the incore detectors and F_Q and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours, and

d) Either the THERMAL POWER level is reduced to $\leq$ 60% of the THERMAL POWER allowable for the reactor coolant pump combination within one hour and within the next 4 hours the High Flux Trip Setpoint is reduced to $\leq$ 70% of the THERMAL POWER allowable for the reactor coolant pump combination, or

e) The remainder of the rods in the group with the inoperable rod are aligned to within ± 6.5% of the inoperable rod within one hour while maintaining the position of the rods within the limits provided in the CORE OPERATING LIMITS REPORT; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each control rod shall be determined to be within the group average height limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the asymmetric rod monitor is inoperable, then verify the individual rod position(s) of the rod(s), with the inoperable asymmetric rod monitor at least once per 4 hours.

4.1.3.1.2 Each control rod not fully inserted shall be determined to be OPERABLE by movement of at least 2% in any one direction at least once every 92 days.

GROUP HEIGHT - AXIAL POWER SHAPING ROD GROUP

LIMITING CONDITION FOR OPERATION

3.1.3.2 All axial power shaping rods (APSR) shall be OPERABLE, unless fully withdrawn, and shall be positioned within ± 6.5% (indicated position) of their group average height.

APPLICABILITY: MODES 1* and 2*.

ACTION:

With a maximum of one APSR inoperable or misaligned from its group average height by more than ± 6.5% (indicated position), operation may continue provided that within 2 hours:

a. The APSR group is positioned such that the misaligned rod is restored to within limits for the group average height, or

b. It is determined that the imbalance limits of Specification 3.2.1 are satisfied and movement of the APSR group is prevented while the rod remains inoperable or misaligned.

SURVEILLANCE REQUIREMENTS

4.1.3.2.1 The position of each APSR rod shall be determined to be within the group average height limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the asymmetric rod monitor is inoperable, then verify the individual rod position(s) of the rod(s), with the inoperable asymmetric rod monitor at least once per 4 hours.

*See Special Test Exceptions 3.10.1 and 3.10.2

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere gaseous or particulate radioactivity at least once per 12 hours.

b. Monitoring the containment sump level and flow indication at least once per 12 hours.

c. Measurement of the CONTROLLED LEAKAGE from the reactor coolant pump seals to the makeup system when the Reactor Coolant System pressure is 2185 ± 20 psig at least once per 31 days.

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours during steady state operation.

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-2 shall be individually demonstrated OPERABLE by verifying leakage testing (or the equivalent) to be within its limit prior to entering MODE 2:

a. After each refueling outage,

b. Whenever the plant has been in COLD SHUTDOWN for 7 days, or more, and if leakage testing has not been performed in the previous 9 months, and

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve.

d. The provisions of Specification 4.0.4 are not applicable for entry into MODES 3 or 4.

4.4.6.2.3 Whenever the integrity of a pressure isolation valve listed in Table 3.4-2 cannot be demonstrated, determine and record the integrity of the high pressure flowpath on a daily basis. Integrity shall be determined by performing either a leakage test of the remaining pressure isolation valve, or a combined leakage test of the remaining pressure isolation valve in a series with the closed motor operated containment isolation valve. In addition, record the position of the closed motor-operated containment isolation valve located in the high pressure piping on a daily basis.

b. At least once per 18 months, or prior to operation after ECCS piping has been drained by verifying that the ECCS piping is full of water by venting the ECCS pump casings and discharge piping high points.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment emergency sump and cause restriction of the pump suction during LOCA conditions. This visual inspection shall be performed:

   1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

   2. For all areas of containment affected by an entry, at least once daily while work is ongoing and again during the final exit after completion of work (containment closeout) when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

   1. Verifying that the interlocks:

      a) Close DH-11 and DH-12 and deenergize the pressurizer heaters, if either DH-11 or DH-12 is open and a simulated reactor coolant system pressure which is greater than the trip setpoint (<438 psig) is applied. The interlock to close DH-11 and/or DH-12 is not required if the valve is closed and 480 V AC power is disconnected from its motor operators.

      b) Prevent the opening of DH-11 and DH-12 when a simulated or actual reactor coolant system pressure which is greater than the trip setpoint (<438 psig) is applied.

   2. a) A visual inspection of the containment emergency sump which verifies that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

      b) Verifying that on a Borated Water Storage Tank (BWST) Low-Low Level interlock trip, with the motor operators for the BWST outlet isolation valves and the containment emergency sump recirculation valves energized, the BWST Outlet Valve HV-DH7A (HV-DH7B) automatically close in $\leq$75 seconds after the operator manually pushes the control switch to open the Containment Emergency Sump Valve HV-DH9A (HV-DH9B) which should be verified to open in $\leq$75 seconds.

   3. Verifying a total leak rate $\leq$ 20 gallons per hour for the LPI system at:

      a) Normal operating pressure or hydrostatic test pressure of $\geq$ 150 psig for those parts of the system downstream of the pump suction isolation valve, and

      b) $\geq$ 45 psig for the piping from the containment emergency sump isolation valve to the pump suction isolation valve.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months by verifying a total leak rate $\leq$ 20 gallons per hour for the system at:

   1. Normal operating pressure or a hydrostatic test pressure of $\geq$ 150 psig for those parts of the system downstream of the pump suction isolation valve, and

   2. $\geq$ 45 psig for the piping from the containment emergency sump isolation valve to the pump suction isolation valve.

d. At least once per 10 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

SPECIAL TEST EXCEPTIONS

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.4 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and shutdown margin provided:

a. Reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s), and

b. All axial power shaping rods are withdrawn to at least 35% (indicated position) and OPERABLE.

APPLICABILITY: MODE 2.

ACTION:

a. With any safety or regulating control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion or the axial power shaping rods not within their withdrawal limits, immediately initiate and continue boration at $\geq$ 25 gpm of 7875 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all safety or regulating control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at $\geq$ 25 gpm of 7875 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The position of each safety, regulating, and axial power shaping rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.4.2 Each safety or regulating control rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 7 days prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 196 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 December 6, 1994, the licensee requested an amendment to the Davis-Besse Nuclear Power Station (DBNPS), Unit 1, operating license to revise the plant Technical Specifications (TS). The proposed changes involve TS 3/4.1.3.1, "Group Height - Safety and Regulating Rod Groups;" TS 3/4.1.3.2, "Group Height - Axial Power Shaping Rod Group;" TS 3/4.4.6.2, "Operational Leakage;" TS 3/4.5.2, "ECCS Subsystems - Tavg equal to or greater than 280°F;" TS 3/4.6.2.1, "Containment Spray System;" and TS 3/4.10.4, "Special Test Exceptions Shutdown Margin." All the proposed TS changes are compatible with DBNPS operating experience and are consistent with NRC guidance provided by Generic Letter (GL) 93-05, "Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation," or NUREG-1430, "Improved Standard Technical Specifications for B&W Plants."

2.0 EVALUATION

The proposed change to TS 4.1.3.1.2 for the Movable Control Assemblies "Group Height - Safety and Regulating Rod Groups," will relax testing requirements from at least once every 31 days to every 92 days. This change is in conformance with the recommendation of GL 93-05, Enclosure 1, Section 4.2.1. Additionally, the DBNPS reliability data for the Control Rod Drive Control System was reviewed and supports extending the surveillance frequency. Since there is no expected adverse effect on safety as a result of extending the surveillance interval, the staff, therefore, concludes that the proposed change is acceptable.

The proposed change to Technical Specification 3/4.1.3.2 will delete Surveillance Requirement (SR) 4.1.3.2.2, that presently requires each Axial Power Shaping Rod (APSR), that is not fully withdrawn, be moved every 31 days and that the movement be at least 2% to verify OPERABILITY. This requirement was deleted from Revision 0 of NUREG-1430. The APSRs have no safety function to insert in order to mitigate any design basis accident. In fact, by the design of the system, these rods are prevented from moving on a reactor trip. SR 4.1.3.2.1, which is not changing, still requires determining the position of each APSR at least once per 12 hours except when the asymmetric rod monitor is inoperable and then at least once every 4 hours. According to the submittal, the DBNPS reliability data for the movement of APSRs was reviewed and supports the removal of this surveillance requirement from the TS. The staff, therefore, concludes that the proposed change is acceptable.

The proposed change to TS 4.4.6.2, for "Operational Leakage," relaxes the requirement to leakage test RCS pressure isolation valves prior to entering MODE 2, whenever the plant has been in COLD SHUTDOWN for 72 hours to whenever the plant has been in COLD SHUTDOWN for 7 days. The proposed change is in conformance with GL 93-05, Enclosure 1, Section 6.1. The DBNPS reliability data for the Control Rod Drive Control System was reviewed and supports extending the 24-hour time period during which to perform the rod full insertion test to 7 days. The staff, therefore, concludes that the proposed change is acceptable.

The proposed change to SR 4.5.2.c.2 for TS 4.5.2, "ECCS Subsystems - Tavg equal to or greater than 280°F," relaxes the inspection requirements for ensuring no debris in containment from "at the completion of each containment entry" to "at least once daily while work is ongoing and again during the final exit after completion of work (containment closeout) when CONTAINMENT INTEGRITY is established." The current DBNPS TS require a containment debris inspection during the final exit for containment closeout. This change does not make substantial changes to the previous requirements, nor does it change the method of performing the surveillance. However, this change will limit the potential for radiation dose to people performing containment inspections when containment integrity is established by limiting the inspection to those areas where work is being performed and ensuring those areas are inspected at least daily. Although the proposed change uses plant specific language, the change meets the intent of GL 93-05, Enclosure 1, Section 7.5. The staff, therefore, concludes that the proposed change is acceptable.

TS 4.6.2.1.d, for the "Containment Spray System," relaxes the SR to perform an air or smoke flow test through the spray header and nozzles from once per 5 years to once per 10 years. This proposed change is in accordance with GL 93-05, Enclosure 1, Section 8.1. As discussed in NUREG-1366, "Improvements to Technical Specifications Surveillance Requirements," Section 8.1, industry experience has shown that the 3 instances where the test detected system problems, that the problems were attributable to plant construction. At the DBNPS, this surveillance test has been satisfactorily performed in the past. Because this system is not used during normal operation, there is no mechanism by which these nozzles could be rendered nonfunctional. The staff, therefore, concludes that the proposed change is acceptable.

TS 4.10.4.2 for "Special Test Exceptions Shutdown Margin" relaxes the SR interval for testing rod insertion capability prior to reducing the shutdown margin from 24 hours to 7 days. This proposed change is in accordance with GL 93-05, Enclosure 1, Section 12. The DBNPS reliability data for the Control Rod Drive Control System was reviewed and supports extending the 24-hour time period during which to perform the rod full insertion test to 7 days.

Based on the review of each proposed change and determination that the proposed changes will not have an adverse effect on safety, the staff concludes that the proposed changes are acceptable and should be approved.

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 and changes surveillance requirements. 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 (60 FR 8757). 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: L. L. Gundrum

Date: March 21, 1995