January 11, 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. 144 TO FACILITY OPERATING LICENSE NO. NPF-3 (TAC NO. 73245)

The Commission has issued Amendment No. 144 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 June 16, 1989 as revised by your submittal dated August 21, 1989.

This amendment relocates the values of cycle-specific limits from the Technical Specifications to the Core Operating Limits Report in accordance with NRC Generic Letter 88-16. The requirements to meet these limits and the associated Action Statements if limits are not met are retained in the Technical Specifications.

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

cc: See next page

Office:	LA/PDIII-3	PM/PDIII-3	PD/PDIII-3	OGC-WF1
Surname:	PKreutzer	TWambach/tg	JHannon	
Date:	12/21/89	12/28/89	12/29/89	1/4/89

DFO1
1/1

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. 144
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 June 16, 1989 as revised August 21, 1989, 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. 144, 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 11, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 144

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
Ia	Ia
-	1-6c
3/4 1-20	3/4 1-20
3/4 1-26	3/4 1-26
3/4 1-27	3/4 1-27
3/4 1-28 through 3/4 1-28d	-
3/4 1-29 through 3/4 1-29d	-
3/4 1-30	3/4 1-30
3/4 1-31	-
3/4 1-32	-
3/4 1-33	3/4 1-33
3/4 1-34	3/4 1-34
3/4 1-35 through 3/4 1-43	-
3/4 2-1	3/4 2-1
3/4 2-2 through 3/4 2-4a	-
3/4 2-9	3/4 2-9
3/4 2-10	3/4 1-10
3/4 2-11	3/4 2-11
3/4 2-12	-
3/4 8-4	3/4 8-4 (no change.)
B 3/4 2-1	B 3/4 2-1
6-16	6-16
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DEFINITIONS

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1.0 DEFINITIONS

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DEFINITIONS

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1.0 DEFINITIONS (Continued)

CORE OPERATING LIMITS REPORT

1.41 The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.7. Plant operation within these core operating limits is addressed in individual specifications.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT - SAFETY AND REGULATING ROD GROUPS

LIMITING CONDITION FOR OPERATIONS

3.1.3.1 All control (safety and regulating) rods shall be OPERABLE and positioned within $\pm$ 6.5% (indicated position) of their group average height.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more control rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within one hour and be in at least HOT STANDBY within 6 hours.

b. With more than one control rod inoperable or misaligned from its group average height by more than $\pm$ 6.5% (indicated position), be in at least HOT STANDBY within 6 hours.

c. With one control rod inoperable due to causes other than addressed by ACTION a, above, or misaligned from its group average height by more than $\pm$ 6.5% (indicated position), POWER OPERATION may continue provided that within one hour either:

   1. The control rod is restored to OPERABLE status within the above alignment requirements, or

   2. The control rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

      a) An analysis of the potential ejected rod worth is performed within 72 hours and the rod worth is determined to be $\leq$ 1.0% Δk at zero power and $\leq$ 0.65% Δk at RATED THERMAL POWER for the remainder of the fuel cycle.

      b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

*See Special Test Exceptions 3.10.1 and 3.10.2.

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.

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 $\pm$ 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 Fault Circuitry is inoperable, then verify the individual rod position(s) of the rod(s), with inoperable Fault Circuitry 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 31 days.

SAFETY ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All safety rods shall be fully withdrawn.

APPLICABILITY: 1* and 2*#.

ACTION:

With a maximum of one safety rod not fully withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, within one hour either:

a. Fully withdraw the rod or

b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each safety rod shall be determined to be fully withdrawn:

a. Within 15 minutes prior to withdrawal of any regulating rod during an approach to reactor criticality.

b. At least once per 12 hours thereafter.

*See Special Test Exception 3.10.1 and 3.10.2.
#With $K_{eff} \geq 1.0$.

REACTIVITY CONTROL SYSTEMS

REGULATING ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The regulating rod groups shall be positioned within the acceptable operating limits for regulating rod position provided in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With the regulating rod groups inserted beyond the operating limits (in a region other than acceptable operation), or with any group sequence or overlap outside the limits provided in the CORE OPERATING LIMITS REPORT except for surveillance testing pursuant to Specification 4.1.3.1.2, either:

a. Restore the regulating groups to within the limits provided in the CORE OPERATING LIMITS REPORT within 2 hours, or

b. Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the rod group position limits provided in the CORE OPERATING LIMITS REPORT within 2 hours, or

c. Be in at least HOT STANDBY within 6 hours.

NOTE: If in unacceptable region, also see Section 3/4.1.1.1.

---

*See Special Test Exception 3.10.1 and 3.10.2.

#With $K_{eff} \geq 1.0$.

REGULATING ROD INSERTION LIMITS

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each regulating group shall be determined to be within the limits provided in the CORE OPERATING LIMITS REPORT at least once every 12 hours except when:

a. The regulating rod insertion limit alarm is inoperable, then verify the groups to be within the insertion limits at least once per 4 hours;

b. The control rod drive sequence alarm is inoperable, then verify the groups to be within the sequence and overlap limits at least once per 4 hours.

ROD PROGRAM

LIMITING CONDITION FOR OPERATION

3.1.3.7 Each control rod assembly (safety, regulating and APSR) shall be programmed to operate in the core location and rod group specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODES 1* and 2*.

ACTION:

With any control rod assembly not programmed to operate as specified above, be in HOT STANDBY within 1 hour.

SURVEILLANCE REQUIREMENTS

4.1.3.7

a. Each control rod assembly shall be demonstrated to be programmed to operate in the specified core location and rod group by:

   1. Selection and actuation from the control room and verification of movement of the proper rod as indicated by both the absolute and relative position indicators:

      a) For all control rod assemblies after the control rod drive patches are locked subsequent to test, reprogramming or maintenance within the panels.

      b) For specifically affected individual rod assemblies following maintenance, test, reconnection or modification of power or instrumentation cables from the control rod drive control system to the control rod drive.

   2. Verifying that each cable that has been disconnected has been properly matched and reconnected to the specified control rod drive.

b. At least once each 7 days, verify that the control rod drive patch panels are locked.

*See Special Test Exceptions 3.10.1 and 3.10.2.

(Next page is 3/4 1-33)

XENON REACTIVITY

LIMITING CONDITION FOR OPERATION

3.1.3.8 THERMAL POWER shall not be increased above the power level cutoff specified in the acceptable operating limits for regulating rod position provided in the CORE OPERATING LIMITS REPORT unless one of the following conditions is satisfied:

a. Xenon reactivity is within 10 percent of the equilibrium value for RATED THERMAL POWER and is approaching stability, or

b. THERMAL POWER has been within a range of 87 to 92 percent of RATED THERMAL POWER for a period exceeding 2 hours in the soluble poison control mode, excluding xenon free start-ups.

APPLICABILITY: MODE 1.

ACTION:

With the requirements of the above specification not satisfied, reduce THERMAL POWER to less than or equal to the power level cutoff within 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.3.8 Xenon reactivity shall be determined to be within 10% of the equilibrium value for RATED THERMAL POWER and to be approaching stability or it shall be determined that the THERMAL POWER has been in the range of 87 to 92% of RATED THERMAL POWER for $\geq$ 2 hours, prior to increasing THERMAL POWER above the power level cutoff.

AXIAL POWER SHAPING ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.9 The axial power shaping rod group shall be within the acceptable operating limits for axial power shaping rod position specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODES 1 and 2*.

ACTION:

With the axial power shaping rod group outside the above insertion limits, either:

a. Restore the axial power shaping rod group to within the limits within 2 hours, or

b. Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the rod group position using the acceptable operating limits provided in the CORE OPERATING LIMITS REPORT within 2 hours, or

c. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.9 The position of the axial power shaping rod group shall be determined to be within the limits provided in the CORE OPERATING LIMITS REPORT at least once every 12 hours except when the axial power shaping rod insertion limit alarm is inoperable, then verify the group to be within the limit provided in the CORE OPERATING LIMITS REPORT at least once every 4 hours.

*With $k_{eff} \geq 1.0$.

(Next page is 3/4 2-1)

3/4.2 POWER DISTRIBUTION LIMITS

AXIAL POWER IMBALANCE

LIMITING CONDITION FOR OPERATION

3.2.1 AXIAL POWER IMBALANCE shall be maintained within the acceptable AXIAL POWER IMBALANCE operating limits provided in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODE 1 above 40% of RATED THERMAL POWER.*

ACTION:

With AXIAL POWER IMBALANCE exceeding the limits specified above, either:

a. Restore the AXIAL POWER IMBALANCE to with the limits provided in the CORE OPERATING LIMITS REPORT within 15 minutes, or

b. Within one hour reduce power until imbalance limits provided in the CORE OPERATING LIMITS REPORT are met or to 40% of RATED THERMAL POWER or less.

SURVEILLANCE REQUIREMENTS

4.2.1 The AXIAL POWER IMBALANCE shall be determined to be within the limits provided in the CORE OPERATING LIMITS REPORT at least once every 12 hours when above 40% of RATED THERMAL POWER except when the AXIAL POWER IMBALANCE alarm is inoperable, then calculate the AXIAL POWER IMBALANCE at least once per hour.

*See Special Test Exception 3.10.1.

QUADRANT POWER TILT

LIMITING CONDITION FOR OPERATION

3.2.4 THE QUADRANT POWER TILT shall not exceed the Steady State Limit for QUADRANT POWER TILT provided in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODE 1 above 15% of RATED THERMAL POWER.*

ACTION:

a. With the QUADRANT POWER TILT determined to exceed the Steady State Limit but less than or equal to the Transient Limit provided in the CORE OPERATING LIMITS REPORT:

   1. Within 2 hours:

      a) Either reduce the QUADRANT POWER TILT to within its Steady State Limit, or

      b) Reduce THERMAL POWER so as not to exceed THERMAL POWER, including power level cutoff, allowable for the reactor coolant pump combination less at least 2% for each 1% of QUADRANT POWER TILT in excess of the Steady State Limit and within 4 hours, reduce the High Flux Trip Setpoint and the Flux-Δ Flux-Flow Trip Setpoint at least 2% for each 1% of QUADRANT POWER TILT in excess of the Steady State Limit.

   2. Verify that the QUADRANT POWER TILT is within its Steady State Limit within 24 hours after exceeding the Steady State Limit or reduce THERMAL POWER to less than 60% of THERMAL POWER allowable for the reactor coolant pump combination within the next 2 hours and reduce the High Flux Trip Setpoint to $\leq$ 65.5% of THERMAL POWER allowable for the reactor coolant pump combination within the next 4 hours.

   3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 60% of THERMAL POWER allowable for the reactor coolant pump combination may proceed provided that the QUADRANT POWER TILT is verified within its Steady State Limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

*See Special Test Exception 3.10.1.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

b. With the QUADRANT POWER TILT determined to exceed the Transient Limit but less than the Maximum Limit provided in the CORE OPERATING LIMITS REPORT, due to misalignment of either a safety, regulating or axial power shaping rod:

1. Reduce THERMAL POWER at least 2% for each 1% of indicated QUADRANT POWER TILT in excess of the Steady State Limit within 30 minutes.

2. Verify that the QUADRANT POWER TILT is within its Transient Limit within 2 hours after exceeding the Transient Limit or reduce THERMAL POWER to less than 60% of THERMAL POWER allowable for the reactor coolant pump combination within the next 2 hours and reduce the High Flux Trip Setpoint to ≤ 65.5% of THERMAL POWER allowable for the reactor coolant pump combination within the next 4 hours.

3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 60% of THERMAL POWER allowable for the reactor coolant pump combination may proceed provided that the QUADRANT POWER TILT is verified within its Steady State Limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

c. With the QUADRANT POWER TILT determined to exceed the Transient Limit but less than the Maximum Limit provided in the CORE OPERATING LIMITS REPORT, due to causes other than the misalignment of either a safety, regulating or axial power shaping rod:

1. Reduce THERMAL POWER to less than 60% of THERMAL POWER allowable for the reactor coolant pump combination within 2 hours and reduce the High Flux Trip Setpoint to ≤ 65.5% of THERMAL POWER allowable for the reactor coolant pump combination within the next 4 hours.

2. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 60% of THERMAL POWER allowable for the reactor coolant pump combination may proceed provided that the QUADRANT POWER TILT is verified within its Steady State Limit at least once per hour for 12 hours or until verified at 95% or greater RATED THERMAL POWER.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

d. With the QUADRANT POWER TILT determined to exceed the Maximum Limit provided in the CORE OPERATING LIMITS REPORT, reduce THERMAL POWER to ≤ 15% of RATED THERMAL POWER within 2 hours.

SURVEILLANCE REQUIREMENTS

4.2.4 The QUADRANT POWER TILT shall be determined to be ≤ the Steady State Limits provided in the CORE OPERATING LIMITS REPORT at least once every 7 days during operation above 15% of RATED THERMAL POWER except when the QUADRANT POWER TILT alarm is inoperable, then the QUADRANT POWER TILT shall be calculated at least once per 12 hours.

(Next page is 3/4 2-13)

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying the fuel level in the day fuel tank.

2. Verifying the fuel level in the fuel storage tank.

3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day tank.

4. Verifying the diesel starts and accelerates up to 900 rpm, preceded by an engine prelube and/or appropriate other warmup procedures.

5. Verifying the generator is synchronized, loaded to $\geq$ 1000 kw, and operates for $\geq$ 60 minutes.

6. Verifying the diesel generator is aligned to provide standby power to the associated essential busses.

7. Verifying that the automatic load sequence timer is OPERABLE with each load sequence time within $\pm$ 10% of its required value.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank is within the acceptable limits specified in Table 1 of ASTM D975-68 when checked for viscosity, water and sediment.

c. At least once per 184 days on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the day fuel tank.

2. Verifying the fuel level in the fuel storage tank.

3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day tank.

4. Verifying the diesel starts from ambient condition and accelerates to at least 900 rpm in $\leq$ 10 seconds.

5. Verifying the generator is synchronized, loaded to $\geq$ 1000 kw, and operates for $\geq$ 60 minutes.

6. Verifying the diesel generator is aligned to provide standby power to the associated essential busses.

7. Verifying that the automatic load sequence timer is OPERABLE with each load sequence time within $\pm$ 10% of its required value.

d. At least once per 18 months during shutdown by:

1. Verifying the generator capability to reject a load equal to the largest single emergency load supplied by the generator without tripping.

2. Simulating a loss of offsite power in conjunction with a safety features actuation system (SFAS) test signal, and:

(a) Verifying de-energization of the essential busses and load shedding from the essential busses.

(b) Verifying the diesel starts from ambient condition on the auto-start signal, energizes the essential busses with permanently connected loads, energizes the auto-connected essential loads through the load sequencer and operates for $\geq$ 5 minutes while its generator is loaded with the essential loads.

(c) Verifying that all diesel generator trips, except engine overspeed and generator differential, are automatically bypassed upon loss of voltage on the essential bus and/or an SFAS test signal.

3. Verifying the diesel generator operates for $\geq$ 60 minutes while loaded to $\geq$ 2000 kw.

4. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000 hour rating of 2838 kw.

e. At least once per 30 months by subjecting the diesels to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service.*

* The provisions of Specification 4.0.2 are not applicable.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (normal operation) and II (incidents of moderate frequency) events by: (a) maintaining the minimum DNRB in the core $\geq$ 1.30 during normal operation and during short term transients, (b) maintaining the peak linear power density $\leq$ 18.4 kW/ft during normal operation, and (c) maintaining the peak power density less than the limits given in the bases to specification 2.1 during short term transients. In addition, the above criteria must be met in order to meet the assumptions used for the loss-of-coolant accidents.

The power imbalance envelope and the insertion limit curves defined in the CORE OPERATING LIMITS REPORT are based on LOCA analyses which have defined the maximum linear heat rate such that the maximum clad temperature will not exceed the Final Acceptance Criteria of 2200°F following a LOCA. Operation outside of the power imbalance envelope alone does not constitute a situation that would cause the Final Acceptance Criteria to be exceeded should a LOCA occur. The power imbalance envelope represents the boundary of operation limited by the Final Acceptance Criteria only if the control rods are at the insertion limits, as defined in the CORE OPERATING LIMITS REPORT and if the steady-state limit QUADRANT POWER TILT exists. Additional conservatism is introduced by application of:

a. Nuclear uncertainty factors.

b. Thermal calibration uncertainty.

c. Fuel densification effects.

d. Hot rod manufacturing tolerance factors.

e. Potential fuel rod bow effects.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensures that the original criteria are met.

The definitions of the design limit nuclear power peaking factors as used in these specifications are as follows:

F_Q Nuclear heat flux hot channel factor, is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and rod dimensions.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod on which minimum DNBR occurs to the average rod power.

It has been determined by extensive analysis of possible operating power shapes that the design limits on nuclear power peaking and on minimum DNBR at full power are met, provided:

$$F_Q \leq 2.93; \quad F^N_{\Delta H} \leq 1.71$$

Power peaking is not a directly observable quantity and therefore limits have been established on the bases of the AXIAL POWER IMBALANCE produced by the power peaking. It has been determined that the above hot channel factor limits will be met provided the following conditions are maintained.

1. Control rods in a single group move together with no individual rod insertion differing by more than $\pm 6.5\%$ (indicated position) from the group average height.

2. Regulating rod groups are sequenced with overlapping groups as required in Specification 3.1.3.6.

3. The regulating rod insertion limits of Specification 3.1.3.6 are maintained.

4. AXIAL POWER IMBALANCE limits are maintained. The AXIAL POWER IMBALANCE is a measure of the difference in power between the top and bottom halves of the core. Calculations of core average axial peaking factors for many plants and measurements from operating plants under a variety of operating conditions have been correlated with AXIAL POWER IMBALANCE. The correlation shows that the design power shape is not exceeded if the AXIAL POWER IMBALANCE is maintained between the limits specified in Specification 3.2.1.

The design limit power peaking factors are the most restrictive calculated at full power for the range from all control rods fully withdrawn to minimum allowable control rod insertion and are the core DNBR design basis. Therefore, for operation at a fraction of RATED THERMAL POWER, the design limits are met. When using incore detectors to make power distribution maps to determine F_Q and $F^N_{\Delta H}$:

a. The measurement of total peaking factor F_Q^{Meas}, shall be increased by 1.4 percent to account for manufacturing tolerances and further increased by 7.5 percent to account for measurement error.

power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL OPERATING REPORT

6.9.1.4 Annual reports covering the activities of the unit during the previous calendar year shall be submitted prior to March 31 of each year.

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions1/, e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (described maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

b. The complete results of steam generator tube inservice inspections (Specification 4.4.5.5.b).

c. The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in

1/ This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

MONTHLY OPERATING REPORT

6.9.1.6 Routine reports of operating statistics, shutdown experience and challenges to the Pressurizer Pilot Operated Relief Valve (PORV) and the Pressurizer Code Safety Valves shall be submitted on a monthly basis to arrive no later than the 15th of each month following the calendar month covered by the report, as follows: The signed original to the Nuclear Regulatory Commission, Document Control Desk, Washington, D. C. 20555, and one copy each to the Region III Administrator and the Davis-Besse Resident Inspector.

CORE OPERATING LIMITS REPORT

6.9.1.7 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle and any remaining part of a reload cycle for the following:

3.1.3.6 Regulating Rod Insertion Limits
3.1.3.7 Rod Program
3.1.3.8 Xenon Reactivity
3.1.3.9 Axial Power Shaping Rod Insertion Limits
3.2.1 AXIAL POWER IMBALANCE
3.2.4 QUADRANT POWER TILT

The analytical methods used to determine the core operating limits addressed by the individual Technical Specifications shall be those previously reviewed and approved by the NRC, specifically:

1) BAW-10122A Rev. 1, "Normal Operating Controls," May 1984

2) BAW-10116A, "Assembly Calculations and Fitted Nuclear Data," May 1977

3) BAW-10117P-A, "Babcock & Wilcox Version of PDQ User's Manual," January 1977

4) BAW-10118A, "Core Calculational Techniques and Procedures," December 1979

5) BAW-10124A, "FLAME 3 - A Three-Dimensional Nodal Code for Calculating Core Reactivity and Power Distributions," August 1976

6) BAW-10125A, "Verification of Three-Dimensional FLAME Code," August 1976

7) BAW-10152A, "NOODLE - A Multi-Dimensional Two-Group Reactor Simulator," June 1985

CORE OPERATING LIMITS REPORT (Continued)

8) BAW-10119, "Power Peaking Nuclear Reliability Factors," June 1977

The methodology for Rod Program received NRC approval in the Safety Evaluation dated January 11, 1990.

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revision or supplements thereto, shall be provided upon issuance for each reload cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

6.9.1.10 Routine Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with the preoperational studies, with operational controls, as appropriate, and with previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the Table and Figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3; and discussion of all analyses in which the LLD required by Table 4.12-1 was not achievable.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 144 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

By letter dated June 16, 1989 (Ref. 1), as amended by letter dated August 21, 1989 (Ref. 2), Toledo Edison Company (the licensee) proposed changes to the Technical Specifications (TS) for the Davis-Besse plant. The proposed changes would modify specifications having cycle-specific parameter limits by replacing the values of those limits with a reference to the Core Operating Limits Report (COLR) for the values of those limits. The proposed changes also include the addition of the COLR to the Definitions section and to the reporting requirements of the Administrative Controls section of the TS. Guidance on the proposed changes was developed by NRC on the basis of the review of a lead-plant proposal submitted on the Oconee plant docket by Duke Power Company. This guidance was provided to all power reactor licensees and applicants by Generic Letter 88-16, dated October 4, 1988 (Ref. 3).

2.0 EVALUATION

The licensee's proposed changes to the TS are in accordance with the guidance provided by Generic Letter 88-16 and are addressed below.

(1) The Definitions section of the TS was modified to include a definition of the Core Operating Limits Report that requires cycle/reload-specific parameter limits to be established on a unit-specific basis in accordance with an NRC approved methodology that maintains the limits of the safety analysis. The definition notes that plant operation within these limits is addressed by individual specifications.

(2) The following specifications were revised to replace the values of cycle-specific parameter limits with a reference to the COLR that provides these limits.

(a) Specification 3.1.3.1, Action c.2e

The statement concerning rod position limits has been revised to explicitly cite the COLR.

(b) Specification 3/4.1.3.6

The regulating rod group position limits for this specification are provided in the COLR.

(c) Specification 3.1.3.7

The control rod assembly location and group for this specification are provided in the COLR.

(d) Specification 3.1.3.8

The power level cut-off for xenon reactivity for this specification is specified in the regulating rod position limits provided in the COLR.

(e) Specification 3.1.3.9

The axial power shaping rods insertion limits are provided in the COLR.

(f) Specification 3/4.2.1

The axial power imbalance limits for this specification are provided in the COLR.

(g) Specification 3.2.4

The quadrant power tilt limits for this specification are provided in the COLR.

The bases of affected specifications have been modified by the licensee to include appropriate references to the COLR. Based on its review, the staff concludes that the changes to these bases are acceptable.

(3) Specification 6.9.1.7 was added to the reporting requirements of the Administrative Controls section of the TS. This specification requires that the COLR be submitted, upon issuance, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector. The report provides the values of cycle-specific parameter limits that are applicable for the current fuel cycle. Furthermore, these specifications require that the values of these limits be established using NRC approved methodologies and be consistent with all applicable limits of the safety analysis. The approved methodologies are the following:

(a) BAW-10122A Rev. 1, "Normal Operating Controls," May 1984.

(b) BAW-10116A, "Assembly Calculations and Fitted Nuclear Data," May 1977.

(c) BAW-10117P-A, "Babcock & Wilcox Version of PDQ User's Manual," January 1977.

(d) BAW-10118A, "Core Calculational Techniques and Procedures," December 1979.

(e) BAW-10124A, "FLAME 3 - A Three-Dimensional Nodal Code for Calculating Core Reactivity and Power Distributions," August 1976.

(f) BAW-10125A, "Verification of Three-Dimensional FLAME Code," August 1976.

(g) BAW-10152A, "NOODLE - A Multi-Dimensional Two-Group Reactor Simulator," June 1985.

(h) BAW-10119, "Power Peaking Nuclear Reliability Factors," June 1977.

(i) The methodology for Rod Program received NRC approval in the Safety Evaluation Report dated September 1989.

Finally, the specification requires that all changes in cycle-specific parameter limits be documented in the COLR before each reload cycle or remaining part of a reload cycle and submitted upon issuance to NRC, prior to operation with the new parameter limits.

On the basis of the review of the above items, the NRC staff concludes that the licensee provided an acceptable response to those items as addressed in the NRC guidance in Generic Letter 88-16 on modifying cycle-specific parameter limits in TS. Because plant operation continues to be limited in accordance with the values of cycle-specific parameter limits that are established using NRC approved methodologies, the NRC staff concludes that this change is administrative in nature and there is no impact on plant safety as a consequence. Accordingly, the staff finds that the proposed changes are acceptable.

As part of the implementation of Generic Letter 88-16, the staff has also reviewed a sample COLR that was provided by the licensee. On the basis of this review, the staff concludes that the format and content of the sample COLR are acceptable.

In addition to implementing the COLR in accordance with Generic Letter 88-16, the licensee presented a methodology for rod program for review. The methodology discusses the criteria and rationale used to determine the designation and location of the control rods in the shutdown, regulating, and axial power shaping rod banks. The control rod designations and locations for each rod group are specified by Specification 3.1.3.7 and provided in the COLR. On the basis of its review, the staff concludes that the methodology for rod program is acceptable.

The staff has reviewed the request by the Toledo Edison Company to modify the Technical Specifications of the Davis-Besse plant that would remove the specific values of some cycle-dependent parameters from the specifications and place the values in a Core Operating Limits Report that would be referenced by the specification. Based on this review, the staff concludes that these Technical Specification modifications are acceptable. It also concludes that the rod program methodology is acceptable.

3.0 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 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). This amendment also involves changes in recordkeeping, reporting or administrative procedures or requirements. Accordingly, with respect to these items, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR §51.22(c)(10). 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

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.

5.0 REFERENCES

1. Letter (#1668) from D.C. Shelton (TECo) to NRC, dated June 16, 1989.

2. Letter (#1691) from D.C. Shelton (TECo) to NRC, dated August 21, 1989.

3. Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," dated October 4, 1988.

Principal Contributor: D. Fieno

Dated: January 11, 1990