Docket No. 50-346

Mr. Donald C. Shelton
Senior 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, Ohio 43449

July 22, 1994

Dear Mr. Shelton:

SUBJECT: AMENDMENT NO. 189 TO FACILITY OPERATING LICENSE NO. NPF-3 (TAC NO. 85289)

The Commission has issued Amendment No. 189 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 18, 1994 as supplemented on June 20, 1994.

The amendment revises TS 2.1.2 (Reactor Core), TS 2.2.1 (Reactor Protection System Setpoints), Bases 2.1.1 and 2.1.2 (Reactor Core), Bases 2.2.1 (Reactor Protection System Instrumentation Setpoints), TS 3.2.2 and 3.2.3 (Power Distribution Limits), Bases 3/4 (Power Distribution Limits), and TS 6.9.1.7 (Administrative Controls, Core Operating Limits Report). This amendment removes cycle-specific limits from TS and relocates them in the Core Operating Limits Report.

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 Garmon West, Jr.

Garmon West, Jr., Assistant Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 189 to License No. NPF-3
2. Safety Evaluation

cc w/enclosures:
See next page

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NAME	MRushbrook	GWest:dy	RStransky	JHannon	CGrimes	U Young
DATE	6/28/94	6/30/94	6/30/94	7/1/	7/13/94	7/20/

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

July 22, 1994

Docket No. 50-346

Mr. Donald C. Shelton
Senior 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, Ohio 43449

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The amendment revises TS 2.1.2 (Reactor Core), TS 2.2.1 (Reactor Protection System Setpoints), Bases 2.1.1 and 2.1.2 (Reactor Core), Bases 2.2.1 (Reactor Protection System Instrumentation Setpoints), TS 3.2.2 and 3.2.3 (Power Distribution Limits), Bases 3/4 (Power Distribution Limits), and TS 6.9.1.7 (Administrative Controls, Core Operating Limits Report). This amendment removes cycle-specific limits from TS and relocates them in the Core Operating Limits Report.

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Sincerely,

Garmon West, Jr.

Garmon West, Jr., Assistant Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 189 to License No. NPF-3
2. Safety Evaluation

cc w/enclosures:
See next page

Mr. Donald C. Shelton
Toledo Edison Company

Davis-Besse Nuclear Power Station
Unit No. 1

cc:

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Columbus, Ohio 43235-2712

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. 189
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 18, 1994, as supplemented on June 20, 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. 189, 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

Garmon West, Jr.

Garmon West, Jr., Assistant 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: July 22, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 189

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
XVI	XVI
2-1	2-1
2-3	2-3
2-5	2-5
2-7	2-7
B 2-1	B 2-1
B 2-2	B 2-2
B 2-5	B 2-5
B 2-6	B 2-6
3/4 2-5	3/4 2-5
3/4 2-7	3/4 2-7
B 3/4 2-2	B 3/4 2-2
6-7	6-7
6-13	6-13
6-14	6-14
6-16	6-16
6-17	6-17

ADMINISTRATIVE CONTROLS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of the reactor coolant core outlet pressure and outlet temperature shall not exceed the safety limit shown in Figure 2.1-1.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of reactor coolant core outlet pressure and outlet temperature has exceeded the safety limit, be in HOT STANDBY within one hour.

REACTOR CORE

2.1.2 The combination of reactor THERMAL POWER and AXIAL POWER IMBALANCE shall not exceed the protective limit shown in the CORE OPERATING LIMITS REPORT for the various combinations of three and four reactor coolant pump operation.

APPLICABILITY: MODE 1.

ACTION:

Whenever the point defined by the combination of Reactor Coolant System flow, AXIAL POWER IMBALANCE and THERMAL POWER has exceeded the appropriate protective limit, be in HOT STANDBY within one hour, and comply with the requirements of Specification 6.7.2.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The Reactor Coolant System pressure shall not exceed 2750 psig.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2 -	Whenever the Reactor Coolant System pressure has exceeded 2750 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within one hour.
MODES 3, 4 and 5 -	Whenever the Reactor Coolant System pressure has exceeded 2750 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

Figure 2.1-2 Reactor Core Safety Limit

DELETED

Table 2.2-1 Reactor Protection System Instrumentation Trip Setpoints

Functional unit	Trip setpoint	Allowable values
1. Manual reactor trip	Not applicable.	Not applicable.
2. High flux	≤104.94% of RATED THERMAL POWER with four pumps operating	≤104.94% of RATED THERMAL POWER with four pumps operating#
	≤80.6% of RATED THERMAL POWER with three pumps operating	≤80.6% of RATED THERMAL POWER with three pumps operating #
3. RC high temperature	≤618°F	≤618°F#
4. Flux -- Δflux/flow[1]	Pump trip setpoints not to exceed the limit lines shown in the CORE OPERATING LIMITS REPORT for four and three pump operation.	Pump allowable values not to exceed the limit lines shown in in the CORE OPERATING LIMITS REPORT for four and three pump operation.
5. RC low pressure[1]	≥1900.0 psig	≥1900.0 psig* ≥1900.0 psig**
6. RC high pressure	≤2355 psig	≤2355.0 psig* ≤2355.0 psig**
7. RC pressure-temperature[1]	≥(16.00 T_{out} °F - 7957.5) psig	≥(16.00 T_{out} °F - 7957.5) psig#
8. High flux/number of RC pumps on[1]	≤55.1% of RATED THERMAL POWER with one pump operating in each loop	≤55.1% of RATED THERMAL POWER with one pump operating in each loop#
	≤0.0% of RATED THERMAL POWER with two pumps operating in one loop and no pumps operating in the other loop	≤0.0% of RATED THERMAL POWER with two pumps operating in one loop and no pumps operating in the other loop#
	≤0.0% of RATED THERMAL POWER with no pumps operating or only one pump operating	≤0.0% of RATED THERMAL POWER with no pumps operating or only one pump operating#
9. Containment pressure high	≤4 psig	≤4 psig#

Figure 2.2-1 Trip Setpoint for Flux -- ΔFlux/Flow

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BASES

2.1.1 AND 2.1.2 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime would result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB using critical heat flux (CHF) correlations. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The B&W-2 and BWC CHF correlations have been developed to predict DNB for axially uniform and non-uniform heat flux distributions. The B&W-2 correlation applies to Mark-B fuel and the BWC correlation applies to all B&W fuel with zircaloy spacer grids. The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30 (B&W-2) and 1.18 (BWC). The value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curve presented in Figure 2.1-1 represents the conditions at which a minimum DNBR equal to or greater than the correlation limit is predicted for the maximum possible thermal power 112% when the reactor coolant flow is 380,000 GPM, which is approximately 108% of design flow rate for four operating reactor coolant pumps. (The minimum required measured flow is 389,500 GPM). This curve is based on the design hot channel factors with potential fuel densification and fuel rod bowing effects.

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 withdrawal, and form the core DNBR design basis.

The CORE OPERATING LIMITS REPORT includes curves for protective limits for AXIAL POWER IMBALANCE and for nuclear overpower based on reactor coolant system flow. A protective limit is a cycle-specific limit that ensures that a safety limit is not exceeded by requiring operation within both the cycle design (operating) limits and the Reactor Protection System setpoints. These protective limit curves reflect the more restrictive of two thermal limits and account for the effects of potential fuel densification and potential fuel rod bow:

1. The DNBR limit produced by a design nuclear power peaking factor as described in the CORE OPERATING LIMITS REPORT or the combination of the radial peak, axial peak, and position of the axial peak that yields no less than the DNBR limit.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limits for all fuel designs during the operating cycle are listed in the CORE OPERATING LIMITS REPORT.

Power peaking is not a directly observable quantity and therefore limits have been established on the basis of the reactor power imbalance produced by the power peaking.

The specified flow rates for the CORE OPERATING LIMITS REPORT curves for protective limits for AXIAL POWER IMBALANCE and for nuclear overpower based on reactor coolant system flow correspond to the analyzed minimum flow rates with four pumps and three pumps, respectively.

The curve of Figure 2.1-1 is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in BASES Figure 2.1. The curves of BASES Figure 2.1 represent the conditions at which a minimum DNBR equal to the DNBR limit is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation or the local quality at the point of minimum DNBR is equal to the corresponding DNB correlation quality limit (+22% (B&W-2) or +26% (BWC)), whichever condition is more restrictive.

RC High Temperature

The RC high temperature trip $\leq$ 618°F prevents the reactor outlet temperature from exceeding the design limits and acts as a backup trip for all power excursion transients.

Flux -- ΔFlux/Flow

The power level trip setpoint produced by the reactor coolant system flow is based on a flux-to-flow ratio which has been established to accommodate flow decreasing transients from high power where protection is not provided by the high flux/number of reactor coolant pumps on trips.

The power level trip setpoint produced by the power-to-flow ratio provides both high power level and low flow protection in the event the reactor power level increases or the reactor coolant flow rate decreases. The power level setpoint produced by the power-to-flow ratio provides overpower DNB protection for all modes of pump operation. For every flow rate there is a maximum permissible power level, and for every power level there is a minimum permissible low flow rate.

For safety calculations the instrumentation errors for the power level were used. Full flow rate is defined as the flow calculated by the heat balance at 100% power. At the time of the calibration the RCS flow will be greater than or equal to the value in Table 3.2-2.

The AXIAL POWER IMBALANCE boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kW/ft limits or DNBR limits. The AXIAL POWER IMBALANCE reduces the power level trip produced by a flux-to-flow ratio such that the boundaries of the figure in the CORE OPERATING LIMITS REPORT are produced.

RC Pressure - Low, High, and Pressure Temperature

The high and low trips are provided to limit the pressure range in which reactor operation is permitted.

During a slow reactivity insertion startup accident from low power or a slow reactivity insertion from high power, the RC high pressure setpoint is reached before the high flux trip setpoint. The trip setpoint for RC high pressure, 2355 psig, has been established to maintain the system pressure below the safety limit, 2750 psig, for any design transient. The RC high pressure trip is backed up by the pressurizer code safety valves for RCS over pressure protection, and is therefore set lower than the set pressure for these valves, $\leq$ 2525 psig. The RC high pressure trip also backs up the high flux trip.

The RC low pressure, 1900.0 psig, and RC pressure-temperature ($16.00\ T_{out}$ - 7957.5) psig, trip setpoints have been established to maintain the DNB ratio greater than or equal to the minimum allowable DNB ratio for those design accidents that result in a pressure reduction. It also prevents reactor operation at pressures below the valid range of DNB correlation limits, protecting against DNB.

High Flux/Number of Reactor Coolant Pumps On

In conjunction with the flux - Δflux/flow trip the high flux/number of reactor coolant pumps on trip prevents the minimum core DNBR from decreasing below the minimum allowable DNB ratio by tripping the reactor due to the loss of reactor coolant pump(s). The pump monitors also restrict the power level for the number of pumps in operation.

POWER DISTRIBUTION LIMITS

NUCLEAR HEAT FLUX HOT CHANNEL FACTOR - F_Q

LIMITING CONDITION FOR OPERATION

3.2.2 F_Q shall be within the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODE 1

ACTION:

With F_Q exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% F_Q exceeds the limit within 15 minutes and similarly reduce the high flux trip setpoint and flux-Δ flux-flow trip setpoint within 4 hours.

b. Demonstrate through incore mapping that F_Q is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a or b, above; subsequent POWER OPERATION may proceed provided that F_Q is demonstrated through incore mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL POWER and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.2.1 F_Q shall be determined to be within its limit by using the incore detectors to obtain a power distribution map:

POWER DISTRIBUTION LIMITS

NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be within the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODE 1.

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% that $F^N_{\Delta H}$ exceeds the limit within 15 minutes and similarly reduce the High Flux Trip Setpoint and Flux - ΔFlux - Flow Trip Setpoint within 4 hours.

b. Demonstrate through in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a or b, above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL POWER and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

$F^N_{\Delta H}$ 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 by compliance with the protective and operating limits in the CORE OPERATING LIMITS REPORT.

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 ±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.

d. Review of proposed tests or experiments determined to involve an unreviewed safety question as defined in 10 CFR 50.59.

e. Review of reports of violations of codes, regulations, orders, Technical Specifications, or Operating License requirements having nuclear safety significance or reports of abnormal degradation of systems designed to contain radioactive material.

f. Review of all proposed changes to the Technical Specifications or the Operating License.

g. Deleted

h. Review of reports of significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect plant safety.

i. Review of the Industrial Security Plan, the Security Training and Qualification Plan, and the Security Contingency Plan, and changes thereto.

j. Review of the Davis-Besse Emergency Plan and changes thereto.

k. Review of items which may constitute potential nuclear safety hazards as identified during review of facility operations.

l. Investigations or analyses of special subjects as requested by the Company Nuclear Review Board.

m. Review of all REPORTABLE EVENTS.

n. Review of all Safety Limit Violation Reports and Protective Limit Violation Reports (Section 6.7).

o. Review of any unplanned, accidental or uncontrolled radioactive releases, evaluation of the event, ensurance that remedial action is identified to prevent recurrence, review of a report covering the evaluation and forwarding of the report to the Plant Manager and to the CNRB.

p. Review of the changes to the OFFSITE DOSE CALCULATION MANUAL.

q. Review of the changes to the PROCESS CONTROL PROGRAM.

r. Review of the Annual Radiological Environmental Operating Report.

s. Review of the Radioactive Effluent Release Report.

t. Review of the Fire Protection Program and changes thereto.

6.7 SAFETY LIMIT VIOLATION OR PROTECTIVE LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The facility shall be placed in at least HOT STANDBY within one hour.

b. The Safety Limit violation shall be reported to the NRC Operations Center by telephone as soon as possible and in all cases within one hour. In addition the Vice President, Nuclear and the CNRB shall be notified within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the SRB. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the CNRB and the Vice President, Nuclear within 14 days of the violation.

6.7.2 The following actions shall be taken in the event the Protective Limit of Specification 2.1.2 is violated:

a. The facility shall be placed in at least HOT STANDBY within one hour.

b. The Protective Limit violation shall be reported to the NRC Operations Center by telephone as soon as possible and in all cases within one hour. In addition the Vice President, Nuclear and the CNRB shall be notified within 24 hours.

c. A Protective Limit Violation Report shall be prepared. The report shall be reviewed by the SRB. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Protective Limit Violation Report shall be submitted to the CNRB and the Vice President, Nuclear within 14 days of the violation.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November, 1972.

b. Refueling operations.

6.8 PROCEDURES AND PROGRAMS (Continued)

c. Surveillance and test activities of safety related equipment.

d. Industrial Security Plant implementation.

e. Davis-Besse Emergency Plan implementation.

f. Fire Protection Program implementation.

g. The radiological environmental monitoring program.

h. The Process Control Program.

i. Offsite Dose Calculation Manual implementation.

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed and approved prior to implementation as set forth in 6.5.3 above.

6.8.3 (deleted)

6.8.4 The following programs shall be established, implemented and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include makeup, letdown, seal injection, seal return, low pressure injection, containment spray, high pressure injection, waste gas, primary sampling and reactor coolant drain systems. The program shall include the following:

(i) Preventive maintenance and/or periodic visual inspection requirements, and

(ii) Integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

(i) Training of personnel,

(ii) Procedures for monitoring, and

(iii) Provisions for maintenance of sampling and analysis equipment.

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:

2.1.2	AXIAL POWER IMBALANCE Protective Limits for Reactor Core Specification 2.1.2
2.2.1	Trip Setpoint for Flux --ΔFlux/Flow for Reactor Protection System Setpoints Specification 2.2.1
3.1.1.3c	Negative Moderator Temperature Coefficient Limit
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
3.2.2	Nuclear Heat Flux Hot Channel Factor, F_Q
3.2.3	Nuclear Enthalpy Rise Hot Channel Factor, $F^N_{\Delta H}$
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, as described in BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses", or any other new NRC-approved analytical methods used to determine core operating limits that are not yet referenced in the applicable approved revision of BAW-10179P-A. The applicable approved revision number for BAW-10179P-A at the time the reload analyses are performed shall be identified in the CORE OPERATING LIMITS REPORT. The CORE OPERATING LIMITS REPORT shall also list any new NRC-approved analytical methods used to determine core operating limits that are not yet referenced in the applicable approved revision of BAW-10179P-A.

CORE OPERATING LIMITS REPORT (Continued)

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 189 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 18, 1994, as supplemented by letter dated June 20, 1994 (Ref. 2), Toledo Edison Company (the licensee) proposed changes to the Technical Specifications (TS) for the Davis-Besse Nuclear Power Station (DBNPS), Unit 1. The proposed changes would modify additional specifications having cycle-specific parameter limits by replacing the values of those limits with a reference to a Core Operating Limits Report (COLR) for the values of those limits in accordance with the guidance provided in Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications." The use of the COLR for DBNPS was previously approved by the NRC.

The June 20, 1994, letter provided supplemental information that did not change the intitial proposed no significant hazards consideration determination.

2.0 EVALUATION

The proposed changes to the TS are in accordance with the guidance provided by GL 88-16 and are addressed below.

(1) The following additional specifications due to the new approved methodology for reload analyses were revised to replace the values of cycle-specific parameter limits with reference to the COLR that provides these limits.

(a) Specification 2.1.2

Axial power imbalance protective limits in Figure 2.1-2 for this specification are specified in the COLR.

(b) Specification 2.2.1

The reactor protective system trip setting limits in Table 2.2.1 Reactor Protection System Instrumentation Trip Setpoints and Figure 2.2.1 Trip Setpoint for flux -- ΔFlux/Flow for this specification are specified in the COLR.

(c) Specification 3.2.2

The nuclear hot channel factor - F_Q limit for this specification is specified in the COLR.

(d) Specification 3.2.3

The nuclear enthalpy rise hot channel factor - $F^N_{\Delta H}$ limit for this specification is specified in the COLR.

The bases of affected specifications have been modified by the licensee to include appropriate reference to the COLR. Based on our review, we conclude that the changes to these bases are acceptable.

(2) Specification 6.9.1.7 is revised to include currently proposed TS changes under the reporting requirements of the Administrative Control 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, this specification requires that NRC-approved methodologies be used in establishing the values of these limits for the relevant specifications and that the values be consistent with all applicable limits of the safety analysis. Those previously reviewed and approved methodologies in COLR TS are deleted and replaced by BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analysis" and its future NRC-approved BAW-10179P-A revision. However, the applicable approved revision number for BAW-10179A shall be listed and identified in the COLR at the time the reload analyses are performed.

On the bases 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 has no impact on plant safety. Accordingly, the staff finds that the proposed changes are acceptable.

We have reviewed the request by Toledo Edison Company to revise the TS of the Davis-Besse Nuclear Power Station, Unit 1 by removing the specific values of additional cycle-dependent parameters from the TS and placing the values in a COLR referenced by the specifications. Based on the review, we conclude that these revisions 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. 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 (59 FR 22014). The amendment also changes a reporting or recordkeeping requirement. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (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 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: T. Huang

Date: July 22, 1994