June 4, 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. 149 TO FACILITY OPERATING LICENSE NO. NPF-3 (TAC NO. 69569)

The Commission has issued Amendment No. 149 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 December 1, 1988.

This amendment reduces the reactor protection system reactor coolant low pressure trip setpoint from greater than or equal to 1983.4 psig to greater than or equal to 1900 psig and correspondingly modifies the reactor protection system variable low pressure setpoint.

By letter dated February 5, 1990, you submitted revised pages for the Bases sections of the Technical Specifications resulting from the core reload for Cycle 7. Specifically, the changes to the Bases incorporate the BWC Critical Heat Flux correlation for the fuel batch being loaded for Cycle 7, in addition to the B&W 2 correlation for the remaining fuel, for Departure from Nucleate Boiling Ratio limit determination. Also, references to Batch 1 fuel are deleted since they are no longer in the core. As stated in 10 CFR 50.36(a), the Bases are not a part of the Technical Specifications and therefore these revised pages do not constitute a change to the Technical Specifications. Your evaluation pursuant to 10 CFR 50.59 concluded that no unreviewed safety question was involved in these revisions. Therefore, no prior approval by the NRC was required. These pages are being issued with this amendment as an administrative convenience. The NRC review of Attachment 4 to the February 5, 1990, letter "TECO Mark-B7 FA Seismic and LOCA Analysis" (B&W Document Identifier 86-1177306-00) is the subject of a separate NRC safety evaluation and letter.

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

cc: See next page

Office:	LA/PDIII-3	PM/PDIII-3	SRXB	PDIII-3	OGC-WF1
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Date:	5/16/90	5/16/90	5/17/90	5/18/90	5/30/90

DOCUMENT NAME: 69569 AMD

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

June 4, 1990

Docket No. 50-346

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Vice President, Nuclear
Toledo Edison Company
Edison Plaza - Stop 712
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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,

Thomas V. Wambach

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Division of Reactor Projects - III, IV, V & Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 149 to License No. NPF-3
2. Safety Evaluation

cc: See next page

Mr. Donald C. Shelton
Toledo Edison Company

Davis-Besse Nuclear Power Station
Unit No. 1

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

TOLEDO EDISON COMPANY

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. 149
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 December 1, 1988, 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.149 , 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 60 days after issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

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: June 4, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 149

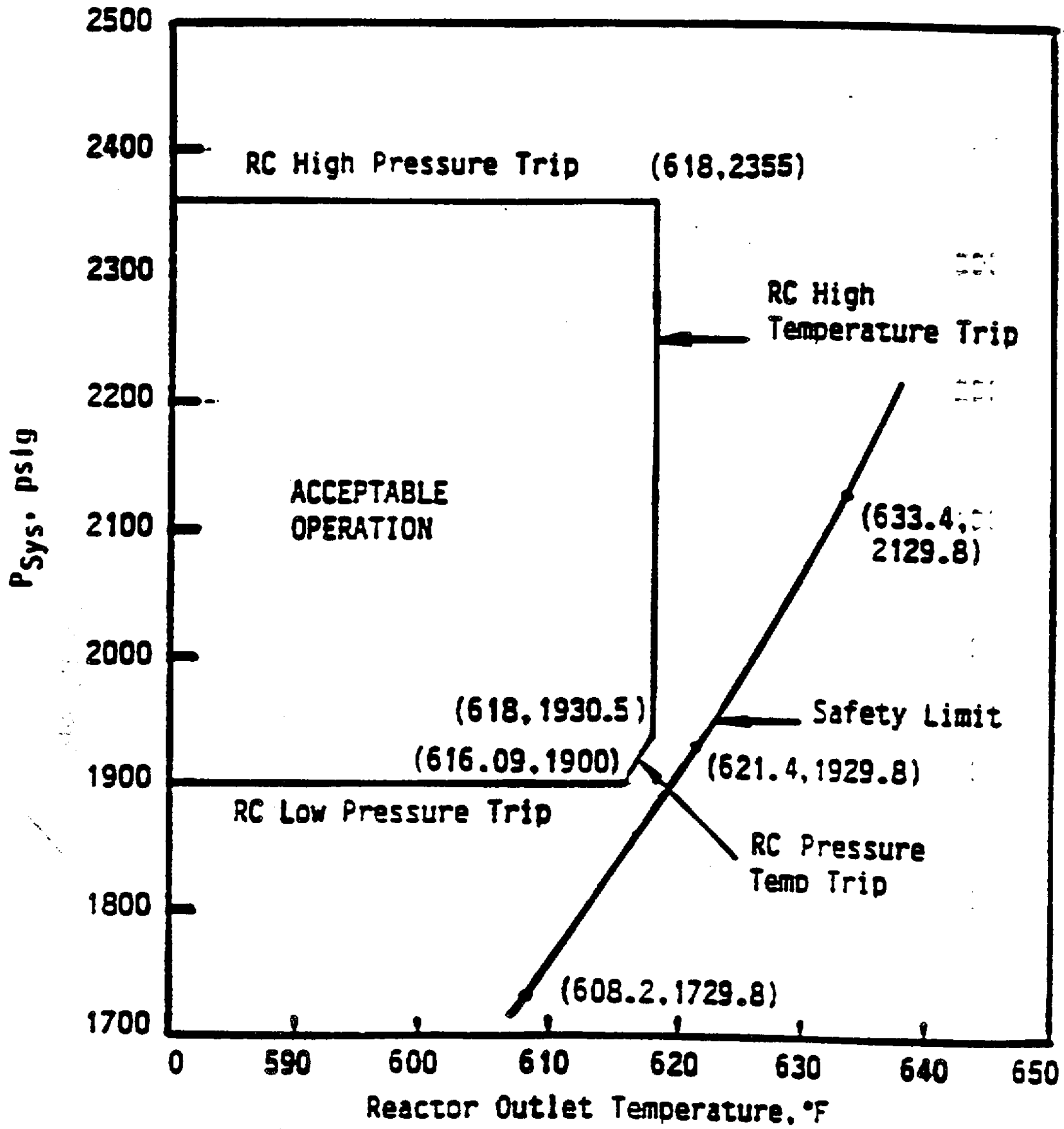
FACILITY OPERATING LICENSE NO. NPF-3

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
2-2	2-2
2-5	2-5
B 2-1	B 2-1
B 2-2	B 2-2
B 2-3	B 2-3
B 2-6	B 2-6
B 3/4 2-1	B 3/4 2-1
B 3/4 2-3	B 3/4 2-3
B 3/4 4-1	B 3/4 4-1

Figure 2.1-1 Reactor Core Safety Limit



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 safety limit shown in Figure 2.1-2 for the various combinations of two, 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 safety limit, be in HOT STANDBY within one hour.

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.

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	$\leq$104.94% of RATED THERMAL POWER with four pumps operating	$\leq$104.94% of RATED THERMAL POWER with four pumps operating#
	$\leq$80.6% of RATED THERMAL POWER with three pumps operating	$\leq$80.6% of RATED THERMAL POWER with three pumps operating#
3. RC high temperature	$\leq$618°F	$\leq$618°F#
4. Flux -- Δflux/flow(1)	Four pump trip setpoint not to exceed the limit line of Figure 2.2-1. For three pump operation, see Figure 2.2-1	Four pump allowable values not to exceed the limit line of Figure 2.2-1#. For three pump operation, see Figure 2.2-1
5. RC low pressure(1)	$\geq$1900.0 psig	$\geq$1900.0 psig* $\geq$1900.0 psig**
6. RC high pressure	$\leq$2355 psig	$\leq$2355.0 psig* $\leq$2355.0 psig**
7. RC pressure-temperature(1)	$\geq$(16.00 T_{out} °F - 7957.5) psig	$\geq$(16.00 T_{out} °F - 7957.5) psig#
8. High flux/number of RC pumps on(1)	$\leq$55.1% of RATED THERMAL POWER with one pump operating in each loop	$\leq$55.1% of RATED THERMAL POWER with one pump operating in each loop#
	$\leq$0.0% of RATED THERMAL POWER with two pumps operating in one loop and no pumps operating in the other loop	$\leq$0.0% of RATED THERMAL POWER with two pumps operating in one loop and no pumps operating in the other loop#
	$\leq$0.0% of RATED THERMAL POWER with no pumps operating or only one pump operating	$\leq$0.0% of RATED THERMAL POWER with no pumps operating or only one pump operating#
9. Containment pressure high	$\leq$4 psig	$\leq$4 psig#

Table 2.2-1. (Cont'd)

(1)Trip may be manually bypassed when RCS pressure ≤1820 psig by actuating shutdown bypass provided that:

a. The high flux trip setpoint is ≤5% of RATED THERMAL POWER.

b. The shutdown bypass high pressure trip setpoint of ≤1820 psig is imposed.

c. The shutdown bypass is removed when RCS pressure >1820 psig.

*Allowable value for CHANNEL FUNCTIONAL TEST.

**Allowable value for CHANNEL CALIBRATION.

#Allowable value for CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION.

2.1 SAFETY LIMITS

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 following hot channel factors with potential fuel densification and fuel rod bowing effects:

$$F_Q = 2.83;\ F^N_{\Delta H} = 1.71;\ F^N_Z = 1.65$$

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 curves of Figure 2.1-2 are based on 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 nuclear power peaking factor of F_Q = 2.83 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 limit is 20.5 kw/ft for all fuel in the core.

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 two curves of Figure 2.1-2 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.

For the curve of BASES Figure 2.1, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 (B&W-2) or 1.18 (BWC) and a local quality at the point of minimum DNBR less than +22% (B&W-2) or +26% (BWC) for that particular reactor coolant pump situation. The DNBR curve for three pump operation is less restrictive than the four pump curve.

2.1.3 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Boiler and Pressure Vessel Code which permits a maximum transient pressure of 110%, 2750 psig, of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to ANSI B 31.7, 1968 Edition, which permits a maximum transient pressure of 110%, 2750 psig, of component design pressure. The Safety Limit of 2750 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The reactor protection system instrumentation trip setpoints specified in Table 2.2-1 are the values at which the reactor trips are set for each parameter. The trip setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits.

The shutdown bypass provides for bypassing certain functions of the reactor protection system in order to permit control rod drive tests, zero power PHYSICS TESTS and certain startup and shutdown procedures. The purpose of the shutdown bypass high pressure trip is to prevent normal operation with shutdown bypass activated. This high pressure trip setpoint is lower than the normal low pressure trip setpoint so that the reactor must be tripped before the bypass is initiated. The high flux trip setpoint of $\leq$5.0% prevents any significant reactor power from being produced. Sufficient natural circulation would be available to remove 5.0% of RATED THERMAL POWER if none of the reactor coolant pumps were operating.

Manual Reactor Trip

The manual reactor trip is a redundant channel to the automatic reactor protection system instrumentation channels and provides manual reactor trip capability.

High Flux

A high flux trip at high power level (neutron flux) provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry.

During normal station operation, reactor trip is initiated when the reactor power level reaches 104.94% of rated power. Due to transient overshoot, heat balance, and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which was used in the safety analysis.

LIMITING SAFETY SYSTEM SETTINGS

BASES

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. Examples of typical power level and low flow rate combinations for the pump situations of Table 2.2-1 that would result in a trip are as follows:

1. Trip would occur when four reactor coolant pumps are operating if power is 108.0% and reactor coolant flow rate is 100% of full flow rate, or flow rate is 92.59% of full flow rate and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is 80.68% and reactor coolant flow rate is 74.7% of full flow rate, or flow rate is 69.44% of full flow rate and power is 75%. Note that the value of 80.6% in Figure 2.2-1 was truncated from the calculated value of 80.68%.

For safety calculations the instrumentation errors for the power level were used. Full flow rate in the above two examples 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 Figure 2.2-1 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.

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 DNBR in the core $\geq$ the minimum allowable DNB ratio 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 ±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.

b. The measurement of enthalpy rise hot channel factor, $F^N_{\Delta H}$, shall be increased by 5 percent to account for measurement error.

For Condition II events, the core is protected from exceeding the values given in the bases to specification 2.1 locally, and from going below the minimum allowable DNB ratio by automatic protection on power, AXIAL POWER IMBALANCE pressure and temperature. Only conditions 1 through 3, above, are mandatory since the AXIAL POWER IMBALANCE is an explicit input to the reactor protection system.

The QUADRANT POWER TILT limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The QUADRANT POWER TILT limit at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts. In the event the tilt is not corrected, the margin for uncertainty on F_Q is reinstated by reducing the power by 2 percent for each percent of tilt in excess of the limit.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the FSAR initial assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR greater than the minimum allowable DNB ratio throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate using delta P instrumentation is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with both reactor coolant loops in operation, and maintain DNBR above the minimum allowable DNB ratio during all normal operations and anticipated transients. With one reactor coolant pump not in operation in one loop, THERMAL POWER is restricted by the Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE, ensuring that the DNBR will be maintained above the minimum allowable DNB ratio 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 equal to the DNB correlation quality limit, whichever is more restrictive.

In MODE 3 when RCS pressure or temperature is higher than the decay heat removal system's design condition (i.e. 330 psig and 350°F), a single reactor coolant loop provides sufficient heat removal capability. The remainder of MODE 3 as well as in MODES 4 and 5 either a single reactor coolant loop or a DHR loop will be sufficient for decay heat removal; but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two DHR loops to be OPERABLE.

Natural circulation flow or the operation of one DHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capacity of operator recognition and control.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psig. Each safety valve is designed to relieve 336,000 lbs per hour of saturated steam at the valve's setpoint.

The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating DHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from any transient.

The relief capacity of the decay heat removal system relief valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that this relief valve is not OPERABLE, reactor coolant system pressure, pressurizer level and make up water inventory is limited and the capability of the high pressure injection system to

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 149 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 application dated December 1, 1988, Toledo Edison Company (the licensee) proposed an amendment to the Davis-Besse Nuclear Power Station (DBNPS) operating license that would revise the Technical Specifications to reduce the reactor protection system (RPS) reactor coolant low pressure trip setpoint from greater than or equal to 1983.4 psig to greater than or equal to 1900 psig. The RPS variable low pressure trip (VLPT) setpoint was also proposed to be correspondingly modified.

2.0 DISCUSSION

Amendment 123 to the DBNPS operating license approving the Cycle 6 operation revised the pressure-temperature safety limit such that the margin between the safety limit and the trip setpoints was significantly increased. The B&W Owners' Group (BWOG) Safety and Performance Improvement Program (SPIP) has identified that up through 1986 there have been five reactor trips on low Reactor Coolant System (RCS) pressure at all B&W plants. Four of these trips have occurred at Davis-Besse. The 1983.4 psig setpoint at Davis-Besse for the RCS low pressure trip has been identified by the BWOG SPIP as a contributing factor in these trips. The other B&W plants have a low pressure RPS setpoint between 1800 and 1900 psig. The VLPT setpoint change is required to accommodate the change in the RCS low pressure trip setpoint and to provide increased operational margin by using some of the increased margin gained from the pressure-temperature safety limit revision in Amendment 123. This resulted from the use of cross-flow methodology for the thermal hydraulic analysis for Cycle 6.

3.0 EVALUATION

The licensee appropriately addressed the areas of core thermal hydraulics, loss of coolant accident (LOCA), and other Updated Safety Analysis Report (USAR) Chapter 15 analyses. The low pressure trip and variable low pressure trip protect the fuel from overheating caused by loss of heat transfer from the fuel resulting from a departure from nucleate boiling (DNB). This occurs when a film of steam forms on the fuel rod which cannot pass heat as efficiently. The pressure-temperature safety limit is set to ensure with 95 percent probability to a 95 percent confidence level that DNB will not occur. The trip setpoints are then selected taking due account of instrument tap location, instrument error allowance, and appropriate margin. The revised RPS low pressure trip setpoint and the VLPT setpoint have no effect on the thermal hydraulic analytical basis for DNB protection provided by the safety limit for normal plant operation. These trips protect against loss of RCS pressure caused by excess cooling of the RCS or loss of coolant inventory from the RCS. For a small range of RCS pressure (1900 psig to 1930.5 psig), the VLPT also protects against excess RCS temperature caused by mismatch of RCS power and secondary side load.

The licensee states and we agree that this reduction in low pressure trip setpoints has no effect on the large break LOCA analysis since no credit is taken for reactor trip in that analysis. For small break LOCAs, where credit is taken for reactor trip, the licensee reports that the time delay in reactor trip caused by the reduction of the setpoint amounts to less than 2 seconds for a break size of 0.04 $ft.^2$ and, of course, less than that for larger size breaks. Since feedwater pumps would still be running during this time, no RCS heatup would occur. The staff agrees that there is no significant impact on the small break LOCA analysis.

The other accidents addressed by the licensee are: break in primary line that penetrates containment, control rod assembly misalignment, feedwater system malfunctions, excessive load increase, steam line breaks, and steam generator tube rupture. The NRC staff agrees that these are the accidents potentially affected by a reduction in these setpoints. The bounding case of a break in a primary coolant line that penetrates containment is a complete severance of the 2½-inch letdown line outside containment upstream of the letdown control valves downstream of the letdown cooler. With the lower low pressure reactor trip setpoint, some additional primary coolant discharge occurs for this event. However, the resultant doses are still well below 10 percent of the 10 CFR Part 100 requirements. The excessive load increase accident is bounded by the steamline break. For lesser magnitude load increases, the analysis performed and approved by the NRC for Cycle 6 used a low pressure setpoint of the proposed 1900 psig. For the steamline break, the lower pressure setpoint reduces the positive reactivity addition to the core caused by cooling since the reactor adds additional heat during the delayed trip. There is some additional energy discharged to the containment for a steamline break inside containment. However, using conservative assumptions, the peak pressure is 22.6 psig, which is below the LOCA pressure and the design pressure for the containment building (36 psig). The remaining accidents were not adversely affected by this reduction in setpoint. The NRC staff agrees with these findings.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a changes 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 changes 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 the 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). 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, 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.

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Dated: June 4, 1990