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Docket No. 50-346

Mr. Lowell E. Roe
Vice President, Facilities
Development
Toledo Edison Company
Edison Plaza
300 Madison Avenue
Toledo, Ohio 43642

Dear Mr. Roe:

The Commission has issued the enclosed Amendment No. 16 to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station, Unit No. 1. The amendment consists of changes to the Technical Specifications in response to your application dated October 23, 1978.

The amendment changes the Technical Specifications concerning the Reactor Core Safety Limit Curve, the Reactor Protection System Instrumentation Trip Setpoints and the maximum Reactor Coolant Hot Leg Temperature for departure from nucleate boiling margin. This amendment also makes minor editorial corrections to Technical Specification Figure 2.1-2 and Table 5.7-1.

7908090678

As a result of the Three Mile Island Nuclear Incident of March 28, 1979, we have a general review of B&W reactors underway which may result in changes in criteria for analysis of transients. Such changes in criteria could result in different Technical Specification requirements. Appropriate action will be taken as may be required to implement any needed Technical Specification changes resulting from this review. Also on May 17, 1979, the Commission issued andOrder confirming your commitment to keep the facility shutdown until certain modifications and changes to procedures are implemented to decrease the likelihood of occurrence of an event similar to the Three Mile Island Incident. The Order requires that Davis-Besse Nuclear Power Station, Unit No. 1 be maintained in a shutdown condition until satisfactory completion of the items in the Order have been confirmed by the Director, Office of Nuclear Reactor Regulation. Therefore, upon issuance of this amendment, operation of the facility can be commenced only after confirmation of completion of the items in the Order.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original signed by
Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosures:
1. Amendment No. 16
2. Safety Evaluation
3. Notice

cc w/enclosures: See next page

*See previous yellow for concurrence

CP / GD

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Docket No. 50-346

Mr. Lowell E. Roe
Vice President, Facilities
Development
Toledo Edison Company
Edison Plaza
300 Madison Avenue
Toledo, Ohio 43652

Dear Mr. Roe:

The Commission has issued the enclosed Amendment No. to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station, Unit No. 1. The amendment consists of changes to the Technical Specifications in response to your application dated October 23, 1978.

The amendment changes the Technical Specifications concerning the Reactor Core Safety Limit Curve, the Reactor Protection System Instrumentation Trip Setpoints and the maximum Reactor Coolant Hot Leg Temperature for departure from nucleate boiling margin. This amendment also makes minor editorial corrections to Technical Specification Figure 2.1-2 and Table 5.7-1.

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Robert W. Reid, Chief
Operating Reactors Branch #4
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cc w/enclosures: See next page

OFFICE	ORB#4:DOR	ORB#4:DOR	C-ORB#4:DOR	STSG	AD-E&P:DOR	OELD
SURNAME	RIngram/cb	GVissing	RReid		B. Grimes	
DATE	5/10/79	5/ /79	5/23/79	5/23/79	5/ /79	5/ /79

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

June 7, 1979

Docket No. 50-346

Mr. Lowell E. Roe
Vice President, Facilities
Development
Toledo Edison Company
Edison Plaza
300 Madison Avenue
Toledo, Ohio 43642

Dear Mr. Roe:

The Commission has issued the enclosed Amendment No. 16 to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station, Unit No. 1. The amendment consists of changes to the Technical Specifications in response to your application dated October 23, 1978.

The amendment changes the Technical Specifications concerning the Reactor Core Safety Limit Curve, the Reactor Protection System Instrumentation Trip Setpoints and the maximum Reactor Coolant Hot Leg Temperature for departure from nucleate boiling margin. This amendment also makes minor editorial corrections to Technical Specification Figure 2.1-2 and Table 5.7-1.

As a result of the Three Mile Island Nuclear Incident of March 28, 1979, we have a general review of B&W reactors underway which may result in changes in criteria for analysis of transients. Such changes in criteria could result in different Technical Specification requirements. Appropriate action will be taken as may be required to implement any needed Technical Specification changes resulting from this review. Also on May 17, 1979, the Commission issued an Order confirming your commitment to keep the facility shutdown until certain modifications and changes to procedures are implemented to decrease the likelihood of occurrence of an event similar to the Three Mile Island Incident. The Order requires that Davis-Besse Nuclear Power Station, Unit No. 1 be maintained in a shutdown condition until satisfactory completion of the items in the Order have been confirmed by the Director, Office of Nuclear Reactor Regulation. Therefore, upon issuance of this amendment, operation of the facility can be commenced only after confirmation of completion of the items in the Order.

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Sincerely,

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosures:
1. Amendment No. 16
2. Safety Evaluation
3. Notice

cc w/enclosures: See next page

Toledo Edison Company

cc w/enclosure(s):

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Commissioners of Ottawa County
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U. S. Environmental Protection Agency
Federal Activities Branch
Region V Office
ATTN: EIS COORDINATOR
230 South Dearborn Street
Chicago, Illinois 60604

cc w/enclosure(s) and incoming
dtd.: 10/23/78

Ohio Department of Health
ATTN: Director of Health
450 East Town Street
Columbus, Ohio 43216

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

THE 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. 16
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 October 23, 1978, 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.

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2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-3 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 16, 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 7, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 16

FACILITY OPERATING LICENSE NO. NPF 3

DOCKET NO. 50-346

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages as indicated. 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.

Pages

2-3
2-5
2-7
2-8
B 2-5
3/4 2-14
3/4 4-1
5-8

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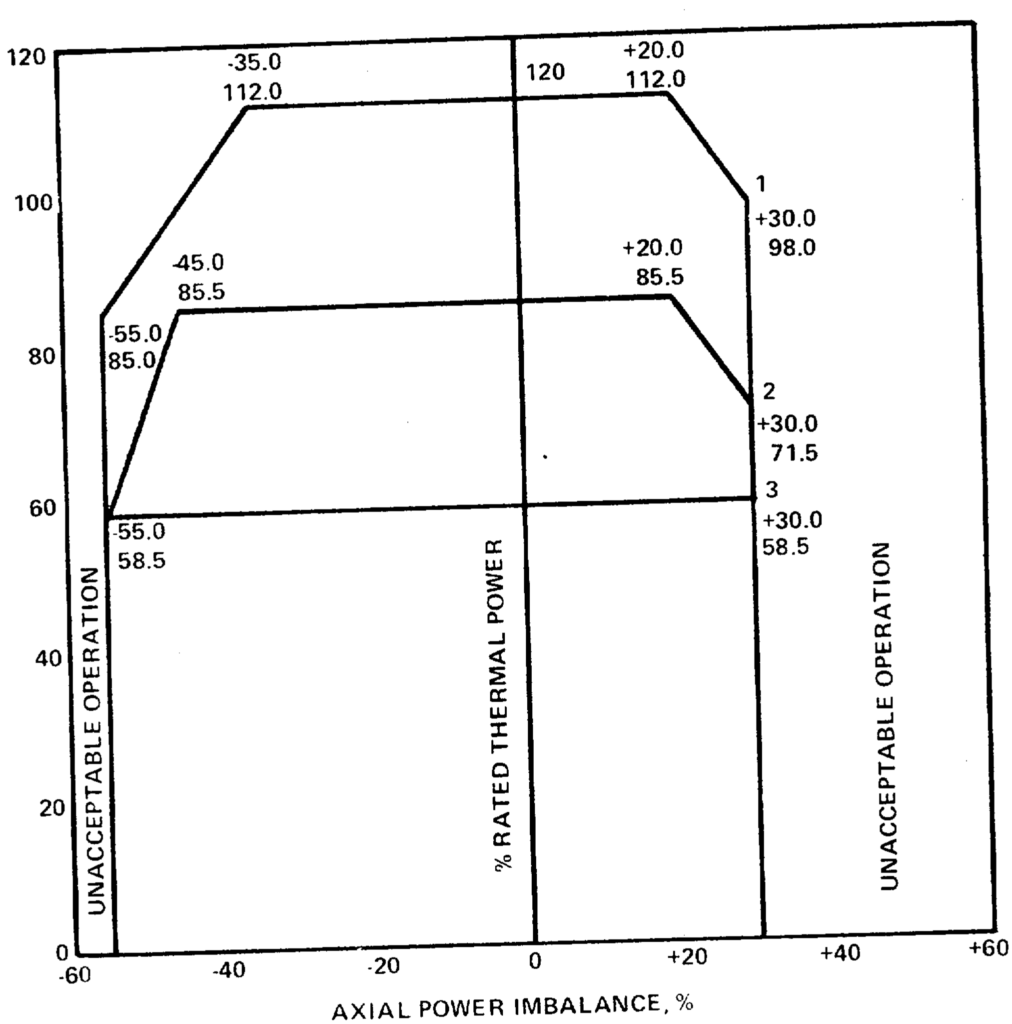

CURVE	REACTOR COOLANT FLOW (GPM)
1	387,200
2	290,100
3	191,000

Figure 2.1-2 Reactor Core Safety Limit

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTION SYSTEM SETPOINTS

2.2.1 The Reactor Protection System instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a Reactor Protection System instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

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	$\leq$ 105.5% of RATED THERMAL POWER with four pumps operating	$\leq$ 105.6% of RATED THERMAL POWER with four pumps operating#
	$\leq$ 78.3% of RATED THERMAL POWER with three pumps operating	$\leq$ 78.4% of RATED THERMAL POWER with three pumps operating#
	$\leq$ 50.6% of RATED THERMAL POWER with one pump operating in each loop	$\leq$ 50.7% of RATED THERMAL POWER with one pump operating in each loop#
3. RC High Temperature	$\leq$ 619°F	$\leq$ 619.08°F#
4. Flux - Δ Flux-Flow(1)	Trip Setpoint not to exceed the limit line of Figure 2.2-1.	Allowable Values not to exceed the limit line of Figure 2.2-2.#
5. RC Low Pressure(1)	$\geq$ 1985 psig	$\geq$ 1984.0 psig* $\geq$ 1976.5 psig**
6. RC High Pressure	$\leq$ 2355 psig	$\leq$ 2356.0 psig* $\leq$ 2363.5 psig**
7. RC Pressure-Temperature(1)	$\geq$ (16.25 T_{out} °F - 7873) psig	$\geq$ (16.25 T_{out} °F - 7873.64) psig#

TABLE 2.2-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
8. High Flux/Number of Reactor Coolant Pumps On(1)	$\leq$ 55.0% of RATED THERMAL POWER with one pump operating in each loop	$\leq$ 55.28% 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	$\leq$ 0.28% of RATED THERMAL POWER with two pumps operating in one loop and no pump operating in the other loop#
	$\leq$ 0.0% of RATED THERMAL POWER with no pumps operating or only one pump operating	$\leq$ 0.28% of RATED THERMAL POWER with no pumps operating or only one pump operating#
9. Containment Pressure High	$\leq$ 4 psig	$\leq$ 4 psig#

(1)Trip may be manually bypassed when RCS pressure $\leq$ 1820 psig by actuating Shutdown Bypass provided that:

a. The High Flux Trip Setpoint is $\leq$ 5% of RATED THERMAL POWER
b. The Shutdown Bypass High Pressure Trip Setpoint of $\leq$ 1820 psig is imposed, and
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

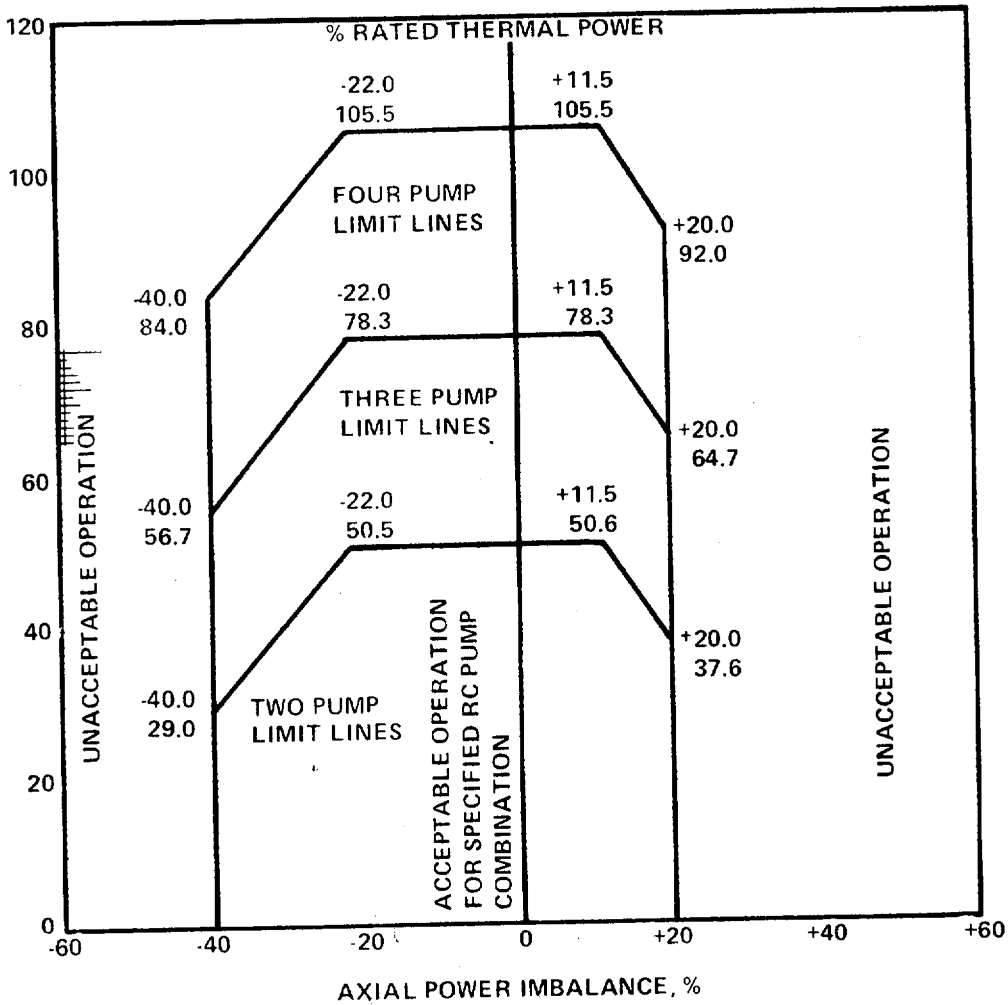

Figure 2.2-1 Trip Setpoint for Flux—ΔFlux—Flow

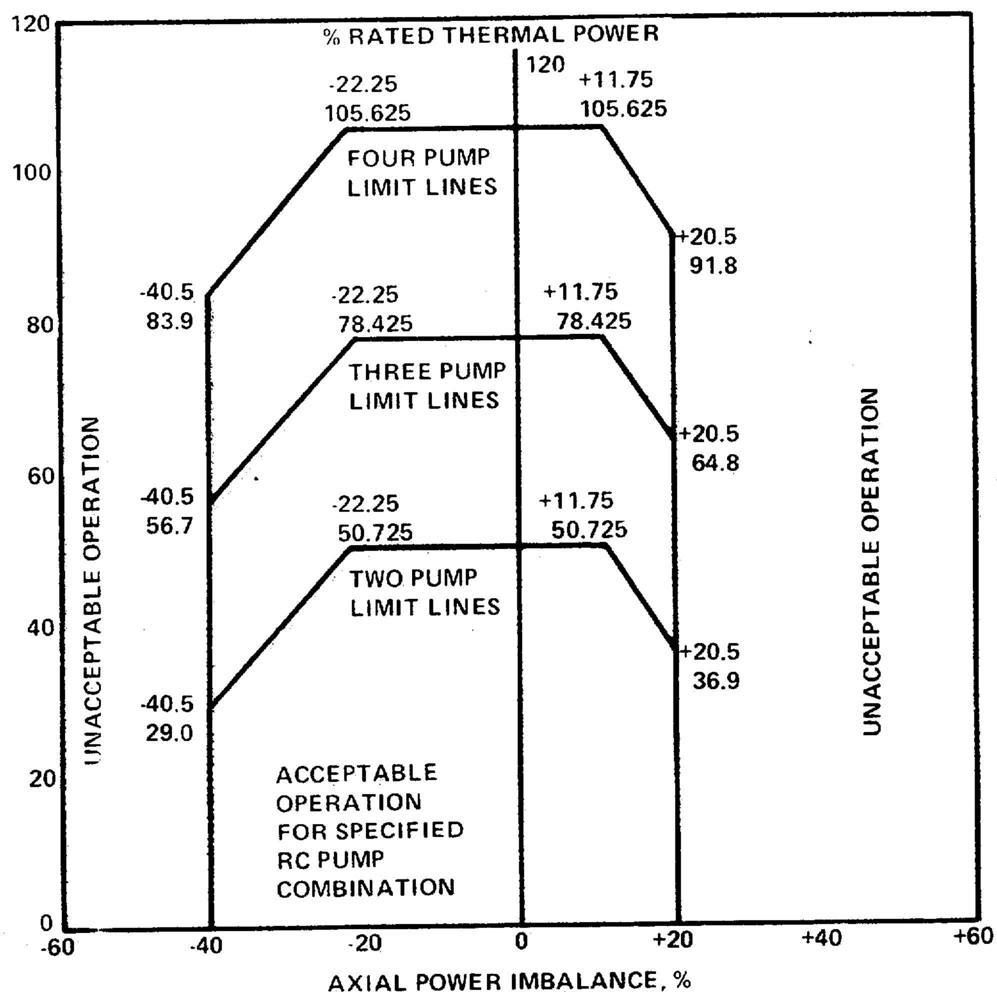

Figure 2.2-2 Allowable Value for Flux—ΔFlux—Flow

RC High Temperature

The RC High Temperature trip $\leq$ 619°F prevents the reactor outlet temperature from exceeding the design limits and acts as a backup trip for all power excursion transients.

Flux - Δ Flux-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. Typical power level and low flow rate combinations for the pump situations of Table 2.2-1 are as follows:

1. Trip would occur when four reactor coolant pumps are operating if power is 105.5% and reactor flow rate is 100%, or flow rate is 94.6% and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is 78.3% and reactor flow rate is 74.7%, or flow rate is 70.9% and power is 75%.

3. Trip would occur when one reactor coolant pump is operating in each loop (total of two pumps operating) if the power is 50.6% and reactor flow rate is 49.0% or flow rate is 46.3% and the power level is 49.0%.

For safety calculations the maximum calibration and instrumentation errors for the power level were used.

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, 2435 psig. The RC High Pressure trip also backs up the High Flux trip.

The RC Low Pressure, 1985 psig, and RC Pressure-Temperature (13.01 T_{out} °F-5973) psig, Trip Setpoints have been established to maintain the DNB ratio greater than or equal to 1.32 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 - Δ Flux-Flow trip the High Flux/Number of Reactor Coolant Pumps On trip prevents the minimum core DNBR from decreasing below 1.32 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.

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant Hot Leg Temperature.

b. Reactor Coolant Pressure

c. Reactor Coolant Flow Rate

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement at least once per 18 months.

TABLE 3.2-1

DNB MARGIN

Parameter	LIMITS Four Reactor Coolant Pumps Operating	 Three Reactor Coolant Pumps Operating	 One Reactor Coolant Pump Operating in Each Loop
Reactor Coolant Hot Leg Temperature T_H °F	$\leq$ 610	$\leq$ 610(1)	$\leq$ 610
Reactor Coolant Pressure, psig.(2)	$\geq$ 2062.7	$\geq$ 2058.7(1)	$\geq$ 2091.4
Reactor Coolant Flow Rate, gpm(3)	$\geq$ 396,880	$\geq$ 297,340	$\geq$ 195,760

(1) Applicable to the loop with 2 Reactor Coolant Pumps Operating.

(2) Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step increase of greater than 10% of RATED THERMAL POWER.

(3) These flows include a flow rate uncertainty of 2.5%.

3/4.4 REACTOR COOLANT SYSTEM

REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1 Both reactor coolant loops and both reactor coolant pumps in each loop shall be in operation.

APPLICABILITY: As noted below, but excluding MODE 6.*

ACTION:

MODES 1 and 2:

a. With one reactor coolant pump not in operation, STARTUP and POWER OPERATION may be initiated and may proceed provided THERMAL POWER is restricted to less than 78.3% of RATED THERMAL POWER and within 4 hours the setpoints for the following trips have been reduced to the values specified in Specification 2.2.1 for operation with three reactor coolant pumps operating:

   1. High Flux
   2. Flux-ΔFlux-Flow

b. With one reactor coolant pump in each loop not in operation, STARTUP and POWER OPERATION may be initiated and may proceed provided THERMAL POWER is restricted to less than 50.6% of RATED THERMAL POWER and within 4 hours the setpoints for the following trips have been reduced to the values specified in Specification 2.2.1 for operation with one reactor coolant pump operating in each loop:

   1. High Flux
   2. Flux-ΔFlux-Flow

*See Special Test Exception 3.10.3.

LIMITING CONDITION FOR OPERATION (Continued)

MODES 3, 4 and 5:

a. Operation may proceed provided at least one reactor coolant loop is in operation with an associated reactor coolant pump or decay heat removal pump.*

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

*All reactor coolant pumps and decay heat removal train pumps may be de-energized for up to 1 hour to accommodate decay heat removal pump switching operations, surveillance testing & pre-operational testing, provided no operations are permitted which could cause dilution of the reactor coolant system boron concentration.

SURVEILLANCE REQUIREMENTS

4.4.1 The Reactor Protective Instrumentation channels specified in the applicable ACTION statement above shall be verified to have had their trip setpoints changed to the values specified in Specification 2.2.1 for the applicable number of reactor coolant pumps operating either:

a. Within 4 hours after switching to a different pump combination if the switch is made while operating, or

b. Prior to reactor criticality if the switch is made while shutdown.

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

Component or System	Cycle or Transient Limit	Design Cycle or Transient
1. Reactor Coolant System	240 Heatup and Cooldown Cycles	70°F to 557°F 557°F to 70°F
2. Reactor Coolant System	160 Step Load Reduction Cycles (Resulting from turbine trip)	100% to 8% RTP*
3. Reactor Coolant System	150 Step Load Reduction Cycles (Resulting from electrical load rejection)	100% to 8% RTP*
4. Reactor Coolant System	40 Reactor Trip Cycles (Resulting from loss of electric power to all RC pumps)	Reactor Trip
5. Reactor Coolant System	160 Reactor Trip Cycles (Resulting from turbine trip without automatic control action)	Reactor Trip
6. Reactor Coolant System	40 Reactor Trip Cycles (Resulting from rod withdrawal accident)	Reactor Trip
7. Once Through Steam Generator	88 Reactor Trip Cycles (Resulting from complete loss of all main feed-water)	Reactor Trip
8. Once Through Steam Generator	40 Reactor Trip Cycles (Resulting from loss of station power)	Reactor Trip
9. Once Through Steam Generator	20 Reactor Trip Cycles (Resulting from loss of feedwater to one steam generator)	Reactor Trip

*RATED THERMAL POWER

TABLE 5.7-1 (Continued)

Component or System	Cycle or Transient Limit	Design Cycle or Transient
10. Once Through Steam Generator	10 Reactor Trip Cycles (Resulting from stuck open turbine bypass valve)	Reactor Trip
11. Reactor Coolant System	40 Rapid Depressurization	2200 psig to 300 psig in one hour
12. Reactor Coolant System	20 Change of Flow Cycles	Loss of one or more RC pumps
13. Reactor Coolant System	20 Hydrostatic Test	Pressurized to $\geq$ 3125 psig
14. Once Through Steam Generator	35 Hydrostatic Tests	Pressurized to $\geq$ 1312.5 psig
15. Reactor Coolant System	40 Test Transients	High Pressure Injection Test
16. Reactor Coolant System	240 Test Transients	Core Flooding Check Valve Test

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 16 TO FACILITY OPERATING LICENSE NO. NPF-3

THE TOLEDO EDISON COMPANY

AND

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

DOCKET NO. 50-346

Introduction

By letter dated October 23, 1978, the Toledo Edison Company (TECO or the licensee) requested an amendment to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station, Unit No. 1 (Davis-Besse 1). The amendment would change the Technical Specifications (TS) concerning the Reactor Core Safety Limit (RCSL) curve, the Reactor Protection System Instrumentation Trip Setpoints (RPSITS), and the maximum Reactor Coolant Hot Leg Temperature for Departure from Nucleate Boiling (DNB) margin. The proposed TS changes are a result of a corrected analysis of the Loss of Flow Accident (LOFA) transient which applies to the as-measured one Reactor Coolant Pump Coastdown curve.

Background

The present TS concerning the RCSL curve, the RPSITS and the maximum Reactor Coolant Hot Leg Temperature for DNB margin resulted from the LOFA analysis using the Babcock and Wilcox (B&W) RADAR code, an NRC-acceptable code for the LOFA transient. In that analysis, the licensee applied a designed one RCP coastdown curve which is not as steep as the actual measured one RCP coastdown curve. The analysis was made to support the changes required as a result of the removal of the Burnable Poison Rod Assemblies (BPRAs), and all but two of the Orifice Rod Assemblies (ORAs)(Amendment No. 11 dated June 16, 1978). After the BPRAs and ORAs were removed, the licensee discovered an error in the B&W analysis of the LOFA transient. The error concerned the use of the as-designed one RCP coastdown curve instead of the as-measured one RCP coastdown curve. The licensee reported this by letter dated July 28, 1978, and took administrative actions to reduce the setpoints for the Reactor Protection System and apply more conservative operating limits. The proposed amendment would implement these new setpoints for the Reactor Protection System and the more conservative operating limits into the TS.

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Evaluation

We have reviewed the licensee's submittal. We find the proposed changes to the TS to be more conservative than the present TS and find that the proposed changes preserve the DNB ratio (DNBR) margin for the loss of flow transient. We conclude that the proposed changes to the TS provide no decrease in the safety margin and are therefore acceptable.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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.

Dated: June 7, 1979

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-346

THE TOLEDO EDISON COMPANY

AND

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 16 to Facility Operating License No. NPF-3, issued to The Toledo Edison Company and The Cleveland Electric Illuminating Company (the licensees), which revised Technical Specifications for operation of the Davis-Besse Nuclear Power Station, Unit No. 1 (the facility) located in Ottawa County, Ohio. The amendment is effective as of its date of issuance.

The amendment changes the Technical Specifications concerning the Reactor Core Safety Limit Curve, the Reactor Protection System Instrumentation Trip Setpoints and the maximum Reactor Coolant Hot Leg Temperature for departure from nucleate boiling margin. This amendment also makes minor editorial corrections to Technical Specifications Figure 2.1-2 and Table 5.7-1.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior

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public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated October 23, 1978, (2) Amendment No. 16 to License No. NPF-3, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C., and at the Ida Rupp Public Library, 310 Madison Street, Port Clinton, Ohio.

A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 7th day of June 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors