November 27, 1985

Docket No. 50-346

DISTRIBUTION
Docket File
NRC PDR
L PDR
PBD-6 Rdg
~~HThompson~~ FMiraglia
OELD
OPA
LHarmon
ACRS-10
~~HOrnstein~~
BGrimes
WJones
~~MVirgilio~~ FOB, PWR-B
RDiggs
RIngram
ADe Agazio
Gray File+4
EJordan
TBarnhart-4
~~EBrach~~ JPartlow
GDick

Mr. Joe Williams, Jr.
Vice President, Nuclear
Toledo Edison Company
Edison Plaza - Stop 712
300 Madison Avenue
Toledo, Ohio 43652

Dear Mr. Williams:

SUBJECT: AMENDMENT NO. 91 TO FACILITY OPERATING LICENSE NO. NPF-3; MINIMUM REACTOR COOLANT SYSTEM FLOW REQUIREMENT

The Commission has issued the enclosed Amendment No. 91 to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station, Unit No. 1. This amendment consists of changes to the Appendix A Technical Specifications (TSs) in response to your application dated February 13, 1985 (No. 1125).

This amendment revises the minimum Reactor Coolant System flow requirement to take credit for the decrease in the core bypass flow resulting from the use of Lump Burnable Poison Rods in Cycle 5 design.

A copy of the Safety Evaluation supporting this amendment is also enclosed. Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Original signed by

George F. Dick, Jr., Project Manager
PWR Project Directorate #6
Division of PWR Licensing-B

Enclosures:
1. Amendment No. 91 to NPF-3
2. Safety Evaluation

cc w/enclosures:
See next page

*See previous white for concurrences:

PBD-6	PBD-6	PBD-6	PBD-6	OELD
RIngram*	GDick;cr*	ADeAgazio*	JStolz*	CBarth*
11/13/85	11/13/85	11/13/85	11/13/85	11/13/85

11/27/85

8512120417 851127
PDR ADOCK 05000346
P PDR

Docket No. 50-346

DISTRIBUTION
Docket File
NRC PDR
L PDR
ORB#4 Rdg
HThompson
OELD
CMiles
LHarmon
ACRS-10
HOrnstein
BGrimes
WJones
MVirgilio
RDiggs
RIngram
ADe Agazio
Gray File+4
EJordan
TBarnhart-4
EBrach

Mr. Joe Williams, Jr.
Vice President, Nuclear
Toledo Edison Company
Edison Plaza - Stop 712
300 Madison Avenue
Toledo, Ohio 43652

Dear Mr. Williams:

SUBJECT: AMENDMENT NO. TO FACILITY OPERATING LICENSE NO. NPF-3; MINIMUM REACTOR COOLANT SYSTEM FLOW REQUIREMENT

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. This amendment consists of changes to the Appendix A Technical Specifications (TSs) in response to your application dated February 13, 1985 (No. 1125).

This amendment revises the minimum Reactor Coolant System flow requirement to take credit for the decrease in the core bypass flow resulting from the use of Lump Burnable Poison Rods in Cycle 5 design.

A copy of the Safety Evaluation supporting this amendment is also enclosed. Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

George F. Dick, Jr., Project Manager
Operating Reactors Branch #4
Division of Licensing

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

cc w/enclosures:
See next page

ORB#4:DL	ORB#4:DL	ORB#4:DL	OELD	AD:OR:DL
RIngram	GDick;cr	JStolz		GLainas
11/13/85	11/14/85	11/18/85	11/26/85	11/ /85

ORB#4:DL
ADeAgazio
11/15/85

11/26/85

Mr. J. Williams
Toledo Edison Company

Davis-Besse Nuclear Power Station
Unit No. 1

cc:
Donald H. Hauser, Esq.
The Cleveland Electric
Illuminating Company
P. O. Box 5000
Cleveland, Ohio 44101

Mr. Robert F. Peters
Manager, Nuclear Licensing
Toledo Edison Company
Edison Plaza
300 Madison Avenue
Toledo, Ohio 43652

Gerald Charnoff, Esq.
Shaw, Pittman, Potts
and Trowbridge
1800 M Street, N.W.
Washington, D.C. 20036

Paul M. Smart, Esq.
Fuller & Henry
300 Madison Avenue
P. O. Box 2088
Toledo, Ohio 43603

Mr. Robert B. Borsum
Babcock & Wilcox
Nuclear Power Generation
Division
Suite 200, 7910 Woodmont Avenue
Bethesda, Maryland 20814

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

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Ohio Department of Health
ATTN: Radiological Health
Program Director
P. O. Box 118
Columbus, Ohio 43216

Attorney General
Department of Attorney
General
30 East Broad Street
Columbus, Ohio 43215

Mr. James W. Harris, Director
Division of Power Generation
Ohio Department of Industrial Relations
2323 West 5th Avenue
P. O. Box 825
Columbus, Ohio 43216

Mr. Harold Kohn, Staff Scientist
Power Siting Commission
361 East Broad Street
Columbus, Ohio 43216

President, Board of
Ottawa County
Port Clinton, Ohio 43452

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. 91
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 February 13, 1985, 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:

8512120418 851127
PDR ADOCK 05000346
P PDR

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 91 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz, Director
PWR Project Directorate #6
Division of PWR Licensing-B

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 27, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 91

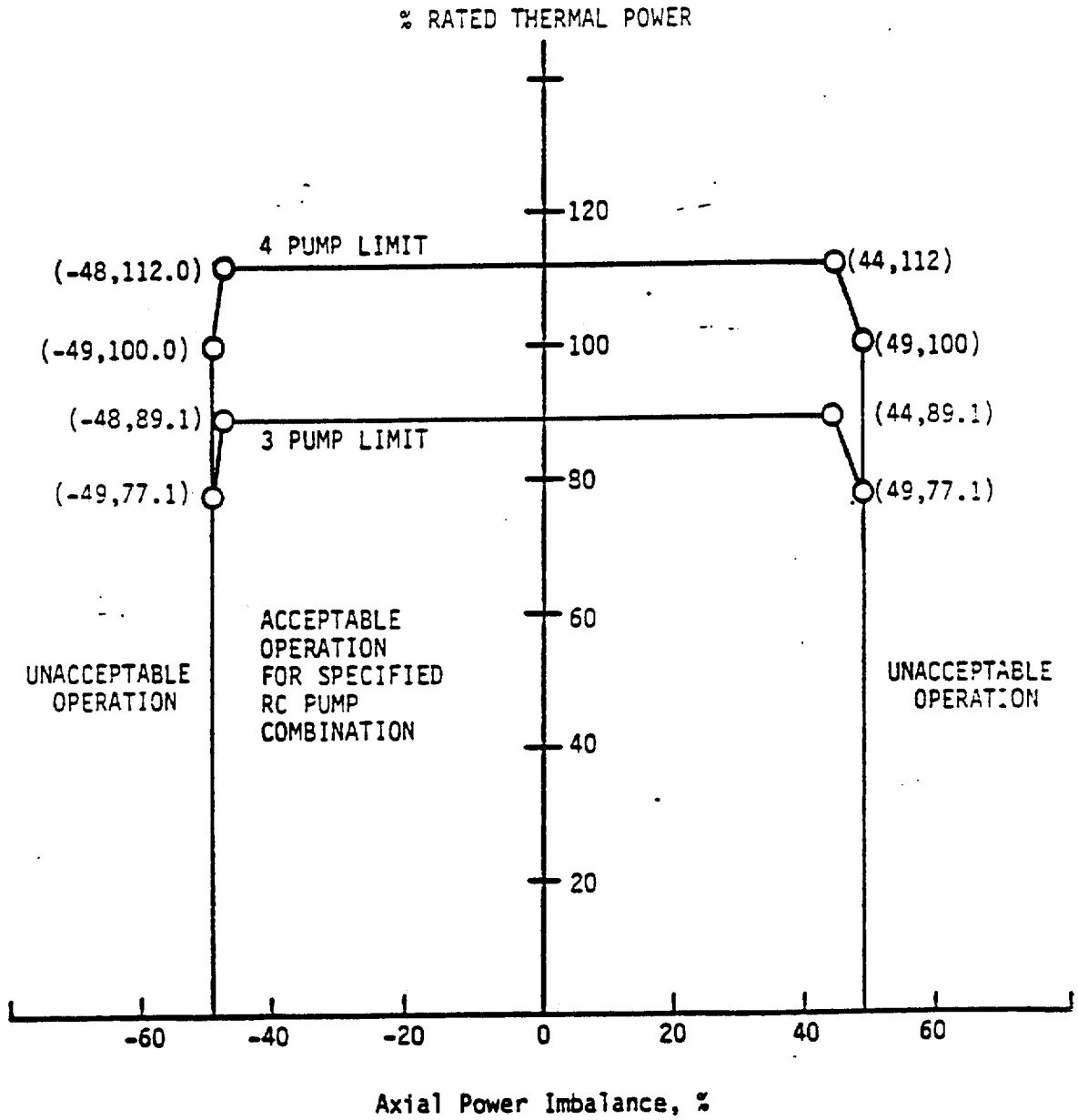
FACILITY OPERATING LICENSE NO. NPF-3

DOCKET NO. 50-346

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain a vertical line indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove	Insert
2-3	2-3
2-7	2-7
3/4 2-14	3/4 2-14
B 2-1	B 2-1
B 2-8	B 2-8

Figure 2.1-2 Reactor Core Safety Limit



PUMPS OPERATING	REACTOR COOLANT FLOW, GPM
4	380,160
3	283,980

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.

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

Curve shows trip setpoint for an approximately 25% flow reduction for three pump operation (283,980 gpm). The actual setpoint will be directly proportional to the actual flow with three pumps.

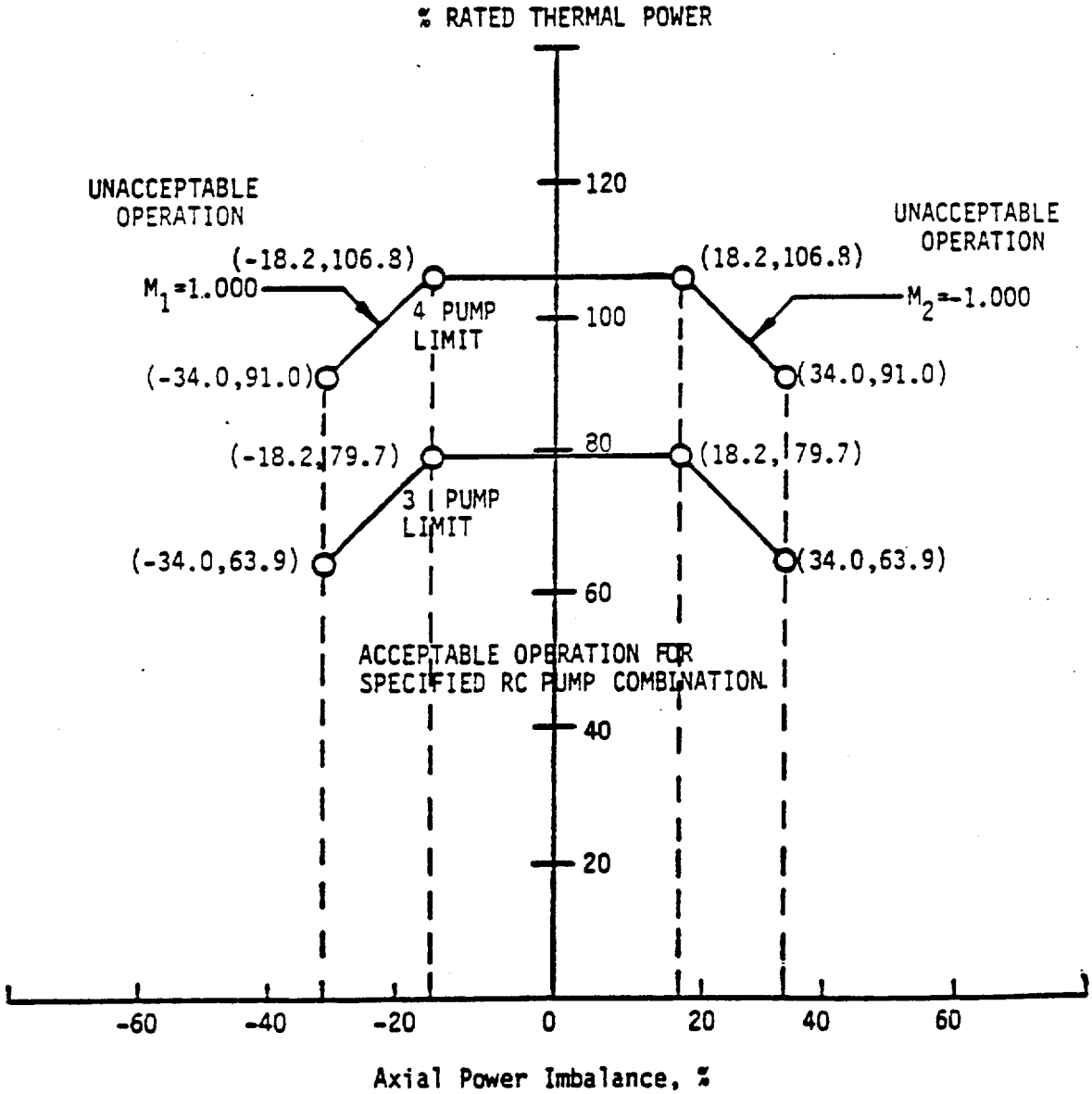


Figure 2.2-2 Allowable Value for Flux-Δ Flux/Flow

DELETED

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:

If parameter a or b above exceeds 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.

If parameter c exceeds its limit, either:
1. Restore the parameter to within its limit within 2 hours, or
2. Limit THERMAL POWER at least 2% below RATED THERMAL POWER for each 1% parameter c is outside its limit for four pump operation within the next 4 hours, or limit THERMAL POWER at least 2% below 75% of RATED THERMAL POWER for each 1% parameter c is outside its limit for 3 pump operation 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	LIMITS: Three Reactor Coolant Pumps Operating
Reactor Coolant Hot Leg Temperature T_H °F	≤ 610	≤ 610[1]
Reactor Coolant Pressure, psig.[2]	≥ 2062.7	≥ 2058.7[1]
Reactor Coolant Flow Rate, gpm[3]	≥ 389,664	≥ 291,080

(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%, and are based on a minimum of 64 lumped burnable poison rod assemblies in place in the core.

The reactor trip envelope appears to approach the safety limits more closely than it actually does because the reactor trip pressures are measured at a location where the indicated pressure is about 30 psi less than core outlet pressure, providing a more conservative margin to the safety limit.

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 1.30 DNBR limit produced by a nuclear power peaking factor of F_Q = 2.56 or the combination of the radial peak, axial peak, and position of the axial peak that yields no less than a 1.30 DNBR.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limits are 20.4 kW/ft for batches 1E, 4B, and 5A and 20.5 kW/ft for batches 5B, 6, and 7.

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 curves 1 and 2 of Figure 2.1-2 correspond to the expected 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 of 1.30 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 +22%, whichever condition is more restrictive. These curves include the potential effects of fuel rod bow and fuel densification.

The DNBR as calculated by the B&W-2 DNB correlation continually increases from point of minimum DNBR, so that the exit DNBR is always higher. Extrapolation of the correlation beyond its published quality range of +22% is justified on the basis of experimental data.

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 through the B&W-2 DNB correlation. The DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. 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 minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This 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 of 1.30 is predicted for the maximum possible thermal power 112% when the reactor coolant flow is 380,160 GPM, which is 108% of design flow rate for four operating reactor coolant pumps. This curve is based on the following hot channel factors with potential fuel densification and fuel rod bowing effects:

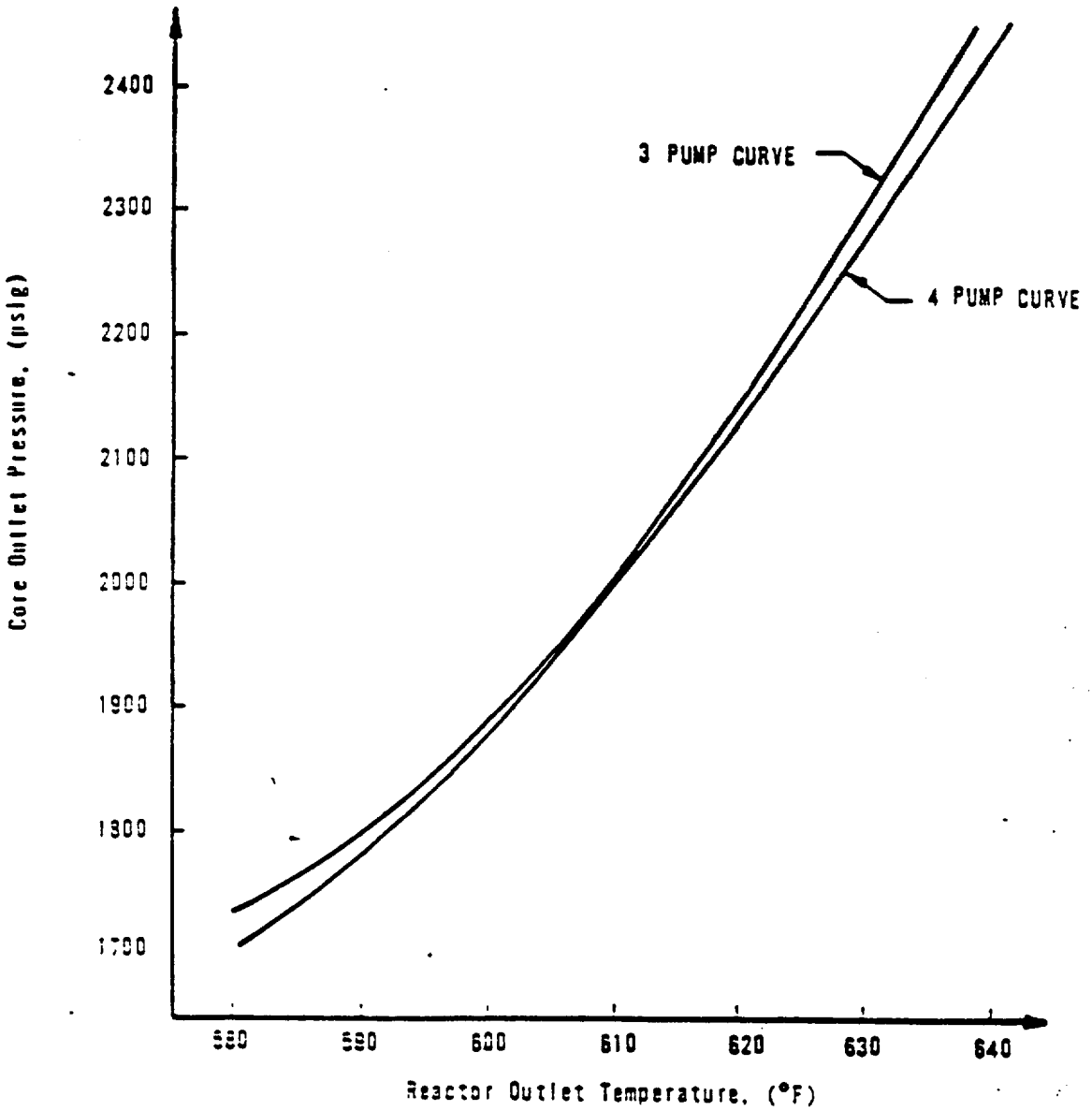
$$F_Q = 2.56; \quad F^N_{\Delta H} = 1.71; \quad F^N_Z = 1.50$$

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.

Containment High Pressure

The Containment High Pressure Trip Setpoint $\leq$ 4 psig, provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the containment vessel or a loss-of-coolant accident, even in the absence of a RC Low Pressure trip.

Bases-Figure 2.1 **Pressure/Temperature Limits at Maximum Allowable Power for Minimum DNBR**



PUMPS	FLOW (GPM)	POWER
4	380,150	112%
3	283,980	89.1%

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 91 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 AND BACKGROUND

By letter dated February 13, 1985 (Ref. 1) Toledo Edison Company, the licensee for Davis-Besse Unit 1, made application for amendment to Facility Operating License No. NPF-3. Changes were proposed to the Davis-Besse Unit 1 Technical Specifications, Appendix A, which involve Figures 2.1-2, 2.2-1, Table 3.2-1 and the Bases. The proposed changes are revisions to the minimum Reactor Coolant System (RCS) flow requirement to take credit for a decrease in core bypass flow as a result of using 64 new Lumped Burnable Poison Rod Assemblies (LBPRAs) in Cycle 5. The licensee presented information to show that with the proposed changes adequate cooling of the reactor core is maintained such that the minimum required Departure From Nucleate Boiling Ratio (DNBR) is maintained.

2.0 EVALUATION

The major bypass flow paths that exist in the Davis-Besse reactor vessel include: (1) empty guide tubes in fuel assemblies, (2) baffle plates, and (3) gaps around the hot leg nozzle in the upper core internals. In Cycle 5, 64 new LBPRAs were inserted which reduced the bypass flow by filling empty guide tubes. The licensee provided the results of a hydraulic analysis which takes into account the decreased bypass flow area because of the insertion of the 64 LBPRAs. This resulted in a decrease of bypass flow from the 10.7% as used in the Cycle 5 Reload Report (no credit was taken for flow area reduction in Cycle 5 from that in Cycle 4) to a value of 8.1%. The decrease in bypass flow results in a net increase in core flow even though there is a slight decrease in system flow from the added resistance of the 64 LBPRAs.

In discussions with the licensee, we requested information on the methods used in arriving at the old and new bypass flow values of 10.7% and 8.1%, respectively. A summary of the core bypass flow methodology, including a flow path model and various combinations of resistances and flow areas, was provided (Ref. 2) for clarification.

The licensee has proposed new values for the minimum RCS flow for both four reactor coolant pump and three reactor coolant pump operation as shown in

8512120419 851127
PDR ADOCK 05000346
P PDR

Table 1 of this Safety Evaluation. These values take into account the increased core coolant flow because of the reduction of bypass flow from 10.7% to 8.1%. New minimum RCS flow values were presented for use in Table 3.2-1 (page 3/4 2-14 of Technical Specification Section 3.2.5). For four pump operation, the minimum RCS flow value was reduced from 396,880 gpm (110% of design flow plus 2.5% uncertainty) to 389,664 gpm (108% of design flow plus 2.5% uncertainty). For three pump operation, the minimum RCS flow value was correspondingly reduced from 297,340 gpm to 291,080 gpm. The minimum DNBR requirement remains unchanged from the Cycle 5 Reload Report. This is because the new value of minimum RCS flow and the core coolant flow associated with it and the 8.1% bypass flow is greater than or equal to the present minimum flow associated with the 10.7% bypass flow.

The licensee stated (Ref. 2) that the measured RCS flow from a heat balance was found to be 406,533 gpm. With a reduction of 2.5% for measurement error, this is 112.6% of the design flow (352,000 gpm) and 104.3% of the proposed Technical Specification minimum RCS flow (380,160 gpm). We have found the 2% reduction (110% to 108% of design flow) in the minimum RCS flow acceptable because of the corresponding increase in core flow from the reduction in bypass flow. In addition, the measured RCS flow indicates that there is additional flow margin available. This evaluation is based on 64 LBPRAs. A decrease in the number of LBPRAs increases the core bypass flow and therefore reduces the active core flow. Therefore, the statement "These flows are based on having a minimum of 64 Lumped Burnable Poison Rod Assemblies in the core.", has been added to footnote (3) of Table 3.2-1, DNB Margin (page 3/4 2-14) of the Technical Specifications.

3.0 EVALUATION OF TECHNICAL SPECIFICATION MODIFICATIONS

The minimum acceptable reactor coolant flow in gpm has been revised as a result of reducing the percent of design flow rate from 110% to 108%. This was possible because of the increase in core flow resulting from a decrease in core bypass flow as discussed in Section 2.0. Table 1 lists the current and proposed Technical Specification flow rates for four and three pump operation as provided in Reference 1. The pertinent Technical Specification pages shown below have been modified accordingly to include the proposed changes presented in Table 1 of this Safety Evaluation.

Page 2-3 (Figure 2.1-2)	(Required System Flow)
Page 2-7 (Figure 2.2-1)	(Required System Flow, 3-pump) - also contains an editorial change to clarify three pump operation represents an "approximate" 25% flow reduction.
Page B 2-1	(Required System Flow, 4-pump; Percent of Design Flow)
Page B 2-8 (Bases Fig.2.1)	(Required System Flow) - also a typo error was corrected to change 89.3% for three pump operation to 89.1%.

Page 3/4 2-14 (Table 3.2-1)	(Required System Flow plus measurement error) - also modified footnote (3) to include minimum number of lumped burnable poison rod assemblies at 64.

4.0 SUMMARY

We have reviewed the changes in flow rate shown on these pages in the Technical Specifications as proposed in Reference 1 and have found them acceptable for the reasons explained in Section 2.0.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. We have determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

6.0 CONCLUSION

We have 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.

Dated: November 27, 1985

The following NRC personnel contributed to this Safety Evaluation:
H. Balukjian

TABLE -1

MINIMUM ACCEPTABLE REACTOR COOLANT FLOW (GPM)

	4-Pump Operation		3-Pump Operation	
	Current TS	Proposed TS	Current TS	Proposed TS
Required System Flow	387,200	380,160	290,100	283,980
Required System Flow Plus 2.5% Measurement Error	396,880	389,664	297,340	291,080
Percent of Design Flow Rate	110%	108%	110%	108%

References

1. Letter from R. P. Crouse, Toledo Edison Company, to John Stolz, NRC, dated February 13, 1985.

2. Letter from R. P. Crouse, Toledo Edison Company, to John Stolz, NRC, dated May 29, 1985. (Proprietary submittal.)