December 28, 1988

Docket No. 50-346
Serial No. DB-88-062

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OGC-WF1	ARM/LFMB
DHagan	PDIII-3 r/f
EJordan	JHannon
	S. BAJWA, TAMS

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. 128 TO FACILITY OPERATING LICENSE NO. NPF-3: HIGH PRESSURE REACTOR TRIP SETPOINT AND PORV SETPOINT (TAC NO. 66727)

The Commission has issued the enclosed Amendment No. 128 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 (TS's) in response to your application dated February 1, 1988 (Serial No. 1464) as supplemented with information contained in a letter dated February 26, 1988 (Serial No. 1496).

This amendment involves changes to TS Sections 2.1.3, 2.2.1, 3/4.3.2.3, 3/4.4.3, and Bases Sections 3/4.3.1, 3/4.3.2, 3/4.4.2, and 3/4.4.3. This amendment revises the TS's to permit increasing the Reactor Coolant System high pressure trip setpoint to 2355 psig from the present value of 2300 psig, to permit increasing the Power Operated Relief Valve (PORV) trip setpoint to 2435 psig from the present value of 2390 psig, and to permit increasing the reactor power level at or below which the Anticipatory Reactor Trip System (ARTS) may be blocked to 45 percent from the present value of 25 percent.

Copies of the related Safety Evaluation, Technical Evaluation Report, and the notice of issuance are enclosed.

Sincerely,

/s/

Albert W. De Agazio, Sr. Project Manager
Project Directorate III-3
Division of Reactor Projects - III, IV, V & Special Projects
Office of Nuclear Reactor Regulation

8901090223 881228
PDR ADOCK 05000346
P PNU

Enclosures:
1. Amendment No.128to License No. NPF-3
2. Safety Evaluation
3. TER (EGG-NTA-8152)
4. Notice of issuance

cc: See next page

DF01
1/1

Office:	LA/PDIII-3	PM/PDIII-3	PD/PDIII-3	OGC-WF1
Surname:	PKreutzer	ADeAgazio/mr	JHannon	BBachmann
Date:	12/20/88	12/20/88	12/20/88	12/22/88

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Project Directorate III-3
Division of Reactor Projects - III, IV, V & Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. to License No. NPF-3
2. Safety Evaluation
3. Technical Evaluation Report
4. Notice of issuance

cc: See next page

Office:	LA/PDIII-3	PM/PDIII-3	PD/PDIII-3	OGC-WF1
Surname:	PKreutzer	ADeAgazio/mr	JHannon	
Date:	/ /88	12/16/88	/ /88	/ /88

Docket No. 50-346
Serial No. DB-88-062

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This amendment involves changes to TS Sections 2.1.3, 2.2.1, 3/4.3.2.3, 3/4.4.3, and Bases Sections 3/4.3.1, 3/4.3.2, 3/4.4.2, and 3/4.4.3. This amendment revises the TS's to permit increasing the Reactor Coolant System high pressure trip setpoint to 2355 psig from the present value of 2300 psig, to permit increasing the Power Operated Relief Valve (PORV) trip setpoint to 2435 psig from the present value of 2390, and to permit increasing the reactor power level at or below which the Anticipatory Reactor Trip System (ARTS) may be blocked to 45 percent from the present value of 25 percent.

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Sincerely,

Albert W. De Agazio, 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. to License No. NPF-3
2. Safety Evaluation
3. Notice of issuance

cc: See next page

Office:	LA/PDIII-3	PM/PDIII-3	PD/PDIII-3	OGC-WF1
Surname:	PKreutzer	ADeAgazio/mr	JHannon	
Date:	12/2/88	12/6/88	12/7/88	/ /88

Mr. Donald C. Shelton
Toledo Edison Company

Davis-Besse Nuclear Power Station
Unit No. 1

cc:
David E. Burke, Esq.
The Cleveland Electric
Illuminating Company
P. O. Box 5000
Cleveland, Ohio 44101

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

Gerald Charnoff, Esq.
Shaw, Pittman, Potts
and Trowbridge
2300 N Street N.W.
Washington, D.C. 20037

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

Mr. Robert B. Borsum
Babcock & Wilcox
Nuclear Power Generation Division
Suite 525, 1700 Rockville Pike
Rockville, Maryland 20852

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

Radiological Health Program
Ohio Department of Health
1224 Kinnear Road
Columbus, Ohio 43212

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

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

Ohio Environmental Protection Agency
361 East Broad Street
Columbus, Ohio 43266-0558

President, Board of
County Commissioners of
Ottawa County
Port Clinton, Ohio 43452

State of Ohio
Public Utilities Commission
180 East Broad Street
Columbus, Ohio 43266-0573

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

TOLEDO EDISON COMPANY

AND

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

DOCKET NO. 50-346

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 128
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 1, 1988, supplemented February 26, 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:

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P PNU

(a) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 128, 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 February 13, 1989.

FOR THE NUCLEAR REGULATORY COMMISSION

for Thomas V. Wambach

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: December 28, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 128

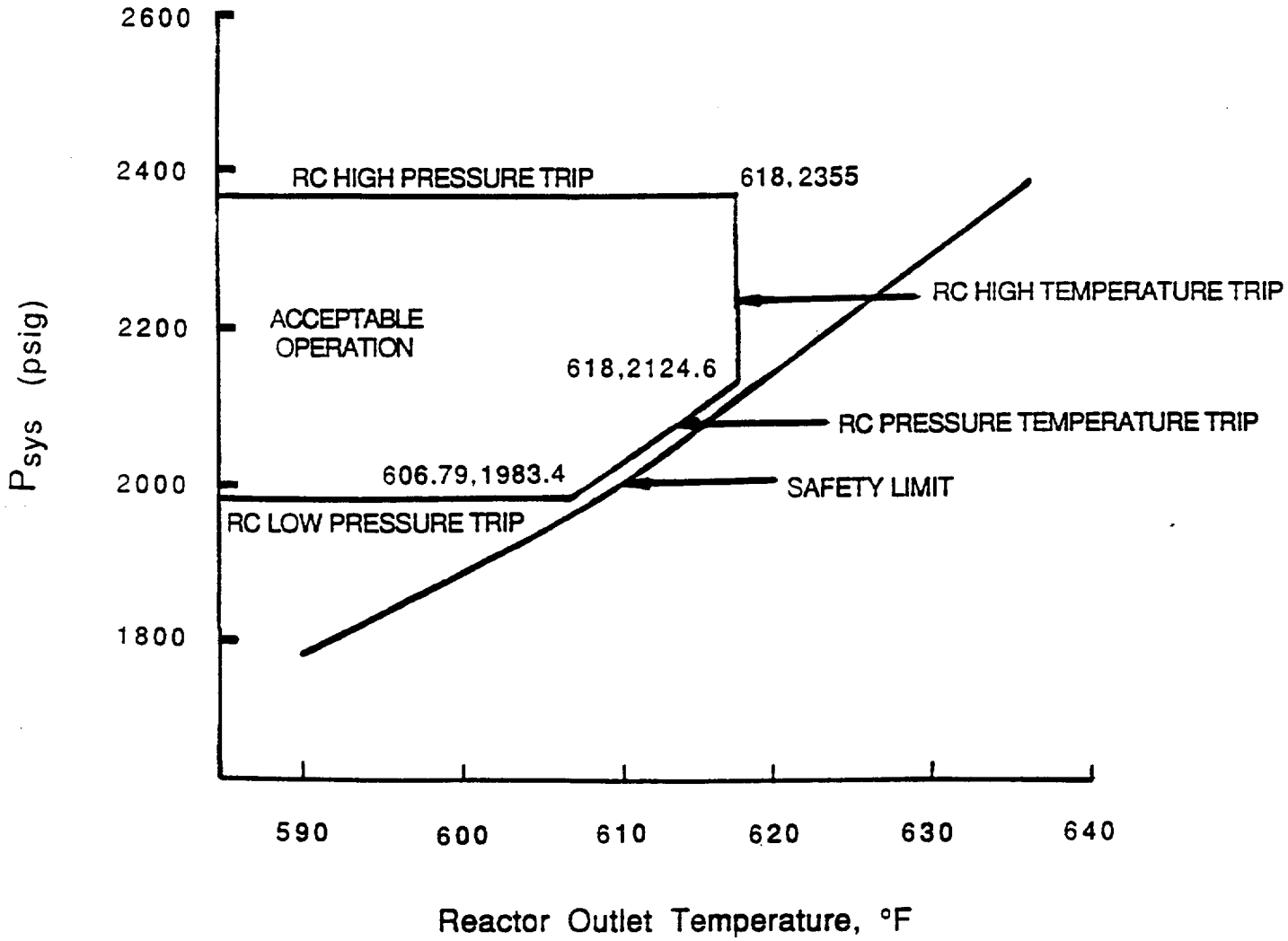
FACILITY OPERATING LICENSE NO. NPF-3

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
3/4 3-30b	3/4 3-30b
3/4 3-30c	3/4 3-30c
3/4 3-30d	3/4 3-30d
3/4 4-4	3/4 4-4
B 3/4 3-1	B 3/4 3-1
B 3/4 4-1	B 3/4 4-1
B 3/4 4-1a	B 3/4 4-1a

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	≤104.94% of RATED THERMAL POWER with four pumps operating	≤104.94% of RATED THERMAL POWER with four pumps operating#
	≤79.7% of RATED THERMAL POWER with three pumps operating	≤79.7% of RATED THERMAL POWER with three pumps operating#
3. RC high temperature	≤618°F	≤618°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 2.2-1#
5. RC low pressure(1)	≥1983.4 psig	≥1983.4 psig* ≥1983.4 psig**
6. RC high pressure	≤2355 psig	≤2355.0 psig* ≤2355.0 psig**
7. RC pressure-temperature(1)	≥(12.60 T_{out} °F - 5662.2) psig	≥(12.60 T_{out} °F - 5662.2) 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#

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.

TABLE 3.3-15

ANTICIPATORY REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Turbine Trip	4	2[a]	3	1[b]	16
2. Trip of Both Main Feed Pump Turbines	4	2	3	1	17
3. Output Logic	4	2	3	1	18

(a) Trip automatically bypassed below 45 percent of RATED THERMAL POWER
(b) Applicable only above 45 percent of RATED THERMAL POWER

ANTICIPATORY REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.3 The Anticipatory Reactor Trip System instrumentation channels of Table 3.3-15 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-15

ACTION: As shown in Table 3.3-15

SURVEILLANCE REQUIREMENTS

4.3.2.3 The Anticipatory Reactor Trip System shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST for the modes and at the frequencies shown in Table 4.3-15.

TABLE 3.3-15 (CONTINUED)

ACTION STATEMENTS

ACTION 16 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirements, restore the inoperable channel to OPERABLE status within 72 hours or reduce reactor power to less than 45 percent of RATED THERMAL POWER within the next 6 hours.

ACTION 17 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirements, restore the inoperable channel to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours.

ACTION 18 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and POWER OPERATION may proceed provided both of the following conditions are satisfied:

a) The control rod drive trip breaker associated with the inoperable channel is placed in the tripped condition within one hour.

b) The Minimum Channels OPERABLE requirement is met; however, one additional control rod drive trip breaker associated with another channel may be tripped for up to 2 hours for surveillance testing per Specification 4.3.2.3, after reclosing the control rod drive trip breaker opened in a) above.

TABLE 4.3-15

ANTICIPATORY REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE IS REQUIRED
1. Turbine Trip[a]	S	Not Applicable	M	1 [b]
2. Main Feed Pump Turbine Trip	S	Not Applicable	M	1
3. Output Logic	Not Applicable	Not Applicable	M	1

(a) Trip automatically bypassed below 45 percent of RATED THERMAL POWER
(b) Applicable only above 45 percent of RATED THERMAL POWER

SAFETY VALVES AND ELECTROMATIC RELIEF VALVE - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting of $\leq$ 2525 PSIG.* When not isolated, the pressurizer electromatic relief valve shall have a trip setpoint of $\geq$ 2435 PSIG and an allowable value of $\geq$ 2435 PSIG.**

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.3 For the pressurizer code safety valves, there are no additional Surveillance Requirements other than those required by Specification 4.0.5. For the pressurizer electromatic relief valve a channel calibration check shall be performed every 18 months.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

** Allowable value for channel calibration check.

LIMITING CONDITION FOR OPERATION

3.4.2 Decay Heat Removal System relief valve DH-4849 shall be OPERABLE with a lift setting of ≤ 330 PSIG* and isolation valves DH-11 and DH-12 open and control power to their valve operators removed.

APPLICABILITY: MODES 4 and 5.

ACTION:

A. With DH-4849 not OPERABLE:

1. Make the valve OPERABLE within eight hours; or

2. a. Within next one hour, disable the capability of both high pressure injection (HPI) pumps to inject water into the reactor coolant system; and

b. Within next eight hours:

1. Disable the automatic transfer of makeup pump suction to the borated water storage tank on low makeup tank level; and

2. Reduce makeup tank level to ≤ 73 inches and reduce reactor coolant system pressure and pressurizer level within the acceptable region on Figures 3.4-2a (in MODE 4) and 3.4-2b (in MODE 5).

B. With DH-11 or DH-12 closed, open DH-21 and DH-23 within one hour.

C. With the control power not removed from DH-11 and DH-12, remove the power to the valve operators at the Motor Control Centers within one hour.

SURVEILLANCE REQUIREMENTS

4.4.2 Decay Heat Removal System relief valve DH-4849 shall be determined OPERABLE:

a. per the surveillance requirements of Specification 4.0.5.

b. at least once per 24 hours by verifying either:

1. isolation valves DH-11 and DH-12 open with control power removed from their valve operators; or

2. valves DH-21 and DH-23 open.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

3/4.3.1 and 3/4.3.2 REACTOR PROTECTION SYSTEM AND SAFETY SYSTEM INSTRUMENTATION

The OPERABILITY of the RPS, SFAS and SFRCS instrumentation systems ensure that 1) the associated action and/or trip will be initiated when the parameter monitored by each channel or combination thereof exceeds its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available for RPS, SFAS and SFRCS purposes from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The measurement of response time at the specified frequencies provides assurance that the RPS, SFAS, and SFRCS action function associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such test demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

An SFRCS channel consists of 1) the sensing device(s), 2) associated logic and output relays (including Isolation of Main Feedwater Non Essential Valves and Turbine Trip), and 3) power sources.

Safety-grade anticipatory reactor trip is initiated by a turbine trip (above 45 percent of RATED THERMAL POWER) or trip of both main feedwater pump turbines. This anticipatory trip will operate in advance of the reactor coolant system high pressure reactor trip to reduce the peak reactor coolant system pressure and thus reduce challenges to the power operated relief valve. This anticipatory reactor trip system was installed to satisfy Item II.K.2.10 of NUREG-0737. The justification for the ARTS turbine trip arming level of 45% is given in BAW-1893, October, 1985.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

3/4.3.3.2 INCORE DETECTORS

The OPERABILITY of the incore detectors ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core. See Bases Figures 3-1 and 3-2 for examples of acceptable minimum incore detector arrangements.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event so that the response of those features important to safety may be evaluated. This capability is required to permit comparison of the measured response to that used in the design basis for the facility. This instrumentation is consistent with the recommendations of Regulatory Guide 1.12 "Instrumentation for Earthquakes," April 1974.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public. This instrumentation is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with both reactor coolant loops in operation, and maintain DNBR above 1.30 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 1.30 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 22%, 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

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and power operated relief valves against water relief.

The low level limit is based on providing enough water volume to prevent a reactor coolant system low pressure condition that would actuate the Reactor Protection System or the Safety Feature Actuation System. The high level limit is based on providing enough steam volume to prevent a pressurizer high level as a result of any transient.

The power operated relief valves and steam bubble function to relieve RCS pressure during all design transients. Operation of the power operated relief valves minimizes the undesirable opening of the spring-loaded pressurizer code safety valves.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these chemistry limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 1 GPM). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads

inject water into the reactor coolant system is disabled to ensure operation within reactor coolant system pressure-temperature limits.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

The pressurizer code safety valves must be set such that the peak Reactor Coolant System pressure does not exceed 110% of design system pressure (2500 psig) or, 2750 psig. The control rod group withdrawal accident will result in the most limiting high pressure in the RCS. The analysis assumes RPS high pressure trip at 2355 psig and the code safety valves open at 2500 psig. The tolerance on the RPS instrument accuracy is 30 psi and, it is +1% for the code safety valve settings. The pressurizer electromatic relief valve was assumed not to open for this transient. The resulting system peak pressure was calculated to be 2700 psig. Therefore, the code safety valve setpoint is conservatively set at $\leq$ 2525 psig which is the maximum pressure of 2500 psig +1% for tolerance.

The pressurizer electromatic relief valve should be set such that it will open before the code safety valves are opened. However, it should not open on any anticipated overpressure transients. BAW-1890, September 1985 identified that the turbine trip from full power would cause the largest overpressure transient. This report demonstrated that with a RPS high pressure trip setpoint of 2355 psig the resulting overshoot in RCS pressure would be limited to 50 psi. Consequently, the minimum PORV setpoint needs to accommodate both the RCS pressure overshoot and the RPS instrument string error of 30 psi.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 128 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 letters dated February 1, 1988 (Ref. 1) and February 26, 1988 (Ref. 2), the Toledo Edison Company proposed to amend the Davis-Besse Nuclear Power Station, Unit No. 1, Appendix A Technical Specifications (TS's). The proposed amendment would raise the Reactor Protection System (RPS) Reactor-Coolant-System (RCS) - high-pressure trip setpoint from 2300 psig to 2355 psig and the raise the minimum pilot operated relief valve (PORV) trip setpoint to at least 2435 psig from the existing value of 2390 psig. Also, the arming threshold for the Anticipatory Reactor Trip (ART) on turbine trip would be raised from the current 25% reactor power level to a level of 45%. These proposed changes are to improve the operational performance of the plant.

The licensee, in providing justification for the requested changes, referred to studies made in 1985 by the Babcock and Wilcox Owners Group (B&WOG). As part of the Transient Assessment Program, the B&WOG had studies performed to improve operational safety and performance through a reduction in the frequency of reactor trips. This effort resulted in two reports submitted for review to the NRC. These reports are:

(1) Justification for Raising Setpoint for Reactor Trip on Pressure," BAW-1890, September 1985 (Ref. 3).

(2) "Basis for Raising Arming Threshold for Anticipatory Reactor Trip on Turbine Trip," BAW-1893, October 1985 (Ref. 4).

The staff acceptances of these reports are documented in evaluations issued on April 22, 1986 (Ref. 5) and April 25, 1986 (Ref. 6), respectively.

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To analyze the effect of the higher safety valve setpoint in combination with a 2355 psig reactor trip setpoint, B&W reanalyzed the control rod assembly withdrawal accident at startup (USAR 15.2.1) which is the most severe overpressure transient at Davis-Besse (Ref. 7).

2.0 EVALUATION

The staff was assisted in this evaluation by EG&G. The EG&G evaluation included a review of the Toledo Edison Company proposed changes and a comparison to the approved B&WOG reports. Also, the control rod assembly withdrawal accident at startup reanalysis was reviewed.

The staff agrees with the EG&G evaluation which concludes that the proposed changes meet the NRC positions established in their review of the B&WOG topical reports and, therefore, meet the applicable regulatory guidance and requirements. Further, it was found that the reanalyzed rod control assembly withdrawal accident at startup (USAR 15.2.1) for Davis-Besse bounds the proposed Technical Specification change. The staff, therefore, concludes that the RCS high-pressure trip setpoint can be increased from 2300 psig to 2355 psig, that the arming threshold for ART on turbine trip can be raised from 25% reactor power to 45% reactor power, and that the PORV minimum trip setpoint can be raised to at least 2435 psig from the existing value of 2390 psig without adverse effects on safety.

3.0 TECHNICAL SPECIFICATION CHANGES

The Technical Specification changes proposed are as follows:

1. Figure 2.1-1, Page 2-2. Reactor Core Safety Limits. The value of the setting for the RCS high-pressure trip would change from 2300 psig to 2355 psig.

2. Table 2.2.-1, Page 2-5. Reactor Protection System Instrumentation Trip Setpoints. The maximum RCS high-pressure setpoint and allowable values for channel functional test and channel calibration would be changed from 2300 psig to 2355 psig.

3. Table 3.3-15, Page 3/4 3-30b. The ARTS Instrumentation footnotes would be modified to account for the change of the arming threshold for ART from 25% power to 45% power.

4. Table 3.3-15, Page 3/4 3-30c. For Action 16 the ART value of 25% of rated thermal power would be changed to 45%.

5. Table 4.3-15, Page 3/4 3-30d. Anticipatory Reactor Trip System Instrumentation Surveillance Requirements. The value shown for the footnotes

(a) and (b) for the arming threshold for the ART would be changed from 25% reactor power to 45% reactor power.

6. TS 3.4.3, Page 3/4 4-4. Reactor Coolant System Limiting Condition for Operation. The minimum value for the trip setpoint of the electromatic relief valve (also referred to as the PORV) would be changed from 2390 psig to 2435 psig and the allowable value is changed from 2390 psig to 2435 psig.

7. TS 3/4.3.1 and 3/4.3.2, Page B 3/4 3-1. Reactor Protection System and Safety System Instrumentation. The value shown for the arming threshold for the ART would be changed from 25% reactor power to 45% reactor power. Also, and editorial addition would be made to state that "The justification for the ARTS turbine trip arming level of 45% is given in BAW-1893, October 1985."

8. Page B 3/4 4-12. Reactor Coolant System Bases. To reflect changes made in a number of values the following were modified. The RPS high-pressure trip would be changed from 2300 psig to 2355 psig. The tolerance for the code safety valve setting accuracy would be changed from +3% to +1% as presented in Reference 1, Attachment 1 Page 7 to correct an inconsistency. The calculated system peak pressure would be changed from 2716 psig to 2700 psig as a result of a reanalysis. An editorial change would modify "anticipated transients" to "anticipated overpressure transients." Also an insert would be made for explanation as follows. "BAW-1890, September 1985 identified that the turbine trip from full power would cause the largest overpressure transient. This report demonstrated that with a RPS high pressure trip setpoint of 2355 psig the resulting overshoot in RCS pressure would be limited to 50 psi. Consequently, the minimum PORV setpoint needs to accommodate both the RCS pressure overshoot and the RPS instrument string error of 30 psi." This provides an explanation for determining the minimum PORV setpoint of 2435 psig. The actual PORV setpoint at the Davis-Besse plant is 2450 psig which is below the code safety valve setpoint of 2525 psig.

The staff has reviewed the proposed changes identified above and has found them to be acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact has been prepared and published in the Federal Register on December 28, 1988 (53 FR 52529). Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

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.

Principal Contributor: H. Balukjian

Dated: December 28, 1988

REFERENCES

1. Letter from D.C. Shelton, Toledo Edison Company, to U.S. Nuclear Regulatory Commission, dated February 1, 1988.

2. Letter from D.C. Shelton, Toledo Edison Company, to U.S. Nuclear Regulatory Commission, dated February 26, 1988.

3. "Justification for Raising Setpoint for Reactor Trip on High Pressure," BAW-1890, September 1985.

4. "Basis for Raising Arming Threshold for Anticipatory Reactor Trip on Turbine Trip," BAW-1893, October 1985.

5. Letter from D.M. Crutchfield, NRC, to J.H. Taylor, Babcock and Wilcox Company, April 22, 1986.

6. Letter from D.M. Crutchfield, NRC, to J.H. Taylor, Babcock and Wilcox Company, April 25, 1986.

7. DB-1 Startup Event Reanalysis, B&W Documents 32-1167957-00 and 36-1167960-02, July 17, 1987 and August 31, 1987.

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U. S. NUCLEAR REGULATORY COMMISSION

TOLEDO EDISON COMPANY, ET AL.

DOCKET NO. 50-346

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 128 to Facility Operating License No. NPF-3, issued to The Toledo Edison Company and The Cleveland Electric Illuminating Company (the licensees), which revised the Technical Specifications (TS's) for operation of the Davis-Besse Nuclear Power Station, Unit No. 1 (the facility) located in Ottawa County, Ohio. The amendment was effective as of the date of its issuance.

The amendment revised the TS's to permit increasing the Reactor Coolant System high pressure trip setpoint to 2355 psig from the present value of 2300 psig, to permit increasing the Power Operated Relief Valve (PORV) trip setpoint to 2435 psig from the present value of 2390 psig, and to permit increasing the reactor power level at or below which the Anticipatory Reactor Trip System (ARTS) may be blocked to 45 percent from the present value of 25 percent.

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.

Notice of Consideration of Issuance of Amendment and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER

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on May 3, 1988 (53 FR 15758). No request for hearing or petition for leave to intervene was filed following this notice.

For further details with respect to this action see (1) the application for amendment dated February 1, 1988 as supplemented February 26, 1988, (2) Amendment No. 128 to License No. NPF-3, (3) the Commission's related Safety Evaluation dated December 28, 1988, (4) Technical Evaluation Report EGG-NTA-8152 dated June 1988 and (5) the Environmental Assessment dated December 8, 1988 (53 FR 52529). All of these items are available for public inspection at the Commission's Public Document Room, 2120 L Street, N.W., Washington, D.C., and the University of Toledo Library, Documents Department, 2801 Bancroft Avenue, Toledo, Ohio 43606.

A copy of items (2), (3) and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention Director, Division of Reactor Projects - III, IV, V and Special Projects.

Dated at Rockville, Maryland this 28th day of December 1988.

FOR THE NUCLEAR REGULATORY COMMISSION

Albert W. De Agazio

Albert W. De Agazio, Sr. Project Manager
Project Directorate III-3
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Idaho
National
Engineering
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Department
of Energy

EGG-NTA-8152
June 1988

TECHNICAL EVALUATION REPORT

REVIEW OF TECHNICAL SPECIFICATION CHANGE REQUEST FOR HIGH PRESSURE REACTOR TRIP, THE MINIMUM PORV SETPOINT, AND THE ARMING THRESHOLD FOR ANTICIPATORY REACTOR TRIP ON TURBINE TRIP BY TOLEDO EDISON FOR DAVIS-BESSE NUCLEAR POWER STATION UNIT 1

C. P. Fineman

EG&G Idaho

Work performed under
DOE Contract
No. DE-AC07-76ID01570

Prepared for the
U.S. NUCLEAR REGULATORY COMMISSION

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NOTICE

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EGG-NTA-8152

TECHNICAL EVALUATION REPORT
REVIEW OF TECHNICAL SPECIFICATION CHANGE REQUEST FOR HIGH PRESSURE REACTOR TRIP, THE MINIMUM PORV SETPOINT, AND THE ARMING THRESHOLD FOR ANTICIPATORY REACTOR TRIP ON TURBINE TRIP BY TOLEDO EDISON FOR DAVIS-BESSE NUCLEAR POWER STATION UNIT 1

Docket No. 50-346

C. P. Fineman

June 1988

Idaho National Engineering Laboratory
EG&G Idaho, Inc.
Idaho Falls, Idaho 83415

Prepared for the
U.S. Nuclear Regulatory Commission
Washington D.C. 20555
Under DOE Contract No. DE-AC07-76ID01570
FIN No. D6039

ABSTRACT

This report documents the EG&G Idaho, Inc., evaluation of the proposal by Toledo Edison to amend the operating license of the Davis-Besse Nuclear Power Station, Unit 1, to raise the reactor trip setpoint on high pressure and to raise the arming threshold for anticipatory reactor trip on turbine trip. A third request was to raise the minimum PORV setpoint from greater than or equal to 2390 psig to greater than 2435 psig. This evaluation found the proposed changes meet the U.S. Nuclear Regulatory Commission's positions established in their review of the Babcock & Wilcox topical reports and are, therefore, acceptable.

FOREWORD

This report is supplied as part of "CE/B&W/W Plant Reactor Systems Generic Items Support" being conducted for the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Reactor Systems Branch, by EG&G Idaho, Inc., NRC Technical Assistance Group.

The U. S. Nuclear Regulatory Commission funded the work under the authorization B&R 20-19-10-11-2, FIN D-6039.

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TECHNICAL EVALUATION REPORT
REVIEW OF TECHNICAL SPECIFICATION CHANGE REQUEST FOR HIGH PRESSURE REACTOR TRIP, THE MINIMUM PORV SETPOINT, AND THE ARMING THRESHOLD FOR ANTICIPATORY REACTOR TRIP ON TURBINE TRIP BY TOLEDO EDISON FOR DAVIS-BESSE NUCLEAR POWER STATION UNIT 1

1.0 INTRODUCTION

In letters from D. C. Shelton, Toledo Edison, to the U. S. Nuclear Regulatory Commission (NRC) dated February 1, 1988[1] and February 26, 1988,[2] Toledo Edison proposed a license amendment to Facility Operating License NPF-3 for Davis-Besse, Unit 1. The proposed amendment would: (1) increase the setpoint for trip (i.e., shutdown) of the reactor on high pressure in the reactor coolant system from 2300 to 2355 psig and (2) increase the arming threshold for anticipatory reactor trip (ART) on turbine trip from 25% of full power to 45% of full power. As a result of the first change, Toledo Edison requested a third technical specification change for Davis-Besse, Unit 1. This change would raise the minimum pilot operated relief valve (PORV) setpoint from greater than or equal to 2390 psig to greater than 2435 psig. The actual PORV setpoint would remain at 2450 psig. In response to a NRC request, Toledo Edison provided additional supporting information by a letter from D. C. Shelton to the NRC dated June 7, 1988 (Reference 3). This report documents the EG&G Idaho review of the proposed amendment and the supporting information.

2.0 BACKGROUND

The Babcock & Wilcox (B&W) Nuclear Steam Supply System (NSSS) was designed with the capability to adjust to minor plant upsets and certain anticipated events such as feedwater transients, rapid load changes, and turbine trips without a reactor trip. The system was designed to initiate a plant runback, upon detection of an upset or equipment malfunction, to a power level consistent with the plant condition and to limit the rise in Reactor Coolant System (RCS) pressure to less than the reactor trip setpoint by opening the pressurizer PORV. Following the TMI-2 accident, the NRC, by IE Bulletin 79-05B,[4] required licensees for all B&W Pressurized Water Reactor (PWR) facilities to make modifications to reduce the number of automatic actuations of the PORVs. The modifications proposed by B&W on behalf of the owners group and accepted by the NRC included (1) raising the PORV setpoint from 2255 to 2450 psig, (2) decreasing the reactor trip on high RCS pressure from 2390 to 2300 psig, and (3) providing an ART for turbine trips above 20% power. In addition in NUREG-0737,[5] the NRC required that B&W demonstrate these modifications: (1) limited the frequency of PORV openings to less than 5% of the total number of overpressure transients (Item II.K.3.7) and (2) limited the probability of a small break Loss of Coolant Accident (LOCA) caused by a stuck open PORV to less than .001 per reactor-year (Item II.K.3.2). B&W submitted a report[6] to demonstrate that the modifications did meet the requirements. The NRC issued a Safety Evaluation Report[7] which concluded that the requirements were met.

Although these modifications met the objectives of reducing challenges to and opening of the PORV, they increased the frequency of reactor trips. Each reactor trip results in a challenge to plant safety systems and any reduction in reactor trip frequency will contribute to overall plant safety as well as plant availability. On behalf of the B&W Owners Group, B&W submitted two reports: (1) Justification for Raising Setpoint for Reactor Trip on High Pressure, BAW-1890, September 1985[8] and (2) Basis for Raising Arming Threshold for Anticipatory Reactor Trip on Turbine Trip, BAW-1893, October 1985.[9] These reports presented B&W's justification for raising the high pressure reactor trip setpoint from

2300 to 2355 psig and increasing the arming threshold for the ART for turbine trips from 20 to 45% power. These reports state the NRC requirements for limiting the frequency of PORV openings and limiting the probability of a small break LOCA due to a stuck open PORV will still be met with these changes. Further, these reports state these changes will reduce the number of reactor trips per reactor year by approximately 10%.

In April 1986, the NRC staff completed its review of these two B&W reports. In their Safety Evaluation Reports (SERs), the staff: (1) reviewed the basis for the proposed changes; (2) reviewed B&W's method of analysis of the effect of the proposed high pressure trip setpoint on PORV openings; (3) compared the results of a Monte Carlo simulation for PORV openings with the NRC requirements contained in NUREG-0737; and (4) reviewed the results of B&W's analysis of the arming threshold for ART. As stated above, the NRC requirements in NUREG-0737 include: (1) the PORV will open in less than 5% of all anticipated overpressure transients (Item II.K.3.7) and (2) the probability of a small break loss of coolant accident caused by a stuck open PORV will be less than 0.001 per reactor-year (Item II.K.3.2). In the SERs,[10,11] the staff concluded on a generic basis that the proposed changes met the NRC requirements and should benefit plants by potentially reducing the reactor trip frequency. Accordingly, the NRC concluded that the B&W reports could be referenced in licensing submittals by the B&W Owners Group members.

3.0 EVALUATION

The licensee in Reference 1 proposed three technical specification changes. These were raising the high pressure reactor trip setpoint from 2300 to 2355 psig, increasing the arming threshold for the ART on turbine trip to 45% power, and increasing the minimum PORV setpoint from greater than or equal to 2390 psig to greater than 2435 psig. The change requests on the high pressure reactor trip setpoint and the ART arming threshold will be discussed first followed by the change request on the minimum PORV setpoint.

The licensee is a member of the B&W Owners Group and in its February 1, 1988 proposal (Reference 1) stated the B&W reports in References 8 and 9 provided the justification for raising the high pressure reactor trip setpoint from 2300 to 2355 psig and increasing the arming threshold for the ART on turbine trip to 45% power. The licensee stated these reports are applicable to Davis-Besse because the available steam bypass, including the first bank of main steam safety valves, is 55% of full steam flow which is conservatively higher than the 43% used in the analysis. The licensee noted the limiting transient in Reference 9 used end of life (EOL) core conditions to represent the temperature moderator coefficient. The licensee stated this was not conservative for Davis-Besse because the moderator coefficient is more negative at EOL than at beginning of life (BOL), and this negative feedback provides a greater core power decrease following a turbine trip. However, the licensee stated that performing the analysis at EOL is satisfactory for Davis-Besse because the ART arming threshold is only to be raised to 45% of full power and the plant, as noted above, has bypass capacity equal to 55% of full steam flow. Therefore, no decrease in core power caused by the temperature moderator coefficient is necessary to for Davis-Besse steaming capacity to terminate the system pressurization transient. This argument is acceptable.

In addition, after reviewing the B&W reports, EG&G Idaho concluded they apply to Davis-Besse because Davis-Besse is a 177A plant for which the reports apply and because the historical data and pressure overshoot distribution included data from Davis-Besse and similar B&W plants. Based

on the information provided the licensee and a review of the B&W reports, EG&G Idaho concluded the data and parameters used in the B&W reports are a conservative bound for Davis-Besse and use of these reports is acceptable.

The important conclusion of the B&W reports which justifies raising the high pressure trip and the ART for turbine trip is that even though both changes will result in a small increase in the probability of opening the PORV, the increase is insignificant compared to the total openings of the PORV from all events. The estimate made in the B&W reports (References 8 and 9) for PORV openings from high pressure events is 1.96×10^{-5} PORV openings per rector year. This estimate assumed a reactor high pressure trip setpoint of 2355 psig, the ART for turbine trip was set at 45%, the PORV setpoint was 2450 psig, the turbine trip frequency remained at 1980 to 1984 levels, and 30% of the runbacks from power levels less than or equal to 45% were unsuccessful. The estimate made in the B&W report (Reference 8) for PORV openings from all events is 8.06×10^{-2} PORV openings per reactor year. The major contributor to the PORV openings was identified as actions by the operators in carrying out plant operation in accordance with the Abnormal Transient Operating Guidelines (ATOG). The 1.96×10^{-5} PORV openings per reactor year from high pressure events applies to Davis-Besse because the B&W reports apply to the plant as discussed in the preceding paragraphs. The 8.06×10^{-2} PORV openings per reactor year from all events applies to Davis-Besse because the plant specific Abnormal Transient Procedures (ATPs) are based on the ATOG. Therefore, for Davis-Besse the PORV openings from high pressure events with the proposed setpoints are insignificant compared with the openings from all events.

Toledo Edison stated in Reference 1 that the original FSAR Chapter 15 analyses used a high pressure trip of 2355 psig but the safety valve setpoint was 2435 psig. The safety valve setpoint is now 2500 psig. To analyze the effect of the higher safety valve setpoint in combination with the 2355 psig reactor trip setpoint, B&W reanalyzed the control rod assembly withdrawal accident at start-up which is the most severe overpressure transient at Davis-Besse.[12] The FSAR safety criteria applied to the accident are primary system pressure less than 2750 psig

and peak thermal power less than 112% of design full power. The analysis results with the 2355 psig reactor trip and 2500 psig safety valve setpoint show a peak pressure of 2700 psig and a peak thermal power of 48.5%. Thus, the safety criteria are still met. Toledo Edison stated these results are also bounded by the original FSAR analysis.

The Licensee in Reference 3 stated additional challenges to the Main Steam Safety Valves (MSSVs) are not expected with the proposed setpoint changes as compared to the openings with the current settings. Toledo Edison provided the following information to support this statement. First, the licensee noted that, even with an ART, plant data shows the MSSVs have opened during a turbine trip from 40% power with the current plant setpoints for arming of the ART and high pressure reactor trip. Toledo Edison provided the following information to clarify the plant data. During a normal plant runback (without reactor trip), the turbine bypass valve has a 920 psig setpoint. Therefore, following a turbine trip without the ART, the turbine bypass valves begin to open almost immediately, limiting the secondary pressure increase. However, following a reactor trip the turbine bypass valve setpoint is 1015 psig (due to control system biasing). Therefore, following a turbine trip with the ART, the turbine bypass valves remain closed until the higher setpoint is reached which pressurizes the secondary. The turbine bypass valves will begin to open only 35 psig below the 1050 psig setpoint for the first bank of MSSVs. Thus, after an ART the secondary pressure only has to increase 35 psig before the first bank of MSSVs is challenged compared to 130 psig without an ART.

Toledo Edison also noted that a B&W Owners Group analysis in Reference 9 showed that all 177A plants should be able to successfully run back without an ART from 30% power with 15% turbine bypass and the current reactor trip setpoint of 2300 psig without challenging the MSSVs. Because Davis-Besse has 25% turbine bypass rather than the 15% bypass assumed in the analysis, Toledo Edison stated it expected the plant to be able to successfully run back from approximately 40% power without challenging the MSSVs.

Taken together, the preceding two paragraphs indicate the new ART arming setpoint could reduce challenges to the MSSVs at less than 40% power. Toledo Edison in a conference call with the NRC Staff and EG&G Idaho, on June 6, 1988, stated this was a possibility but it did not have the data or analysis to completely justify that statement. However, Toledo Edison did think the reasoning supported the conclusion that the changes would not increase the challenges to the MSSVs. EG&G Idaho agrees with this conclusion. With respect to the 40 to 45% power range, Toledo Edison stated in the June 6, 1988 conference call that the first bank of MSSVs would probably lift at the current (25%) or proposed (45%) arming threshold for the ART. Therefore, the number of challenges to the MSSVs in this power range would be the same with either ART arming threshold.

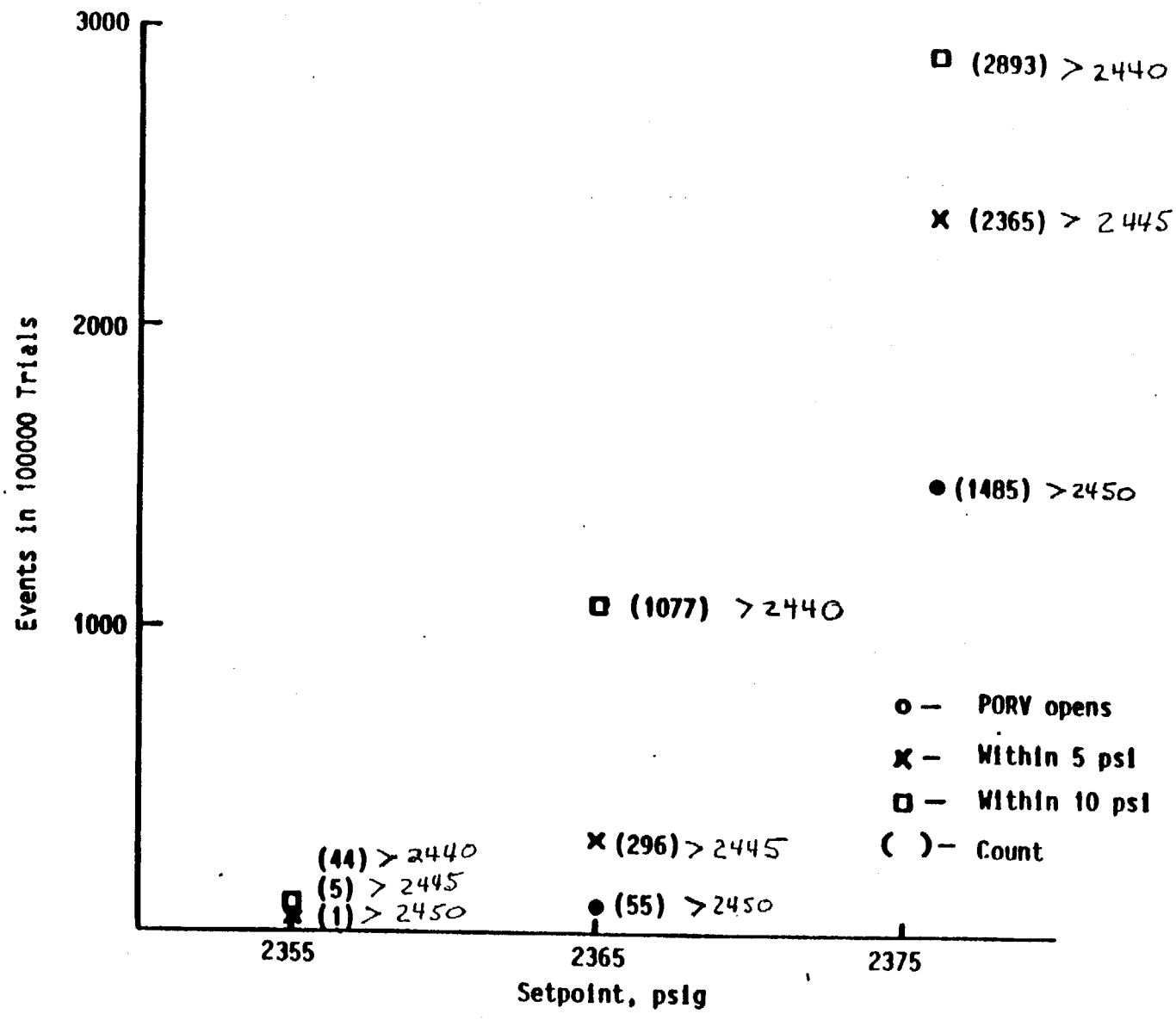
The final change request made by Toledo Edison was to raise the minimum PORV setpoint. During a conference call with the NRC Staff and EG&G Idaho on June 17, 1988, Toledo Edison clarified the actual technical specification change would raise the minimum PORV setpoint from greater than or equal to 2390 psig to greater than 2435 psig. Toledo Edison stated the actual setpoint would remain at 2450 psig. The basis for this change was stated to be the requirement in Technical Specification Bases 3/4.4.3 that the PORV should not open during any anticipated overpressure transient. This means the the minimum PORV setpoint needs to accommodate a 30 psi instrument string error and, based on Reference 9, a high pressure reactor trip setpoint overshoot of 50 psia. This resulted in a maximum indicated pressure of 2435 psig. The old minimum PORV setpoint of 2390 psig no longer met this requirement. Changing the minimum PORV setpoint to be greater than 2435 psig will meet this requirement because the minimum setpoint will greater than the maximum indicated pressure.

During the review of the above change, EG&G Idaho noted that the PORV setpoint assumed in the analyses of References 8 and 9 was 2450 psig. By allowing a minimum PORV setpoint of 2435 psig in the Technical Specifications, the PORV theoretically could be set to open at a pressure below that analyzed in References 8 and 9 and, as a result, these references would no longer apply to Davis-Besse. Reference 8 was reviewed to determine whether it contained any data that would allow extension of

the report to the minimum PORV setpoint of 2435 psig. Figure 4.4-1 of Reference 8, presented the results of Monte Carlo simulations (number of PORV openings per 100,000 trials) assuming reactor trip setpoints of 2355, 2365, and 2375 psi. This figure is shown on the next page. The figure also shows the number of simulations where the pressure was calculated to increase to within 5 and 10 psi of the 2450 psi PORV setpoint. While there is not a direct correlation between the effects of raising the reactor trip setpoint 10 psi and reducing the PORV setpoint a similar amount, by using the results of the Monte Carlo simulations presented in the figure, it was possible to conservatively estimate the PORV opening frequency for overpressure transients at the 2435 psig minimum setpoint.

A conservative estimate of PORV opening frequency assuming a 2435 psig setpoint was developed as follows. First, with a 2355 psig reactor trip setpoint, it was noted that 44 simulations came within 10 psi of the 2450 psig setpoint. That is with a reactor trip setpoint of 2355 psig, if the PORV setpoint had been 2440 psig, the PORV would have opened 44 times in 100,000 trials. Second, it was noted that with a reactor trip setpoint of 2365 psig, the PORV opened 55 times in 100,000 trials at the 2450 psig setpoint. This implies that increasing the reactor trip setpoint 10 psi results in a larger number of PORV openings than decreasing the PORV setpoint 10 psi. Third, it was noted that for the 2365 psig reactor trip setpoint, 296 simulations came within 5 psi of the 2450 psig setpoint. That is with a reactor trip setpoint of 2365 psig, if the PORV setpoint had been 2445 psig, the PORV would have opened 296 times in 100,000 trials. Finally, using the result that a 10 psi increase in the reactor trip setpoint results in a larger number of PORV openings than a 10 psi reduction in the PORV setpoint, it was concluded that 296 openings in 100,000 trial would conservatively bound the number of PORV openings for a reactor trip setpoint of 2355 psig and a PORV setpoint of 2435 psig.

This value of 296 PORV openings in 100,000 trials was used to determine whether the 2435 psig setpoint met the NRC requirements in NUREG-0737, Items II.K.3.2 and II.K.3.7. With respect to Item II.K.3.7, it is noted that a PORV opening frequency of 2.96×10^{-3} is much less



Results of Monte Carlo Simulations from BAW-1890

than the NRC requirement of 5.0×10^{-2}. Therefore, Item II.K.3.7 is met at the minimum setpoint of 2435 psig. With respect to Item II.K.3.2, the calculations in Reference 10, Section II.D, were repeated with the PORV opening frequency of 2.96×10^{-3} and assuming 1.96 high pressure reactor trips per year (Reference 11, Section II.C). The probability of a SBLOCA due to a stuck open PORV was calculated to be 1.73×10^{-3}. This is comparable to the 1.6×10^{-3} calculated in Reference 10 and falls within the WASH-1400[13] range of 10^{-2} and 10^{-4} for the probability of a SBLOCA. Therefore, Item II.K.3.2 is met at the minimum setpoint of 2435 psig.

Therefore, the change to raise the minimum PORV setpoint from greater than or equal to 2390 psig to greater than 2435 psig is considered acceptable because, with a minimum setpoint greater than 2435 psig, the PORV should not open during anticipated overpressure transients (the setpoint accommodates the 30 psi instrument error and 50 psi pressure overshoot), the actual PORV setpoint will remain at 2450 psig, and NUREG-0737, Items II.K.3.2 and II.K.3.7, are satisfied at the minimum setpoint of 2435 psig.

4. CONCLUSIONS

Proposed changes to the following Davis-Besse technical specifications were reviewed:

Section 2.1.3, Reactor Coolant System Pressure, Figure 2.2-1

Section 2.2.1, Reactor Protection System Setpoints, Table 2.2-1

Section 3/4.3.2.3, Anticipatory Trip System Instrumentation, Tables 3.3-15 and 4.3-15

Section 3/4.4.3, Safety Valves and Electromatic Relief Valve-Operating Bases

Sections 3/4.3.1 and 3/4.3.2, Reactor Protection System and Safety System Instrumentation

Bases Sections 3/4.4.2 and 3/4.4.3, Safety Valves

The proposed changes would (1) increase the setpoint for trip of the reactor on high pressure in the reactor coolant system from 2300 to 2355 psig, (2) increase the arming threshold for ART on turbine trip from 25% of full power to 45% of full power, and (3) raise the minimum PORV setpoint from greater than or equal to 2390 psig to greater than 2435 psig. The actual PORV setpoint would remain at 2450 psig. As discussed in the preceding section, the proposed changes meet the NRC positions established in their review of the B&W topical reports and, therefore, meet the applicable regulatory guidance and requirements. Therefore, the proposed changes are acceptable.

5. REFERENCES

1. Letter from D. C. Shelton, Toledo Edison, to U. S. Nuclear Regulatory Commission, Serial No. 1464, February 1, 1988.

2. Letter from D. C. Shelton, Toledo Edison, to U. S. Nuclear Regulatory Commission, Serial No. 1496, February 26, 1988.

3. Letter from D. C. Shelton, Toledo Edison, to U. S. Nuclear Regulatory Commission, Serial No. 1538, June 7, 1988.

4. IE Bulletin 79-05B, "Nuclear-Incident at Three Mile Island - Supplement," April 2, 1979.

5. Clarification of TMI Action Plan Requirements, NUREG-0737, November 1980.

6. Babcock & Wilcox Report 12-1122779 Rev. 1, Report on PORV Opening Probability and Justification for Present Systems and Setpoints, January 1981.

7. NRC Memorandum from F. H. Rowsome to G. C. Laines, "Safety Evaluation of the B&W Licensees' Responses to TMI Action II.K.3.2," August 24, 1983.

8. Justification for Raising Setpoint for Reactor Trip on High Pressure, BAW-1890, September 1985.

9. Basis for Raising Arming Threshold for Anticipatory Rector Trip on Turbine Trip, BAW-1893, October 1985.

10. Letter from D. M. Crutchfield, NRC, to J. H. Taylor, B&W, "Acceptance for Referencing of Licensing Topical Report BAW-1890, Justification for Raising Setpoint for Reactor Trip on High Pressure," April 27, 1986.

11. Letter from D. M. Crutchfield, NRC, to J. H. Taylor, B&W, "Acceptance for Referencing of Licensing Topical Report BAW-1893, Basis for Raising Threshold for Anticipatory Reactor Trip on Turbine Trip," April 25, 1986.

12. DB-1 Startup Event Reanalysis, B&W Documents 32-1167957-00 and 36-1167960-02, July 17, 1987 and August 31, 1987.

13. Reactor Safety Study - An Assessment of Accident Risks in U. S. Commercial Nuclear Power Plants, WASH-1400, 1975.

NRC FORM 335 (12-84) NRCM 1102 3201, 3202

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EG&G Idaho, Inc.
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This report documents the EG&G Idaho, Inc., evaluation of the proposal by Toledo Edison to amend the operating license of the Davis-Besse Nuclear Power Station, Unit 1, to raise the reactor trip setpoint on high pressure and to raise the arming threshold for anticipatory reactor trip on turbine trip. A third request was to raise the minimum PORV setpoint from greater than or equal to 2390 psig to greater than 2435 psig. This evaluation found the proposed changes meet the U.S. Nuclear Regulatory Commission's positions established in their review of the Babcock and Wilcox topical reports and are, therefore, acceptable.

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