MARCH 2 4 1981

Docket No. 50-346

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U.S. NUCLEAR REGULATORY COMMISSION

Mr. Richard P. Crouse
Vice President, Nuclear
Toledo Edison Company
Edison Plaza
300 Madison Avenue
Toledo, Ohio 43652

Dear Mr. Crouse:

The Commission has issued the enclosed Amendment No. 37 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 filing dated September 16, 1980.

This amendment adds Technical Specifications which incorporate certain of the Three Mile Island Unit No. 2 Lessons Learned Category "A" requirements.

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

Your submittal was in response to our model Technical Specifications enclosed in our letter of July 2, 1980. In our letter we also stated that your license should be amended by adding license conditions related to a System Integrity Measurements Program and Improved Iodine Monitoring Program. Although your submittal did not address these conditions, it is our position that your license should be amended to include these programs. In lieu of license conditions, the program requirements should be included in Section 6.0, "Administrative Controls", of the Davis-Besse Technical Specifications. You are requested to propose an amendment to your license to include these programs within 30 days of receipt of this letter or provide your reasons for excluding them.

Sincerely,

Original signed by

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

8103300481

Enclosures & cc: See next page

*SEE PREVIOUS WHITE FOR CONCURRENCES

OFFICE	ORB#4:DL	ORB#4:DL	C-ORB#4:DL	AD:OR:DL	OELD		
SURNAME	DGarner/cb	RIngram*	JStolz	TNovak*	MRothschild*		
DATE	3/2?/81	1/2/81	3/13/81	1/9/81	1/14/81		

Enclosures:
1. Amendment No. 37
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

Concur in amendment Fed. Reg. Notice & [illegible], subject to editorial correction of letter (as noted on letter) MWR

OFFICE	ORB#4:DL	ORB#4:DL	C-ORB#4:DL	AD-OR:DL	OELD		
SURNAME	DGarner/cb	RIngram	RReid	TNovak	M. Rothschild		
DATE	~~12~~/5/~~80~~81	~~12~~/2/~~80~~81	~~12~~/ /~~80~~81	~~12~~/9/81	~~12~~/14/~~80~~81		

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

March 24, 1981

Docket No. 50-346

Mr. Richard P. Crouse
Vice President, Nuclear
Toledo Edison Company
Edison Plaza
300 Madison Avenue
Toledo, Ohio 43652

Dear Mr. Crouse:

The Commission has issued the enclosed Amendment No. 37 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 filing dated September 16, 1980.

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Your submittal was in response to our model Technical Specifications enclosed in our letter of July 2, 1980. In our letter we also stated that your license should be amended by adding license conditions related to a System Integrity Measurements Program and Improved Iodine Monitoring Program. Although your submittal did not address these conditions, it is our position that your license should be amended to include these programs. In lieu of license conditions, the program requirements should be included in Section 6.0, "Administrative Controls", of the Davis-Besse Technical Specifications. You are requested to propose an amendment to your license to include these programs within 30 days of receipt of this letter or provide your reasons for excluding them.

Sincerely,

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures & cc: See next page

Enclosures:
1. Amendment No. 37
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
Port Clinton, Ohio 43452

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Oak Harbor, Ohio 43449

Director, Criteria and Standards
Division
Office of Radiation Programs (ANR-460)
U. S. Environmental Protection Agency
Washington, D. C. 20460

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.:
9/16/80

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. 37
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 September 16, 1980, 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.

8103300 489

2. Accordingly, Facility Operating License No. NPF-3 is hereby amended as indicated below and by changes to the Technical Specifications as indicated in the attachment to this license amendment:

   Revise paragraph 2.C.(2) to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 37, 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

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 24, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 37

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

1-8

3/4 3-10

3/4 3-11

3/4 3-12

3/4 3-21

3/4 3-22

3/4 3-27

3/4 3-48

3/4 3-50

6-4

6-5

TABLE 1.1

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	%RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	$\geq$ 0.99	> 5%	$\geq$ 280°F
2. STARTUP	$\geq$ 0.99	$\leq$ 5%	$\geq$ 280°F
3. HOT STANDBY	< 0.99	0	$\geq$ 280°F
4. HOT SHUTDOWN	< 0.99	0	280°F > T_{avg} > 200°F
5. COLD SHUTDOWN	< 0.99	0	$\leq$ 200°F
6. REFUELING**	$\leq$ 0.95	0	$\leq$ 140°F

*Excluding decay heat.

**Reactor vessel head unbolted or removed and fuel in the vessel.

TABLE 1.2

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 6 months.*
R	At least once per 18 months.*
S/U	Prior to each reactor startup.
N.A.	Not applicable.

* In these Technical Specifications, 6 months is defined to be 184 days, and 18 months is defined to be 550 days.

3/4.3.2 SAFETY SYSTEM INSTRUMENTATION

SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.1 The Safety Features Actuation System (SFAS) functional units shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With a SFAS functional unit trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the functional unit inoperable and apply the applicable ACTION requirement of Table 3.3-3, until the functional unit is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint value.

b. With a SFAS functional unit inoperable, take the action shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1.1 Each SFAS functional unit shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST during the MODES and at the frequencies shown in Table 4.3-2.

4.3.2.1.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of functional units affected by bypass operation. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each functional unit affected by bypass operation.

4.3.2.1.3 The SAFETY FEATURES RESPONSE TIME of each SFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one functional unit per function such that all functional units are tested at least once every N times 18 months where N is the total number of redundant functional units in a specific SFAS function as shown in the "Total No. of Units" Column of Table 3.3-3.

TABLE 3.3-3

SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF UNITS	UNITS TO TRIP	MINIMUM UNITS OPERABLE	APPLICABLE MODES	ACTION
1. INSTRUMENT STRINGS					
a. Containment Radiation - High	4	2	3	1,2,3,4,6****	9#
b. Containment Pressure - High	4	2	3	1, 2, 3	9#
c. Containment Pressure - High-High	4	2	3	1, 2, 3	9#
d. RCS Pressure - Low	4	2	3	1, 2, 3*	9#
e. RCS Pressure - Low-Low	4	2	3	1, 2, 3**	9#
f. BWST Level - Low	4	2	3	1, 2, 3	9#
2. OUTPUT LOGIC					
a. Incident Level #1: Containment Isolation	2	1	2	1,2,3,4,6****	10
b. Incident Level #2: High Pressure Injection and Starting Diesel Generators	2	1	2	1, 2, 3, 4	10
c. Incident Level #3: Low Pressure Injection	2	1	2	1, 2, 3, 4	10
d. Incident Level #4: Containment Spray	2	1	2	1, 2, 3, 4	10
e. Incident Level #5: Containment Sump Recirculation	2	1	2	1, 2, 3, 4	10

TABLE 3.3-3

SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF UNITS	UNITS TO TRIP	MINIMUM UNITS OPERABLE	APPLICABLE MODES	ACTION
3. MANUAL ACTUATION					
a. SFAS (except Containment Spray and Emergency Sump Recirculation)	2	2	2	1,2,3,4,6****	11
b. Containment Spray	2	2	2	1, 2, 3, 4	11
4. SEQUENCE LOGIC CHANNELS	4	2***	4	1, 2, 3, 4	9#
5. INTERLOCK CHANNELS					
a. Decay Heat Isolation Valve	1	1	1	1, 2, 3, 4, 5	12#
b. Pressurizer Heaters	2	2	2	3,4,5	13#

TABLE 3.3-3 (Continued)

TABLE NOTATION

* Trip function may be bypassed in this MODE with RCS pressure below 1800 psig. Bypass shall be automatically removed when RCS pressure exceeds 1800 psig.

** Trip function may be bypassed in this MODE with RCS pressure below 600 psig. Bypass shall be automatically removed when RCS pressure exceeds 600 psig.

*** One must be in SFAS Channels #1 or #3, the other must be in Channels #2 or #4.

**** This instrumentation must be OPERABLE during core alterations or movement of irradiated fuel within the containment to meet the requirements of Tech. Spec. 3.9.4.

\# The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 9 - With the number of OPERABLE functional units one less than the Total Number of Units, startup and/or power operation may proceed provided both of the following conditions are satisfied:

a. The inoperable functional unit is placed in the tripped condition within one hour.

b. The Minimum Units OPERABLE requirement is met; however, one additional functional unit may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

ACTION 10 - With any component in the Output Logic inoperable, trip the associated components within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 11 - With the number of OPERABLE Units one less than the Total Number of Units, restore the inoperable functional unit to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 12 - a. With less than the Mimimum Units OPERABLE and reactor coolant pressure $\geq$ 438 psig, both Decay Heat Isolation Valves (DH11 and DH12) shall be verified closed.

b. With Less than the Minimum Units OPERABLE and reactor coolant pressure < 438 psig operation may continue; however, the functional unit shall be OPERABLE prior to increasing reactor coolant pressure above 438 psig.

ACTION 13 - With less than the Minimum Units OPERABLE and reactor coolant pressure <438 psig, operation may continue; however, the functional unit shall be OPERABLE prior to increasing reactor coolant pressure above 438 psig, or the inoperable functional unit shall be placed in the tripped state.

TABLE 4.3-2

SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. INSTRUMENT STRINGS				
a. Containment Radiation - High	S	R	M	1,2,3,4,6#
b. Containment Pressure - High	S	R	M(2)	1, 2, 3
c. Containment Pressure - High-High	S	R	M(2)	1, 2, 3
d. RCS Pressure - Low	S	R	M	1, 2, 3
e. RCS Pressure - Low-Low	S	R	M	1, 2, 3
f. BWST Level - Low	S	R	M	1, 2, 3
2. OUTPUT LOGIC				
a. Incident Level #1: Containment Isolation	S	R	M	1,2,3,4,6#
b. Incident Level #2: High Pressure Injection and Starting Diesel Generators	S	R	M	1, 2, 3, 4
c. Incident Level #3: Low Pressure Injection	S	R	M	1, 2, 3, 4
d. Incident Level #4: Containment Spray	S	R	M	1, 2, 3, 4
e. Incident Level #5: Containment Sump Recirculation	S	R	M	1, 2, 3, 4
3. MANUAL ACTUATION				
a. SFAS (Except Containment Spray and Emergency Sump Recirculation)	NA	NA	M(1)	1,2,3,4,6#
b. Containment Spray	NA	NA	M(1)	1, 2, 3
4. SEQUENCE LOGIC CHANNELS	S	NA	M	1, 2, 3, 4

TABLE 4.3-2

SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
5. INTERLOCK CHANNELS				
a. Decay Heat Isolation Valve	S	R	**	1, 2, 3, 4, 5
b. Pressurizer Heater	S	R	**	3,4,5

**See Specification 4.5.2.d.1

TABLE NOTATION

(1) Manual actuation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual safeguards actuation shall receive a CHANNEL FUNCTIONAL TEST at least once per 31 days.

(2) The CHANNEL FUNCTIONAL TEST shall include exercising the transmitter by applying either vacuum or pressure to the appropriate side of the transmitter. The provisions of Section 3.0.3 are not applicable for the first test of each channel following the first refueling outage.

The surveillance requirements of Section 4.9.4 apply during core alterations or movement of irradiated fuel within the containment.

TABLE 3.3-11 (Continued)

TABLE NOTATION

* May be bypassed when steam pressure is below 650 psig. Bypass shall be automatically removed when the steam pressure exceeds 650 psig.

The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 13 - With the number of OPERABLE Channels one less than the Total Number of Channels, startup and/or power operation may proceed until performance of the next required CHANNEL FUNCTIONAL TEST provided the inoperable section of the channel is placed in the tripped condition within 1 hour.

ACTION 14 - With the number of OPERABLE Channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

TABLE 3.3-12

STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNITS	TRIP SETPOINTS	ALLOWABLE VALUES
1. Steam Line Pressure - Low	$\geq$ 591.6 psig	$\geq$ 591.6 psig* $\geq$ 586.6 psig**
2. Steam Generator Level - Low(1)	$\geq$ 20" H_2O	$\geq$ 20" H_2O* $\geq$ 18.87" H_2O**
3. Steam Generator Feedwater Differential Pressure - High(2)	$\leq$ 197.6 psid	$\leq$ 197.6 psid* $\leq$ 199.6 psid**
4. Reactor Coolant Pumps - Loss of	High $\leq$ 1384.6 amps Low $\geq$ 106.5 amps	$\leq$ 1384.6 amps# $\geq$ 106.5 amps#

(1)Measured above the lower steam generator tubesheet.

(2)Where differential pressure is steam generator minus feedwater pressure.

*Allowable Value for CHANNEL FUNCTIONAL TEST

**Allowable Value for CHANNEL CALIBRATION

#Allowable Value for CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION

TABLE 3.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE
1. SG Outlet Steam Pressure	1/Steam Generator
2. RC Loop Outlet Temperature	2/Loop
3. RC Loop Pressure	2/Loop
4. Pressurizer Level	2
5. SG Startup Range Level	2/Steam Generator
6. Auxiliary Feedwater Status	1/AFW System
7. Containment Vessel Hydrogen	2
8. Containment Vessel Post-Accident Radiation	2
9. Containment Vessel Isolation Status	1/Valve
10. SFAS Status	1/Channel
11. Safety Features Equipment Status	1/System
12. RPS Status	1/Channel
13. SFRCS Status	1/Channel
14. High Pressure Injection Flow	1/Channel

TABLE 3.3-10 (Continued)

POST-ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE
15. Low Pressure Injection (DHR) Flow	1/Channel
16. HPI System Pump and Valve Status	1/System
17. LPI System Pump and Valve Status	1/System
18. Containment Spray Pump and Valve Status	1/System
19. Core Flood Valve Status	1/System
20. BWST Valve Status	1/System
21. Containment Emergency Sump Valve Status	1/Valve
22. Containment Air Recirculation Fan Status	1/Fan
23. Containment Air Cooling Fan Status	1/Fan
24. EVS Fan and Damper Status	1/System
25. Auxiliary Feedwater Flow Rate	1/Steam Generator
26. RC System Subcooling Margin Monitor	1
27. PORV Position Indicator	1
28. PORV Block Valve Position Indicator	1
29. Safety Valve Position Indicator	1/Valve
30. BWST Level	3

TABLE 4.3-10

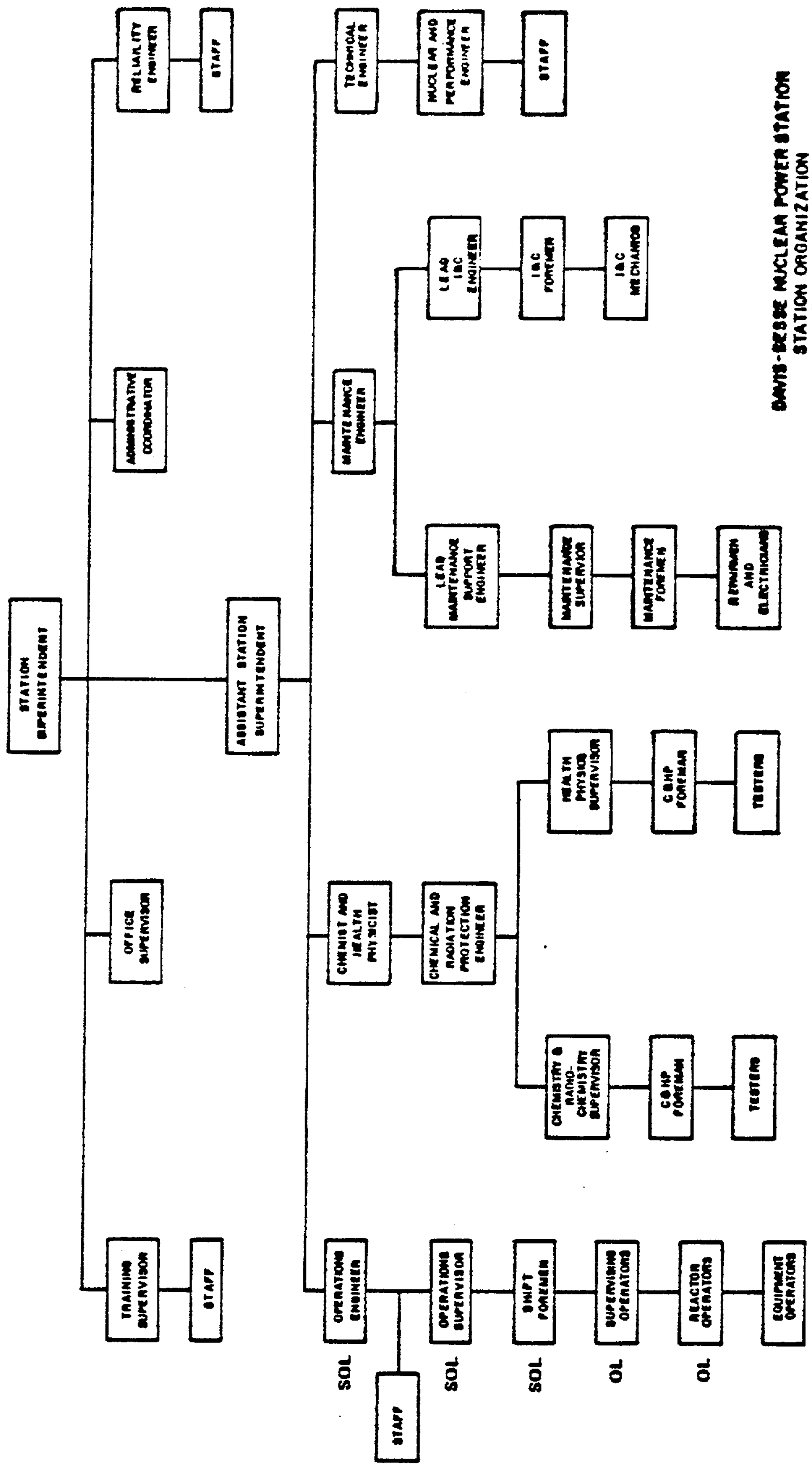
POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. SG Outlet Steam Pressure	M	R
2. RC Loop Outlet Temperature	M	R
3. RC Loop Pressure	M	R
4. Pressurizer Level	M	R
5. SG Startup Range Level	M	R
6. Auxiliary Feedwater Status	M	NA
7. Containment Vessel Hydrogen	M	R
8. Containment Vessel Post-Accident Radiation	M	R
9. Containment Vessel Isolation Status	M	NA
10. SFAS Status	M	NA
11. Safety Features Equipment Status	M	NA
12. RPS Status	M	NA
13. SFRCS Status	M	NA
14. High Pressure Injection Flow	M	R

TABLE 4.3-10 (Continued)

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
15. Low Pressure Injection (DHR) Flow	M	R
16. HPI System Pump and Valve Status	M	NA
17. LPI System Pump and Valve Status	M	NA
18. Containment Spray Pump and Valve Status	M	NA
19. Core Flood Valve Status	M	NA
20. BWST Valve Status	M	NA
21. Containment Emergency Sump Valve Status	M	NA
22. Containment Air Recirculation Fan Status	M	NA
23. Containment Air Cooling Fan Status	M	NA
24. EVS Fan and Damper Status	M	NA
25. Auxiliary Feedwater Flow Rate	M	R
26. RC System Subcooling Margin Monitor	M	R
27. PORV Position Indicator	M	R
28. PORV Block Valve Position Indicator	M	R
29. Pressurizer Safety Valve Position Indicator	M	R
30. BWST Level	S	R



DAVIS-BESSE NUCLEAR POWER STATION
STATION ORGANIZATION
FIGURE 6.2-2

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION#

LICENSE CATEGORY	APPLICABLE MODES	
	1, 2, 3 & 4	5 & 6
SOL	1	1*
OL	2	1
Non-Licensed	2	1
Shift Technical Advison	1	None Required

*Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling supervising CORE ALTERATIONS.

#Shift crew composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the (Radiation Protection Manager) who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975 and (2) the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the (position title) and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Fire Marshall and shall meet or exceed the requirements of Section 27 of the NFPA Code-1976.

6.5 REVIEW AND AUDIT

6.5.1 STATION REVIEW BOARD (SRB)

FUNCTION

6.5.1.1 The Station Review Board (SRB) shall function to advise the Station Superintendent on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The Station Review Board shall be composed of the:

Chairman:	Assistant Station Superintendent
Member:	Operations Engineer
Member:	Technical Engineer
Member:	Maintenance Engineer
Member:	Lead Instrument and Control Engineer
Member:	Nuclear and Performance Engineer
Member:	Chemist and Health Physicist
Member:	Reliability Engineer
Member:	Station Superintendent

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the SRB Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in SRB activities at any one time.

MEETING FREQUENCY

6.5.1.4 The SRB shall meet at least once per calendar month and as convened by the SRB Chairman or his designated alternate.

QUORUM

6.5.1.5 A quorum of the SRB shall consist of the Chairman or his designated alternate and four members including alternates.

RESPONSIBILITIES

6.5.1.6 The Station Review Board shall be responsible for:

a. Review of 1) all procedures required by Specification 6.8 and changes thereto, 2) any other proposed procedures or changes thereto as determined by the Station Superintendent to affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear safety.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 37 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

I. INTRODUCTION

By letter dated September 16, 1980, the Toledo Edison Company (the licensee) proposed changes to the Technical Specifications (TSs) appended to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station, Unit No. 1. The changes involve the incorporation of certain of the TMI-2 Lessons Learned Category "A" requirements. The licensee's request is in direct response to the NRC staff's letter dated July 2, 1980.

II. BACKGROUND INFORMATION

By our letter dated September 13, 1979, we issued to all operating nuclear power plants requirements established as a result of our review of the TMI-2 accident. Certain of these requirements, designated Lessons Learned Category "A" requirements, were to have been completed by the licensee prior to any operation subsequent to January 1, 1980. Our evaluation of the licensee's compliance with these Category "A" items was attached to our letter to the licensee dated May 6, 1980.

In order to provide reasonable assurance that operating reactor facilities are maintained within the limits determined acceptable following the implementation of the TMI-2 Lessons Learned Category "A" items, we requested that licensees amend their TSs to incorporate additional Limiting Conditions of Operation and Surveillance Requirements, as appropriate. This request was transmitted to all licensees on July 2, 1980. Included therein were model specifications that we had determined to be acceptable. The licensee's application is in direct response to our request. Each of the issues identified by the NRC staff and the licensee's response is discussed in the Evaluation below.

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III. EVALUATION

Emergency Power Supply Requirements

The pressurizer water level indicators are important in a post-accident situation. Adequate emergency power supplies add assurance of post-accident functioning of these components. The licensee has the requisite emergency power supplies. The existing TSs already provide appropriate surveillance and actions for the pressurizer level indicators and thus are acceptable.

Direct Indication of Valve Position

The licensee has provided a direct indication of power operated relief valve (PORV) and safety valve position in the control room. These indications are a diagnostic aid for the plant operator and provide no automatic action. The licensee has provided TSs with a 31-day channel check and an 18-month channel calibration requirement; thus, the TSs are acceptable and they meet our July 2, 1980 model TS criteria.

Instrumentation for Inadequate Core Cooling

The licensee has installed an instrument system to detect the effects of low reactor coolant level and inadequate core cooling. These instruments, subcooling meters, receive and process data from existing plant instrumentation. We previously reviewed this system in our Safety Evaluation dated May 6, 1980. The licensee submitted TSs with a 31-day channel check and an 18-month channel calibration requirement and actions to be taken in the event of component inoperability. We conclude the TSs are acceptable as they meet our July 2, 1980 model TS criteria.

Diverse Containment Isolation

The licensee currently has containment isolation system design so that diverse parameters will be sensed to ensure automatic isolation of non-essential systems under postulated accident conditions. These parameters are low reactor coolant system pressure and high containment pressure. We have reviewed this system in our Lessons Learned Category "A" Safety Evaluation dated May 6, 1980. The design is such that it does not result in the automatic loss of containment isolation after the containment isolation signal is reset. Reopening of containment isolation would require deliberate operator action. The licensee's current TSs list each affected containment isolation valve and provide for the appropriate surveillance and actions in the event of component inoperability; therefore, we conclude that the TSs are acceptable.

Auxiliary (Emergency) Feedwater Flow Indication

The licensee has installed auxiliary feedwater flow indication that meets our testability and vital power requirements. We reviewed this system in our Safety Evaluation dated May 6, 1980. The licensee has proposed a TS with 31-day channel check and 18-month channel calibration requirements. We find this TS acceptable as it meets the criteria of our July 2, 1980 model TS criteria.

Shift Technical Advisor (STA)

Our request indicated that the TSs related to minimum shift manning should be revised to reflect the augmentation of an STA. The licensee's application would add one STA to each shift to perform the function of accident assessment. The individual performing this function will have at least a bachelor's degree or equivalent in a scientific or engineering discipline with special training in plant design, and response and analysis of the plant for transients and accidents. Based on our review, we find the licensee's submittal to satisfy our requirements and is acceptable.

IV. 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 the amendment.

V. CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because that 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.

Date: March 24, 1981

7590-01

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. 37 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.

This amendment adds Technical Specifications which incorporate certain of the Three Mile Island Unit No. 2 Lessons Learned Category "A" requirements.

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 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 the issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated September 16, 1980, (2) Amendment No. 37 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, NW, Washington, DC, 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, DC 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 24th day of March 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing