June 1, 1993

Docket Nos. 50-334
and 50-412

Mr. J. D. Sieber, Senior Vice President
and Chief Nuclear Officer
Nuclear Power Division
Duquesne Light Company
Post Office Box 4
Shippingport, Pennsylvania 15077-0004

Dear Mr. Sieber:

SUBJECT: ISSUANCE OF AMENDMENT NOS.172AND51 TO FACILITY OPERATING LICENSES DPR-66 AND NPF-73, IN RESPONSE TO CHANGE REQUEST NOS. 208/74, REDUCED THERMAL DESIGN FLOW (TAC NOS. M85819 AND M85820)

The Commission has issued the enclosed Amendment Nos.172 and 51 to Facility Operating License Nos. DPR-66 and NPF-73 for the Beaver Valley Power Station, Unit Nos. 1 and 2. The amendments consist of changes to the Technical Specifications in response to your application dated February 19, 1993, as supplemented March 31 and April 19, 1993.

The amendments revise the Appendix A Technical Specifications (TSs) relating to reactor thermal design flow (TDF). The amendments reduce the minimum required TDF by about 1.5 percent.

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

Sincerely,

/S/
Gordon E. Edison, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 172 to DPR-66
2. Amendment No. 51 to NPF-73
3. Safety Evaluation

cc w/enclosures:
See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 1, 1993

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SUBJECT: ISSUANCE OF AMENDMENT NOS. 172 AND 51 TO FACILITY OPERATING LICENSES DPR-66 AND NPF-73, IN RESPONSE TO CHANGE REQUEST NOS. 208/74, REDUCED THERMAL DESIGN FLOW (TAC NOS. M85819 AND M85820)

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The amendments revise the Appendix A Technical Specifications (TSs) relating to reactor thermal design flow (TDF). The amendments reduce the minimum required TDF by about 1.5 percent.

A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

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Gordon E. Edison, Senior Project Manager
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Office of Nuclear Reactor Regulation

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1. Amendment No. 172 to DPR-66
2. Amendment No. 51 to NPF-73
3. Safety Evaluation

cc w/enclosures:
See next page

Mr. J. D. Sieber
Duquesne Light Company

Beaver Valley Power Station
Units 1 & 2

cc:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DUQUESNE LIGHT COMPANY

OHIO EDISON COMPANY

PENNSYLVANIA POWER COMPANY

DOCKET NO. 50-334

BEAVER VALLEY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 172
License No. DPR-66

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Duquesne Light Company, et al. (the licensee) dated February 19, 1993 as supplemented March 31, and April 19, 1993, 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. DPR-66 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 172 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance, to be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Director
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 1, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 172

FACILITY OPERATING LICENSE NO. DPR-66

DOCKET NO. 50-334

Replace the following pages of Appendix A Technical Specifications, with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
XXV	XXV
2-1	2-1
2-2	2-2
2-3	---
2-4	---
2-6	2-6
B 2-4	B 2-4
3/4 2-12	3/4 2-12
3/4 2-13	3/4 2-13

Figure Index

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1 for 3 loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

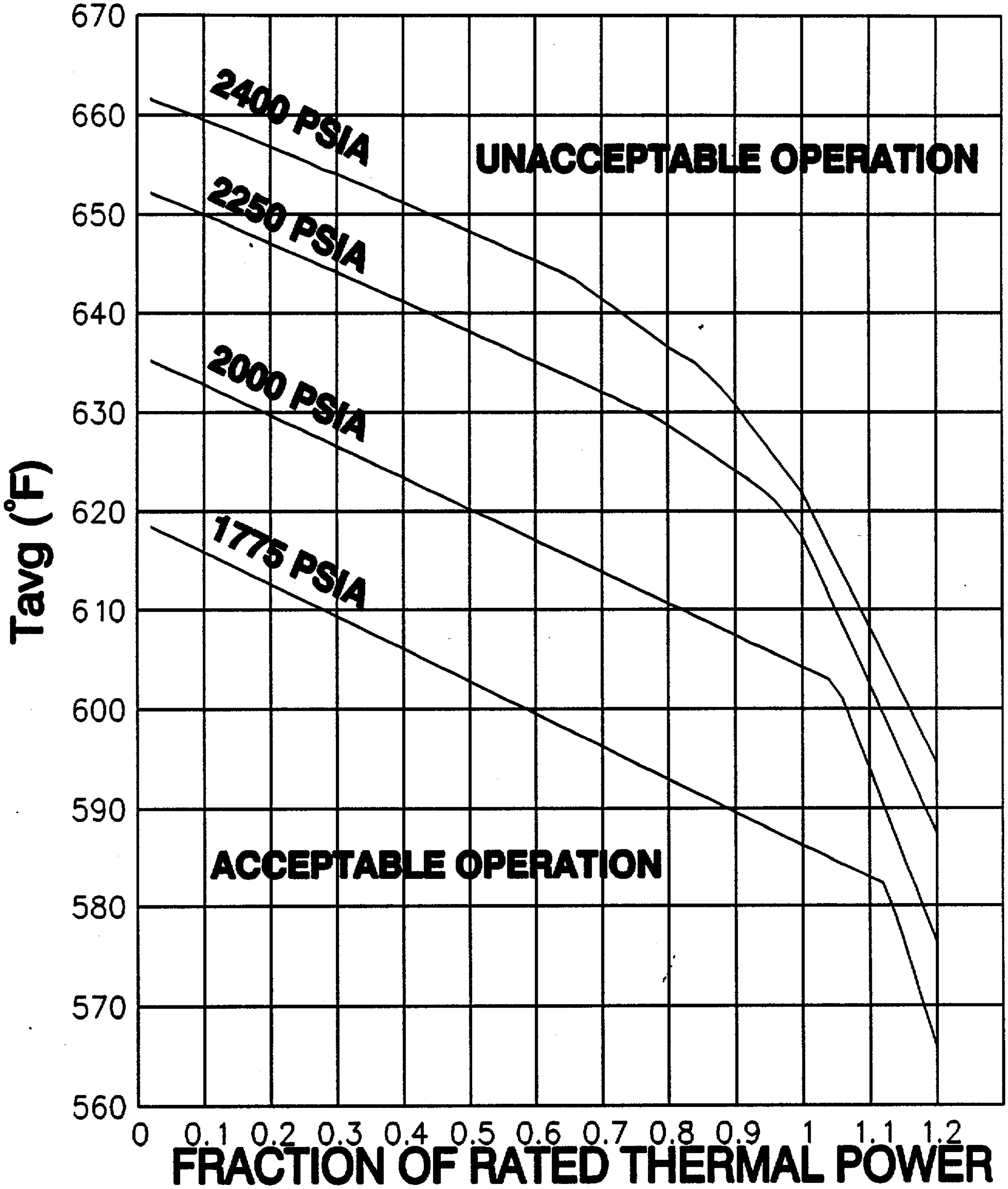

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT
THREE LOOP OPERATION

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - ≤ 25% of RATED THERMAL POWER	Low Setpoint - ≤ 27.3% of RATED THERMAL POWER
	High Setpoint - ≤ 109% of RATED THERMAL POWER	High Setpoint - ≤ 111.3% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 seconds	≤ 6.3% of RATED THERMAL POWER with a time constant ≥ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 seconds	≤ 6.3% of RATED THERMAL POWER with a time constant ≥ 2 seconds
5. Intermediate Range, Neutron Flux	≤ 25% of RATED THERMAL POWER	≤ 31.1% of RATED THERMAL POWER
6. Source Range, Neutron Flux	≤ 10^5 counts per second	≤ 1.4 x 10^5 counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 4
9. Pressurizer Pressure--Low	≥ 1945 psig	≥ 1934 psig
10. Pressurizer Pressure--High	≤ 2385 psig	≤ 2394 psig
11. Pressurizer Water Level--High	≤ 92% of instrument span	≤ 93.9% of instrument span
12. Loss of Flow	≥ 90% of design flow* per loop	≥ 89.0% of design flow* per loop

*Design flow is 87,200 gpm per loop.

The Power Range Negative Rate trip provides protection to ensure that the minimum DNBR is maintained above the design DNBR limit for control rod drop accidents. At high power a single or multiple rod drop accident could cause flux peaking which, when in conjunction with nuclear power being maintained equivalent to turbine power by action of the automatic rod control system, could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor. For those transients on which reactor trip on power range negative rate trip is not postulated, it is shown that the minimum DNBR is greater than the design DNBR limit.

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor start-up. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range Channels will initiate a reactor trip at about 10^{+5} counts per second unless manually blocked when P-6 becomes active. The Intermediate Range Channels will initiate a reactor trip at a current level proportional to approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This setpoint includes corrections for changes in density and heat capacity of water with temperature and dynamic compensation for piping delays from the core to the loop temperature detectors. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown on Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

a. Reactor Coolant System T_{avg}

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

APPLICABILITY: MODE 1[2].

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

---

(1) The values presented in Table 3.2-1 correspond to analytical limits used in the safety analyses.

(2) The provisions of Specification 4.0.4 are not applicable for Reactor Coolant System total flow rate to allow a calorimetric flow measurement and the calibration of the Reactor Coolant System total flow rate indicators.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	3 Loops In Operation
Reactor Coolant System T_{avg}	$\leq$ 580.7°F
Pressurizer Pressure	$\geq$ 2220 psia(1)
Reactor Coolant System Total Flow Rate	$\geq$ 261,600 gpm

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

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DUQUESNE LIGHT COMPANY

OHIO EDISON COMPANY

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

THE TOLEDO EDISON COMPANY

DOCKET NO. 50-412

BEAVER VALLEY POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 51
License No. NPF-73

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Duquesne Light Company, et al. (the licensee) dated February 19, 1993, as supplemented March 31, and April 19, 1993, 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-73 is hereby amended to read as follows:

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

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 51 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated in the license. DLCO shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance, to be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Director
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 1, 1993

ATTACHMENT TO LICENSE AMENDMENT NO.51

FACILITY OPERATING LICENSE NO. NPF-73

DOCKET NO. 50-412

Replace the following pages of Appendix A, Technical Specifications, with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
2-2	2-2
2-4	2-4
3/4 2-11	3/4 2-11
3/4 2-12	3/4 2-12

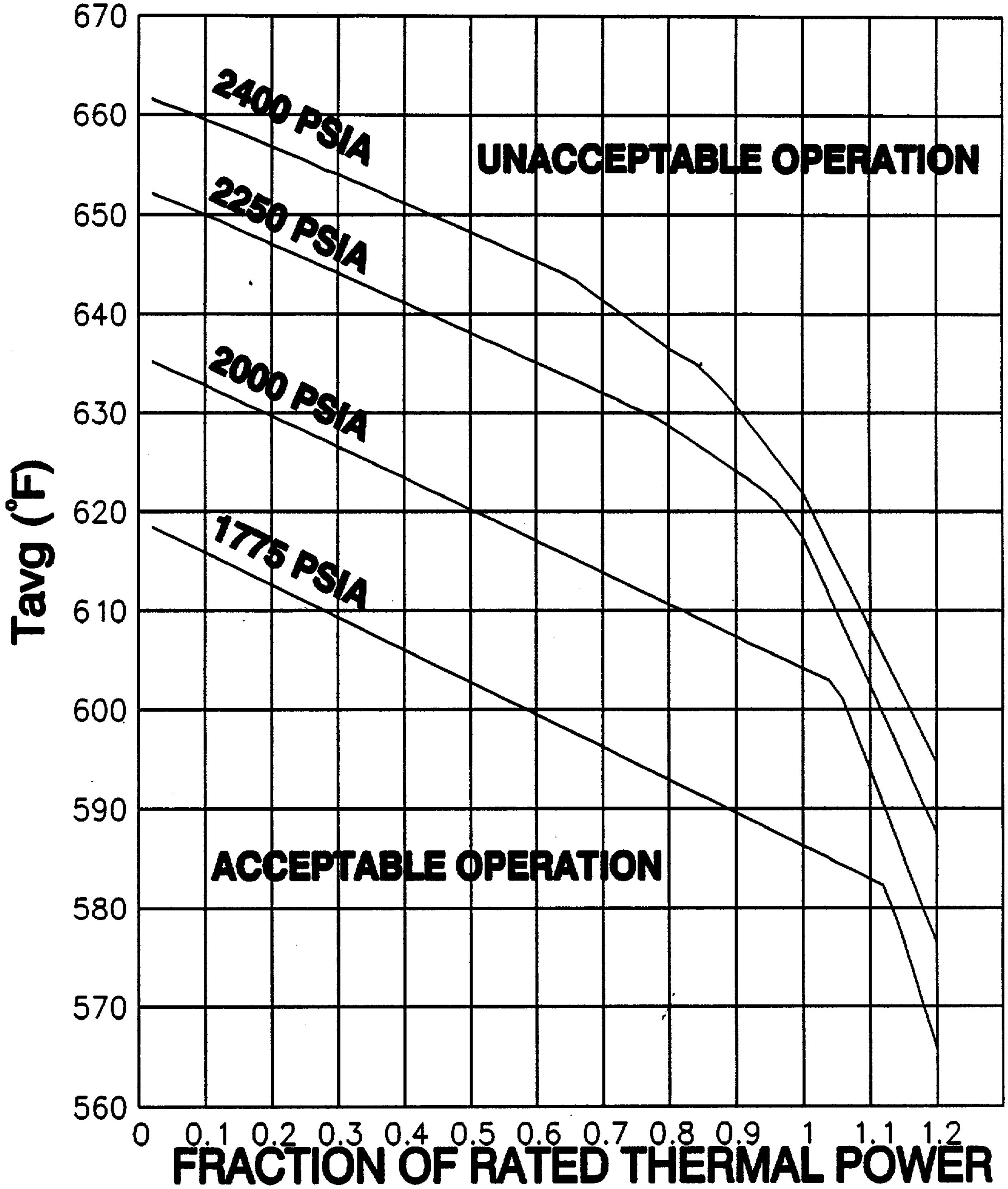

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT THREE LOOP OPERATION

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWANCE (TA)	Z	S	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N/A	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux,					
a. High Setpoint	7.5	4.56	0	≤ 109% of RTP*	≤ 111.1% of RTP*
b. Low Setpoint	8.3	4.56	0	≤ 25% RTP*	≤ 27.1% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	1.6	0.50	0	≤ 5% of RTP* with a time constant ≥ 2 seconds	≤ 6.3% of RTP* with a time constant ≥ 2 seconds
4. Power Range, Neutron Flux High Negative Rate	1.6	0.50	0	≤ 5% of RTP* with a time constant ≥ 2 seconds	≤ 6.3% of RTP* with a time constant ≥ 2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.41	0	≤ 25% RTP*	≤ 30.9% of RTP*
6. Source Range, Neutron Flux	17.0	10.01	0	≤ 10^5 cps	≤ 1.4 x 10^5 cps
7. Overtemperature ΔT	7.0	5.10	See Note 5	See Note 1	See Note 2
8. Overpower ΔT	4.9	1.71	1.49	See Note 3	See Note 4
9. Pressurizer Pressure-Low	3.1	0.71	1.67	≥ 1945 psig***	≥ 1935 psig***
10. Pressurizer Pressure-High	6.2	4.96	0.67	≤ 2375 psig	≤ 2383 psig
11. Pressurizer Water Level-High	8.0	2.18	1.67	≤ 92% of instrument span	≤ 93.8% of instrument span
12. Loss of Flow	2.5	1.39	0.60	≥ 90% of loop design flow**	≥ 88.9% of loop design flow**

* = RATED THERMAL POWER
** Loop design flow = 87,200 gpm
*** Time constants utilized in the lead-lag controller for Pressurizer Pressure-Low are 2 seconds for lead and 1 second for lag. Channel calibration shall ensure that these time constants are adjusted to those values.

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

a. Reactor Coolant System T_{avg}

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

APPLICABILITY: MODE 1(2).

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

---

(1) The values presented in Table 3.2-1 correspond to analytical limits used in the safety analyses.

(2) The provisions of Specification 4.0.4 are not applicable for Reactor Coolant System total flow rate to allow a calorimetric flow measurement and the calibration of the Reactor Coolant System total flow rate indicators.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	3 Loops In Operation
Reactor Coolant System T_{avg}	$\leq$ 580.2°F
Pressurizer Pressure	$\geq$ 2220 psia[(1)]
Reactor Coolant System Total Flow Rate	$\geq$ 261,600 gpm

---

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

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 172 TO FACILITY OPERATING LICENSE NO. DPR-66

AMENDMENT NO. 51 TO FACILITY OPERATING LICENSE NO. NPF-73

DUQUESNE LIGHT COMPANY
OHIO EDISON COMPANY
PENNSYLVANIA POWER COMPANY
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY
THE TOLEDO EDISON COMPANY

BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-334 AND 50-412

1. INTRODUCTION

By letter dated February 19, 1993, Duquesne Light Company (the licensee) submitted proposed revisions to the Technical Specifications (TS) for Beaver Valley Units 1 and 2 to support a 1.5 percent reduction in minimum Reactor Coolant System (RCS) total flow rate. Additional proposed revisions of an administrative nature were also submitted. A conference call with the licensee was held on March 17, 1993, and a request for additional information dated March 19, 1993 was subsequently transmitted. The licensee responded to this request by letter dated March 31, 1993. A subsequent conference call was held on April 12, 1993. The licensee responded to open items discussed in the call by letter dated April 19, 1993. The March 31, and April 19, 1993, submittals provided additional information that did not change the initial no significant hazards consideration determination.

The licensee's request to reduce minimum RCS total flow rate is in anticipation of the need to plug or sleeve defective steam generator tubes, the extent of which will be determined for Unit 1 during the current refueling/steam generator inspection outage. The proposed 1.5 percent reduction is expected to accommodate future levels of steam generator tube plugging/sleeving predicted by the licensee for both units.

To reflect the proposed reduction in flow for Units 1 and 2, the design flow per loop specified in TS Table 2.2-1 is changed from 88,500 gpm to 87,200 gpm. Additionally, the allowable value associated with Table item 12 ("loss of flow") is changed from 88.9% to 89.0% for Unit 1, and from 88.8% to 88.9% for Unit 2. In Table 3.2-1 for Unit 1, total flow rate is changed from 265,500 gpm to 261,600 gpm, while for Unit 2, this change is from 270,850 gpm to 261,600 gpm. The difference reflects the fact that the current Unit 2 value includes a 2.0% flow uncertainty whereas the Unit 1 value does not. The footnote in Table 3.2-1 (Unit 2) that refers to the inclusion of flow uncertainty will be deleted to maintain consistency with Unit 1. Flow uncertainty for Unit 2 will be accounted for administratively, as it currently is for Unit 1, through the operating procedures. Further, the proposed addition of footnote (1) to TS 3.2.5 for Units 1 and 2 clarifies that the values in Table 3.2-1 are intended to be analysis limits and not indicated values.

To support the proposed reduction in flow, the exit boiling portion of the core thermal limits illustrated in Figure 2.1-1 required revision for Units 1 and 2. Additionally, Figures 2.1-2 and 2.1-3 (and all references to them) have been deleted for Unit 1. These figures, which provide core thermal limits for two-loop operation, are extraneous because plant operation with less than three loops is not permitted under the current license. Similarly, the limits pertaining to two-loop operation in Table 3.2-1 for Unit 1 have been deleted. The Unit 2 TS do not address two-loop operation.

A proposed revision to TS 3.2.5 for Unit 1 has been made to allow entry into Mode 1 in the event the surveillance interval for RCS total flow rate extends beyond the required 18-month interval specified in TS 4.2.5.2. The flow rate must be determined by measurement at a reactor power of at least 90 percent. With the plant shut down, current limiting condition for operation (LCO) 3.2.5 would prevent (through TS 4.0.4) commencement of power operation and thus the execution of the surveillance. Therefore, an exclusion to TS 4.0.4 for this case has been added in the form of a footnote to TS 3.2.5. A similar change is made for Unit 2.

Finally, the limit on Tavg specified in Table 3.2-1 has been reduced from ≤581 °F to ≤580.7 °F for Unit 1, and from ≤580.3 °F to ≤580.2 °F for Unit 2.

2.0 EVALUATION

Minimum RCS total flow rate is a critical input parameter to the analyses presented in Chapters 14 and 15 of the licensee's Updated Final Safety Analysis Reports (UFSAR). Accordingly, to support operation of Units 1 and 2 under the proposed reduction in flow, the impact of this reduction on these analyses must be evaluated. The licensee's above-referenced submittals provide a summary of these evaluations and assessments of the following: 1) whether all acceptance criteria continue to be met, 2) whether the current core thermal limits remain bounding, 3) whether current setpoints set forth in the TS continue to provide adequate protection, and 4) whether the performance of key components and systems remains acceptable.

For the non-loss of coolant accident (LOCA) transients, the current licensing basis supports a maximum tube plugging level of 20 percent. The re-evaluations consider only the effects of a 1.5 percent reduction in RCS total flow rate and continue to support a 20 percent plugging level. For those transients with departure from nucleate boiling (DNB) acceptance criteria, the licensee has determined through sensitivity studies that the proposed reduction in flow will reduce the departure from nucleate boiling ratio (DNBR) by 2.4 percent or less. The current retained DNBR margin for Units 1 and 2 is 9.9 percent (based on a safety analysis DNBR limit of 1.33 and a design limit DNBR of 1.21). To account for the adverse impact of flow reduction on DNBR, the licensee has taken a 2.4 percent penalty against this retained margin for all relevant non-LOCA transients. Any resulting analysis margins for the specific events (i.e., based on the difference between the minimum DNBR for an event and the safety analysis limit) were not credited.

With regard to the impact of the proposed flow reduction on core thermal limits, the licensee's evaluation indicates that the DNBR-limited segment of each thermal limit line in the current TS remains bounding. As noted above, the retained DNBR margin is large enough to accommodate the penalty assessed against it due to flow reduction. The vessel exit boiling segment of each thermal limit line was revised, however, to reflect the 1.5 percent reduction in flow. The effect is less than a 1°F decrease in exit boiling limits. Additionally, the licensee confirmed that the current OT delta-T and OP delta-T setpoint equations continue to provide protection for the revised core thermal limits.

For those non-LOCA transients which are not DNB related or for which other acceptance criteria in addition to DNB are relevant, the licensee has examined the effects of the proposed reduction in flow. To determine if the design basis continues to be met, existing sensitivity data as well as sensitivity analyses performed to support the proposed reduction in flow were employed. The licensee has presented, for each of the relevant transients, results which indicate that adequate margin to the applicable acceptance criteria currently exists to accommodate the effect of reduced flow.

The large and small break LOCA events (LBLOCA and SBLOCA) and steam generator tube rupture (SGTR) event were re-evaluated to determine if the relevant acceptance criteria continue to be met under the proposed flow reduction. The SGTR re-evaluation (at a tube plugging level of 20 percent for each unit) indicated a slight increase in break flow and calculated radiation dose. However, based on conservative assumptions made in the FSAR analyses of record regarding coolant activity (which is unaffected by the proposed flow reduction) the licensee has stated that the FSAR results remain bounding.

For Unit 1, the analysis of record for the LBLOCA limiting break case indicates a calculated peak cladding temperature (PCT) of 2149 °F. This is based on a BASH re-analysis at a 20 percent tube plugging level. A total penalty of 4 °F has been assessed against this value to account for fuel rod backfill initial pressure uncertainty and, as specified in the present amendment request, a revised RCS Tavg uncertainty. This results in a cumulative PCT of 2153 °F.

The analysis of record for the limiting break SBLOCA indicates a PCT of 1802 °F and is based on a NOTRUMP analysis at a 10 percent plugging level. Excluding penalties associated with the present amendment request, a net penalty of 380 °F (consisting of various permanent and interim penalties and benefits) has been assessed against this value to date. The penalties associated with the proposed amendment total 15 °F and are due to a revised RCS Tavg uncertainty (5 °F) and a corresponding Burst/Blockage SPIKE interim penalty (10 °F). The result is a cumulative PCT of 2197 °F. For Unit 1, the licensee reports that no change in RCS Tavg is predicted as a direct result of the proposed flow reduction and, therefore, no corresponding PCT penalty or benefit is incurred for either the LBLOCA or SBLOCA case.

The Unit 2 analysis of record for the LBLOCA limiting break case is based on a BART analysis at a 5 percent plugging level and indicates a PCT of 2120 °F. Including penalties assessed against this value prior to the present amendment request, the most recent cumulative PCT is 2191 °F. The licensee's April 19, 1993 submittal identifies a 25 °F PCT benefit related to the WREFLOOD structural heat model, which reduces the cumulative PCT to 2166 °F. The licensee reports that, for Unit 2, the proposed flow reduction results in a small decrease in RCS Tavg for both the LBLOCA and SBLOCA cases. Because existing BART data indicates that higher values of RCS Tavg are limiting for the LBLOCA, a PCT benefit would thus result. The licensee, however, has not quantified or credited this benefit.

For the Unit 2 SBLOCA, the analysis of record for the limiting break case is based on a NOTRUMP analysis at a 5 percent plugging level and indicates a PCT of 1399 °F. Including penalties assessed against this value prior to the present amendment request, the most recent cumulative PCT is 2119 °F. Existing NOTRUMP sensitivity data indicates that either direction (i.e., an increase or decrease) can be limiting for RCS Tavg. For the proposed flow reduction and resulting small decrease in RCS Tavg, these data show that a 1 °F change in PCT could result. The licensee has conservatively taken this 1 °F change as a penalty. Additionally, the licensee has discovered discrepancies between the RCS Tavg input values to the LBLOCA and SBLOCA analyses, and the current values. These discrepancies, due to error and to evolutionary changes in the plant, are such that the analyses values exceed the current values. This would have a beneficial effect on the LBLOCA PCT since a higher RCS Tavg is limiting. However, this benefit was not quantified or credited. For the SBLOCA, on the other hand, a PCT penalty of 20 °F was assessed for conservatism. A corresponding 36 °F Burst/Blockage SPIKE interim penalty was then taken, bringing the cumulative PCT to 2176 °F.

The above results for LBLOCA and SBLOCA indicate that Units 1 and 2 may be small break limited. Further, the SBLOCA results for both units and the LBLOCA results for Unit 2 reflect significant changes from the analyses of record, in the context of 10 CFR 50.46. In all cases, however, the above results indicate that the acceptance criterion of a cumulative PCT of less than 2200 °F has been met. The licensee has also examined the effect of the proposed flow reduction on the LOCA mass and energy release calculations and has concluded that the current FSAR analyses remain bounding.

The effect of the proposed flow reduction on key NSSS components, the steam generator, key heat exchangers, valves and pumps, and fluid systems has been evaluated by the licensee to confirm that their operation continues to remain in compliance with the applicable acceptance criteria, codes, and standards. In all cases for Units 1 and 2, the results of these evaluations indicated continued acceptable performance under reduced flow conditions.

On the basis of the above evaluation, we find that with regard to the proposed TS revisions, the licensee has provided adequate supporting evaluations to demonstrate that:

1) For all non-LOCA and LOCA events addressed in Chapters 14/15 of the UFSARs for Units 1 and 2, the relevant acceptance criteria continue to be met for operation at the proposed reduced RCS total flow rate.

2) The core thermal limits have been revised to bound operation of Units 1 and 2 at the reduced flow rate.

3) The current OT delta-T and OP delta-T setpoints provide adequate plant protection at the reduced flow rate.

4) Performance of key systems and components continues to remain acceptable at the reduced flow conditions.

Therefore, we find the proposed TS revisions to be acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (58 FR 16224). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

The Commission 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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