July 12, 1989

Docket Nos. 50-334
and 50-412

Mr. J. D. Sieber, Vice President
Nuclear Group
Duquesne Light Company
Post Office Box 4
Shippingport, Pennsylvania 15077

Dear Mr. Sieber:

SUBJECT: BEAVER VALLEY UNITS 1 AND 2 - ISSUANCE OF AMENDMENT (TAC NOS. 72989 AND 72990)

The Commission has issued the enclosed Amendment No. 142 to Facility Operating License No. DPR-66 for the Beaver Valley Power Station, Unit 1, and Amendment No. 18 for Facility Operating License No. NPF-73 for Unit 2, in response to your application dated April 21, 1989.

The amendments revise the Technical Specifications of both units to delete Table 4.5-5, "Reactor Vessel Material Irradiation Surveillance Schedule," and associated surveillance requirement 4.4.9.1.c. This table will be included in the Updated Final Safety Analysis Report of each unit. Meanwhile, there is no change in the reactor vessel material surveillance program, which will continue to be governed by 10 CFR 50, Appendix H.

A copy of the associated Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

Signed by

Peter S. Tam, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 142 to DPR-66
2. Amendment No. 18 to NPF-73
3. Safety Evaluation

cc w/enclosures:
See next page

[AMEND TAC 72989]

LA:PDI-4	PM:PDI-4 PST	PD:PDI-4	OGC	BC:EMTB
SNorris	PTam:lm	JStolz	MYoung	CCheng
06/21/89	06/21/89	~~06/ /89~~ 07 12	07/10/89	6/21/89

8907200323 890712
PDR ADOCK 05000334
P PDC

DF01
111

cc:

Jay E. Silberg, Esquire
Shaw, Pittman, Potts and Trowbridge
2300 N Street, N.W.
Washington, DC 20037

Kenny Grada, Manager
Nuclear Safety
Duquesne Light Company
P. O. Box 4
Shippingport, Pennsylvania 15077

John A. Lee, Esquire
Duquesne Light Company
One Oxford Centre
301 Grant Street
Pittsburgh, Pennsylvania 15279

W.F. Carmichael, Commissioner
Department of Labor
1800 Washington Street East
Charleston, West Virginia 25305

John D. Borrows
Director, Utilities Department
Public Utilities Commission
180 East Broad Street
Columbus, Ohio 43266-0573

Director, Pennsylvania Emergency
Management Agency
Post Office Box 3321
Harrisburg, Pennsylvania 17105-3321

Bureau of Radiation Protection
Pennsylvania Department of
Environmental Resources
ATTN: R. Janati
Post Office Box 2063
Harrisburg, Pennsylvania 17120

Mayor of the Borrough of
Shippingport
Post Office Box 3
Shippingport, Pennsylvania 15077

Ashley C. Schannauer
Assistant City Solicitor
City of Pittsburgh
313 City-County Building
Pittsburgh, Pennsylvania 15219

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406

Resident Inspector
U.S. Nuclear Regulatory Commission
Post Office Box 181
Shippingport, Pennsylvania 15077

DATED: July 12, 1983
AMENDMENT NOS. 143 AND 18 TO FACILITY OPERATING LICENSE NOS. DPR-66 AND NPF-73

Docket File
NRC PDR
Local PDR
Plant File
S. Varga (14E4)
B. Boger (14A2)
J. Stolz
S. Norris
P. Tam
OGC
D. Hagan (MNBB 3302)
E. Jordan (MNBB 3302)
B. Grimes (9A2)
T. Meek(8) (P1-137)
W. Jones (P-130A)
J. Calvo (11F23)
ACRS (10)
GPA/PA
ARM/LFMB

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

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. 142
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 April 21, 1989 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. 142, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective on issuance, to be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: July 12, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 142

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
3/4 4-23	3/4 4-23
3/4 4-26	--
B 3/4 4-10	B 3/4 4-10
3/4 4-25	3/4 4-25

SURVEILLANCE REQUIREMENTS

4.4.9.1

a. The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

b. The Reactor Coolant System temperature and pressure conditions shall be determined to be to the right of the criticality limit line within 15 minutes prior to achieving reactor criticality.

vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50 for reactor criticality and for inservice leak and hydrostatic testing.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in UFSAR Table 4.5-3 to assure compliance with the requirements of Appendix H to 10 CFR 50.

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORV's or an RCS vent opening of greater than 3.14 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are ≤ 275°F. Either PORV has adequate relieving capability to protect the RCS from over-pressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator ≤ 25°F above the RCS cold leg temperature or (2) the start of a charging pump and its injection into a water solid RCS.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a(g)(6)(i).

3/4.4.11 RELIEF VALVES

The relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to ensure the ability to seal this possible RCS leakage path.

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: WELD METAL
COPPER CONTENT: 0.31 WT%
PHOSPHORUS CONTENT: 0.015 WT%
RT_{NDT} INITIAL: 0°F
RT_{NDT} AFTER 9.5 EFPY: 1/4T, 274°F
3/4T, 137°F

CURVE APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 9.5 EFPY AND CONTAINS MARGINS OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS

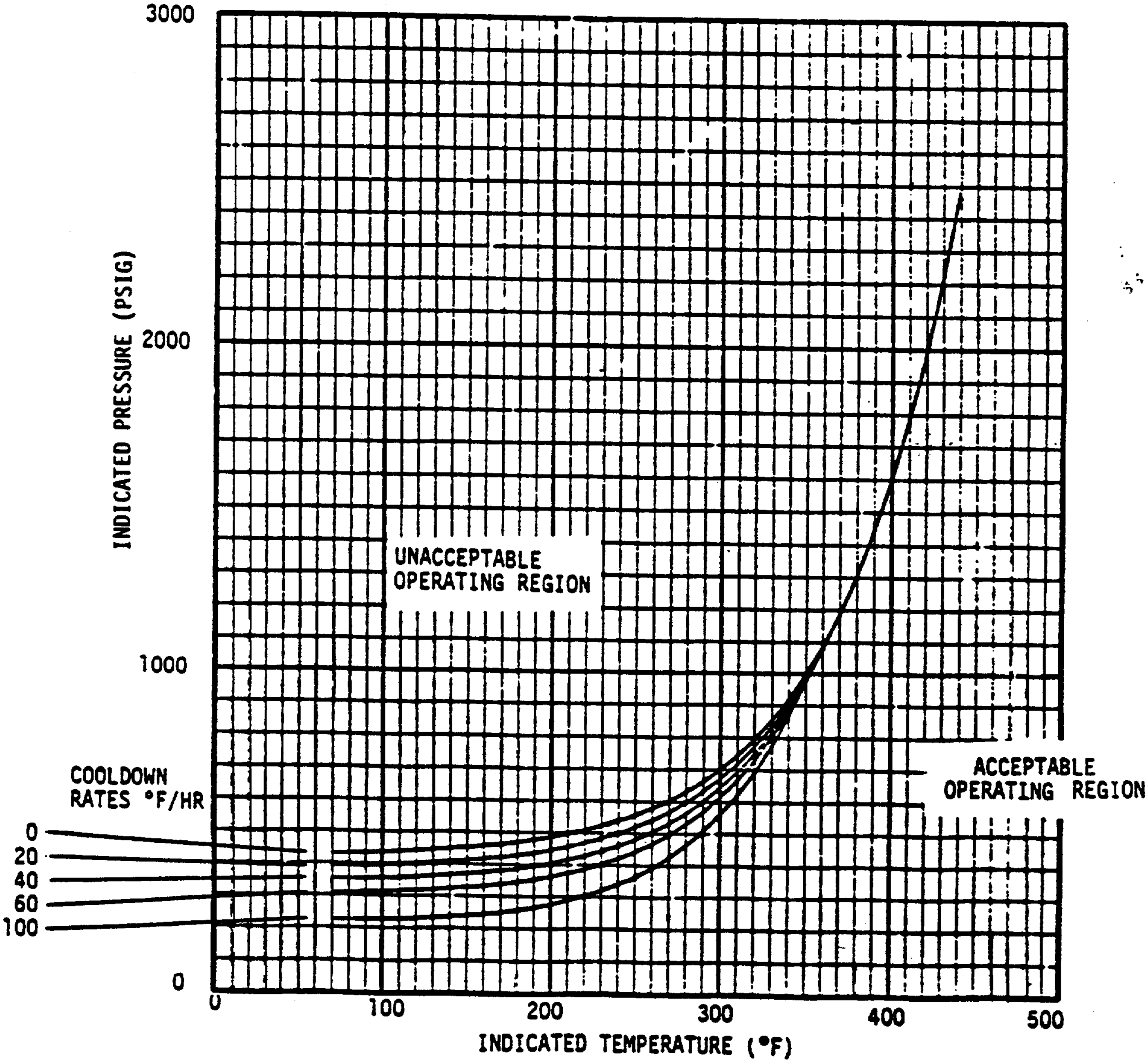

FIGURE 3.4-3 BEAVER VALLEY UNIT NO. 1 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS APPLICABLE FOR THE FIRST 9.5 EFPY

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

DUQUESNE LIGHT COMPANY

OHIO EDISON COMPANY

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

THE TOLEDO EDISON COMPANY

DOCKET NO. 50-412

BEAVER VALLEY POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 18
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 April 21, 1989 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) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 18, 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 on issuance, to be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 12, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 18

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
3/4 4-30	3/4 4-30
3/4 4-33	--
B 3/4 4-14	B 3/4 4-14
B 3/4 4-15	B 3/4 4-15
3/4 4-32	3/4 4-32

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 60°F in any 1-hour period,

b. A maximum cooldown of 100°F in any 1-hour period, and

c. A maximum temperature change of $\leq$ 5°F in any 1-hour period during hydrostatic testing operations above system design pressure.

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

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an analysis to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1

a. The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

b. The Reactor Coolant System temperature and pressure conditions shall be determined to be to the right of the criticality limit line within 15 minutes prior to achieving reactor criticality.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50 for reactor criticality and for inservice leak and hydrostatic testing.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in UFSAR Table 5.3-6 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on the pressurizer heatup and cooldown rates and auxiliary spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs or an RCS vent opening of greater than 3.14 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are ≤ 350°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator ≤ 50°F above the RCS cold leg temperature or (2) the start of a charging pump and its injection into a water solid RCS.

OVERPRESSURE PROTECTION SYSTEMS

The Maximum Allowed PORV Setpoint for the Overpressure Protection Systems (OPPS) is derived by analysis which models the performance of the OPPS assuming various mass input and heat input transients. Operation with a PORV setpoint less than or equal to the maximum setpoint ensures that nominal 10 EFPY Appendix G limits will not be violated with consideration for: (1) a maximum pressure overshoot beyond the PORV setpoint which can occur as a result of time delays in signal processing and valve opening; (2) a 50°F heat transport effect made possible by the geometrical relationship of the RHR suction line and the RCS wide range temperature indicator used for OPPS; (3) instrument uncertainties; and (4) single failure. To ensure mass and heat input transients more severe than those assumed cannot occur, Technical Specifications require lockout of all but one centrifugal charging pump while in MODES 4, 5, and 6 with the reactor vessel head installed and disallow start of an RCP if secondary coolant temperature is more than 50°F above reactor coolant temperature. Exceptions to these requirements are acceptable as described below.

Operation above 350°F but less than 375°F with only one centrifugal charging pump OPERABLE is allowed for up to 4 hours. As shown by analysis LOCAs occurring at low temperature, low pressure conditions can be successfully mitigated by the operation of a single centrifugal charging pump and a single LHSI pump with no credit for accumulator injection. Given the short time duration

OVERPRESSURE PROTECTION SYSTEMS (Continued)

that the condition of having only one centrifugal charging pump OPERABLE is allowed and the probability of a LOCA occurring during this time, the failure of the single centrifugal charging pump is not assumed.

Operation below 350°F but greater than 325°F with all centrifugal charging pumps OPERABLE is allowed for up to 4 hours. During low pressure, low temperature operation all automatic Safety Injection actuation signals are blocked. In normal conditions a single failure of the ESF actuation circuitry will result in the starting of at most one train of Safety Injection (one centrifugal charging pump, and one LHSI pump). For temperatures above 325°F, an overpressure event occurring as a result of starting these two pumps can be successfully mitigated by operation of both PORVs without exceeding Appendix G limit. Given the short time duration that this condition is allowed and the low probability of a single failure causing an overpressure event during this time, the single failure of a PORV is not assumed. Initiation of both trains of Safety Injection during this 4-hour time frame due to operator error or a single failure occurring during testing of a redundant channel are not considered to be credible accidents.

The maximum allowed PORV setpoint for the Overpressure Protection System will be updated based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50, Appendix H and in accordance with the schedule in UFSAR Table 5.3-6.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a (g)(6)(i).

MATERIAL PROPERTY BASIS

Controlling Material	: Plate Metal
Copper Content	: Conservatively Assumed to be 0.10 wt%
Phosphorus Content	: 0.010 wt%
RT_{NDT} Initial	: 60°F
RT_{NDT} After 10 EFPY	: 1/4T, 139°F
	3/4T, 114°F

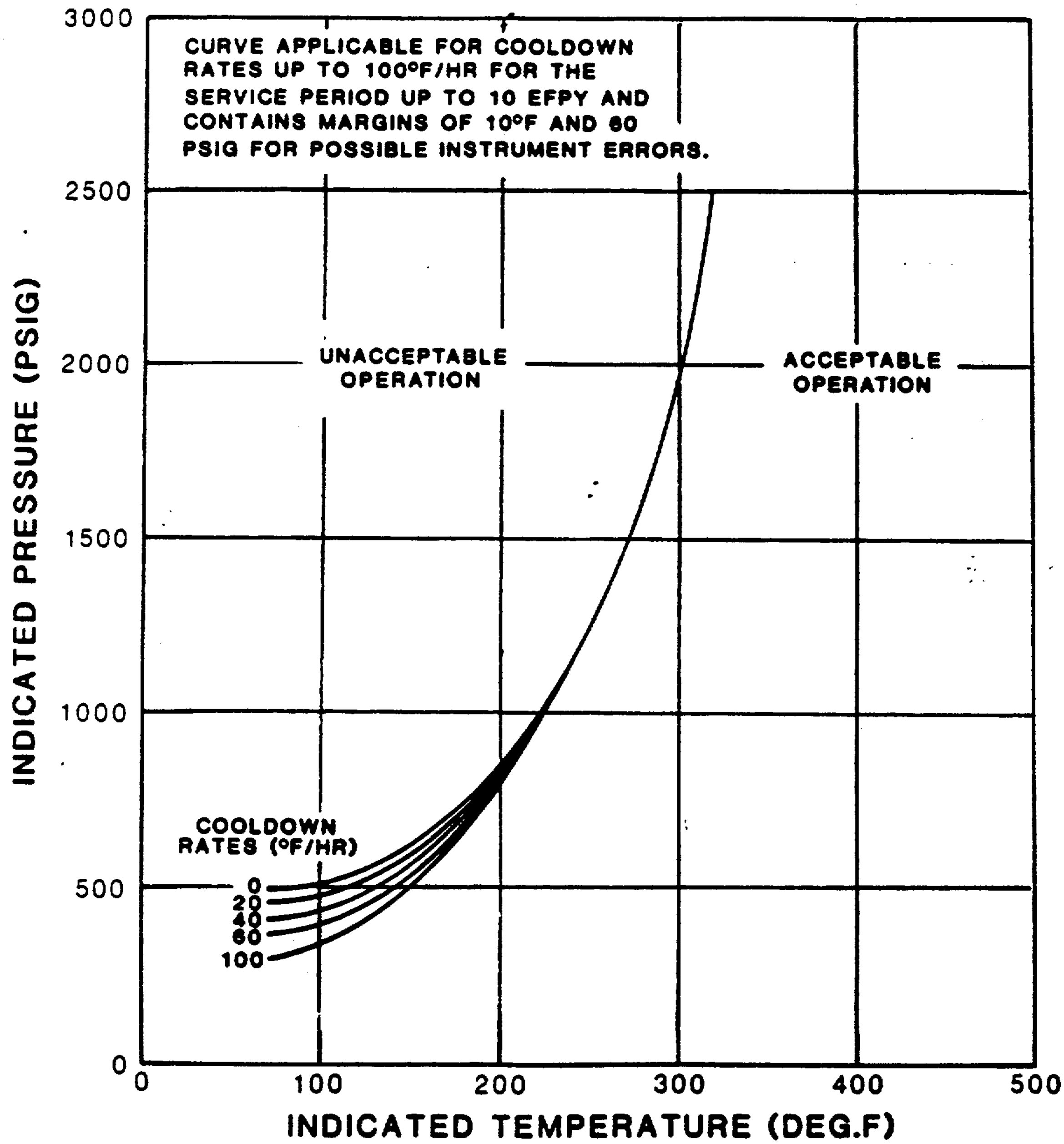

FIGURE 3.4-3 BEAVER VALLEY UNIT NO. 2 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS APPLICABLE FOR THE FIRST 10 EFPY

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 142 TO FACILITY OPERATING LICENSE NO. DPR-66

AMENDMENT NO. 18 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, UNITS NO. 1 AND NO. 2

DOCKET NOS. 50-334 AND 50-412

INTRODUCTION

Surveillance requirement 4.4.9.1.c of the Beaver Valley Technical Specifications requires the removal and examination of the reactor vessel surveillance capsules to update the plant heatup and cooldown curves. Table 4.4-5 "Reactor Vessel Material Irradiation Surveillance Schedule" imposes the capsule removal schedule including vessel location, lead factors and estimated capsule fluence. By letter dated April 21, 1989, Duquesne Light Company (the licensee, acting as agent for the above utilities) requested that both specification 4.4.9.1.c and Table 4.4-5 be deleted. Our review of that request follows.

DISCUSSION AND EVALUATION

10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements" imposes requirements on light water reactor licensees to monitor changes in the fracture toughness properties of reactor vessels. The changes are caused by neutron irradiation and the thermal environment. Under the program, fracture toughness test data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the reactor vessel. A proposed withdrawal schedule, which is the subject of Table 4.4-5 and specification 4.4.9.1.c of the Technical Specifications (TS), must be submitted for staff approval. Therefore, Table 4.4-5 of the units' TS contains the staff-approved specimen withdrawal schedule.

The licensee proposed to relocate Table 4.4-5 from the TS to the Updated Final Safety Analysis Report (UFSAR) of each unit. The licensee provided draft UFSAR pages to show this. Such relocation does not change the subject surveillance requirement or the design of specimen capsules in any way.

10 CFR 50.36(c)(3) specifies that surveillance requirements, such as that for reactor vessel material, be included in the technical specifications. By this amendment, we approve relocation, but not substantive change, of a surveillance requirement. The specimen withdrawal schedule in Table 4.4-5 will continue to exist as required by 10 CFR 50, Appendix H, and will be controlled by inclusion in the UFSAR; any future changes to the specimen withdrawal schedule will require prior staff approval, as specified in Section II.B.3, Appendix H of 10 CFR 50. Hence, the reactor vessel surveillance program can be controlled by means other than inclusion in the plant technical specifications.

We therefore find the deletion of Table 4.4-5 and specification 4.4.9.1.c acceptable.

The associated bases of specification 4.4.9.1.c is also revised to reflect the relocation of Table 4.4-5 to UFSAR Table 4.3-6. This change is also acceptable.

ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. We have determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be release offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. We have previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these 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 these amendments.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and (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.

Dated: July 12, 1989

Principal Contributor: Peter S. Tam