January 15, 1993

Docket No. 50-334

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

Dear Mr. Sieber:

SUBJECT: ISSUANCE OF AMENDMENT NO.169 TO FACILITY OPERATING LICENSE DRP-66, REVISION OF HEATUP/COOLDOWN CURVES (P/T LIMITS) - (TAC NO. M80559)

The Commission has issued the enclosed Amendment No. 169 to Facility Operating License No. DPR-66 for the Beaver Valley Power Station, Unit No. 1, in response to your application dated July 15, 1991, as modified by letters dated September 30, 1991, December 13, 1991, and November 2, 1992.

The amendment revises the Appendix A Technical Specifications (TSs) relating to reactor overpressure protection. Changes are made to TSs involving Reactor Coolant Pump Startup, Reactor Coolant System Pressure - Temperature Limits, and Reactor Coolant System Overpressure Protection System. Specifically, the amendment 1) revises TS 3.4.1.6 to ensure that the limiting conditions specified for pressurizer level and secondary water temperature of each steam generator are met to mitigate the consequences of starting a reactor coolant pump, 2) revises Figures 3.4-2 and 3.4-3 (TS 3.4.9.1) to extend the applicability of the heatup and cooldown curves to 16 effective full power years (EFPY), and 3) revises the enable temperature for the overpressure protection system specified in TS 3.4.9.3 to be consistent with the new 16 EFPY curves. Bases Section 3/4 4.1 also is revised to reflect the changes to the TSs mentioned above.

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.

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.169to DPR-66
2. Safety Evaluation

cc w/enclosures:
See next page

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Contingent on change in 1st para of SER

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

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Sincerely,

G E Edison

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. 169 to DPR-66
2. Safety Evaluation

cc w/enclosures:
See next page

cc:

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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. 169
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 July 15, 1991, as modified by letters dated September 30, 1991, December 13, 1991, and November 2, 1992, 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.

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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. 169, 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: January 15, 1993

ATTACHMENT TO LICENSE AMENDMENT NO.169

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 4-4a	3/4 4-4a
3/4 4-24	3/4 4-24
3/4 4-25	3/4 4-25
3/4 4-27a	3/4 4-27a
B 3/4 4-1a	B 3/4 4-1a

REACTOR COOLANT PUMP STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.6 If both OPPS PORV's are not OPERABLE, an idle reactor coolant pump in a non-isolated loop shall not be started, unless:

1. The actual pressurizer water level is less than 60 percent (840 ft^3), and

2. The secondary water temperature* of each steam generator is less than 25°F above each of the in-service RCS cold leg temperatures.

APPLICABILITY: When the temperature of one or more of the non-isolated loop cold legs is less than or equal to the enable temperature set forth in Specification 3.4.9.3.

ACTION:

With the pressurizer water level greater than or equal to 60 percent or the temperature of the steam generator in the loop associated with the reactor coolant pump being started greater than or equal to 25°F above the cold leg temperature of the other non-isolated loops, suspend the startup of the reactor coolant pump.

SURVEILLANCE REQUIREMENTS

4.4.1.6.1 The pressurizer water volume and the secondary water temperature of the non-isolated steam generators shall be determined within ten minutes prior to starting a reactor coolant pump.

---

* The secondary water temperature is to be verified by direct measurement of the fluid temperature, or contact temperature readings on the steam generator secondary, or blowdown piping after purging of stagnant water within the piping.

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: INTERMEDIATE SHELL PLATE B6607-2

RT_{NDT} AFTER 16 EFPY : 1/4T, 224°F
3/4T, 188°F

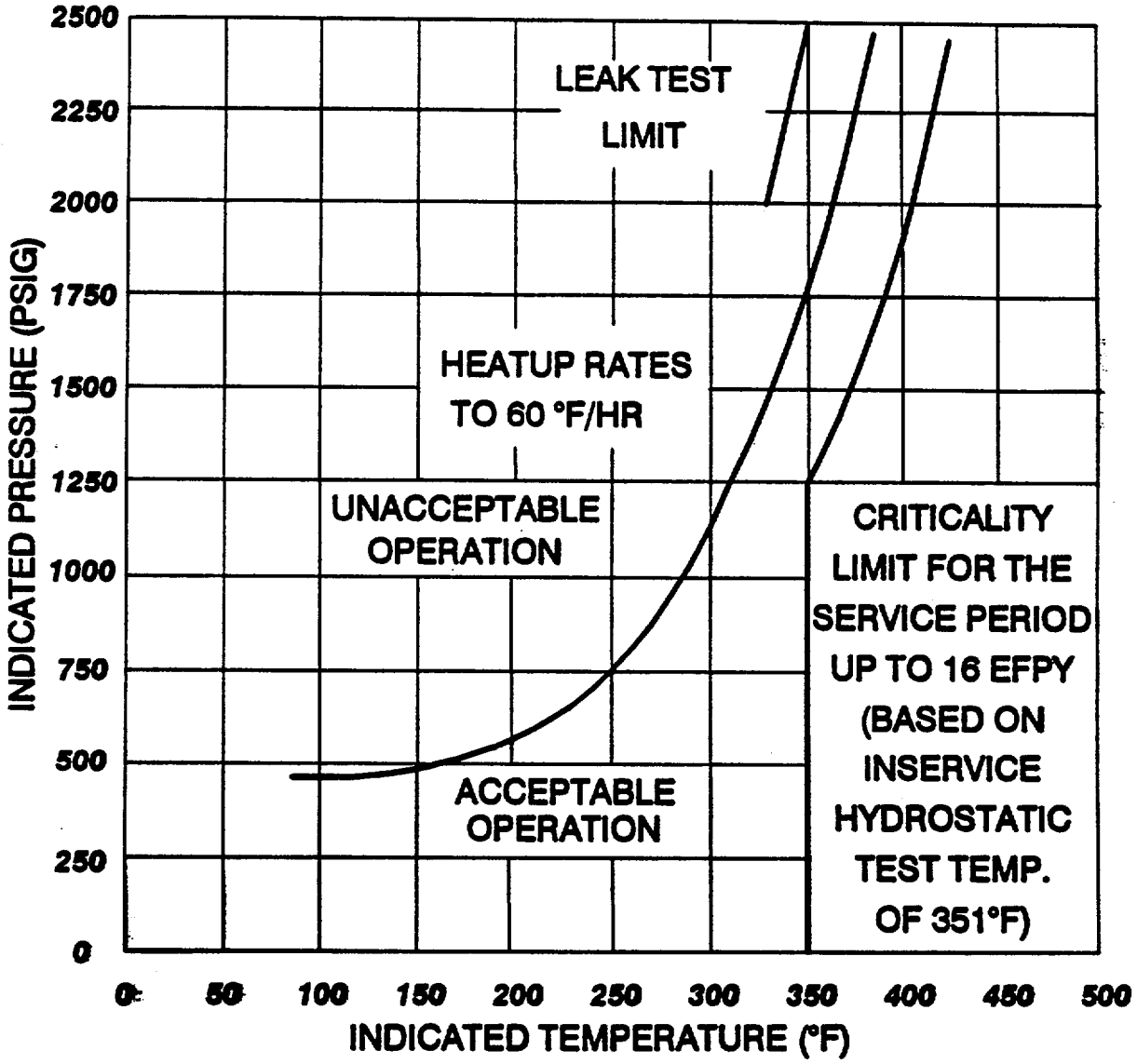

FIGURE 3.4-2

Beaver Valley Unit 1 Reactor Coolant System Heatup Limitations Applicable for the First 16 EFPY

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: INTERMEDIATE SHELL PLATE B6607-2

RT_{NDT} AFTER 16 EFPY : 1/4T, 224°F
3/4T, 188 °F

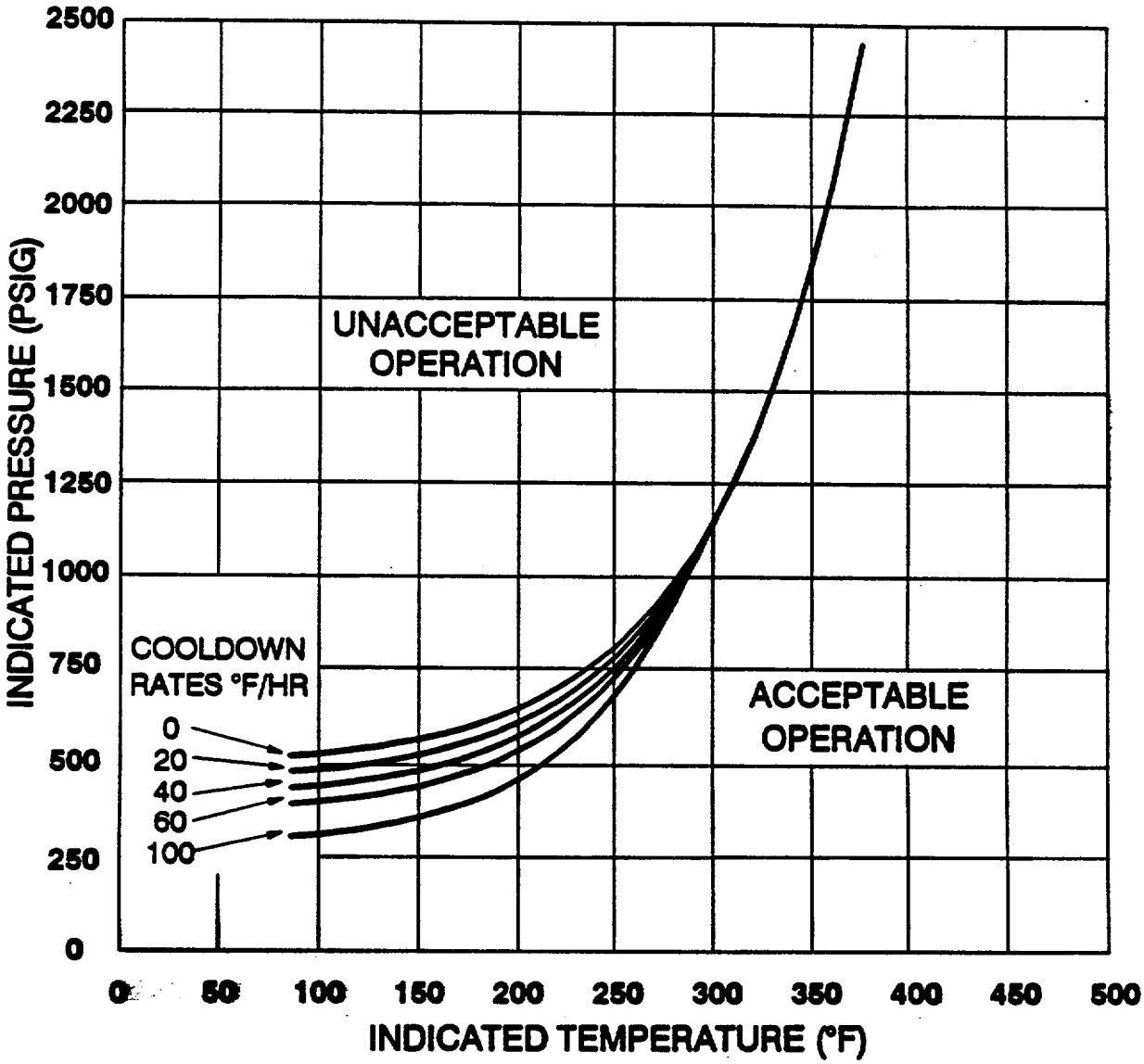

FIGURE 3.4-3
Beaver Valley Unit 1 Reactor Coolant System Cooldown Limitations Applicable for the First 16 EFPY

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following overpressure protection systems shall be OPERABLE:

a. Two power operated relief valves (PORVs) with a nominal trip setpoint of less than or equal to 432 psig, or

b. A reactor coolant system vent of greater than or equal to 3.14 square inches.

APPLICABILITY: When the temperature of one or more of the non-isolated RCS cold legs is less than or equal to an enable temperature of 329°F.

ACTION:

a. With one PORV inoperable, either restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through a 3.14 square inch vent(s) within the next 12 hours; maintain the RCS in a vented condition until both PORVs have been restored to OPERABLE status. Refer to Technical Specification 3.4.1.6 for further limitations.

b. With both PORV's inoperable, depressurize and vent the RCS through a 3.14 square inch vent(s) within 12 hours; maintain the RCS in a vented condition until both PORVs have been restored to OPERABLE status.

c. The provisions of specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENT

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE BY:

3/4.4.1 REACTOR COOLANT LOOPS, (continued)

of Appendix G by (1) restricting the water level in the pressurizer and thereby providing a volume for the primary coolant to expand into and (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 25°F above each of the RCS cold leg temperatures.

Power is removed from the isolated loop stop valves (hot leg and cold leg) to ensure that no reactivity addition to the core can occur while the loop is isolated due to inadvertent opening of the isolated loop stop valves. Isolated loop startup is limited to Modes 5 and 6 in accordance with the NRC SER on N-1 loop operation. Verification of the isolated loop boron concentration prior to opening the isolated loop stop valves provides a reassurance of the adequacy of the shutdown margin in the remainder of the system. Restoration of power to the hot leg stop valve allows opening this valve to complete the recirculation flowpath in conjunction with the relief line bypassing the cold leg stop valve and ensures adequate mixing in the isolated loop. This enables the temperature and boron concentration of the isolated loop to be brought to equilibrium with the remainder of the system. Limiting the temperature differential between the isolated loop and the remainder of the system prior to opening the cold leg stop valve prevents any significant reactivity effects due to cool water addition to the core.

Startup of an idle loop will inject cool water from the loop into the core. The reactivity transient resulting from this cool water injection is minimized by delaying isolated loop startup until its temperature is within 20°F of the operating loops. Making the reactor subcritical prior to loop startup prevents any power spike which could result from this cool water induced reactivity transient.

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 2735 psig. Each safety valve is designed to relieve 345,000 lbs. per hour of saturated steam at the valve set point. The relief capacity of a single safety valve is adequate to

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT 169 TO FACILITY OPERATING LICENSE NO. DPR-66

DUQUESNE LIGHT COMPANY

OHIO EDISON COMPANY

PENNSYLVANIA POWER COMPANY

BEAVER VALLEY POWER STATION, UNIT NO. 1

DOCKET NO. 50-334

1.0 INTRODUCTION

The reactor coolant system (RCS) pressure-temperature (P/T) limits during plant heat-up and cooldown for Beaver Valley Power Station, Unit No. 1 are specified in Technical Specification (TS) Figures 3.4-2 and 3.4-3. The requirements for the low temperature overpressure protection (LTOP) system are specified in TS 3.4.9.3. An associated restriction on the start-up of a reactor coolant pump is specified in TS 3.4.1.6 and is consistent with the assumptions used in the analysis supporting the LTOP setpoint.

By letter dated July 15, 1991 (Ref. 1), September 30, 1991 (Ref. 2), December 13, 1991 (Ref. 3), and November 2, 1992 (Ref. 4), Duquesne Light Company proposed amendments to the P/T curves in TS Figures 3.4-2 (for heatup) and 3.4-3 (for cooldown) and changes in the LTOP system setpoints. The proposed changes replace the current heatup and cooldown curves applicable to 9.5 effective full power years (EFPY) of operation, with curves applicable to 16 EFPY. The new curves were developed based on the guidance provided in Revision 2 of Regulatory Guide 1.99. Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Effect on Plant Operations," recommends RG 1.99, Rev. 2, be used in calculating P/T limits, unless the use of different methods can be justified. The proposed changes to TS 3.4.1.6 add a more stringent limitation by requiring both conditions 1 and 2 of that TS to apply, rather than just one of them. In addition, the action statement in TS 3.4.1.6 proposes to clarify the action to be consistent with the stated LCO conditions 1 and 2. The supplemental letters provided minor revisions to the application but they do not change the initial proposed no significant hazards consideration determination and do not expand the scope of the original Federal Register notice.

2.0 EVALUATION

Materials Considerations

To evaluate the P/T limits, the staff uses the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the ASTM Standards and the ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); RG 1.99, Rev. 2; NUREG-0800 Standard Review Plan (SRP) Section 5.3.2; and Generic Letter (GL) 88-11.

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Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide Technical Specifications for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the Technical Specifications. The P/T limits are among the limiting conditions of operation in the Technical Specifications for all commercial nuclear plants in the U.S. Appendices G and H of 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits.

An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy Upper Shelf Energy (USE). GL 88-11 requested that licensees and permittees use the methods in RG 1.99, Rev. 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H of 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline.

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the Beaver Valley Unit 1 reactor vessel. The amount of neutron irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff has determined that the material with the highest ART at 16 EFPY at 1/4T (T = reactor vessel beltline thickness) was the lower shell plate B6903-1 with 0.20% copper (Cu), 0.54% nickel (Ni), and an initial RT_{ndt} of 27°F. The material with the highest ART at 16 EFPY at 3/4T was the intermediate shell plate (B6607-2) with 0.14% Cu, 0.62% Ni, and an initial RT_{ndt} of 73°F.

The licensee has removed three surveillance capsules (V, U, and W) from Beaver Valley Unit 1 (References 5, 6, 7, 8, 9). The results from capsule V were published in Westinghouse Report WCAP-9860; the results from capsule U in WCAP-10867; and the results from capsule W in WCAP-12005. All surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline material at 1/4T, plate B6903-1, the staff calculated the ART at 16 EFPY to be 226.0°F. For the 3/4T limiting material, plate B6607-2, the staff calculated the ART at 16 EFPY to be 188.7°F. The staff used a fluence of 1.32E19 n/cm^2 for 1/4T and 5.11E18 n/cm^2 for 3/4T.

The ART of plate B6903-1 was determined by Section 2 of RG 1.99, Rev. 2, because that material was in the surveillance capsules. The ART for plate B6607-2 was calculated per Section 1 of RG 1.99, Rev. 2, because it was not in the surveillance capsules.

The licensee used the method in RG 1.99, Rev. 2, to calculate an ART of 224°F at 16 EFPY at 1/4T and 188°F at 3/4T for the same limiting plate materials. The staff judges that a difference of 2°F between the licensee's ART of 224°F and the staff's ART of 226°F at 1/4T is acceptable. Substituting the ART of 226°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on the flange reference temperature of 60°F, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

LTOP Considerations

In the current TS, when the temperature of one or more of the non-isolated RCS cold legs is less than or equal to 292°F, LTOP is provided by either of the pressurizer power operated relief valves (PORVs) with a lift setting of less than or equal to 444 psig or an RCS vent greater than or equal to 3.14 square inches. The operability of a PORV or an RCS vent of greater than or equal to 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 the RCS cold leg temperature is less than or equal to 292°F. Each PORV has adequate relieving capacity to protect the RCS against P/T limits when the transient is limited to either (1) the start of an idle RCP with secondary water temperature in the steam generator less than or equal to 25°F above the RCS cold leg temperature or (2) the mass addition transient due to one centrifugal charging pump in operation with maximum charging flow into the RCS, assuming the letdown line is isolated.

By letter dated July 15, 1991, the licensee proposed a change in the PORV setting from 444 psig to 432 psig. The new PORV setting is based on the results of an analysis corresponding to the 16 EFPY P/T curves with a correction of 4 psig to address a non-conservatism identified in the new steam

generator tube plugging analysis. The licensee also proposed changing the LTOP enable temperature from 292°F to 314°F. However, after reviewing the information submitted by the licensee, the staff concluded that the newly proposed enable temperature was not sufficiently conservative since the temperature difference between the bulk fluid and the vessel metal was not factored into the licensee's analysis. In response to a staff request to consider this effect, the licensee, in its letter dated November 2, 1992, proposed a revised enable temperature of 329°F. This enable temperature was calculated by the licensee's contractor, Westinghouse, as described in topical report FDRT-SRPLO-266(92) "Methodology for calculating Enable Temperature Set Point for Beaver Valley Unit 1," dated September 1992. The staff has reviewed the Westinghouse methodology and concluded that it leads to the most conservative enable temperature, which is:

> the reactor coolant fluid temperature corresponding to vessel metal temperature of at least RT_{NDT}+90°F at the beltline location that is controlling in the 10 CFR Part 50 Appendix G fracture toughness limit calculations.

The licensee's proposed changes in technical specifications 3.4.1.6, 3.4.9.3 and associated bases sections reflect the TS changes discussed above. The staff finds that the changes are based on applicable regulatory guidance in Standard Review Plan 5.2.2 (Revision 2), are reasonably conservative and are acceptable.

In addition, the staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 16 EFPY because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The proposed limits also satisfy Generic Letter 88-11 because the licensee used the method in RG 1.99, Rev. 2, to calculate the ART. Hence, the proposed P/T limits may be incorporated into the Beaver Valley Unit 1 Technical Specifications.

3.0 STATE CONSIDERATION

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

4.0 ENVIRONMENTAL CONSIDERATION

This amendment changes 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 amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (56 FR 57894). Accordingly, the amendment

meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: J. Tsao
C. Liang

Date: January 15, 1993

6.0 REFERENCES

1. Letter from J. D. Sieber (DQLCo) to USNRC Document Control Desk, Subject: Beaver Valley Power Station, Unit 1, Proposed Operating License Change Request 191 (TAC No. 80559), July 15,1991.

2. Letter from J. D. Sieber (Duquesne Light Company) to NRC, "Beaver Valley Power Station, Unit No. 1, Docket No. 50-334, License No. DPR-66, Proposed Operating License Change Request No. 191 correction," dated September 30, 1991.

3. Letter from J. D. Sieber (DLC) to NRC, "Beaver Valley Power Station, Unit No. 1, Docket No. 50-334, License No. DPR-66, Response to Information Request dated November 14, 1991," dated December 13, 1991.

4. Letter from J. D. Sieber (Duquesne Light Company) to NRC, "Beaver Valley Power Station, Unit No. 1, Docket No. 50-334, License No. DPR-66, Proposed Operating License Change Request No. 191, Revision 1," dated November 2, 1992.

5. Letter from J. D. Sieber (DL) to USNRC Document Control Desk, Subject: Beaver Valley Power Station, Unit 1, Reactor Vessel Capsule W Test Results Report (WCAP-12005), January 24, 1989.

6. R. S. Boggs et al, "Analysis of Capsule U from the Duquesne Light Company Beaver Valley Unit Reactor Vessel Radiation Surveillance Program, WCAP-10867," Westinghouse Electric Corporation, September, 1985

7. Letter from J. J. Carey (DL) to S. A. Varga (USNRC), subject: Beaver Valley Power Station, Unit 1, Reactor Vessel Irradiation Specimen Test Report (WCAP-9860), October 15, 1981.

8. Letter from C. N. Dunn (DL) to R. W. Reid (USNRC), subject: Beaver Valley Power Station, Unit 1, Reactor Vessel Material Surveillance Program, July 21, 1977.

9. Beaver Valley Final Safety Analysis Report