July 12, 1977

Docket No.: 50-334

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Duquesne Light Company
ATTN: Mr. C. N. Dunn, Vice President
Operations Division
435 Sixth Avenue
Pittsburgh, Pennsylvania 15219

Gentlemen:

The Commission has issued the enclosed Amendment No. 9 to Facility Operating License No. DPR-66 for the Beaver Valley Power Station Unit No. 1. The amendment consists of changes to the Technical Specifications in response to your application dated February 11, 1977.

This amendment incorporates into the Technical Specifications changes resulting from a reanalysis of ECCS performance with the following assumptions: (1) upper head fluid temperature equal to hot leg temperatures; (2) one percent steam generator tube plugging; and (3) the rod bow linear power penalty removed for burnups less than 24,000 MWD/MTU. Changes to the Technical Specifications include raising the total nuclear peaking factor to 2.32 and a deletion of the use of the Axial Power Distribution Monitoring System for in-core monitoring of axial power.

We have evaluated your reanalysis of ECCS performance submitted with your November 30, 1976 letter, in response to the Commission's Order of August 27, 1976. We have found your reanalysis to be acceptable and to adequately consider revised upper head temperatures. Effective upon issuance of this amendment, the Commission's Order for Modification of License dated August 27, 1976, relative to Facility Operating License No. DPR-66, is terminated.

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

Sincerely,

Original Signed by

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Const. 1

Enclosures and cc: See next page

OFFICE	ORB#4:DOR	ORB#4:DOR	C-RSB-OT:DOR	STSG:DOR	OELD	C-ORB#4:DOR
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DATE	7/5/77	7/5/77	7/ /77	7/5/77	7/11/77	7/12/77

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☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

Enclosures:
1. Amendment No. 9
2. Safety Evaluation
3. Notice

cc w/enclosures: See next page

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SURNAME ➤						
DATE ➤						

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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. 9
License No. DPR-66

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

   A. The application for amendment by Duquesne Light Company, filed on behalf of itself, Ohio Edison Company, and Pennsylvania Power Company (the licensees) dated February 11, 1977, 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 Appendices A and B, as revised through Amendment No. 9, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 12, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 9

FACILITY OPERATING LICENSE NO. DPR-66

DOCKET NO. 50-334

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

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3/4 2-2A (deleted)
3/4 2-3A (deleted)
3/4 2-4A (deleted)
3/4 2-1B
3/4 2-2B
3/4 2-3B
3/4 2-4B
3/4 2-5
3/4 2-6a
3/4 2-14 (deleted)
3/4 2-15 (deleted)
3/4 2-16 (deleted)
3/4 3-47 (deleted)
3/4 3-48 (deleted)
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3/4.2 POWER DISTRIBUTION LIMITS

AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within a ± 7% target band (flux difference units) about the target flux difference.

APPLICABILITY: MODE 1 ABOVE 50% RATED THERMAL POWER*

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the ± 7% target band about the target flux difference and with THERMAL POWER:

1. Above 90% of RATED THERMAL POWER, within 15 minutes:

a) Either restore the indicated AFD to within the target band limits, or

b) Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER.

2. Between 50% and 90% of RATED THERMAL POWER:

a) POWER OPERATION may continue provided:

1) The indicated AFD has not been outside of the ± 7% target band for more than 1# hour penalty deviation cumulative during the previous 24 hours, and

2) The indicated AFD is within the limits shown on Figure 3.2-1. Otherwise, reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.

b) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1.1 provided the indicated AFD is maintained within the limits of Figure 3.2-1. A total of 16 hours operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

* See Special Test Exception 3.10.2

\# A 2 hour penalty deviation is permissible during tests performed as part of the Augmented Startup Test Program.

LIMITING CONDITION FOR OPERATION (Continued)

b. THERMAL POWER shall not be increased above 90% of RATED THERMAL POWER unless the indicated AFD is within the ± 7% target band and ACTION 2.a) 1), above has been satisfied.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD has not been outside of the ± 7% target band for more than 1 hour penalty deviation cumulative during the previous 24 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AXIAL FLUX DIFFERENCE shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1. At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2. At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

b. Monitoring and logging the indicated AXIAL FLUX DIFFERENCE for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AXIAL FLUX DIFFERENCE Monitor Alarm is inoperable. The logged values of the indicated AXIAL FLUX DIFFERENCE shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its ± 7% target band when at least 2 of 4 or 2 of 3 OPERABLE excore channels are indicating the AFD to be outside the target band. POWER OPERATION outside of the ± 7% target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

b. One-half minute penalty deviation for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels below 50% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.1.3 The target flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days with all part length control rods fully withdrawn. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 The target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference pursuant to 4.2.1.3 above or by linear interpolation between the most recently measured value and 0 percent at the end of the cycle life. The provisions of Specification 4.0.4 are not applicable.

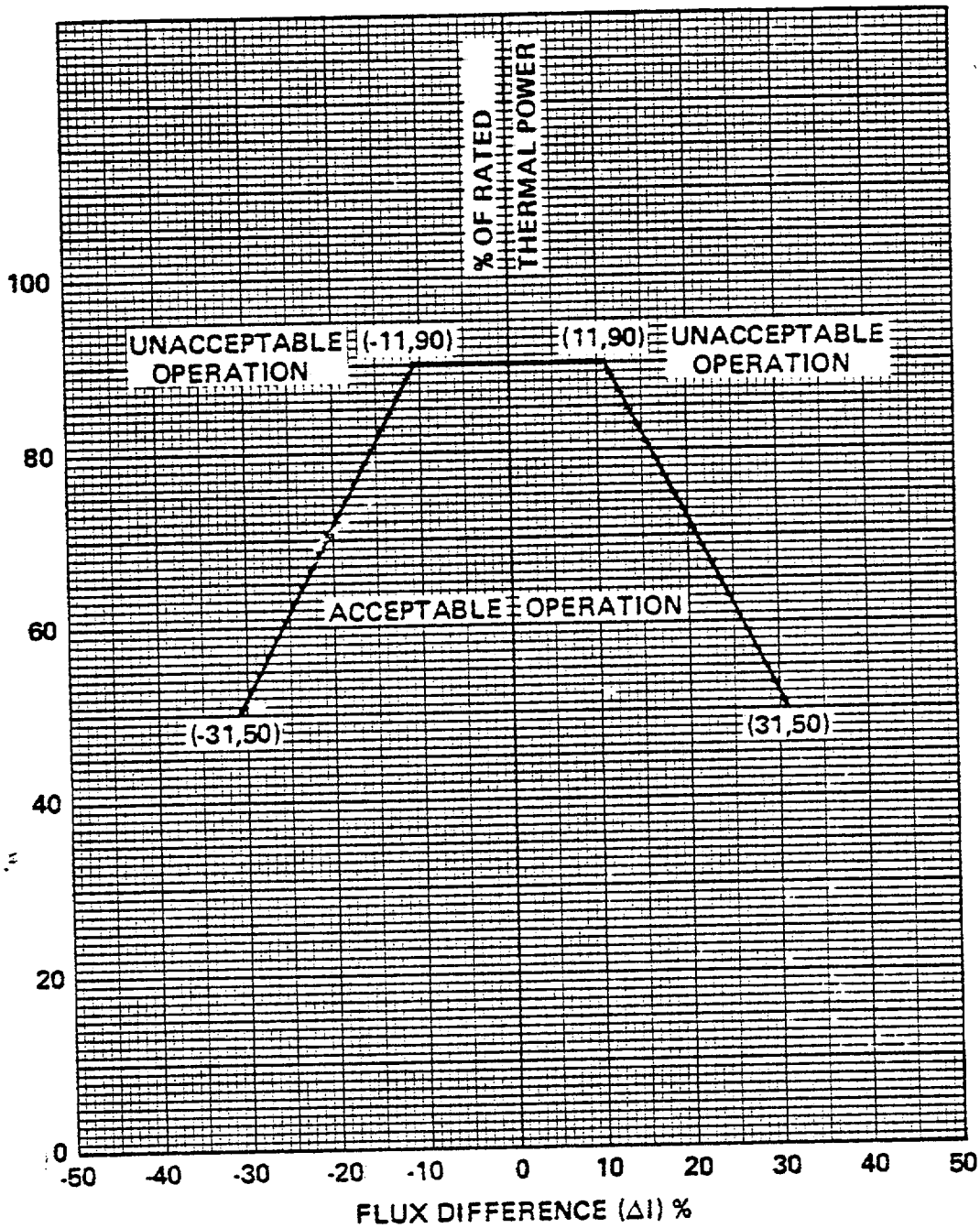

FIGURE 3.2-1 AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER

HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \left[\frac{2.32}{P}\right] [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq [(4.64)] [K(Z)] \text{ for } P \leq 0.5$$

where $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

and K(Z) is the function obtained from Figure 3.2-2 for a given core height location.

APPLICABILITY: MODE 1

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similiarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit. The Overpower ΔT Trip Setpoint reduction shall be performed with the reactor subcritcal.

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 F_{xy} shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured F_{xy} component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.

c. Comparing the F_{xy} computed (F_{xy}^C) obtained in b, above to:

1. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in e and f below, and

2. The relationship:

$$F_{xy}^L = F_{xy}^{RTP} \; [1+0.2(1-P)]$$

where F_{xy}^L is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP} and P is the fraction of RATED THERMAL POWER at which F_{xy} was measured.

d. Remeasuring F_{xy} according to the following schedule:

1. When F_{xy}^C is greater than the F_{xy}^{RTP} limit for the appropriate measured core plane but less than the F_{xy}^L relationship, additional power distribution maps shall be taken and F_{xy}^C compared to F_{xy}^{RTP} and F_{xy}^L :

a) Either within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^C was last determined, or

b) At least once per 31 EFPD, whichever occurs first.

2. When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limits for RATED THERMAL POWER within specific core planes shall be:

1. $F_{xy}^{RTP} \leq 1.71$ for all core planes containing bank "D" control rods and/or any part length rods, and
2. $F_{xy}^{RTP} \leq 1.55$ for all unrodded core planes.

f. The F_{xy} limits of e, above, are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:

1. Lower core region from 0 to 15%, inclusive.
2. Upper core region from 85 to 100% inclusive.
3. Grid plane regions at 17.8 ± 2%, 32.1 ± 2%, 46.3 ± 2%, 60.6 ± 2% and 74.9 ± 2%, inclusive.
4. Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the bank "D" or part length control rods.

g. With F_{xy}^{C} exceeding F_{xy}^{L}:

1. The $F_Q(Z)$ limit shall be reduced at least 1% for each 1% F_{xy}^{C} exceeds F_{xy}^{L}, and
2. The effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit.

4.2.2.3 When $F_Q(Z)$ is measured pursuant to Specification 4.10.2.2, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS		
	3 Loops In Operation	2 Loops In Operation & Loop Stop Valves Open	2 Loops In Operation & Isolated Loop Stop Valves Closed
Reactor Coolant System T_{avg}	$\leq$ 581°F	$\leq$ 570°F	$\leq$ 570°F
Pressurizer Pressure	$\geq$ 2220 psia*	$\geq$ 2220 psia*	$\geq$ 2220 psia*
Reactor Coolant System Total Flow Rate	$\geq$ 265,500 gpm	$\geq$ 189,000 gpm	$\geq$ 187,800 gpm

*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.

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core $\geq$ 1.30 during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature & cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Conditions I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of hot channel factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope of 2.32 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions with the part length control rods withdrawn from the core. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are

obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

Although it is intended that the plant will be operated with the AXIAL FLUX DIFFERENCE within the ± 7% target band about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1 hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels between 50% & 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% & 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the target band & the THERMAL POWER is greater than 90% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% & 90% & 15% & 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour & 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band near the beginning of core life.

a. abnormal perturbations in the radial power shape, such as from rod misalignment, effect $F^N_{\Delta H}$ more directly than F_Q,

b. although rod movement has a direct influence upon limiting F_Q to within its limit, such control is not readily available to limit $F^N_{\Delta H}$, and

c. errors in prediction for control power shape detected during startup physics tests can be compensated for in F_Q by restriting axial flux distributions. This compensation for $F^N_{\Delta H}$ is less readily available.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted. The limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3 percent for each percent of tilt in excess of 1.0.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient & accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 9 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

Introduction

By letter dated February 11, 1977, Duquesne Light Company (DLC) requested approval of proposed Technical Specification changes for Beaver Valley Power Station Unit No. 1 (BVPS-1) relating to the Total Nuclear Peaking Factor (F_Q) and the Axial Power Distribution Monitoring System (APDMS).

Discussion

On August 27, 1976, we issued an Order for Modification of License for BVPS-1 which modified the Technical Specification limit for the F_Q to 2.22. The Order was issued because recent operating data gathered at the Connecticut Yankee facility indicated that reactor vessel upper head fluid temperatures were higher than reactor inlet water temperatures for reactor vessels similar in design to that in operation at BVPS-1. These higher upper head temperatures would have the effect of increasing the calculated peak clad temperature in the event of a loss of coolant accident (LOCA). Our Order required DLC to submit a reevaluation of emergency core cooling system (ECCS) performance with an approved Westinghouse Model with an appropriate correction for upper head water temperature.

Nuclear Peaking Factor

By letter dated November 30, 1976, DLC submitted the requested re-evaluation of the ECCS performance. The reevaluation used the October 1975 version of the Westinghouse ECCS Evaluation Model[1, 2, 3, 4]

and assumed the upper head fluid temperature equal to the fluid outlet (hot leg) temperature. The reanalysis supersedes the previously performed ECCS evaluation for a large break LOCA[5] which used the October 1975 version of the evaluation model but which assumed that the upper head fluid temperature was equal to cold leg temperature. A reanalysis of the small break LOCA was not required since the small breaks are relatively insensitive to the upper head fluid temperature change[6]. The reevaluation of the ECCS performance in Westinghouse plants was required because recent experimental data had indicated that the actual temperature in the upper reactor vessel head was in the range of the cold leg temperature plus 50 to 75 percent of the difference between vessel inlet and outlet temperatures[7]. Westinghouse has proposed to undertake a program designed to measure this temperature in the operating reactors. In the meantime, the NRC decided conservatively to request all the Westinghouse plant licensees to reevaluate the ECCS performance with upper head fluid temperature equal to the fluid outlet temperature.

The ECCS analysis provided by DLC was performed with the following input parameters:

Licensed Core Power	-	102% of 2652 MWt
Peak Linear Power	-	102% of 12.07 kw/ft
Nuclear Peaking Factor	-	2.32
Accumulator Water Volume	-	1025 $ft.^3$ each

The analysis consisted of the evaluation of ECCS performance for double ended cold leg guillotine break (DECLG) with a discharge coefficient C_D of 0.4 and assumed one percent uniform plugging of steam generator tubes. Also the rod bow linear power penalty, which was explicitly incorporated in the previous analysis, has been removed. The removal of the rod bow penalty results from a recent Westinghouse study[8] which shows that for burnups less than or equal to 24,000 MWD/MTU, the rod bow penalty is equal to or less than 5.7%. This rod bow penalty is adequately accounted for statistically in the nuclear and engineering penalties incorporated into the original LOCA analysis. DLC identified the break with $C_D = 0.4$ as the critical break with respect to limiting values of peak cladding temperature and local zirconium water reaction from the previous break spectrum analysis which used the October 1975 version of the evaluation model but which was performed with upper leg fluid temperature equal to the cold leg temperature[5]. DLC justified

the use of this analysis for determining critical break size by referencing the generic sensitivity study performed by Westinghouse[6] in which it was shown that the change in the upper head fluid temperature does not affect the critical break size. The peak cladding temperature and the maximum local zirconium water reaction corresponding to the critical break identified by DLC are 2041°F and 5.05 percent, respectively.

Based on our review, we conclude that the results of the ECCS reanalysis, using the October 1975 version of the Westinghouse ECCS Evaluation Model with upper head temperature equal to the outlet (hot leg) temperature, are conservative relative to the 10CFR50.46 criteria. However, the ECCS analysis was limited to one percent plugging of steam generator tubes and an ECCS reevaluation by DLC would be required in order for BVPS-1 to operate above this limit. Also, the removal of the linear power penalty put the upper limit on fuel burnup at 24,000 MWD/MTU. With the above mentioned considerations, we find the results of the reanalysis submitted by DLC acceptable, and since the F_Q assumed in the reanalysis was 2.32, this value is acceptable and should replace the value of 2.22 presently in the Technical Specifications.

Axial Power Distribution Monitoring System

By their letter dated February 11, 1977, DLC also requested that the BVPS-1 Technical Specifications be amended to delete the required use of the APDMS below an average core burnup of 3500 MWD/MTU.

The basis for this request is an increase of the value of the F_Q at rated power to 2.32. The increased permissible linear heat rate allowed from increasing F_Q to 2.32 is about 4% and corresponds, for invariant radial peaking factors, to an increase of the permissible axial peaking factor (F_Z) of about 4%. DLC states that, using a constant axial control strategy, the core axial power distribution can be controlled to the less restrictive value of F_Z using excore instrumentation alone; hence, there is adequate justification for deletion of incore monitoring requirements using the APDMS.

Constant axial control (CAC) strategy has been described in the open literature[10] and defended in Westinghouse proprietary topical report WCAP 8385[11]. Validity of the CAC strategy at beginning of cycle has been confirmed by work performed at Brookhaven[12]. These studies show that using CAC (e.g., a target axial offset and a 5% flux difference control band about the target axial offset), the realizable reactor power shapes, and hence magnitude and location of limiting conditions,

will be less severe than the initial conditions used in the safety analysis. The studies address base load and load follow operation. The studies show on a generic basis that presuming use of CAC strategy, monitoring the reactor by excore instrumentation alone will adequately assume that the limiting value of F_Q of 2.32 will not be exceeded.

A previous ECCS analysis for BVPS-1[9] supported a value of F_Q less than or equal to 2.23. (As discussed above, this value was reduced by an Order to 2.22.) Based on an F_Q less than or equal to 2.23, a plant specific CAC analysis for cycle 1 was submitted[13] and reviewed[9]. The plant specific analysis assumed a $\pm$ 7% flux difference monitoring band. The analysis supported use of the CAC strategy with monitoring on excore instrumentation alone for core average fuel exposure greater than or equal to 3500 MWD/MTU. For core burnup less than 3500 MWD/MTU additional monitoring using the APDMS was required.

We concluded above that an F_Q of 2.32 is acceptable for BVPS-1. The revised value of F_Q is consistent with the generic Westinghouse topical report[11] which justifies monitoring on excores alone presuming a $\pm$ 5% flux difference monitoring band. The plant specific analysis which we previously reviewed[9] assumed a $\pm$ 7% flux difference monitoring band. The extended monitoring band increases the realizable power distributions, most notably the axial power shapes, that must be considered as initial conditions to the safety analysis. This plant specific analysis for fuel exposure greater than 3500 MWD/MTU has explicitly verified the safety of the expanded monitoring band presuming excore monitoring alone. For core exposure less than 3500 MWD/MTU, a 7% flux difference monitoring band and a value of F_Q of 2.23, this analysis showed that monitoring of the axial power distribution using the incore APDMS would be required above 95.5% of rated power. Review of this analysis shows sufficient margin with the revised value of F_Q of 2.32 to justify deletion of required use of the APDMS at exposures less than 3500 MWD/MTU.

Based on review of available analysis, we conclude that operation of BVPS-1 during cycle 1 using a constant axial control strategy and a 7% flux difference monitoring band with excore monitoring alone is acceptable, and that use of the incore APDMS is no longer necessary and its required use may be deleted from the Technical Specifications.

Our review of Reference 11 is nearing completion. Assuming acceptance of this topical report, operation beyond cycle 1 with a 5% flux difference monitoring band could be approved on a generic basis. However, if DLC wishes to operate BVPS-1 beyond cycle 1 with a 7% flux difference monitoring band, a plant and cycle specific analysis will be required.

Technical Specifications

We have reviewed the Technical Specification changes relating to raising the F_Q to 2.32 and deleting the APDMS which DLC proposed by letter dated February 11, 1977. We have made some changes to DLC's submittal and DLC has agreed to the changes. The proposed Technical Specifications reflect a testing and surveillance program along with the limiting conditions for operation that provides assurance that the F_Q upper bound limit of 2.32 will not be exceeded. We also conclude that the proposed Technical Specifications are acceptable and consistent with those of other facilities operating with similar systems and found acceptable by the staff.

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 this amendment.

Conclusion

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

Dated: July 12, 1977

REFERENCES

1. WCAP-8339, "Westinghouse ECCS Evaluation Model - Summary," July 1974.

2. WCAP-8471 (Proprietary), WCAP-8472 (Nonproprietary), "Westinghouse ECCS Evaluation Model - Supplementary Information," April 1, 1975.

3. WCAP-8622 (Proprietary), WCAP-8623 (Nonproprietary) "Westinghouse ECCS Evaluation Model - October 1975 Version," November 1975.

4. Letter NS-CE-924, C. Eicheldinger (Westinghouse) to D. B. Vassallo (NRC) dated January 23, 1976.

5. FSAR, Beaver Valley Power Station, Unit 1, Amendment 21, June 1976.

6. WCAP-8853, "Westinghouse ECCS - Three-Loop Plant (17x17) Sensitivity Studies," dated September 1976.

7. Letter NS-CE-1163, C. Eicheldinger (Westinghouse) to V. Stello (NRC), dated August 13, 1976.

8. NRC - "Interim Safety Evaluation Report on Westinghouse Fuel Bowing," dated April 1976.

9. Supplement No. 3, FSAR, Beaver Valley Power Station, Unit 1, March 19, 1976.

10. R. J. Sipush, et. al.,"Load-Follow Demonstrations Employing Constant Axial Offset Power-Distribution Control Procedures ,"Nuclear Technology, Vol. 31, October 1976.

11. T. Morita, et . al.,"Topical Report Power Distribution Control and Load Following Procedures", Westinghouse WCAP-8385, September, 1974.

12. A. Buslick, et.al., "Power Peaking During Load Following Using Westinghouse Constant Axial Offset Power-Distribution Control," BNL-NUREG-22477, January, 1977.

13. Letter to D. B. Vassallo (NRC) from E. J. Woolever (Duquesne Light Company) February 26, 1976, "Beaver Valley Power Station-Unit No. 1 Power Distribution Limits."

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-334

DUQUESNE LIGHT COMPANY

OHIO EDISON COMPANY

PENNSYLVANIA POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 9 to Facility Operating License No. DPR-66, issued to Duquesne Light Company, Ohio Edison Company, and Pennsylvania Power Company (the licensees), for operation of the Beaver Valley Power Station Unit No. 1 (the facility) located in Beaver County, Pennsylvania. The amendment is effective as of its date of issuance.

The amendment incorporates into the Technical Specifications changes resulting from a reanalysis of ECCS performance with the following assumptions: (1) upper head fluid temperature equal to hot leg temperatures; (2) one percent steam generator tube plugging; and (3) the rod bow linear power penalty removed for burnups less than 24,000 MWD/MTU. Changes to the Technical Specifications include raising the total nuclear peaking factor to 2.32 and a deletion of the use of the Axial Power Distribution Monitoring System for in-core monitoring of axial power. This amendment also terminates the Commission's Order for Modification of License dated August 27, 1976.

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 issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated February 11, 1977, (2) Amendment No. 9 to License No. DPR-66, 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, N.W., Washington, D. C. and at the Beaver Area Memorial Library, 100 College Avenue, Beaver, Pennsylvania.

A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, 12th day of July 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

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