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JUN 14 1982

Docket No. 50-334

Mr. J. J. Carey, Vice President
Nuclear Division
Duquesne Light Company
435 Sixth Avenue
Pittsburgh, Pennsylvania 15219

Dear Mr. Carey:

The Commission has issued the enclosed Amendment No. 51 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 transmitted by letters dated February 23, 1982, as supplemented April 21, April 26, May 5, May 26, June 1 and June 10, 1982.

The amendment modifies the Technical Specifications on rod misalignment, core peaking factor limits and associated parameters.

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

Sincerely,

Original signed by:

Peter S. Tam, Project Manager
Operating Reactors Branch No. 1
Division of Licensing

Enclosures:
1. Amendment No. 51 to DPR-66
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

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Duquesne Light Company

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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. 51
License No. DPR-66

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

   A. The application for amendment by Duquesne Light Company, Ohio Edison Company, and Pennsylvania Power Company (the licensees) dated February 23, 1982, as amended by letters dated April 21, April 26, May 5, May 26, June 1 and June 10, 1982, 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 containined in Appendices A and B, as revised through Amendment No. 51, 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

Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: JUN 14 1982

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 51 TO FACILITY OPERATING LICENSE NO. DPR-66

DOCKET NO. 50-334

Revise Appendix A as follows:

Remove Pages	Insert Pages
2-6	2-6
3/4 1-18	3/4 1-18
3/4 1-19	3/4 1-19
---	3/4 1-19A
3/4 1-20	3/4 1-20
3/4 1-20A	3/4 1-20A
---	3/4 1-20B
3/4 1-21	3/4 1-21
3/4 1-22	3/4 1-22
3/4 1-23	3/4 1-23
---	3/4 1-23A
3/4 10-2	3/4 10-2
3/4 10-6	3/4 10-6
B3/4 1-3	B3/4 1-3
B3/4 1-4	B3/4 1-4
3/4 2-6A	3/4 2-6A
3/4 2-8	3/4 2-8
---	3/4 2-9A
B3/4 2-5	B3/4 2-5
6-16	6-16
6-17	6-17

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 - ≤ 26% of RATED THERMAL POWER
	High Setpoint - ≤ 109% of RATED THERMAL POWER	High Setpoint - ≤ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 seconds	≤ 5.5% 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	≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 seconds
5. Intermediate Range, Neutron Flux	≤ 25% of RATED THERMAL POWER	≤ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	≤ 10^5 counts per second	≤ 1.3×10^5 counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 3
9. Pressurizer Pressure--Low	≥ 1945 psig	≥ 1935 psig
10. Pressurizer Pressure--High	≤ 2385 psig	≤ 2395 psig
11. Pressurizer Water Level--High	≤ 92% of instrument span	≤ 93% of instrument span
12. Loss of Flow	≥ 90% of design flow per loop	≥ 89% of design flow per loop

*Design flow is 88,500 gpm per loop.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods shall be OPERABLE and positioned within ± 12 steps (indicated position, as determined in accordance with Specification 3.1.3.2) corresponding to their respective group demand counter position.

APPLICABILITY: MODES 1* and 2*

ACTION:

a. With one or more full length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With more than one full length rod inoperable or misaligned from the group demand counter position by more than ± 12 steps (indicated position determined in accordance with Specification 3.1.3.2), be in HOT STANDBY within 6 hours.

c. With one full length rod trippable but inoperable due to causes other than addressed by ACTION a, above, or misaligned from its group demand counter position by more than ± 12 steps (indicated position determined in accordance with Specification 3.1.3.2), POWER OPERATION may continue provided that within one hour either:

   1. The rod is restored to OPERABLE status within the above alignment requirements, or

   2. The rod is declared inoperable and the remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limits of Figures (3.1-1) and (3.1-2); the THERMAL POWER level shall be restricted pursuant to Specification (3.1.3.6) during subsequent operation, or

   3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

      a) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the hour and, within the next 4 hours the high neutron flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER.

b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F_{\Delta H}$ are verified to be within their limits within 72 hours.

d) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 Each shutdown and control rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

*See Special Test Exceptions 3.10.2 and 3.10.4

TABLE 3.1-1

ACCIDENT ANALYSES REQUIRING REEVALUATION IN THE EVENT OF AN INOPERABLE FULL OR PART LENGTH ROD

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Loss of Reactor Coolant From Small Ruptured Pipes Or From Cracked Large Pipes Which Actuates The Emergency Core Cooling System

Single Rod Cluster Control Assembly Withdrawal At Full Power

Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)

Major Secondary Systems Pipe Rupture

Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

POSITION INDICATION SYSTEMS-OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 The shutdown and control rod position indication system shall be OPERABLE as follows:

- Group Demand Counter(1), 1 per group
- Individual analog rod position instrument channel, 1 per rod, ± 12 steps(1) accuracy
- Automatic Rod Position Deviation Monitor(2), setpoints ≤ 12 steps

(1) During the first hour during and following rod motion, the group demand counter is the primary indicator of precise rod position information , with the analog channels displaying general rod movement information. For power levels below 50%, a 1-hour thermal soak time is allowed before the analog channels are required to perform within the specified accuracy.

(2) For power levels below 50% a one hour thermal soak time is allowed. Therefore, if a Rod Position Deviation Monitor alarm clears itself within this hour, the alarm is considered invalid.

APPLICABILITY: MODES 1 and 2*

ACTION:

a. If the Rod Position Indicating System indicates(2) a potentially misaligned rod(s), this indication shall be verified immediately (within 15 minutes) by measuring the analog rod position channel primary voltage. If this meausrement confirms that a rod is misaligned, Specification 3.1.3.1.3.c is applicable.

b. With a maximum of one group demand position indicator per bank inoperable either:

1. Verify that all rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 12 steps (indicated position) of each other at least once per 8 hours, or

2.. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

c. With a maximum of one analog rod position indicator per bank inoperable, either:

1. Determine rod position for the affected rod by measuring the detector primary voltage, as follows:

a. Immediately

b. If the associated rod moved greater than 6 steps (greater than 12 steps if all the rods in the group have been determined to be within 6 steps of group demand counter indicator by primary voltage measurements within the previous 4 hours)

c. At 4 hour intervals if the affected rod(s) are not fully inserted or withdrawn

d. At 24 hour intervals if the affected rod(s) are fully inserted or withdrawn, or,

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

d. With the Automatic Rod Deviation Monitor inoperable, POWER OPERATION may continue provided that the deviation between the indicated positions is checked by the operator at least once per 4 hours. The provisions of Specification 3.0.4 are not applicable.

4.1.3.2.1 Each of the group demand counters shall be determined to be OPERABLE by:

a. Performing a CHANNEL CHECK by the group demand counters within a bank, and observing proper overlap of the indicated positions, at least once per 7 days; and

b. Performing a CHANNEL CHECK by an intercomparison between the control bank benchboard indicators and the logic solid state indicators in the logic cabinet, and determining their agreement within ± 2 steps, at least once per 92 days.

4.1.3.2.2 Each of the analog rod position indicators shall be determined to be OPERABLE by:

a. Performing a CHANNEL CHECK by intercompaison** between each analog rod position indicator and its corresponding group demand counter at least once per 24 hours.

b. Performing a CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at least once per 18 months.

4.1.3.2.3 The Automatic Rod Deviation Monitor shall be determined to be OPERABLE by performing a functional test at least once per 7 days, and the deviation between the position indicated by the individual analog rod position instrument channel and the position indicated by the corresponding group demand counter shall be checked** manually for each rod at least once per 24 hours.

*For Core PHYSICS TESTING in Mode 2, primary detector voltage measurements may be used to determine the position of rods in shutdown banks A and B and control banks A and B for the purpose of satisfying Specification 3.1.3.2. During Mode 2 operation, rod position indicators for shutdown banks A and B and control banks A and B may deviate from the group demand indicators by greater than ± 12 steps during reactor startup and shutdown operations, while rods are being withdrawn or inserted. If the rod position indicators for shutdown banks A and B and control banks A and B deviate by greater than ± 12 steps from the group demand indicator, rod withdrawal or insertion may continue until the desired group height is achieved. When the desired group height is achieved, a one hour soak time is allowed below 50% reactor power to permit stabilization of the rod position analog indicators. To attain thermal equilibrium during the one hour soak time, the absolute value of rod motion shall not exceed 6 steps.

**For power levels below 50% one hour thermal "soak time" is permitted. During this soak time, the absolute valve of rod motion is limited to six steps.

POSITION INDICATION SYSTEM-SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3. The group step demand counter and at least two analog rod position indication channels per group shall be OPERABLE for each group of shutdown or control rods not fully inserted.

APPLICABILITY: MODES 3* , 4* and 5*

ACTION:

With less than the above required position indicator(s) OPERABLE, immediately insert all rods or, immediately open the reactor trip system breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required position indicator(s) shall be determined to be OPERABLE by performance of Specifications 4.1.3.2.1 and 4.1.3.2.2.

*With the reactor trip system breakers in the closed position.

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) rod drop time from the fully withdrawn position shall be $\leq$ 2.2 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. $T_{avg} \geq 541^{\circ}F$, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODE 3.

ACTION:

a. With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the rod drop times within limits but determined with 2 reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to:

1. $\leq$ 61% of RATED THERMAL POWER when the reactor coolant stop valves in the nonoperating loop are open, or

2. $\leq$ 66% of RATED THERMAL POWER when the reactor coolant stop valves in the nonoperating loop are closed.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of full length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head.

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once per 18 months.

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be fully withdrawn.

APPLICABILITY: MODES 1* and 2*#

ACTION:

With a maximum of one shutdown rod not fully withdrawn, except for surveillance testing pursuant to Specification (4.1.3.1.1), within one hour either:

a. Fully withdraw the rod, or

b. Declare the rod to be inoperable and apply Specification (3.1.3.1).

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined to be fully withdrawn by use of the group demand counters, and verified by the analog rod position indicators**

a. Within 15 minutes prior to withdrawal of any rods in control banks A, B, C or D during an approach to reactor criticality, and

b. At least once per 24 hours thereafter.

*See Special Test Exception 3.10.2 and 3.10.4
**For power levels below 50% one hour thermal "soak time" is permitted. During this soak time, the absolute value of rod motion is limited to six steps.
#With Keff $\geq$ 1.0

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as shown in Figures 3.1-1 and 3.1-2.

APPLICABILITY: MODES 1* and 2*#

ACTION:

With the control banks inserted beyond the above insertion limits, except for surveillance testing pursuant to Specification (4.1.3.1.1), either:

a. Restore the control banks to within the limits within 2 hours, or

b. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the group position using the above figures, or

c. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 When the Rod Insertion Limit Monitor is OPERABLE, the deviation between the position indicated by the individual analog rod position instrument channel and the position indicated by the corresponding group demand counter shall be checked** manually for each rod at least once per 24 hours. When the Rod Insertion Limit Monitor is inoperable, the deviation bewteen indicated positions shall be checked** manually at least once per 4 hours.

*See Special Test Exception 3.10.2 and 3.10.3
#with $K_{eff} \geq 1.0$
**For power levels below 50%, one hour thermal "soak time" is permitted. During this soak time, the absolute valve of rod motion is limited to six steps.

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided;

a. The THERMAL POWER is maintained ≤85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1

ACTION:

With any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 are suspended, either;

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be ≤85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided;

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

b. The reactor trip setpoints on the OPERABLE Intermediate and Power Range Nuclear Channels are set at $\leq$25% of RATED THERMAL POWER.

APPLICABILITY: MODE 2.

ACTION:

With the THERMAL POWER >5% of RATED THERMAL POWER, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be $\leq$5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.4.2 Each Intermediate and Power Range Channel shall be subjected to a CHANNEL FUNTIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

3/4.1.2 BORATION SYSTEMS (continued)

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from all operating conditions of 1.0% Δk/k after xenon decay and cooldown to 200°F. The maximum boration capability requirements occur at EOL from full power equilibrium xenon conditions and requires 11,336 gallons of 7000 ppm borated water from the boric acid storage tanks or 49,917 gallons of 2000 ppm borated water from the refueling water storage tank.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

The boration capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% Δk/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either 5000 gallons of 7000 ppm borated water from the boric acid storage tanks or 175,000 gallons of 2000 ppm borated water from the refueling water storage tank.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the movable control assemblies is established by observing rod motion and determining that rods are positioned within ± 12 steps (indicated position), of the respective group demand counter position. The OPERABILITY of the rod position indication system is established by appropriate periodic CHANNEL CHECKS, CHANNEL FUNCTIONAL TESTS and CHANNEL CALIBRATIONS. OPERABILITY of the control rod position indicators is required to determine control rod position and thereby ensure compliance with the control rod alignment and insertion limits. The OPERABLE condition for the analog rod position indicators is defined as being capable of indicating rod position within ± 12 steps of the associated group demand indicator. For power levels below 50 percent, the specifications of this section permit a one hour stabilization period to permit stabilization of known thermal drift in the analog rod position indicator channels. During this stabilization period, greater reliance is placed upon the group demand position indicators to determine rod position. Above 50 percent power, rod motion is not expected to induce thermal transients of sufficient magnitude to exceed the rod position indicator instrument accuracy of ± 12 steps. Limited use of rod position indication primary detector voltages is allowed as a backup method of determining control rod positions. Comparison of the group demand indicator to the calibration curve is sufficient to allow determination that a control rod is indeed misaligned from its bank when primary voltage measurements are used. Comparison of the group demand counters to the bank insertion limits with verification of rod position with the analog rod

position indicators (after thermal soak after rod motion) is sufficient verification that the control rods are above the insertion limits below 50 percent power. Above 50 percent power, reliance is placed on the analog rod position indicator channels to assure that control rods are above the insertion limits.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

Continuous monitoring of rod position with respect to insertion limits and rod deviation is provided by the rod insertion limit monitor and rod deviation monitor, respectively. OPERABILITY of the rod deviation monitor is verified by a functional test at least once per 7 days and by compairson of the indicated analog positions versus the respective group demand counters at least once per 24 hours. If the rod deviation monitor or the rod insertion limit monitor is INOPERABLE, the frequency of manual comparison of indicated rod position is increased to an interval of at least once per 4 hours.

An evaluation of DNB and test data from experiments of fuel rod bowing in subchannels containing thimble cells has identified that it is appropriate to impose a penalty factor to the accident analyses DNBR results. Accordingly, a thimble cell rod bow penalty as a function of fuel burnup, is applied to the measured values of the enthalpy rise hot channel factor, $F^N_{\Delta H}$. Additional in-plant operational data and DNB tests designed to quantify the effects of rod bow have provided a basis for reducing the penalty applied to the enthalpy rise hot channel factor to account for rod bow effects. The necessary penalty is applied as a function of fuel burnup (See Figure 3.2-3).

The radial peaking factor F_{xy} (Z) is measured periodically to provide assurance that the hot channel factor, F_Q (Z), remains within its limit. The F_{xy} limit for Rated Thermal Power (F^{RTP}_{xy}) as provided in the Radial Peaking Factor Limit Report per specification 6.9.1.10 was determined from expected power control maneuvers over the full range of burnup conditions in the core.

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.

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 maximum allowed, power by 3 percent for each percent of tilt in excess of 1.0.

SURVEILLANCE REQUIREMENTS (Continued)

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} limit for Rated Thermal Power (F_{xy}^{RTP}) shall be provided for all core planes containing bank "D" control rods and all unrodded core planes in a Radial Peaking Factor Limit Report per specification 6.9.1.10.

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%, 46.4 ± 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" control rods.

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

NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$$F^N_{\Delta H} \leq 1.55\ [1 + 0.2\ (1-P)]\ \ [1-RBP(BU)]$$

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

RBP (BU) = Rod Bow Penalty as a function of region average burnup as shown in Figure 3.2-4, where a region is defined as those assemblies with the same loading date (reloads) or enrichment (first cores).

APPLICABILITY: MODE 1

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours,

b. Demonstrate thru in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL power and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

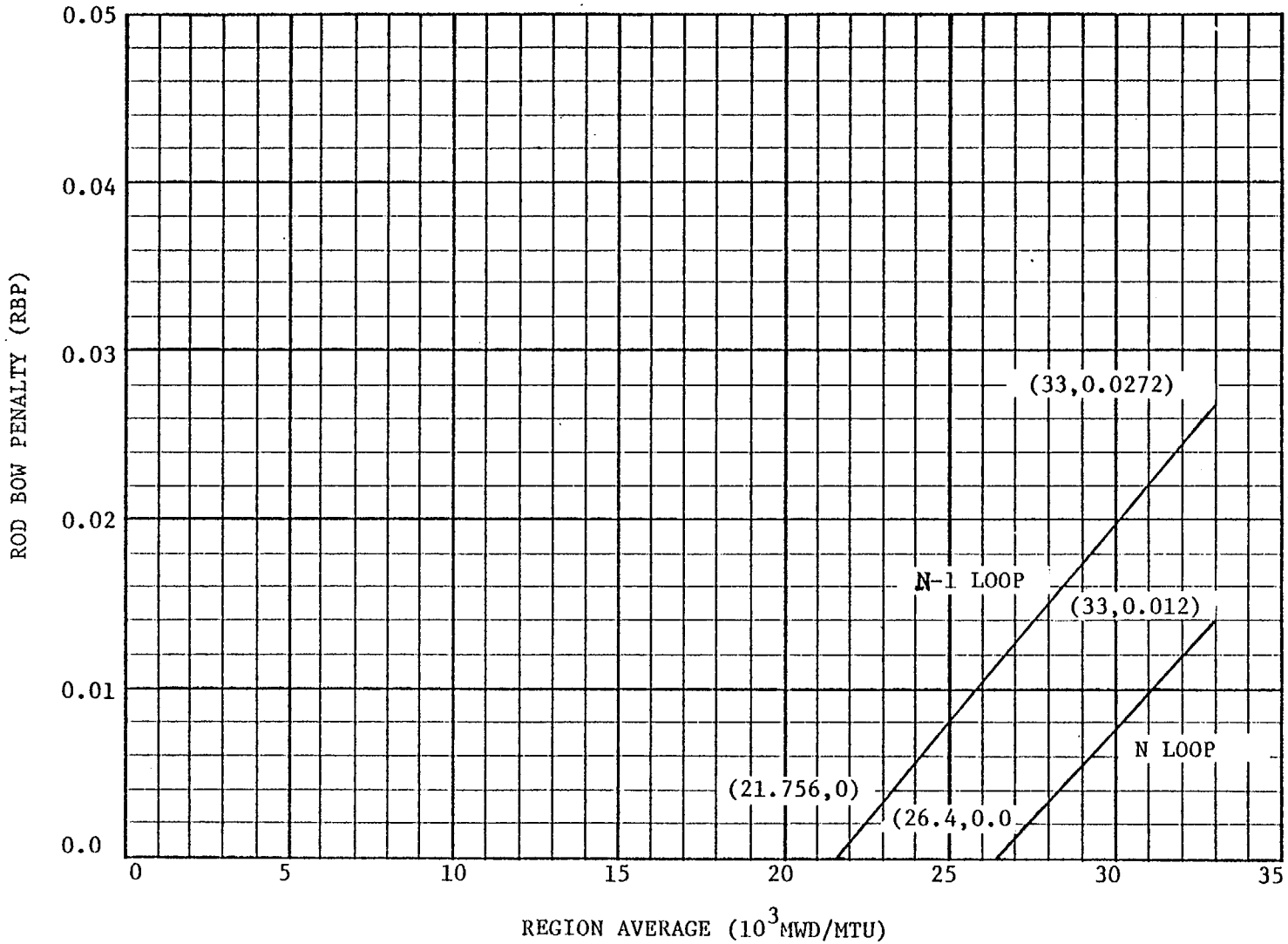

Figure 3.2-4

ROD BOW PENALTY AS A FUNCTION OF BURNUP

i. Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than that assumed in the accident analyses in the safety analysis report or technical specifications bases; or discovery during plant life of conditions not specifically considered in the safety analysis report or technical specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

THIRTY-DAY WRITTEN REPORT

6.9.1.9 The types of events listed below shall be the subject of written reports to the Director of the Regional Office within 30 days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system of engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

d. Abnormal degradation of systems other than those specified in 6.9.1.8.c above, designed to contain radioactive material resulting from the fission process.

RADIAL PEAKING FACTOR LIMIT REPORT

6.9.1.10 The F_{xy} limit for Rated Thermal Power (F_{xy}^{RTP}) shall be provided to the Director of the Regional Office of Inspection and Enforcement, with a copy to the Director, Nuclear Reactor Regulation , Attention Chief of the Core Performance Branch, U. S. Nuclear Regulatory Commission, Washington, DC 20555 for all core planes containing bank "D" control rods and all unrodded core planes at least 60 days prior to cycle initial criticality. In the event that the limit would be submitted at some other time during core life, it will be submitted 60 days

prior to the date the limit would become effective unless otherwise exempted by the Commission.

Any information needed to support F_{xy}^{RTP} will be by request from the NRC and need not be included in this report.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Director of the Office of Inspection and Enforcement Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Inservice Inspection Program Reviews, Specifications 4.4.10.1 and 4.4.10.2.

b. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

c. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.3.3.

d. Inoperable Meteorological Monitoring Instrumentation, Specification 3.3.3.4.

e. Seismic event analysis, Specification 4.3.3.3.2.

f. Sealed source leakage in excess of limits, Specification 4.7.9.1.3.

g. Fire Detection Instrumentation, Specification 3.3.3.6.

h. Fire Suppression Systems, Specification 3.7.14.1, 3.7.14.2, and 3.7.14.3.

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE OCCURRENCES submitted to the Commission.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 51 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 23, 1982, Duquesne Light Company (the licensee) informed NRC that it had completed its review of the safety and operational aspects of the design for Cycle 3 of the Beaver Valley Power Station Unit No. 1. Also, by letter on the same date the licensee submitted a request for amendment to the Technical Specifications for Cycle 3 and beyond. Since that time, the licensee has amended the February 23 request by letters dated May 5, May 26, June 1, and June 10, 1982. In addition, the staff has had several conversations with the licensee and received additional information submitted by letters dated April 21 and April 26, 1982. This evaluation addresses the proposed Technical Specification changes.

Power Range Neutron Flux - High Positive Rate and High Negative Rate Trip Setpoints

The licensee has requested changes in the Power Range, Neutron Flux, High Positive Rate and High Negative Rate values in Table 2.2-1 Reactor Trip System Instrumentation Trip Setpoints, to the original values. The present values were an interim solution when a potential problem following dropped rod events was first discovered in early 1979. This interim solution was reevaluated later in 1979 and an interim procedural position limiting rod insertion when the plant is above 90% power in automatic control was adopted. Westinghouse has reevaluated the setpoints of the flux rate trip setpoints considering the 1979 interim procedural position and has found that the original setpoints are appropriate.

On this basis, we find the requested change to Table 2.2-1 of the Beaver Valley Technical Specifications acceptable.

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$F_{\Delta H}$ and Partial Power Multiplier

The licensee requested the $F_{\Delta H}$ value of 1.5355 be changed to 1.55. The 1.5355 value included a penalty for rod bow. Rod bow is now accounted for by a burnup dependent rod bow penalty which has been included in the proposed Technical Specification change. We find this change to account for rod bow acceptable. The licensee also requested that the partial power multiplier be changed from 0.2 to 0.3. However, insufficient justication has been submitted by the licensee and we cannot accept the change of value of the partial power multiplier, which thus remains at 0.2. Sufficient justification has been submitted for the $F_{\Delta H}$ value of 1.55 and we find this acceptable.

F_{xy}

A change to the Technical Specifications on F_{xy} was requested to remove the cycle-dependent values of F_{xy} as a function of core height and provide these F_{xy} values by means of a Peaking Factor Report. It is anticipated that F_{xy} will change from cycle to cycle and this Technical Specification change would eliminate the necessity of making a Technical Specification change for each reload. A similar change has been approved for Farley Units 1 and 2. The Radial Peaking Factor Limit Report will be submitted at least 60 days prior to initial criticality for each cycle. This report was submitted on March 9, 1982 for the Beaver Valley Cycle 3, meeting the 60-day requirement. Based on the information submitted by the licensee, we find the proposed change involving F_{xy} acceptable.

The licensee requested that the partial power multiplier be changed from 0.2 to 0.3 for F_{xy} also. We consider this change to be inappropriate and since the licensee has not submitted adequate justification, this change has not been found acceptable.

Control Rod Position Indication System

Background

The characteristics of interest are of two general types. First, the channels have non-linearity in the steady-state response. Second, the channels display a time-dependent (transient) response due to thermal effects in the detector assembly.

A typical steady-state calibration curve is an arc-shaped curve, with the indicated position low at the near full-in and near full-out extremities and the indicated position high in the mid-travel region. For most rods, but not all, the Zero and Span adjustments allow the steady-state calibration curve to be fitted within the $\pm$ 12 steps acceptance band under hot zero-power conditions.

Prior to the start of operating cycle #2, the licensee reported some difficulties in attaining the required performance from the analog system of rod position indication. The NRC approved (Reference 9) certain actions as an acceptable solution for that operating cycle. Those actions were interim; long-term solutions were to be arrived at by the licensee during the 12-18 months duration of cycle #2.

The rod position detector is a linear variable transformer consisting of primary and secondary coils alternately stacked on a stainless steel cylindrical tube. An extension shaft from the rod drive mechanism extends up into the tube and serves as the variable "core" for the transformer. With a constant a.c. current source applied to the primary windings, the position of the rod drive extension shaft changes the primary to secondary coupling and produces a secondary voltage that is directly related to rod position

The transient response for the RPI's is typically of the "over shoot" type. That is, if the rod is being pulled out, the RPI indication will show a greater withdrawal and later settle (at thermal equilibrium) back to the steady-state value; if the rod is being inserted, the initial indication is greater insertion than actual. The magnitude of this thermal transient response appears empirically to be insignificant in the region of the lowest one-third of rod travel. However, near the fully withdrawn positions, this transient response at some plants can be as great as 25 steps. The time constant of the thermal recovery toward the steady-state value varies with rod location radially across the core and has values between 10 and 15 minutes. "Settling Times" (or "soak times") of 20 to 45 minutes have been observed before steady-state is reached.

Evaluation

The February 1982 submittal reported that a core physics analysis had been performed for operating cycle #3 which indicated that a wider rod misalignment tolerance may be adequate (i.e., ± 32 steps vs. ± 24 steps). Staff review of this approach indicated technical difficulties.

The April 1982 submittal provided additional technical information and modified the approach to provide a three-step tolerance, as had been suggested by W. The modified approach proposed a wide instrument error tolerance (± 16 steps) near the full-in position (0-30 steps), a "reference curve" for the intermediate positions, and the standard ± 12 steps tolerance for the 150-228 step range. The staff considered this approach and concluded that it would be awkward and could introduce human factors engineering questions that could outweigh the benefits of the approach.

The May 5, 1982 submittal withdrew both expanded tolerance approaches and proposed certain design and operational changes, within the original ± 12 step instrument error allowance. Our review of this proposal indicated

general acceptability, but three questions remained. The licensee's May 26, 1982 submittal addressed these concerns. The licensee provided additional information on June 1, 1982, and revised Technical Specifications (and bases) on June 10, 1982.

To address the rod position indicators (RPI) concerns, the licensee has proposed a multi-faceted program. Each aspect is described and discussed below.

1. The group demand counters will be the primary indicators for precise rod position information, with the individual analog RPI's displaying general information regarding rod motion, especially during the first hour. We have previously approved, for another W PWR facility, this shift of which rod position indicator is to be considered the primary indicator (see References 11, 12). The bases for the approval are (a) that a potentially misaligned rod that is undetected for an hour or so is not an unacceptable risk, (b) the operating experience over the years with the control rod drive system and the demand counter has indicated a very high reliability, while the experience with the analog indicators has been plagued with less-than-desirable performance, and (c) the demand counters can be checked periodically to confirm that the rods did in fact go to the position requested by the demand counters. We find this approach acceptable for this facility also.

2. The licensee has proposed to develop "custom made" meter scales for the analog indicators that exactly correspond to the individual calibrations of the analog position channels. These custom made scales will, in effect, eliminate most of the steady state errors that are due to the non-linearities of the detector response. We believe this approach is sound and desirable, and should help restore operator reliance in the analog indicators.

3. After the individual rod position channels are calibrated at hot, zero-power conditions, the steady-state calibration shifts slightly as full power is approached. These shifts are predictable and reproducible, and have values of three to six steps (i.e., 1 to 2-1/2%) depending upon the radial position of the rod in the core. The exact values of these shifts must be determined empirically for each rod. We are allowing the licensee to make minor corrections to the channel calibrations after reaching full power to accomodate the steady state shifts due to power ascension. We are requiring, since the calibration adjustments (zero and span) are interdependent, that following any such correction, the channel will be tested at least one intermediate position and at least one low position to confirm that the overall calibration has not been adversely affected by the correction.

4. The deviation between the position indicated by the demand counter and by the individual analog position channels will be periodically

monitored to detect any potentially misaligned rod. When the plant process computer is available, the Rod Position Deviation Monitor will provide continuous and automatic monitoring. Additionally, the deviation will be monitored manually by the reactor operator on at least a daily basis, as a backup cross-check. When the automatic monitor is not available, the formally-required manual surveillance will be increased to a minimum of every four hours. As a matter of practice, the operators will be monitoring the rod positions much more frequently. If this deviation monitoring should indicate that a rod is possibly misaligned, the primary detector voltage of the position channel will be measured immediately (less than 15 minutes) to confirm the validity of the indication. If the primary voltage indicates a rod position that is different from the demand counter by 12 steps or more, the rod will be declared to be misaligned and the appropriate Action Statement of the Technical Specification will be invoked. This level of surveillance on the potential deviation of the rods and planned actions are the maximum practical and provide timely protection for the core physics parameters.

During our review of the licensee's proposed program, we identified 3 additional questions. First, while the custom made meter scales will improve the accuracy of the rod position information presented by these meters, the voltage inputs from the position channels to the process computer which generates the Rod Deviation Alarm will continue to suffer from both steady-state non-linearities and thermal transient responses. This condition is expected to generate frequent spurious control-room alarms, which are quite undesirable. The licensee has determined that the computer has the capacity to use a curve-fitting process in determining rod position from the input voltage from the analog channels. The licensee has therefore agreed to provide a curve fit for each rod of up to as much as a fifth-order equation, as may be necessary to achieve a curve that matches the calibration curve within $\pm$ 2 steps error. The use of curve-fitting eliminates steady state error from causing spurious alarms. Alarms generated by thermal transients are discussed further below. The use of the curve-fitting has previously been reviewed and approved for another W PWR (see Reference 11) and its acceptable for this facility also.

Second, are there methods and procedures to demonstrate the accuracy of the group demand step counters? It has been generally presumed that the accuracy of these counters is probably no worse than $\pm$ 2 steps. The basis for this value is the possibility that the mechanical counter could skip one step as a group is withdrawn and could skip another step as the group is re-inserted, resulting in a possible 2-step error at the end of the rod motion cycle. The licensee and Westinghouse have stated (Reference 13) that there are no physical provisions in the design of the rod drive control system that could be used on a practical basis to prove the 2-step accuracy presumption. However, the licensee has proposed two surveillance actions intended to provide assurance that the counters are operable and within reasonable accuracy. By comparing the two counters within a bank and by comparing the mechanical counters to an electronic counter, differencies due to mechanical action of the counter

can be detected. We have determined that this level of surveillance is the maximum practical available for this system design, and does provide reasonable assurance both that the most likely source of error will be detected promptly and that no gross errors are present.

Third, manual surveillance action of the analog rod position indicators is to take place daily after 1-hour soak time following rod motion, to allow thermal equilibrium to be established. Our question was that, realizing that with the automatic reactor control system in operation the rods are typically re-adjusted a few steps every few minutes, can the manual surveillance actually be performed? Upon reconsidering this point, the licensee has stated that, above 50% power, rod motions are usually small and are, hence, not expected to induce significant thermal transients in the analog rod position channels. Therefore, above 50% the allowance of a 1-hour soak time is not needed. Below 50%, a 6-step limit ("absolute" value) is placed on rod motion during the soak time. This band of 6-steps is ample for typical automatic rod motion and still retains at least quasi-equilibrium conditions at the rod position detectors. Additionally, the small rod motions that typically occur above 50% power are not expected to generate false Rod Position Deviation alarms, and hence spurious alarms due to transient effects should not have a significant impact on the operator. We believe these adjustments in the proposed Technical Specifications will make them more workable, provide reasonable operational flexibility and retain the original safety intent.

After resolving these three questions, we recognized that we may have made the computer-generated Rod Deviation alarm so quiet that it is no longer self-testing and could, in fact, be in a failed-silent condition for a long time. Therefore, periodic functional testing appears necessary. Typically, alarms are tested on a monthly basis. The licensee has proposed to test this alarm on a weekly basis. In consideration of typical plant process computer availability factors across the nation, we believe that weekly testing is appropriate for this feature. The test will be performed by injecting a test signal either on the demand side or on the analog position instrumentation side to demonstrate that the plant computer remains capable of recognizing a deviation of 12 steps or more.

We are continuing to allow that detector primary voltage measurements be used in lieu of an operable analog rod position indicator channel. However, in view of the system improvements that have been made, we are returning to the original limit of 1 channel per bank for which voltage measurements are acceptable. Further, we believe that the number of potential Licensee Event Reports will not be an onerous burden. The licensee has agreed with the reasonableness of this step.

Along the same line, we are requiring that, if the manual surveillance of the analog indicators or the automatic deviation monitor indicates a misaligned rod, a confirmation measurement will be made by measuring the channel primary voltage prior to declaring the rod to be misaligned and invoking Action Statement 3.1.3.1.c.

Our review also encompassed the need for rod position indication during plant-shutdown modes of operation. We are requiring that, in addition to the demand counters, at least two RPI's per rod group be operable. The purpose of rod position information during shutdown modes is to protect the shutdown margin and hence prevent an inadvertent criticality incident. We recognize that with the reactor system substantially subcritical during the shutdown modes (i.e., $K \ll 1.0$), that given the noise level that typically exists on the source range (startup) nuclear instruments, the increase in the subcritical multiplication due to withdrawing a shutdown bank may not be readily discernible. Therefore a bank could be believed to have been withdrawn, given credit for shutdown margin calculation purposes, but not available for shutdown margin if it is not actually withdrawn. Therefore, analog indicators will serve to verify any real rod withdrawal. By requiring that RPI's in each rod group be operable, we assure that both groups in the bank actually received power and did in fact move. By requiring more than one RPI in each group to be operable, we assure that a single malfunctioning instrument will not give the operator incorrect and misleading information. By these requirements, we conclude that the rod position input to shutdown margin calculations will be valid.

The June 10, 1982 submittal provided revised technical specification changes to implement the engineering solutions that had been reached between us and the licensee. The changes include a requirement that the plant computer (i.e., the automatic Rod Deviation Monitor) be operable. In consideration of the availability factors for nuclear power plant process computers, we are granting an exemption to the standard specification 3.0.4 for this requirement. The purpose is to allow the plant to startup when the computer is not available, but anytime the computer is not available to provide its service, additional actions will be provided manually as compensatory measures.

The revised Technical Specifications also define the overall rod position indication system as including the demand counters, the analog position indicators, and the automatic rod position deviation monitor. This definition causes certain requirements to be shifted from specification 3.1.3.1 to specification 3.1.3.2, but results in a more logical arrangement.

The specific Technical Specification page changes covered by this evaluation are included as Attachment "A."

Conclusion

The licensee has proposed to retain the $\pm$ 24 steps potential rod misalignment limit, which is consistent with original licensing basis for this plant. The licensee agrees that the proper accuracy (potential uncertainty) to be associated with the rod position instrumentation is $\pm$ 12 steps and hence remedial action will be taken when the indicated deviation is 12 steps or greater.

The licensee has proposed a comprehensive approach of design changes and operational changes to the rod position indication system. We conclude that this approach will provide a satisfactory method of determining the rod positions within the allocated accuracy of 12 steps. Further, the approach reduces the potential for misleading information and spurious alarms that could generate ambiguity and lead to erroneous actions by the reactor operator. Therefore, we conclude that the licensee's technical proposal and associated Technical Specification changes are acceptable.

Unless operational problems are encountered during this operating cycle, we consider the licensee's proposal to be an acceptable long-term solution for this particular plant.

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.

Date: June 14, 1982

Principal Contributors:

J. T. Beard
D. Hoffman
M. Chatterton

References

1. Letter: J. Carey (Duquesne Light) to S. Varga (NRC), dated February 23, 1982

2. Letter: J. Carey (Duquesne Light) to R. C. Haynes (NRC), dated March 9, 1982

3. Letter: J. Sieber (Duquesne Light) to S. Varga (NRC), dated April 21, 1982

4. Letter: J. Carey (Duquesne Light) to S. Varga (NRC), dated April 26, 1982

5. Letter: J. Carey (Duquesne Light) to S. Varga (NRC), dated May 5, 1982

6. Letter: J. Carey (Duquesne Light) to S. Varga (NRC), dated May 28, 1982

7. Letter: J. Carey (Duquesne Light) to S. Varga (NRC), dated June 1, 1982

8. Letter: J. Carey (Duquesne Light) to S. Varga (NRC), dated June 10, 1982

9. Memo: J. Olshinski (NRC) to S. Varga (NRC), dated November 18, 1980.

11. Memo: G. Lainas (NRC) to T. Novak (NRC), dated April 1, 1982

12. Memo: L. Rubenstein (NRC) to T. Novak (NRC), dated May 20, 1982

13. Telephone Conference: J. Sieber , K. Grada, (Duquesne Light); J. Jenkins (Westinghouse); J. T. Beard, D. Wigginton, P. Tam, D. Hoffman (NRC), May 28, 1982

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. 51 to Facility Operating License No. DPR-66 issued to Duquesne Light Company, Ohio Edison Company, and Pennsylvania Power Company (the licensees), which revised Technical Specifications for operation of the Beaver Valley Power Station, Unit No. 1 (the facility) located in Beaver County, Pennsylvania. The amendment is effective as of the date of issuance.

The amendment modifies the Technical Specifications on rod misalignment, core peaking factor limits and associated parameters.

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 this 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.

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For further details with respect to this action, see (1) the application for amendment dated February 23, 1982 (amended by letters dated April 21, April 26, May 5, May 26, June 1, and June 10, 1982), (2) Amendment No. 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 B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, Pennsylvania 15001. 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 Licensing.

Dated at Bethesda, Maryland, this 14th day of June 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing