Docket File

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

September 25, 1990

Docket No. 50-412
Serial No. BV-90-011

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

Dear Mr. Sieber:

SUBJECT: AMENDMENT NO. 33 TO FACILITY OPERATING LICENSE NPF-73: REMOVAL OF HOT LEG AND COLD LEG RTD BYPASS SYSTEM - CHANGE REQUEST NO. 38 (TAC NO. 76512)

The Commission has issued the enclosed Amendment No. 33 to Facility Operating License No. NPF-73 for the Beaver Valley Power Station, Unit 2, in response to your application dated April 16, 1990 (Change Request No. 38).

The amendment modifies the Appendix A Technical Specifications (TSs) to remove existing requirements on the reactor coolant resistance temperature detector (RTD) manifold instrumentation, and to replace them with requirements for fast-response thermowell-mounted RTDs.

This amendment contains several changes that were not requested in your April 16, 1990, application but which were discussed with Mr. Sovick of your staff during a telephone conversation on August 22, 1990. Your application overlooked reference to the RTD Manifold Instrumentation at two locations (pages 2-7 and 2-9) in Table 2.2-1, "Reactor Trip System Instrumentation Trip Setpoints Notation". It was agreed with Mr. Sovick that the phrase "by RTD Manifold Instrumentation" should be deleted at both locations from the notation for delta-T for clarity since that manifold instrumentation is to be removed. Also discussed with Mr. Sovick were clarifications, associated with the manifold instrumentation deletion, to Bases page B 2-4 for both the Overtemperature delta-T and Overpower delta-T trips.

DF01

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

Sincerely,

Albert W. De Agazio

Albert W. De Agazio, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 33 to NPF-73
2. Safety Evaluation

cc w/enclosures:
See next page

September 25, 1990

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

Sincerely,

original signed by Albert W. De Agazio

Albert W. De Agazio, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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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

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

THE TOLEDO EDISON COMPANY

DOCKET NO. 50-412

BEAVER VALLEY POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 33
License No. NPF-73

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

   A. The application for amendment by Duquesne Light Company, et al. (the licensee) dated April 16, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-73 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 33 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated in the license. DLCO shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance, to be implemented prior to restart from the 1990 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 25, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 33

FACILITY OPERATING LICENSE NO. NPF-73

DOCKET NO. 50-412

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

Remove	Insert
2-4	2-4
2-7	2-7
2-8	2-8
2-9	2-9
2-10	2-10
3/4 2-12	3/4 2-12
3/4 3-6	3/4 3-6
3/4 3-8	3/4 3-8
B 2-4	B 2-4

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlock Setpoints shall be consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

a. With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 2.2-1 adjust the Setpoint consistent with the Trip Setpoint value.

b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Value column of Table 2.2-1, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 2.2-1 and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

EQUATION 2.2-1 $\quad Z + R + S \leq TA$

where:

Z = The value for column Z of Table 2.2-1 for the affected channel,

R = The "as measured" value (in percent span) of rack error for the affected channel,

S = Either the "as measured" value (in percent span) of the sensor error, or the value of Column S (Sensor Error) of Table 2.2-1 for the affected channel, and

TA = The value from Column TA (Total Allowance in % of span) of Table 2.2-1 for the affected channel.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWANCE (TA)	Z	S	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux					
a. High Setpoint	7.5	4.56	0	$\leq$ 109% of RTP*	$\leq$ 111.1% of RTP*
b. Low Setpoint	8.3	4.56	0	$\leq$25% of RTP*	$\leq$27.1% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	1.6	0.50	0	$\leq$ 5% of RTP* with a time constant $\geq$ 2 seconds	$\leq$ 6.3% of RTP* with a time constant $\geq$ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	1.6	0.50	0	$\leq$ 5% of RTP* with a time constant $\geq$ 2 seconds	$\leq$ 6.3% of RTP* with a time constant $\geq$ 2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.41	0	$\leq$ 25% of RTP*	$\leq$ 30.9% of RTP*
6. Source Range, Neutron Flux	17.0	10.01	0	$\leq 10^5$ cps	$\leq 1.4 \times 10^5$ cps
7. Overtemperature ΔT	7.0	5.10	See Note 5	See Note 1	See Note 2
8. Overpower ΔT	4.9	1.71	1.49	See Note 3	See Note 4
9. Pressurizer Pressure-Low	3.1	0.71	1.67	$\geq$ 1945 psig***	$\geq$ 1935 psig***
10. Pressurizer Pressure-High	6.2	4.96	0.67	$\leq$ 2375 psig	$\leq$ 2383 psig
11. Pressurizer Water Level-High	8.0	2.18	1.67	$\leq$ 92% of instrument span	$\leq$ 93.8% of instrument span
12. Loss of Flow	2.5	1.39	0.60	$\geq$ 90% of loop design flow**	$\geq$ 88.8% of loop design flow**

* = RATED THERMAL POWER
** Loop design flow = 88,500 gpm
*** Time constants utilized in the lead-lag controller for Pressurizer Pressure-Low are 2 seconds for lead and 1 second for lag. Channel calibration shall ensure that these time constants are adjusted to those values.

specified in Table 2.2-1, in percent span, from the analysis assumptions. Use of Equation 2.2-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for REPORTABLE EVENTS.

The methodology to derive the trip setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the trip setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Power Range, Neutron Flux

The Power Range, Neutron Flux channel high setpoint provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The low setpoint provides redundant protection in the power range for a power excursion beginning from low power. The trip associated with the low setpoint may be manually bypassed when P-10 is active (two of the four power range channels indicate a power level of above approximately 10 percent of RATED THERMAL POWER) and is automatically reinstated when P-10 becomes inactive (three of the four channels indicate a power level below approximately 10 percent of RATED THERMAL POWER).

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from partial power.

The Power Range Negative Rate trip provides protection to ensure that the minimum DNBR is maintained above 1.30 for control rod drop accidents. At high power a multiple rod drop accident could cause local flux peaking which, when in conjunction with nuclear power being maintained equivalent to turbine power by action of the automatic rod control system, could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate trip for those control rod drop accidents for which DNBRs will be greater than 1.30.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range Channels will initiate a reactor trip at about 10^{+5} counts per second unless manually blocked when P-6 becomes active. The intermediate range channels will initiate a reactor trip at a current level proportional to approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. Although no explicit credit was taken for operation of the Source Range Channels in the accident analyses, operability requirements in the Technical Specifications will ensure that the Source Range Channels are available to mitigate the consequences of an inadvertent control bank withdrawal in MODES 3, 4 and 5.

Overtemperature ΔT

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

Overpower ΔT

The Overpower ΔT reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature ΔT protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for changes in density and heat capacity of water with temperature, and dynamic compensation for transport, thermowell, and RTD response time delays from the core to RTD output indication. The Overpower ΔT trip provides protection to mitigate the consequences of various size steam line breaks as reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Release."

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS
NOTATION

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau^1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \left\{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T\left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K_3 (P - P') - f_1 (\Delta I)\right\}$$

Where: ΔT = Measured ΔT;

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead-lag compensator on measured ΔT;

τ_1 , τ_2 = Time constants utilized in lead-lag compensator for ΔT, $\tau_1 = 8$ s, $\tau_2 = 3$ s;

$\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT;

τ_3 = Time constants utilized in the lag compensator for ΔT, $\tau_3 = 0$ s;

ΔT_o = Indicated ΔT at RATED THERMAL POWER;

K_1 = 1.28;

K_2 = 0.017/°F;

$\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = The function generated by the lead-lag compensator for T_{avg} dynamic compensation;

τ_4, τ_5 = Time constants utilized in lead-lag compensator for T_{avg}, $\tau_4 = 30$ s, $\tau_5 = 4$ s;

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS
NOTATION (Continued)

T	=	Average temperature, °F;
$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0 s;
T'	=	$\leq$ 576.2°F (Nominal T_{avg} at RATED THERMAL POWER);
K_3	=	0.00082;
P	=	Pressurizer Pressure, psig;
P'	=	2235 psig (Nominal RCS operating pressure);
S	=	Laplace transform operator, s^{-1};

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) For $q_t - q_b$ between -33% and +9%, $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) For each percent that the magnitude of $q_t - q_b$ exceeds -33%, the ΔT Trip Setpoint shall be automatically reduced by 2.52% of its value at RATED THERMAL POWER; and

(iii) For each percent that the magnitude $q_t - q_b$ exceeds +9%, the ΔT Trip Setpoint shall be automatically reduced by 1.75% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.6% of ΔT span.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS
NOTATION (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau^1 S)}{(1 + \tau_2 S)} \left(\frac{1}{(1 + \tau_3 S)}\right) \leq \Delta T_o \left\{K_4 - K_5 \left(\frac{\tau_7 S}{(1 + \tau_7 S)}\right) \left(\frac{1}{(1 + \tau_6 S)}\right) T - K_6 \left[T \left(\frac{1}{(1 + \tau_6 S)}\right) - T''\right] - f_2(\Delta i)\right\}$$

Where: ΔT	=	Measured ΔT;
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, τ_1 = 8 s, τ_2 = 3s;
$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constant utilized in the lag compensator for ΔT, τ_3 = 0 s;
ΔT_o	=	Indicated ΔT at RATED THERMAL POWER;
K_4	=	1.0781;
K_5	=	0.02/°F for increasing average temperature and 0 for decreasing average temperature;
$\frac{\tau_7 S}{1 + \tau_7 S}$	=	The function generated by the rate-lag compensator for T_{avg} dynamic compensation;
τ_7	=	Time constant utilized in rate-lag compensator for T_{avg}, τ_7 = 10 s;

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS
NOTATION (Continued)

$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0 s;
K_6	=	0.0012/°F for $T > T''$ and $K_6 = 0$ for $T \leq T''$;
T	=	Average Temperature, °F;
T"	=	Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, ≤576.2°F);
S	=	Laplace transform operator, s^{-1}; and
$f_2(\Delta I)$	=	0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.5% of ΔT span.

NOTE 5: The sensor error for temperature is 1.49% and 0.73% of span for pressure.

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

a. Reactor Coolant System T_{avg}

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

APPLICABILITY: MODE 1*

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

4.2.5.1.2 The provisions of Specification 4.0.3 and 4.0.4 are not applicable for the reactor startups following the initial fueling for Reactor Coolant System total flow rate to allow a calorimetric flow measurement and the calibration of the Reactor Coolant System total flow rate indicators.

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

*The provisions of Specification 3.0.2 are not applicable for the reactor startup following the initial fueling for Reactor Coolant System total flow rate to allow a calorimetric flow measurement and the calibration of the Reactor Coolant System total flow rate indicators.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	3 Loops in Operation
Reactor Coolant System T_{avg}	$\leq$ 580.3°F
Pressurizer Pressure	$\geq$ 2220 psia*
Reactor Coolant System Total Flow Rate	$\geq$ 274,800 gpm**

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

**Includes a 3.5% flow measurement uncertainty.

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the reactor trip system breakers in the closed position and the control rod drive system capable of rod withdrawal.

(1) Trip function may be manually bypassed in this MODE above P-10.

(2) Trip function may be manually bypassed in this MODE above P-6.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels and with the THERMAL POWER level:

a. Less than or equal to 5% of RATED THERMAL POWER, place the inoperable channel in the tripped condition within 1 hour and restore the inoperable channel to OPERABLE status within 24 hours after increasing THERMAL POWER above 5% of RATED THERMAL POWER; otherwise reduce thermal power to less than 5% RATED THERMAL POWER within the following 6 hours.

b. Above 5% of RATED THERMAL POWER, operation may continue provided all of the following conditions are satisfied:

   1. The inoperable channel is placed in the tripped condition within 1 hour.

   2. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.1.1.

   3. Either, THERMAL POWER is restricted to $\leq$75% of RATED THERMAL and the Power Range, Neutron Flux trip setpoint is reduced to $\leq$85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.C.

ACTION 3 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below P-6, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 setpoint.

TABLE 3.3-1 (Continued)

b. Above P-6 but below 5% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 5% of RATED THERMAL POWER.

c. Above 5% of RATED THERMAL POWER, POWER OPERATION may continue.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below P-6, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above P-6 setpoint and suspend positive reactivity operations.

b. Above P-6, operation may continue.

ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor Trip System breakers, suspend all operations involving positive reactivity changes and verify Valve 2CHS-91 is closed and secured in position within the next hour.

ACTION 6 - This Action is not used.

ACTION 7 - With the number of OPERABLE channels* one less than the Total Number of Channels and with the THERMAL POWER level:

a. Less than or equal to 5% of RATED THERMAL POWER, place the inoperable channel in the tripped condition within 1 hour; restore the inoperable channel to operable status within 24 hours after increasing THERMAL POWER above 5% of RATED THERMAL POWER; otherwise reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the following 6 hours.

b. Above 5% of RATED THERMAL POWER, place the inoperable channel in the tripped condition within 1 hour; operation may continue until performance of the next required CHANNEL FUNCTIONAL TEST.

ACTION 8 - With the number of OPERABLE channels one less than the Total Number of Channels and with the THERMAL POWER level above P-9, place the inoperable channel in the tripped condition within 1 hour; operation may continue until performance of the next required CHANNEL FUNCTIONAL TEST.

ACTION 9 - This ACTION is not used

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*An OPERABLE hot leg channel consists of: 1) three RTD's per hot leg, or 2) two RTD's per hot leg with the failed RTD disconnected and the required bias applied.

TABLE 3.3-1 (Continued)

ACTION 10 - This Action is not used.

ACTION 11 - With less than the Minimum Number of Channels OPERABLE, operation may continue provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 12 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours and/or open the reactor trip breakers.

ACTION 39 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

ACTION 40 -

a. With one of the diverse trip features (undervoltage or shunt trip attachment) of a reactor trip breaker inoperable, restore the diverse trip feature to OPERABLE status within 48 hours or declare the breaker inoperable and be in HOT STANDBY within the next 6 hours. Neither breaker shall be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

b. With one reactor trip breaker inoperable as a result of something other than an inoperable diverse trip feature, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 44 - With less than the Minimum Number of channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	NOT APPLICABLE
2. Power Range, Neutron Flux	$\leq$ 0.5 seconds*
3. Power Range, Neutron Flux, High Positive Rate	NOT APPLICABLE
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 0.5 seconds*
5. Intermediate Range, Neutron Flux	NOT APPLICABLE
6. Source Range, Neutron Flux (Below P-10)	NOT APPLICABLE
7. Overtemperature ΔT	$\leq$ 6.0 seconds*
8. Overpower ΔT	$\leq$ 6.0 seconds*
9. Pressurizer Pressure--Low (Above P-7)	$\leq$ 2.0 seconds
10. Pressurizer Pressure--High	$\leq$ 2.0 seconds
11. Pressurizer Water Level--High (Above P-7)	NOT APPLICABLE
12. Loss of Flow - Single Loop (Above P-8)	$\leq$ 1.0 seconds
13. Loss of Flow - Two Loop (Above P-7 and below P-8)	$\leq$ 1.0 seconds
14. Steam Generator Water Level--Low-Low (Loop Stop Valves Open)	$\leq$ 2.0 seconds
15. DELETED.	
16. Undervoltage-Reactor Coolant Pumps (Above P-7)	$\leq$ 1.5 seconds
17. Underfrequency-Reactor Coolant Pumps (Above P-7)	$\leq$ 0.9 seconds

*Neutron detectors are exempt from response time testing. Response time shall be measured from detector output or input of first electronic component in channel.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 33 TO FACILITY OPERATING LICENSE NO. NPF-73

DUQUESNE LIGHT COMPANY

OHIO EDISON COMPANY

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

THE TOLEDO EDISON COMPANY

BEAVER VALLEY POWER STATION, UNIT NO. 2

DOCKET NO. 50-412

1.0 INTRODUCTION

By letter dated April 16, 1990, Duquesne Light Company proposed a revision to the Beaver Valley Unit 2 Appendix A Technical Specifications (TSs) Table 2.2-1, 3.2-1, 3.3-1 and Table 3.3-2. The proposed revision would allow removal of the existing Resistance Temperature Detector (RTD) bypass manifold system and replacement with thermowell-mounted fast-response RTDs located directly in the reactor coolant system piping. This plant modification affects the FSAR response time design basis for overtemperature/overpower Delta T and loss of flow reactor trip functions. The Beaver Valley Unit 2 TSs are also modified to incorporate the revised response times and/or instrument uncertainties into the applicable reactor trip functions. The reactor protection system T average and Delta T input methodology along with the system inputs of T average and Delta T to the plant control system are also revised.

Based on a telephone call with the licensee on August 22, 1990, additional changes were made. These changes were administrative in nature (deletions of references to the system to be removed) to achieve consistency in the TSs. They do not affect the substance of the amendment, nor do they alter the staff's proposed no significant hazards consideration finding.

2.0 DISCUSSION

2.1 Existing System

Currently, the T-hot and T-cold RTDs used for reactor control and protection are inserted into reactor coolant bypass loops. Separate bypass loops are provided for each reactor coolant loop such that individual loop temperature signals may be developed for use in the reactor control and protection systems. A bypass loop from the hot leg side of each steam generator to the intermediate leg is used for the T hot RTDs. Another bypass loop from the cold leg side of the reactor coolant pump to the intermediate leg is used for the T cold RTDs. Both T-hot and T-cold manifolds empty through a common header to the intermediate leg between the steam generator and reactor coolant pump. The RTDs are located within manifolds and are inserted directly into the reactor coolant bypass flow without thermowells. The bypass manifold system limits high velocity coolant flow to the RTDs and compensates for the temperature streaming effects present in the hot leg piping. For each T-hot bypass loop, flow is provided by three scoop tubes located at 120 degree intervals around the hot leg. Because of the mixing effects of the reactor coolant pump only one scoop tube connection is required for bypass flow to the T-cold bypass manifold.

The output from the bypass loop RTDs provides the signals necessary to calculate the average loop temperature (T average) and the loop differential temperature (Delta T). The T average an Delta T signals are then input to the reactor protection system and engineered safety features actuation systems. The input of T average and Delta T signals to the plant control system are derived from a separate set of bypass loop RTDs and T average and Delta T calculations.

2.1 Modified System

The modified system hot leg temperature measurement for each loop will be obtained using three fast-response, narrow-range, single-element RTDs. These RTDs will be mounted in thermowells within the three existing hot leg bypass scoop penetrations. Each bypass scoop will be modified such that reactor coolant will flow in through the existing holes of the bypass scoop past the RTD/thermowell and out through a new hole machined in the bottom of the bypass scoop. This modified RTD arrangement performs the same sampling/temperature averaging function as the bypass manifold system.

The cold-leg measurements are obtained by one fast-response, narrow-range, dual-element RTD located at the discharge of the reactor coolant pump. This RTD is also mounted in a thermowell within the existing cold leg bypass manifold penetration. Because temperature gradients in the cold leg are eliminated by the mixing action of the reactor coolant pumps, only one RTD is necessary for cold-leg temperature measurement. As in the hot leg, the bypass manifold penetration nozzle will be modified to accept the RTD thermowell. Additionally, the bypass manifold return line will be capped at the nozzle on the intermediate leg.

Neither of these modifications will affect the single existing wide-range RTDs installed in each hot and cold leg of the reactor coolant system. The RTDs will continue to provide hot and cold leg temperature information for reactor startup, shutdown or post-accident monitoring.

To accomplish the hot-leg temperature averaging function previously done by the bypass manifold system, the modified hot-leg RTD temperature signals are electronically averaged in the reactor protection system. This averaged T-hot signal will then be used with the T-cold signal to calculate reactor coolant loop Delta T and T average values utilized in the reactor protection and control systems.

The present bypass system uses separate dedicated RTDs for the control and protection systems. However, the modified system thermowell-mounted RTDs are used for both protection and control. This Class 1E and Non Class 1E interface will require the use of isolation devices for the control system T average and Delta T signals derived from the reactor protection system.

Additionally, the T average and Delta T signals used in the control grade logic are input into a median signal selector (MSS) in lieu of the high auctioneered T average or Delta T signal used by the present plant control system. The MSS selects the signal that is between the highest and lowest values of the three T average and Delta T loop inputs. By selecting the median value the MSS provides the plant control system with a valid T average and Delta T value. The MSS will also preserve the functional independence between control and protection systems that now share common sensors within RPS.

To ensure proper operation of the median signal selector the existing manual switches that allow for defeating a T average or Delta T signal from a single loop will be eliminated. Also, the conversion to thermowell-mounted RTDs will result in the elimination of the control grade RTDs and their associated control board indicators. The protection system channels will now provide inputs to the control system through isolators and the MSS. Existing control board alarms, indicators and T average and Delta T deviation alarms will provide the means to identify some RTD failures. A failure of a hot leg RTD can be handled by having plant personnel disconnect the affected channel and rescaling the electronics to average the remaining two hot leg inputs. A bias value is then added to the T-hot average signal to compensate for the failed RTD and maintain a value comparable with the previous three RTD average. This bias value is developed per procedure/TS requirements using data recorded at 100% power and during normal protection system surveillances.

2.3 Evaluation

Based on our review the modified RTD system is not functionally different from the unmodified system except for the use of three RTDs instead of one. The reactor protection or engineered safety features actuation systems will operate as before. The FSAR as documented in section 7 for Beaver Valley Unit 2 remains valid. The additional electronics for averaging the three T-hot RTD signals are to be qualified to the same level as the existing 7300 electronics. The isolation devices are also standard 7300 series equipment and were previously reviewed under WCAP 8892A. The RTD qualification will satisfy the requirement of 10 CFR 50.49. Based on the above, the staff finds the proposed plant modification to replace the RTD bypass manifold system with thermowell mounted, fast response RTDs located directly in the reactor coolant system piping to be acceptable.

To support the modifications required to eliminate the RTD bypass manifold system, changes to the Beaver Valley Unit 2 Technical Specifications were proposed. These TS revisions are a result of the difference in the uncertainties between the thermowell mounted RTDs and the bypass manifold system. Revised values for allowable values of Z, S and TA for Overtemperature Delta T, Overpower Delta-T and Loss of Flow were calculated using essentially the same setpoint methodology as previously approved by the staff. Evaluations performed by the licensee determined that the revised values of allowable values, Z and S remain valid for the proposed bypass system elimination. The remaining TS changes revise Table 3.2-1 (DNB parameters) and reactor coolant T average from 580.2 to 580.3 degrees to reflect revised RCS temperature uncertainties. The staff finds the proposed changes to be acceptable based on our review of the licensee submittal and the Beaver Valley Unit 2 TS.

3.0 ENVIRONMENTAL CONSIDERATION

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

4.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: September 25, 1990

Principal Contributor:

Clifford K. Doutt