September 10, 1997

Mr. J. E. Cross
President-Generation Group
Duquesne Light Company
Post Office Box 4
Shippingport, PA 15077

SUBJECT: BEAVER VALLEY POWER STATION, UNIT NO. 1 (M98136)

Dear Mr. Cross:

The Commission has issued the enclosed Amendment No. 205 to Facility Operating License No. DPR-66 for the Beaver Valley Power Station, Unit No. 1. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated March 10, 1997, which submitted Proposed Operating License Change Request No. 240.

The amendment modifies the TSs by reducing the reactor coolant system specific activity limits in accordance with the NRC's guidance provided in Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes by Outside Diameter Stress Corrosion Cracking." The definition of DOSE EQUIVALENT I-131 is replaced with the Improved Standard TS definition in the first sentence and an equation is added based on dose conversion factors derived from the International Commission on Radiation Protection (ICRP) ICRP-30. TS 3.4.8, Specific Activity, is revised by reducing the DOSE EQUIVALENT I-131 limit from 1.0 μCi/gram to 0.35 μCi/gram for the 48-hour limit and from 60 μCi/gram to 21 μCi/gram for the maximum instantaneous limit. Item 4.a in TS Table 4.4-12, Primary Coolant Specific Activity Sample and Analysis Program, TS Figure 3.4-1, and the Bases for TS 3/4.4.8 are also modified to reflect the reduced DOSE EQUIVALENT I-131 limit.

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

Sincerely,
/S/
Donald S. Brinkman, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-334

Enclosures: 1. Amendment No. 205 to DPR-66
2. Safety Evaluation

cc w/encls: See next page

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OFFICIAL RECORD COPY DOCUMENT NAME: BV98136.AMD

SE dated 8/1/97 was provided and no major changes were made.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 10, 1997

Mr. J. E. Cross
President-Generation Group
Duquesne Light Company
Post Office Box 4
Shippingport, PA 15077

SUBJECT: BEAVER VALLEY POWER STATION, UNIT NO. 1 (TAC NO. M98136)

Dear Mr. Cross:

The Commission has issued the enclosed Amendment No.205 to Facility Operating License No. DPR-66 for the Beaver Valley Power Station, Unit No. 1. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated March 10, 1997, which submitted Proposed Operating License Change Request No. 240.

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Sincerely,

Donald S. Brinkman

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Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-334

Enclosures: 1. Amendment No. 205 to DPR-66
2. Safety Evaluation

cc w/encls: See next page

J. E. Cross
Duquesne Light Company

Beaver Valley Power Station
Units 1 & 2

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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. 205
License No. DPR-66

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

   A. The application for amendment by Duquesne Light Company, et al. (the licensee) dated March 10, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-66 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No.205 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance, to be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 10, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 205

FACILITY OPERATING LICENSE NO. DPR-66

DOCKET NO. 50-334

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

Remove	Insert
1-4	1-4
3/4 4-18	3/4 4-18
3/4 4-20	3/4 4-20
3/4 4-21	3/4 4-21
B 3/4 4-2b	B 3/4 4-2b
---	B 3/4 4-2c
B 3/4 4-3g	B 3/4 4-3g
B 3/4 4-4	B 3/4 4-4
B 3/4 4-5	B 3/4 4-5
B 3/4 7-3	B 3/4 7-3

c. Pressure Boundary LEAKAGE

Pressure Boundary LEAKAGE shall be LEAKAGE (except steam generator tube LEAKAGE) through a nonisolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

1.15 THROUGH 1.17 (DELETED)

QUADRANT POWER TILT RATIO (QPTR)

1.18 QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.

DOSE EQUIVALENT I-131

1.19 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The DOSE EQUIVALENT I-131 is calculated with the following equation:

$$C_{I-131_{D.E.}} = C_{I-131} + \frac{C_{I-132}}{170} + \frac{C_{I-133}}{6} + \frac{C_{I-134}}{1000} + \frac{C_{I-135}}{34}$$

Where "C" is the concentration, in microcuries/gram of the iodine isotopes. This equation is based on dose conversion factors derived from ICRP-30.

STAGGERED TEST BASIS

1.20 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals;

b. The testing of one (1) system, subsystem, train or other designated component at the beginning of each subinterval.

FREQUENCY NOTATION

1.21 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the primary coolant shall be limited to:

a. ≤ 0.35 μCi/gram DOSE EQUIVALENT I-131, and

b. ≤ $100/\overline{E}$ μCi/gram.

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

ACTION:

MODES 1, 2, and 3*

a. With the specific activity of the primary coolant > 0.35 μCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in HOT STANDBY with T_{avg} < 500°F within 6 hours.

b. With the specific activity of the primary coolant > $100/\overline{E}$ μCi/gram, be in HOT STANDBY with T_{avg} < 500°F within 6 hours.

MODES 1, 2, 3, 4 and 5

a. With the specific activity of the primary coolant > 0.35 μCi/gram DOSE EQUIVALENT I-131 or > $100/\overline{E}$ μCi/gram, perform the sampling and analysis requirement of item 4a of Table 4.4-12 until the specific activity of the primary coolant is restored to within its limits.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-12.

* With T_{avg} ≥ 500°F

(next page is 3/4 4-20)

TABLE 4.4-12

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	MINIMUM FREQUENCY	MODES IN WHICH SURVEILLANCE REQUIRED
1. Gross Activity Determination	3 times per 7 days with a maximum time of 72 hours between samples.	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days	1,
3. Radiochemical for $\overline{E}$ Determination	1 per 6 months	1,
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 0.35 µCi/gram DOSE EQUIVALENT I-131 or 100/$\overline{E}$ µCi/gram, and	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 & 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.	1, 2, 3

#Until the specific activity of the primary coolant system is restored within its limits.

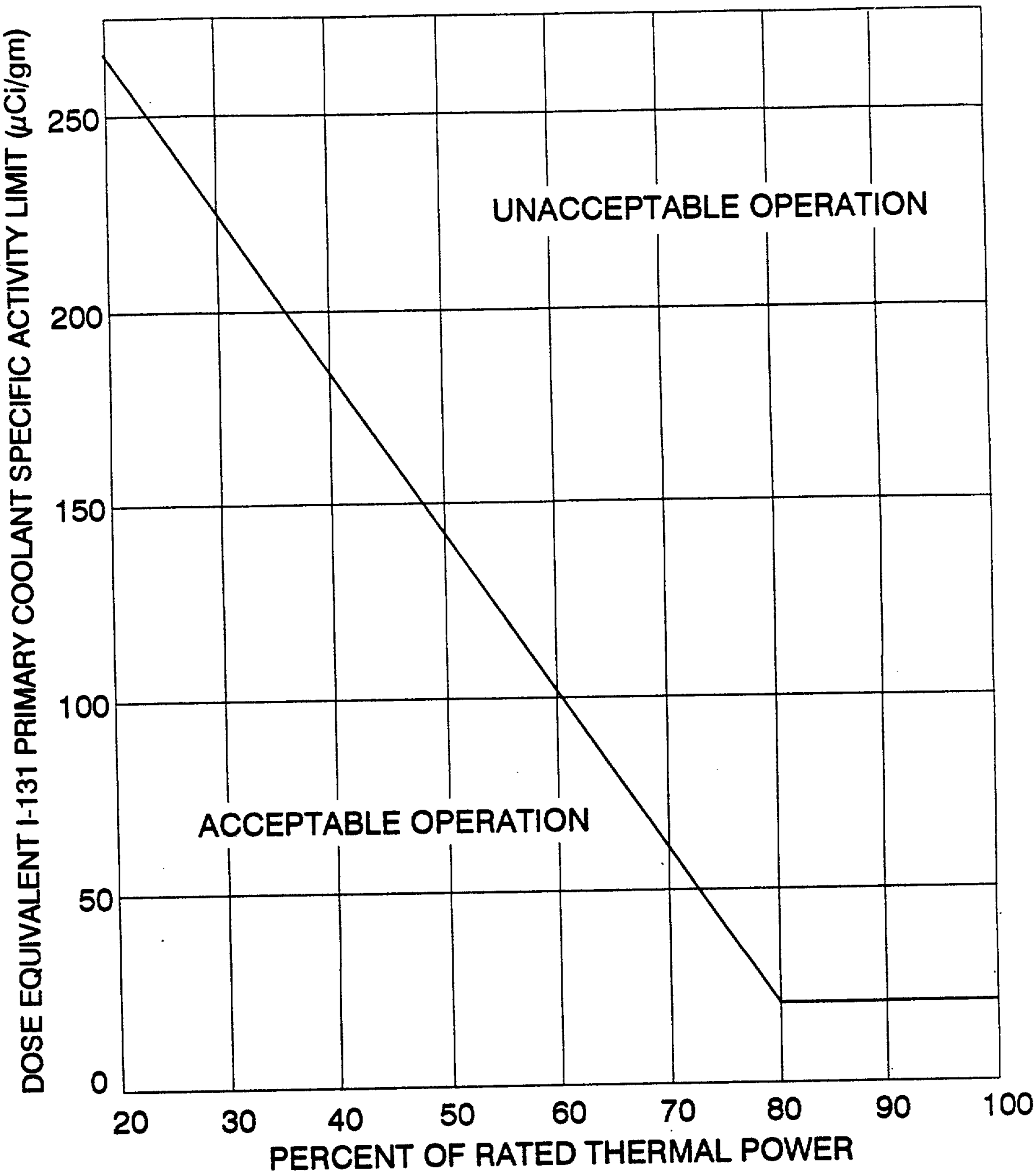

FIGURE 3.4-1

DOSE EQUIVALENT I-131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity > 0.35 μCi/gram Dose Equivalent I-131

3/4.4.5 STEAM GENERATORS (Continued)

correlation and then the subsequent derivation of the voltage repair limit from the structural limit (which is then implemented by this surveillance).

The voltage structural limit is the voltage from the burst pressure/bobbin voltage correlation, at the 95-percent prediction interval curve reduced to account for the lower 95/95-percent tolerance bound for tubing material properties at 650°F (i.e., the 95-percent LTL curve). The voltage structural limit must be adjusted downward to account for potential degradation growth during an operating interval and to account for NDE uncertainty. The upper voltage repair limit; V_{URL}, is determined from the structural voltage limit by applying the following equation:

$$V_{URL} = V_{SL} - V_{Gr} - V_{NDE}$$

where V_{Gr} represents the allowance for degradation growth between inspections and V_{NDE} represents the allowance for potential sources of error in the measurement of the bobbin coil voltage. Further discussion of the assumptions necessary to determine the voltage repair limit are discussed in GL 95-05.

Safety analyses were performed pursuant to Generic Letter 95-05 to determine the maximum MSLB-induced primary-to-secondary leak rate that could occur without offsite doses exceeding a small fraction of 10 CFR 100 (concurrent iodine spike), 10 CFR 100 (pre-accident iodine spike), and without control room doses exceeding GDC-19. The current value of this allowable leak rate and a summary of the analyses are provided in Section 14.2.5 of the UFSAR.

The mid-cycle equation in SR 4.4.5.4.10.d should only be used during unplanned inspections in which eddy current data is acquired for indications at the tube support plates.

SR 4.4.5.5 implements several reporting requirements recommended by GL 95-05 for situations which the NRC wants to be notified prior to returning the SGs to service. For the purposes of this reporting requirement, leakage and conditional burst probability can be calculated based on the as-found voltage distribution rather than the projected end-of-cycle (EOC) voltage distribution (refer to GL 95-05 for more information) when it is not practical to complete these calculations using the projected EOC voltage distributions prior to returning the SGs to service. Note that if leakage and conditional burst probability were calculated using the measured EOC voltage distribution for the purposes of addressing the GL section 6.a.1 and 6.a.3 reporting criteria, then the results of the projected EOC voltage distribution should be provided per the GL section 6.b (c) criteria.

3/4.4.5 STEAM GENERATORS (Continued)

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission pursuant to Specification 6.6 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

LCO (Continued)

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Primary-to-Secondary LEAKAGE through Any One SG

Operating experience at PWR plants has shown that sudden increases in leak rate are often precursors to larger tube failures. Maintaining an operating LEAKAGE limit of 150 gpd per steam generator will minimize the potential for a large LEAKAGE event at power. This operating LEAKAGE limit is more restrictive than the operating LEAKAGE limit in standardized technical specifications. This provides additional margin to accommodate a tube flaw which might grow at a greater than expected rate or unexpectedly extend outside the thickness of the tube support plate. This reduced LEAKAGE limit, in conjunction with a leak rate monitoring program, provides additional assurance that this precursor LEAKAGE will be detected and the plant shut down in a timely manner.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The primary coolant specific activity is limited in order to maintain offsite and control room operator doses associated with postulated accidents within applicable requirements. Specifically, the 0.35 μCi/gm DOSE EQUIVALENT I-131 limit ensures that the offsite dose does not exceed a small fraction of 10 CFR Part 100 guidelines and that control room operator thyroid dose does not exceed GDC-19 in the event of primary-to-secondary leakage induced by a main steam line break.

BASES

3/4.4.8 SPECIFIC ACTIVITY (Continued)

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity > 0.35 μCi/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 0.35 μCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limits shown on Figure 3.4-1 must be restricted to ensure that assumptions made in the UFSAR accident analyses are not exceeded.

Reducing Tavg to < 500°F minimizes the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. This action also reduces the pressure differential across the steam generator tubes reducing the probability and magnitude of main steam line break accident induced primary-to-secondary leakage. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4.1.4 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal-induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that steam releases to the environment will not be significant contributors to radioactivity releases resulting from analyzed accidents. Many of the analyzed accidents assume that a loss of auxiliary AC power occurs, making the main condenser unavailable for plant cooldown, and making it necessary to dump steam to the environment via SG atmospheric dump valves. Maintaining secondary system specific activity within the limits ensures that these releases, in conjunction with other releases associated with the accident, will be within applicable dose criteria.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 205 LICENSE NOS. DPR-66

DUQUESNE LIGHT COMPANY

OHIO EDISON COMPANY

PENNSYLVANIA POWER COMPANY

BEAVER VALLEY POWER STATION, UNIT NO.1

DOCKET NO. 50-334

1.0 INTRODUCTION

By letter dated March 10, 1997, the Duquesne Light Company (the licensee) submitted a request for changes to the Beaver Valley Power Station, Unit No. 1, (BVPS-1) Technical Specifications (TSs). The requested changes would modify the TSs by reducing the reactor coolant system (RCS) specific activity limits in accordance with the Generic Letter (GL) 95-05, ¯Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes by Outside Diameter Stress Corrosion Cracking.Ÿ The definition of DOSE EQUIVALENT I-131 would be replaced with the Improved Standard TS definition wording in the first sentence and an equation would be added based on dose conversion factors derived from the International Commission on Radiation Protection (ICRP) ICRP-30. TS 3.4.8, Specific Activity, would be revised by reducing the DOSE EQUIVALENT I-131 limit from 1.0 μCi/gram to 0.35 μCi/gram for the 48-hour limit and from 60 μCi/gram to 21 μCi/gram for the maximum instantaneous limit. Item 4.a in TS Table 4.4-12, Primary Coolant Specific Activity Sample and Analysis Program, TS Figure 3.4-1, and the Bases for TS 3/4.4.8 would also be modified to reflect the reduced DOSE EQUIVALENT I-131 limit.

2.0 EVALUATION

The licensee performed an assessment of the radiological dose consequences of a main steam line break accident in support of its amendment request to reduce the RCS specific activity limits from 1.0 μCi/gram to 0.35 μCi/gram for the 48-hour limit and from 60 μCi/gram to 21 μCi/gram for the maximum instantaneous limit (in accordance with Generic Letter 95-05). Prior to this amendment request, BVPS-1 had been approved to operate based upon a 4.5 gpm primary to secondary leak initiated by an accident in the faulted steam generator and the TS allowable value for primary to secondary leakage from each intact steam generator of 150 gpd per steam generator. As part of this amendment request, the licensee is proposing to change the faulted steam generator leakage from 4.5 gpm to 11.75 gpm. The licensee is also proposing to change the allowable activity levels of DOSE EQUIVALENT I-131 in the primary coolant from 1.0 μCi/gram to 0.35 μCi/gram for the 48-hour limit and from 60 μCi/gram to 21 μCi/gram for the maximum instantaneous limit. The licensee found the radiological dose consequences acceptable, assuming an allowable activity level of DOSE EQUIVALENT I-131 in the secondary coolant of 0.1 μCi/gram.

The NRC staff has independently calculated the doses resulting from a main steam line break accident using the methodology associated with SRP 15.1.5, Appendix A. The staff performed two separate assessments. One was based upon a pre-existing iodine spike activity level of 21 μCi/gram of DOSE EQUIVALENT I-131 and the other was based upon an accident initiated iodine spike. For the accident initiated spike, the staff assumed that the primary coolant activity level was 0.35 μCi/gram of DOSE EQUIVALENT I-131. The accident initiated an increase in the release rate of iodine from the fuel by a factor of 500 over the release rate to maintain an activity level of 0.35 μCi/gram of DOSE EQUIVALENT I-131 in primary coolant. For these two cases, the staff calculated doses for individuals located at the Exclusion Area Boundary (EAB) and at the Low-Population Zone (LPZ). The control room operator's thyroid dose was also calculated. The parameters which were utilized in the staff's assessment are presented in Table 1. The doses calculated by the staff are presented in Table 2.

The staff's calculations showed that the thyroid doses for the EAB and LPZ would be less than the guidelines established by Standard Review Plan (SRP) 15.1.5, Appendix A of NUREG-0800. The control room operator thyroid dose would be less than the guidelines of SRP 6.4 of NUREG-0800. Therefore, the staff concluded that, based upon an acceptance criterion of 300 rem thyroid at the EAB and LPZ for the pre-existing spike case and an acceptance criterion of 30 rem thyroid dose for the accident initiated spike case and for the control room operator dose assessments, a leak rate of 11.75 gpm is an acceptable limit for the maximum primary to secondary leakage initiated in the faulted steam generator by the main steam line break accident.

3.0 SUMMARY

The acceptance criteria guidelines of SRP 15.1.5, Appendix A are 300 rem thyroid at the EAB and LPZ for the pre-existing spike case and 30 rem thyroid for accident initiated spike case and for the control room operator dose. The NRC staff calculated doses for the proposed TS and Bases changes are within these guidelines; therefore, the proposed TS and Bases changes are acceptable.

TABLE 1
INPUT PARAMETERS FOR BVPS-1 EVALUATION OF MAIN STEAMLINE BREAK ACCIDENT

1. Primary Coolant Concentration of 21 μCi/gram of DOSE EQUIVALENT I-131

Pre-existing Spike Value (μCi/gram)

I-131 =	16.23
I-132 =	5.66
I-133 =	26.09
I-134 =	3.50
I-135 =	13.55

2. Volume of Primary Coolant and Secondary Coolant.

Primary Coolant Volume (ft^3)	9,387
Primary Coolant Temperature (°F)	577.0
Secondary Coolant Steam Volume (ft^3)	3,788
Secondary Coolant Liquid Volume (ft^3)	2,080
Secondary Coolant Steam Temperature (°F)	516.8
Secondary Coolant Feedwater Temperature (°F)	436.9

3. TS Limits for DOSE EQUIVALENT I-131 in the Primary and Secondary Coolant.

Maximum Instantaneous DOSE EQUIVALENT I-131 Concentration (μCi/gram)	21
Primary Coolant DOSE EQUIVALENT I-131 Concentration (μCi/gram)	0.35
Secondary Coolant DOSE EQUIVALENT I-131 Concentration (μCi/gram)	0.1

4. TS Value for the Primary to Secondary Leak Rate.

Primary to secondary leak rate, maximum any SG (gpd)	150
Primary to secondary leak rate, total all SGs (gpd)	450

5. Maximum Primary to Secondary Leak Rate to the Faulted and Intact SGs.

Faulted SG (gpm)	11.75
Intact SGs (gpm/SG)	0.1

6. Iodine Partition Factor

Faulted SG	1.0
Intact SG	0.01

7. Steam Released to the Environment

Faulted SG (0 - 30 minutes)	150,000 lbs
Intact SGs (0 - 2 hours)	375,000 lbs
Intact SGs (2 - 8 hours)	705,393 lbs

8. Letdown Flow Rate 60 gpm

9. Release Rate for 0.35 μCi/gram of DOSE EQUIVALENT I-131

Release Rate (Ci/hr)	500X Release Rate (Ci/hr)
I-131 = 2.79	1395
I-132 = 6.22	3110
I-133 = 6.87	3435
I-134 = 9.16	4580
I-135 = 6.64	3319

10. Atmospheric Dispersion Factors

	sec/m^3
EAB (0-2 hours)	1.30×10^{-3}
LPZ (0-8 hours)	1.30×10^{-4}
Control Room (0-8 hours)	2.42×10^{-3}

11. Control Room Parameters

Filter Efficiency (%)	
Air intake filter	95
Volume (ft^3)	1.73×10^5
Makeup flow (cfm)	690
Recirculation Flow (cfm)	0
Unfiltered Inleakage (cfm)	10
Occupancy Factors	
0-1 day	1.0
1-4 days	0.6
4-30 days	0.4

Table 2 - THYROID DOSES FROM BVPS-1 MAIN STEAM LINE BREAK ACCIDENT

	DOSE	
LOCATION	Pre-Existing Spike (rem)	Accident-Initiated Spike** (rem)
EAB	39.9*	20.9
LPZ	14.5*	22.7
Control Room **	28.6	27.2

* Acceptance Criterion = 300 rem thyroid
** Acceptance Criterion = 30 rem thyroid

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (62 FR 24985). 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.

6.0 CONCLUSION

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

Principal Contributor: C. Hinson

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