MISSISSIPPI POWER & LIGHT COMPANY
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June 16, 1983

NUCLEAR PRODUCTION DEPARTMENT

Mr. Harold R. Denton, Director
U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D. C. 20555

Dear Mr. Denton:

SUBJECT: Grand Gulf Nuclear Station
Unit 1
Docket No. 50-416
License No. NPF-13
File 0260/L-860.0
Control Room Envelope Leak Tightness Test - Attachment 1 to Operating License, Items 1.c and 3
AECM-83/0333

Items 1.c and 3 in Attachment 1 to the facility operating license for the Grand Gulf Nuclear Station (GGNS) (License No. NPF-13, as amended) require:

1.c. Resolve issue regarding maintaining the control room at greater than 1/8 inch water pressure.

3. Prior to exceeding 1 percent power, the licensee shall complete to the satisfaction of the NRC the following open items and construction deficiencies: a) control room envelope leak tightness test.

In the process of satisfying these conditions, Mississippi Power & Light (MP&L) has taken the following steps:

1. Control room inleakage has been substantially reduced from that noted in testing conducted prior to licensing of GGNS.

2. Allowable inleakage has been increased, and an analysis has been performed to demonstrate that doses to an operator following an accident meet the applicable requirements of the Standard Review Plan (SRP), Section 6.4 (NUREG-0800).

The original GGNS licensing bases for control room habitability is provided in FSAR Section 6.4. The methodology for the control room dose assessment is described in FSAR Section 15.6.5. This dose assessment considers a control room infiltration rate of 263 cfm.

The dose assessment was revised using an allowable inleakage rate of 590 cfm and modified meteorological assumptions. The results indicate the control room doses to be within the limits of the SRP. In accordance with the SRP, the infiltration rate used in the analysis should be no less than one-half of the measured leakage from the control room when it is pressurized to 1/8" wg.

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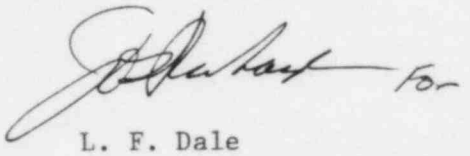
With one exception, the revised analysis uses the same methodology as that described in FSAR Section 15.6.5. Two minutes after the design basis accident, all of the leakage from the primary containment is assumed to be processed through the Standby Gas Treatment System (SGTS) which releases from an elevated vent at elevation 266'-0". This results in reduced dispersion factors for the releases after two minutes since the original FSAR analysis conservatively assumed a ground level release event after the SGTS went into operation.

On March 8, 1983, in test CZ51PT01 Revision 1, the control room was pressurized to 1/8" wg and a leaktightness test successfully completed. Based on the revised analysis (discussed in the justification attached) and the test results, MP&L considers License Conditions 1.c and 3 from Attachment 1 to be resolved.

Yours truly,



L. F. Dale
Manager of Nuclear Services

FGB/SHH/JDR:sap
Attachment

cc: Mr. J. B. Richard (w/o)
Mr. R. B. McGehee (w/o)
Mr. T. B. Conner (w/o)
Mr. G. B. Taylor (w/o)

Mr. Richard C. DeYoung, Director (w/a)
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REVISED CONTROL ROOM HABITABILITY ANALYSIS

The basic model employed for the X/Qs within a building wake at the Grand Gulf control room emergency air intake following an accident is given in FSAR 2.3.4.3:

$$X/Q = \frac{K_c}{A\bar{u}} \qquad (1)$$

Where A = cross-sectional building area perpendicular to the wind direction, m^2

$\bar{u}$ = average wind speed within the wake, m/sec

K_c = nondimensional concentration coefficient

K_c is a function of nondimensional space coordinates (x/L, y/L, and X/L,) building configuration, wind direction, and source configuration. The K_c field for a given building configuration, source configuration, and wind direction is considered to be invariant. Accordingly, K_c values determined by wind tunnel tests with a model structure are expected to be the same as those that would be obtained with a geometrically similar building in the full-scale atmosphere in the same wind direction, with a similar leak. The X/Q values at the control room emergency intake due to radiological releases are calculated based on K_c values extrapolated from wind tunnel tests on geometrically similar building configurations.

The methodology employed for estimating the X/Q values for different time periods, i.e., 0-2, 2-8, 8-24, 24-96 and 96-720 hours, immediately following an accidental release from the containment is described in Section 2.3.4.3 of the FSAR.

The revised analysis uses the same dose assessment methodology as that described in FSAR Section 15.6.5. This methodology considers the same holdup and filtration conditions due to Standby Gas Treatment System processing following a drawdown period of two minutes after the design basis accident. The revised analysis, however, takes credit for the fact that, following drawdown, the releases are through an elevated vent to elevation 266'-0". This assumption produces an increased dispersion of the release.

The differences between the parameters used in the revised analysis and those used in the original FSAR analysis are outlined below:

	FSAR	Revised
1.	Kc value of 3 taken from references 2 & 4	Kc value of 1.5 taken from references 1 & 2
2.	Area of 3723 m^2	Area of 3321 m^2
3.	two years of meteorological data	three years of data

	FSAR	Revised
4.	X/Q percentile distribution based on Ref. 3	distribution calculated based on meteorological data
5.	X/Qs presented in FSAR Table 15.6-12	X/Qs recalculated and are shown in Table 1

Items 1 and 2 change in order to reflect the elevated release. Item 3 is changed in consideration of greater availability of data. The fourth difference is due to a revised method for handling meteorological averaging. The new X/Qs result from the changes in 1-4.

A comparison of the X/Qs used in the FSAR analysis and the revised analysis is presented in Table 1. The association of the specific percentile X/Q values with the averaging time periods is given in Table 2.

RESULTS

Using the dose assessment methodology described in Chapter 15 of the FSAR, the elevated release assumption contributes to lower X/Q values and thus lower control room doses. The revised doses during the 30 day period following a design basis event are presented in Table 3. These doses were calculated considering 590 cfm infiltration into the control room and are below SRP 6.4 limits (NUREG-0800).

REFERENCES

1. Halitsky, J., Golden, J., Halpern, P., (1963): "Wind Tunnel Tests of Gas Diffusion From a Leak in the Shell of a Nuclear Power Reactor and From a Nearby Stack," New York University, Department of Met. & Ocean, GSL Rep. 63-2 under USWB Contract Cwb-10321.

2. Halitsky, J., (1963): "Gas Diffusion Near Buildings," ASHRAE Trans. 69 pp. 464-484.

3. Murphy, K. G., Campe, K. M., (1974): "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19," 13th AEC Air Cleaning Conference.

4. Slade, D. H., Meteorology and Atomic Energy, U. S. Atomic Energy Commission, Office of Information Services, 1968.

TABLE 1

COMPARISON OF X/Q FOR FSAR AND REVISED ANALYSES*

Averaging Time Period	FSAR Analysis (sec/m^3)	Revised Analysis (sec/m^3)
0 - 2 min	1.34×10^{-3}	1.34×10^{-3}
2 min - 8 hr	1.34×10^{-3}	6.74×10^{-4}
8 hr - 24 hr	7.86×10^{-4}	2.04×10^{-4}
24 hr - 96 hr	2.51×10^{-4}	1.10×10^{-4}
96 hr - 720 hr	3.00×10^{-5}	0**

* Corrected for control room occupancies given in FSAR Table 15.6-13.

** In the old method of calculating X/Qs for the time periods after 8 hours, the initial 5 percentile X/Q was reduced by a factor based on wind frequency in a given direction. The new method considers the actual meteorological data base; a X/Q of zero is assigned to the 40 percentile X/Q because the wind does not blow in a direction which can cause control room doses for 40 percent of the year.

TABLE 2

X/Q PERCENTILE RELATIONSHIP TO AVERAGING TIME PERIOD

Averaging Time Period (hrs)	Corresponding Percentile X/Q (%)
0 - 2 min.	5
2 min. - 8	5
8 - 24	10
24 - 96	20
96 - 720	40

TABLE 3

CONTROL ROOM DOSES

Type	Dose (Rem)	Allowable Dose (Rem)
Thyroid*	29.2	30
Whole Body	1.04	5
Beta-Skin	15.5	30

* Limiting Dose