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May 12, 1996

FREEDOM OF INFORMATION
ACT REQUEST
FOIA-96-207
Rec'd 5-16-96

Division of Freedom of Information
and Publication Services
Office of Administration and Resources Management
Nuclear Regulatory Commission
Washington, D.C. 20555

Re: Freedom of Information request

Dear Sir or Madam:

Pursuant to the Freedom of Information Act, I hereby request copies of the following:

-Reports of inspection conducted by the Atomic Energy Commission at General Public Utilities Corp.'s Three Mile Island Unit 1 for the following dates: April 7, 1972, July 11-14, 1972, and March 26-28, 1973. These are not in the Public Document Room.

--Reports from meetings between AEC and GPU personnel on May 1 and May 8, 1973. These are not in the PDR.

--Any other documents in your national or regional files pertaining to Three Mile Island Unit 1 between 1966 and March 27, 1979, which have not yet been placed in the PDR.

--Any correspondence between the AEC and Gilbert Associates, Inc., of Reading, Pennsylvania, pertaining to TMI Unit 1 from the period 1966-March 27, 1979. Gilbert was the architect/engineer of TMI Unit 1.

--Any correspondence between the AEC and United Engineers and Contractors pertaining to TMI Unit 1 during the period 1966-March 27, 1979. UAE was the general contractor for TMI Unit 1.

Pursuant to the Freedom of Information Act, I also request "Representative of the News Media" status, which entitles me to a waiver of search fees and 100 pages of free copying. These documents are needed for a book I am writing on the history of General Public Utilities Corp. I am an established freelance writer, the author of one previous book. *Unseen Danger: A Tragedy of People, Government and the Centralia Mine Fire*, was published in 1986 by University of Pennsylvania Press. In addition, I have been a newspaper journalist for 20 years.

Thank you for your attention to this request.

Sincerely,

David DeKok

David DeKok

THREE MILE ISLAND NUCLEAR STATION UNIT I

REACTOR CONTAINMENT BUILDING

INTEGRATED LEAK RATE TEST

APRIL 1977

METROPOLITAN EDISON COMPANY

SUBSIDIARY OF GENERAL PUBLIC UTILITIES CORPORATION

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1.0 SYNOPSIS

The Three Mile Island Nuclear Station Unit 1 reactor containment building was subjected to a periodic integrated leak rate test during the period from April 16, 1977 to April 19, 1977. The purpose of this test was to demonstrate the acceptability of the building leakage rate at an internal pressure 50.6 psig (P_a). Testing was performed in accordance with the requirements of 10 CFR 50, Appendix J and ANSI N45.4-1972.

The measured leakage rate based on the mass point method of analysis was found to be 0.042 percent by weight per day at 50.6 psig. The leakage rate at the upper bound of the 95 percent confidence interval is 0.052 percent by weight per day which is well below the allowable leakage rate of 0.075 percent by weight per day at 50.6 psig.

The final leakage rate of 0.042 percent by weight per day was obtained after adjustments were made and the test was restarted. The initial building leakage rate indicated was in excess of 0.1 percent by weight per day. The adjustments made consisted of tightening mechanical joints and packings.

Since the industrial cooler system was in operation during the integrated leak rate test, addition of the local leakage rate of the system isolation valves (RB-V2* and RB-V7) to the measured integrated leakage rate must be considered. The combined local leakage rate of both these isolation valves was 0.007 percent by weight per day. The addition of this value increases the total integrated leakage rate to 0.049 percent by weight per day.

The supplemental instrumentation verification at P_a was 1.0 percent, well within the 25 percent requirement of 10 CFR 50, Appendix J, Section III A.3.b.

All testing was performed by Metropolitan Edison Company with the technical assistance of Gilbert Associates, Inc. Procedural and calculational methods were witnessed by Nuclear Regulatory Commission personnel and audited by the Metropolitan Edison Company site Quality Control staff.

2.0 INTRODUCTION

The objective of the periodic integrated leak rate test was the verification of the overall leak tightness of the reactor containment building at the calculated design basis accident pressure of 50.6 psig. The allowable leakage is defined by the design basis accident applied in the safety analysis in accordance with site exposure guidelines specified by 10 CFR 100. For Three Mile Island Nuclear Station Unit 1, the maximum allowable integrated leakage rate at the design basis accident pressure of 50.6 psig (P_a) is 0.10 percent by weight per day (L_a).

Testing was performed in accordance with the procedural requirements as stated in Metropolitan Edison Company Three Mile Island Nuclear Station Unit 1 Surveillance Procedure 1303-6.1. This procedure was recommended for approval by the Three Mile Island Nuclear Station Unit 1 Plant Operations Review Committee and approved by the Unit Superintendent prior to the commencement of the test.

The combined local leakage rates from the reactor containment building isolation valves and penetrations required to be tested by 10 CFR 50, Appendix J, was less than 60 percent of the maximum allowable leakage rate (L_a) at 50.6 psig prior to the commencement of the integrated leak rate test (Refer to Appendix D).

Leakage rate testing was accomplished at the pressure level of 50.6 psig for a period of 24 hours. The 24 hour period was followed by an 8 hour supplemental test for a verification of test instrumentation. During the 32 hour period of testing, the reactor containment building internal temperature was maintained at $72.0 \pm 0.3^{\circ}F$.

3.0 ACCEPTANCE CRITERIA

Acceptance criteria established prior to the test and as specified by 10 CFR 50, Appendix J and ANSI N45.4-1972 are as follows:

a. The measured leakage rate (L_{am}) at the calculated design basis accident pressure of 50.6 psig (P_a) shall be less than 75 percent of the maximum allowable leakage rate (L_a), specified as 0.10 percent by weight of the building atmosphere per day. The acceptance criteria is determined as follows:

$$L_a = 0.10\%/\text{day}$$

$$0.75L_a = 0.075\%/\text{day}$$

b. The test instrumentation shall be verified by means of a supplemental test. Agreement between the containment leakage measured during the Type A test and the containment leakage determined during the supplemental test shall be within 25 percent of L_a.

4.0 TEST INSTRUMENTATION

4.1 SUMMARY OF INSTRUMENTS

The sensor locations were the same as those used for the preoperational ILRT in 1974. Test instruments employed are described, by system, in the following subsections.

4.1.1 Temperature Indicating System

Overall system accuracy: $\pm$ 0.19^{o}F

Overall system repeatability: $\pm$ 0.19^{o}F

Components:

a. Resistance Temperature Detectors

Quantity	24
Manufacturer	Rosemount
Type	Model 104 AAN, 100 ohm, platinum
Range, ^{o}F	60-110
Accuracy, ^{o}F	$\pm$ 0.1
Repeatability, ^{o}F	$\pm$ 0.1

b. Bridge Cards

Quantity	24
Manufacturer	Rosemount
Type	Model 440-L3
Range, ^{o}F	60-110
Accuracy, ^{o}F	$\pm$ 0.25% of span
Repeatability, ^{o}F	$\pm$ 0.25% of span

c. Digital Indicator

Quantity	1
Manufacturer	Weston
Type	Model 1230 *
Range, °F	60-110
Accuracy, °F	± 0.1
Repeatability, °F	± 0.1

* Modified for direct digital temperature readout

4.1.2 Dewpoint Indicating System

Overall system accuracy: ± 1.12°F

Overall system repeatability: ± 0.52°F

Components:

a. Dewcell Elements

Quantity	10
Manufacturer	Foxboro
Type	Model 2711AG, 18 carat gold
Range, °F	0-100
Accuracy, °F	± 1.0
Repeatability, °F	± 0.5

b. Dewpoint Recorder

Quantity	1
Manufacturer	Foxboro
Type	Model Y/ERB12
Range, °F	0-100
Accuracy, °F	± 0.5% of span
Repeatability, °F	± 0.15% of span

4.1.3 Pressure Monitoring System

Overall system accuracy: ± 0.015% of indicated pressure

Overall system repeatability: ± 0.001 psia

Precision Pressure Gauges

Quantity	2
Manufacturer	Texas Instruments
Type	Model 145-01
Range, psia	0-100
Accuracy, psia	± 0.015% of indicated pressure
Repeatability, psia	± 0.001% of full scale

4.1.4 Supplemental Test Flow Monitoring System

Overall system accuracy: ± 1% of full scale

Flow meter

Quantity	1
Manufacturer	Brooks
Type	Model 1114-08
Range, scfh at 0 psig and 100°F	30.9 - 309
Accuracy, scfh	± 1% of full scale

4.2 CALIBRATION CHECKS

Temperature, dewpoint, pressure and flow measuring systems were checked for calibration before the test in accordance with Metropolitan Edison Company Procedure 1430-Y-23, as recommended by ANSI N45.4-1972, Section 6.2 and 6.3. The results of the calibration checks are on file at Three Mile Island Nuclear Station Unit 1. The supplemental test at 50.6 psig confirmed the instrumentation acceptability.

4.3 INSTRUMENTATION PERFORMANCE

Prior to the start of the integrated leak rate test, one dewcell began indicating a dewpoint temperature approximately $30^{\circ}F$ lower than the other 9 dewcells. This dewcell was eliminated from future readings. The remaining 9 dewcells performed well at all times and provided more than adequate coverage of the containment. The temperature, pressure, and flow measuring systems performed well throughout the test.

4.4 SYSTEMATIC ERROR ANALYSIS

Systematic error, in this test, is induced by the operation of the temperature indicating system, dewpoint indicating system and the pressure indicating system.

Justification of instrumentation selection was accomplished, using manufacturer's accuracy and repeatability tolerances stated in Section 4.1, by computing the figure of merit as follows.

The leakage rate, in weight percent per day (%/day), based on an interval of measurement of 24 hour duration is

$$L = 100 \left[1 - \frac{P_{24} T_o}{P_o T_{24}}\right] \text{ \%/day}$$

where:

$P_o = P_{To} - P_{wvo}$, psia - partial pressure of air at start

$P_{24} = P_{T24} - P_{wv24}$, psia = partial pressure of air at finish

T_o = building mean ambient internal temperature at start, $^{\circ}R$

T_{24} = building mean ambient internal temperature at finish, $^{\circ}R$

The change, or uncertainty in L due to uncertainties in the systematic measured variables is given by

$$\delta L = 100 \left[\left(\frac{\partial L}{\partial P_{24}} \tau P_{24}\right)^2 + \left(\frac{\partial L}{\partial P_o} \tau P_o\right)^2 + \left(\frac{\partial L}{\partial T_o} \tau T_o\right)^2 + \left(\frac{\partial L}{\partial T_{24}} \tau T_{24}\right)^2 \right]^{1/2}$$

where τ is the systematic error for each variable. The error in L after differentiation is

$$e_L = 100 \left[\left(\frac{T_o \; e_{P24}}{P_o \; T_{24}}\right)^2 + \left(\frac{P_{24} T_o \; e_{Po}}{P_o^2 \; T_{24}}\right)^2 + \left(\frac{P_{24} \; e_{To}}{P_o \; T_{24}}\right)^2 + \left(\frac{P_{24} T_o \; e_{T24}}{P_o \; T_{24}^2}\right)^2 \right]^{1/2}$$

where:

$e_{Po} = \tau P_o$

$e_{P24} = \tau P_{24}$

$e_{To} = \tau T_o$

$e_{T24} = \tau T_{24}$

Since the values of T_o and T_{24} are essentially the same, within $0.28^\circ F$, and P_o and P_{24} are essentially the same, within 0.002 psia, let $T_o = T_{24}$, $P_o = P_{24}$, $e_{Po} = e_{P24} = e_p$ and $e_{To} = e_{T24} = e_T$. The systematic error in L then reduces to

$$e_L = 141.4 \left[\left(\frac{e_p}{P_o}\right)^2 + \left(\frac{e_T}{T_o}\right)^2 \right]^{1/2} \qquad (1)$$

where the error in pressure (e_p) may be expressed as

$$e_p = (e_{p_a}{}^2 + e_{p_b}{}^2)^{1/2}$$

and

e_{P_a} = error induced by the precision pressure gauges, or

$$e_{P_a} = \pm \frac{(0.00015)(65.340)}{(2)^{1/2}} \text{ psia}$$

$$e_{P_a} = \pm 0.0069 \text{ psia}$$

and

e_{P_b} = error induced by the dewcells, or

$$e_{P_b} = \pm \frac{1.12}{(9)^{1/2}} \ {}^{o}F$$

$$e_{P_b} = \pm 0.373^{o}F$$

From steam tables, at a dewpoint of $65^{o}F$, the pressure equivalent to $\pm 0.373^{o}F$ is

$$e_{P_b} = \pm 0.0039 \text{ psia}$$

Therefore,

$$e_P = [(0.0069)^2 + (0.0039)^2]^{1/2} \text{ psia}$$

$$e_P = \pm 0.0079 \text{ psia}$$

The error in temperature (e_T) may be expressed as

$$e_T = \pm \frac{0.19}{(24)^{1/2}} \ {}^{o}F$$

$$e_T = \pm 0.0388^{o}F$$

Hence, for values at 50.6 psig,

P_o = 65.340 psia

T_o = 531.42^{o}R

and substitution into equation (1) yields

$$e_L = 141.4 \left[\left(\frac{0.0029}{65.340}\right)^2 + \left(\frac{0.0388}{531.42}\right)^2\right]^{\frac{1}{2}}$$

$$e_L = \pm\ 0.020\%/\text{day}$$

The maximum expected systematic error (figure of merit) of the test instrumentation is e_L.

If equation (1) is solved using previously stated repeatability values, the figure of merit is calcuated to be

$$e_L = \pm\ 0.011\%/\text{day}$$

Containment leakage rate computations are a function of changes in temperature and pressure relative to each other, not absolute values. Therefore, the repeatability error analysis is more meaningful.

A conclusion reached from the above calculation was that the instrumentation selected yielded an error value five times less than the allowable leakage rate value of 0.10 percent per day and that the instrumentation combination was of sufficient sensitivity for this test. The e_L values are not based on a statistical analysis of leakage rate calculations and are used strictly for instrumentation selection.

4.5 SUPPLEMENTAL VERIFICATION

In addition to the calibration checks described in Section 4.2, test instrumentation operation was verified by a supplemental test subsequent to the completion of the 24 hour leakage rate test. This test consisted of imposing a known calibrated leakage rate on the reactor containment building. After the flow rate was established, it was not altered for the duration of the test.

During the supplemental test, the measured leakage rate was

$$L_c = L_{v'} + L_o$$

where,

L_c = measured composite leakage rate consisting of the reactor building leakage rate plus the imposed leakage rate

L_o = imposed leakage rate

$L_{v'}$ = leakage rate of the reactor building during the supplemental test phase

Rearranging the above equation,

$$L_{v'} = L_c - L_o$$

The reactor containment building leakage during the supplemental test can be calculated by subtracting the known superimposed leakage rate from the measured composite leakage rate.

The reactor containment building leakage rate during the supplemental test ($L_{v'}$) was then compared to the measured reactor containment building leakage rate during the preceding 24 hour test (L_{am}) to determine instrumentation acceptability. Instrumentation is considered acceptable if the difference between the two building leakage rates is within 25 percent of the maximum allowable leakage rate (L_a).

5.0 TEST PROCEDURE

5.1 PREREQUISITES

Prior to commencement of reactor containment building pressurization, the following basic prerequisites were satisfied:

a. Proper operation of all test instrumentation was verified.

b. All reactor containment building isolation valves were closed using the normal mode of operation. All associated system valves were placed in post-accident positions.

c. Equipment within the reactor containment building, subject to damage, was protected from external differential pressures.

d. Portions of fluid systems which, under post-accident conditions become extensions of the containment boundary, were drained and vented.

e. The penetration pressurization and fluid block systems were depressurized. Gauges were installed at penetration pressurization manifolds to provide means for detection of leakage into the system. These gauges were removed and the manifolds were vented prior to the start of the test.

f. Pressure gauges were installed on closed systems within containment to provide means for detection of leakage into such systems.

g. Local leakage rate testing of containment isolation valves and penetrations was concluded.

h. Potential pressure sources were removed or isolated from the containment.

i. All accessible liner weld channels (approximately 35 percent of the total) were vented to the containment atmosphere.

j. A general inspection of the accessible interior and exterior areas of the containment was completed.

5.2 GENERAL DISCUSSION

Following the satisfaction of the prerequisites stated in Section 5.1,
the reactor containment building pressurization was initiated at a
rate of approximately 2.5 psi per hour. Building internal
temperature was maintained at approximately 72°F. Building pressure
temperature were monitored half hourly and the amperage required
recirculation unit fans (AH-E-1A, 1B and 1C) was monitored
leak rate testing was initiated at the 50.6 psig pressure
-three hours elapsed between reaching the 50.6 psig
nd the recording of official data. For
24 hour leak test and the 8 hour supplemental
nal containment temperature was maintained
by varying the industrial cooler cooling
inment recirculation fan unit cooler
ccurred at half-hour intervals

the two precision gaug
lculated.

b. The twenty-four RTD temperatures were recorded and the average calculated.

c. The nine dewpoint values were recorded. The average of the nine values was converted to vapor pressure using steam tables. This permitted correction of the total pressure to the partial pressure of air by subtracting the vapor pressure.

The use of vapor pressure (P_{wv}), average temperature (T) and the total pressure (P_T) is described in more detail in Section 6.1. All original data is on file at Three Mile Island Nuclear Station Unit 1.

The plot of average temperature and weight of air was performed half hourly (See Appendix B). Atmospheric weather conditions were clear from 1000 on April 16, 1977 to 1830 on April 18, 1977. From 1900 on April 18, 1977 to 1530 on April 19, 1977, the weather conditions were cloudy.

When convenient, the available half-hourly values of P_{wv}, T and P_T were transmitted via on-site portable computer terminal to the Gilbert Associates, Inc. home office for analysis using the CLERCAL computer program. Computer program results, including a least squares fit of the data, were returned to the site via the terminal. A final computer run was made after data for a full 24 hour period was available.

Subsequent to the 24 hour leak test, a superimposed leakage rate as established for an additional 8 hour period. During this time, perature, pressure and vapor pressure were monitored as described e.

h. Potential pressure sources were removed or isolated from the containment.

i. All accessible liner weld channels (approximately 35 percent of the total) were vented to the containment atmosphere.

j. A general inspection of the accessible interior and exterior areas of the containment was completed.

5.2 GENERAL DISCUSSION

Following the satisfaction of the prerequisites stated in Section 5.1, the reactor containment building pressurization was initiated at a rate of approximately 2.5 psi per hour. Building internal temperature was maintained at approximately 72°F. Building pressure and temperature were monitored half hourly and the amperage required by the recirculation unit fans (AH-E-1A, 1B and 1C) was monitored hourly. Leak rate testing was initiated at the 50.6 psig pressure level. Forty-three hours elapsed between reaching the 50.6 psig pressure level and the recording of official data. For the duration of the 24 hour leak test and the 8 hour supplemental test, the average internal containment temperature was maintained within a band of $\pm$ 0.3°F by varying the industrial cooler cooling water flow rate to the containment recirculation fan unit coolers.

During the test the following occurred at half-hour intervals (See Appendix A):

a. Pressures indicated by each of the two precision gauges were recorded and the average calculated.

b. The twenty-four RTD temperatures were recorded and the average calculated.

c. The nine dewpoint values were recorded. The average of the nine values was converted to vapor pressure using steam tables. This permitted correction of the total pressure to the partial pressure of air by subtracting the vapor pressure.

The use of vapor pressure (P_{wv}), average temperature (T) and the total pressure (P_T) is described in more detail in Section 6.1. All original data is on file at Three Mile Island Nuclear Station Unit 1.

The plot of average temperature and weight of air was performed half hourly (See Appendix B). Atmospheric weather conditions were clear from 1000 on April 16, 1977 to 1830 on April 18, 1977. From 1900 on April 18, 1977 to 1530 on April 19, 1977, the weather conditions were cloudy.

When convenient, the available half-hourly values of P_{wv}, T and P_T were transmitted via on-site portable computer terminal to the Gilbert Associates, Inc. home office for analysis using the CLERCAL computer program. Computer program results, including a least squares fit of the data, were returned to the site via the terminal. A final computer run was made after data for a full 24 hour period was available.

Subsequent to the 24 hour leak test, a superimposed leakage rate was established for an additional 8 hour period. During this time, temperature, pressure and vapor pressure were monitored as described above.

5.3 TEST PERFORMANCE

5.3.1 Pressurization Phase

Pressurization of the reactor building containment was started on April 15, 1977 at 0500. The pressurization rate was approximately 2.5 psi per hour. When containment internal pressure reached 12 psig, at 1120 on April 15, 1977, pressurization was secured. An inspection team entered containment to perform the 12 psig inspection. During pressurization to the 12 psig pressure level, the Leak Rate Test System air dryer drain and the cyclone separator drain were not functioning properly. Pressurization was secured while temporary bypasses were installed. While at the 12 psig pressure level, these drains were repaired. The 12 psig internal inspection was completely satisfactorily and pressurization was restarted at 1336 on April 15, 1977.

During pressurization to the 50.6 psig pressure level, the following observations were made:

a. Several penetration pressurization manifold isolation valves were suspected of leaking. The main header was then vented to ensure the penetration pressurization system would remain depressurized.

b. A buildup of pressure on several of the pressure gauges installed on penetration pressurization manifolds indicated a small amount of leakage from the fuel transfer tube flanges, the personnel and emergency airlock door seals, and manifold "J".

c. A small amount of water leakage was noticed from Nuclear Services Closed Cycle Cooling Water valves NS-V4 and NS-V15.

d. One leak rate test dewcell began to indicate a dewpoint temperature approximately 30°F lower than the remaining nine dewcells. This dewcell was eliminated from data collection.

When containment internal pressure reached 50.7 to 50.8 psig, at 0600 on April 16, 1977, pressurization was secured. Temperature was controlled by throttling the industrial cooler pump discharge valve, RB-V18D, which supplies cooling water to the recirculation fan units cooling coils. All penetration pressurization system temporary manifold pressure gauges were removed.

5.3.2 Integrated Leak Rate Testing Phase

After waiting 4 hours, leak rate testing was started. Temperature had stabilized at approximately 72°F. From 1000 on April 16, 1977 until 0500 on April 18, 1977, an excessive leakage rate was indicated by the data collected. The weight of containment air and the average containment temperature versus time for this time period are presented in Appendix B, Exhibits 1 and 2. During this time, the following sequence of events took place:

a. At 1200 on April 16, 1977, the leakage rate, based on two hours of data, was 0.151 percent by weight per day. This established a baseline for the mass point versus time graph. Plant auxiliary operators were sent on routine leak detection. There was no cause for immediate concern since only a limited amount of data had been collected.

b. Subsequent mass points were following approximately the same trend as previously reported. Pressure gauges were installed on manifolds "J", "N" and "O" of the penetration pressurization system for leakage detection. Plant auxiliary operators were again dispatched for leak detection.

c. At 1930 on April 16, 1977, the leakage rate, based on nine and one-half hours of data, was 0.199 percent by weight per day. The pressure gauge on manifold "O" (Fuel Transfer Tube Flanges) was replaced with a flow indicator.

d. The fluid block line to valve IC-V4 was isolated and vent valve FB-V122 was opened. Leakage through this path was evident.

e. The purge valves and the access lock doors were soap-checked and no leakage was indicated. A bonnet/packing leak on penetration pressurization system valve PP-V46 and reducer leaks on the reactor building pressure sensing lines near BS-V37C and BS-V37D were found and repaired.

f. An investigation revealed that several of the automatic fluid block initiation valves, specified to be open, were closed. All automatic fluid block initiation valves were opened.

g. Additional flow indicators were placed on the main steam lines from steam generator A (OTSG A) and steam generator B (OTSG B). At 2400 on April 16, 1977, the following leakages were indicated:

Location	Leakage
Manifold "O"	160 sccm
OTSG A	850 sccm
OTSG B	0 sccm
WDG-V4	700 sccm

h. Since the amount of leakage found was insignificant compared to the leakage indicated by the data (250,000 sccm), leak detection continued. At 0245 on April 17, 1977, the reactor containment building was repressurized to between 50.7 and 50.8 psig.

i. Indicated leakage from OTSG A had increased to 1000 sccm. The fluid block line to IC-V4 was opened and no pressure buildup in manifolds "N" and "O" was observed.

j. As leak detection continued, the measured containment leakage rates were as follows:

Date	Time Interval	Leakage Rate	95% Confidence
4-17	0300-0700	0.120%/day	0.051%/day
4-17	0300-1400	0.126%/day	0.009%/day

k. A valve lineup verification was performed and no deviations were found. A systematic quadrant by quadrant check of penetrations and isolation valves failed to identify any significant leakage. The following adjustments were made on April 17, 1977:

1) Fittings and connections in the leak rate test panel were tightened.

2) Flanges downstream of LR-V2 and LR-V3 were tightened.

Minor amounts of leakage were evident at the following locations:

1) LR-V2 and LR-V3 packing
2) Purge supply interspace.
3) Personnel airlock
4) Purge exhaust interspace
5) OTSG A

l. At 2230 on April 17, pressurization of the secondary side of OTSG A was begun to determine if a change in the indicated reactor containment building leakage could be detected. With the OTSG A at 16 psig, it was decided to depressurize OTSG A since the data prior to 2230 had indicated an upward trend in the mass points. At 0210 on April 18, 1977, OTSG A was depressurized and it was decided to collect and evaluate a full 24 hours of data.

m. At 1030 on April 18, 1977, it was noted that approximately 5 psig pressure had built-up between the seals of the emergency airlock and the personnel airlock. Vents were opened, the pressure was bled off and the vents were left open to allow a leakage path to exist.

n. The containment leakage rate measured from 0300 to 1130 on April 18, 1977 was 0.097 percent by weight per day with an upper bound 95 percent confidence of 0.017 percent by weight per day.

o. At 1320 on April 18, 1977, the concrete shield for the equipment hatch was put in place.

p. The measured containment leakage rate from 0300 to 1600 on April 18, 1977 was 0.103 percent by weight per day with an upper bound 95 percent confidence of 0.009 percent by weight per day.

q. Subsequent to 1600 on April 18, 1977 a shift in the trend of the containment mass points occurred.

r. An acceptable leakage rate of 0.042 percent by weight per day was obtained from 0500 on April 18, 1977 to 0500 on April 19, 1977.

Due to the lack of any local leakage rate determinations prior to the adjustments mentioned in Section 5.3.2.k., the initial unsatisfactory leakage rate indications must be assumed to constitute a failed test.

Nevertheless, since an extensive search failed to identify any significant sources of leakage, it is unlikely that the initial measured leakage rate values, which were in excess of 0.10 percent by weight per day, were true measurements of leakage from the reactor containment building to the outside atmosphere. Two possible explanations for the initial results are:

a. There was leakage into volumes internal to the containment building. The internal volumes may have been (1) the reactor coolant system, since a slow steady decrease in the pressurizer level was noted throughout the test with no corresponding increase in reactor building sump level, and/or (2) the volume between isolation barriers. Additionally, there may have been air entrainment into the concrete and insulation material inside the containment. However, the length of time that the excessive leakage rate was present and the abrupt rather than gradual change in the leakage rate do not tend to support this explanation entirely.

b. The apparent leakage was the result of a diurnal effect. The heating of the containment during the day and the cooling of the containment during the night would cause a change in the containment internal pressure due to the expansion/contraction of the containment without a corresponding detectable change in the containment internal temperature. However, the data, as presented in Appendix B, Exhibits 1 and 2, does not appear to totally support an explanation based on diurnal effects.

5.3.3 Supplemental Leakage Rate Test Phase

After the 24 hour integrated leak rate test data was obtained and evaluated, and the leakage rate found to be acceptable, and a release permit had been obtained, a known leak rate was imposed on the reactor containment building through a calibrated flowmeter for a period of 8 hours.

5.3.4 Depressurization Phase

After all required data was obtained and evaluated, and the supplemental test results were found to be acceptable, and permission from the health physics department and unit superintendent was obtained, depressurization of the reactor containment building was started. A post test inspection of the building revealed no unusual findings.

6.0 METHODS OF ANALYSIS

6.1 GENERAL DISCUSSION

The absolute method of leakage rate determination was employed during testing at the 50.6 psig pressure level. The Gilbert Associates, Inc. CLERCAL computer code calculates the percent per day leakage rate using the mass point method of data analysis. The results presented are based on the mass point method.

The mass point method of computing leakage rates uses the following ideal gas law equation to calculate the weight of air inside containment for each half hour:

$$W = \frac{144\ PV}{RT} = \frac{KP}{T}$$

where,

W = mass of air inside containment, lbm

$K = 144\ V/R = 5.3983 \times 10^6\ \frac{\text{lbm} - {}^{o}\text{R} - \text{in.}^2}{\text{lbf}}$

P = partial pressure of air, psia

T = average internal containment temperature, ^{o}R

$V = 2.0 \times 10^6\ ft^3$

The partial pressure of air, P, is calculated as follows:

$$P = \frac{P_{T1} + P_{T2}}{2} - P_{wv}$$

where,

P_{T1} = true corrected total pressure from PI-390, psia

P_{T2} = true corrected total pressure from PI-391, psia

P_{wv} = partial pressure of water vapor determined by averaging the nine dewpoint temperatures and converting to vapor pressure with the use of steam tables, psia

The average internal containment temperature, T is calculated as follows:

$$T = \frac{\text{sum of 24 RTD's}}{24} + 459.69^{o}R$$

The weight of air is plotted versus time for the 24 hour test and for the 8 hour supplemental test. The Gilbert Associates, Inc. CLERCAL computer code fits the locus of these points to a straight line using a linear least squares fit. The equation of the linear least squares fit line is of the form $W = W_o + W_1 t$ where W_1 is the slope in lbm per hour and W_o is the weight at time zero.

The least squares parameters are calculated as follows:

$$W_o = \frac{\Sigma\, t_i^{\,2} \;\; \Sigma\, W_i - \Sigma\, t_i \;\Sigma\, t_i\, W_i}{S_{xx}}$$

$$W_1 = \frac{N\, \Sigma\, t_i\, W_i - \;\Sigma\, t_i \;\; \Sigma\, W_i}{S_{xx}}$$

where,

$$S_{xx} = N\, \Sigma\, t_i^{\,2} - (\Sigma\, t_i)^2$$

The weight percent leakage per day can then be determined from the following equation:

$$\text{wt. \%/day} = \frac{-2400\, W_1}{W_o}$$

where the negative sign is used since W_1 is a negative slope to express the leakage rate as a positive quantity.

6.2 STATISTICAL EVALUATION

After performing the least squares fit, the CLERCAL computer code calculates the following statistical parameters:

a. Standard error of confidence for the curve fit (S_e).

b. Limits of the 95 percent confidence interval for the curve fit.

c. Limits of the 95 percent confidence interval for the leakage rate (C_L).

The significance of the measured leakage rate can then be evaluated in view of the number of data points exceeding the limits of the 95 percent confidence interval and by the magnitude of the upper bound of the 95 percent confidence interval for the leakage rate.

Standard error of confidence is defined as follows:

$$S_e = \left[\frac{\Sigma \left[W_i - (W_o + W_1 \ t_i)\right]^2}{N-2} \right]^{\frac{1}{2}}$$

where,

W_i = observed mass of air

$(W_o + W_1 \ t_i)$ = least squares calculated mass of air

N = number of data points

This parameter is an expression of the difference between an observed a..d a calculated (least squares) mass point. The 95 percent confidence interval of the fit is twice the st..ndar⁴ error of confidence ($2S_e$). The "degree-of-fit" is evaluated by determining the number of data points, W_i, not falling in the interval $(W_o + W_1 t) \pm 2S_e$.

The 95 percent confidence limit for the mass leakage rate is calculated as follows:

$$C_L = t_{95}\, S_e \left[\frac{N}{S_{xx}} + \frac{S_{xx} + (\Sigma\, t_i)^2}{NS_{xx}}\right]^{1/2}$$

where,

t_{95} = Student's t distribution with N-2 degrees of freedom

This parameter is an expression of the uncertainty in the measured leakage rate.

7.0 DISCUSSION OF RESULTS

7.1 RESULTS AT P_a

Data obtained during the integrated leak rate test at P_a indicated the following maximum changes (highest reading to lowest reading) during the 24 hour test period:

Variable	Maximum Change
P_T	0.026 psia
P_{wv}	0.011 psia
T	0.47°F

The method used in calculating the mass point leakage rate is defined in Section 6.0. The result of this calculation is a mass point leakage rate of 0.042 %/day.

The 95 percent confidence limit associated with this leakage rate is 0.010 percent per day. Thus, the leakage rate at the upper bound of the 95 percent confidence interval becomes

$$L_{am} = 0.042 + 0.010$$

$$L_{am} = 0.052 \text{ \%/day}$$

The measured leakage rate and the measured leakage rate at the upper bound of the 95 percent confidence level are well below the acceptance criteria of 0.075 percent per day (0.75 L_a). A comparison of each of the observed weights with the weights calculated using the least squares line reveals only one of the forty-nine data points does not lie within the 95 percent confidence interval. Therefore, reactor containment building leakage at the calculated design basis accident pressure (P_a) of 50.6 psig is considered to be acceptable.

7.2 SUPPLEMENTAL TEST RESULTS

After conclusion of the 24 hour test at 50.6 psig, flowmeter FI-111 was placed in service and a flow rate of 207 SCFH was established. This flow rate is equivalent to a leakage rate of 0.056 percent per day. After the flow was established, it was not altered for the duration of the supplemental test.

The measured leakage rate (L_c) during the supplemental test was calculated to be 0.099 percent per day using the mass point method of analysis. The 95 percent confidence interval associated with this leakage rate is 0.020 percent per day. None of the 25 data points is out of confidence.

The building leakage rate during the supplemental test is then determined as follows:

$$L_{v'} = L_c - L_o$$

$$L_{v'} = 0.099\%/\text{day} - 0.056\ \%/\text{day}$$

$$L_{v'} = 0.043\%/\text{day}$$

Comparing this leakage rate with the building leakage rate measured during the 24 hour test yields the following:

$$\frac{|L_{am} - L_{v'}|}{L_a} = \frac{|(0.042) - (0.043)|}{0.10} = 0.01$$

The building leakage rates agree within 1.0 percent of L_a which is well below the acceptance criteria of 25 percent of L_a. Therefore, the acceptability of the test instrumentation is considered to have been verified.

8.0 TYPE B AND C LEAKAGE RATE HISTORIES

Refer to Appendicies C, D and E for the report on Type B and C testing performed since the previous Type A test.

9.0 REFERENCES

1. SP 1303-6.1, "Reactor Building Integrated Leak Rate Test", Metropolitan Edison Company Surveillance Procedure.

2. Code of Federal Regulations, Title 10, Part 50, Appendix J, (1-1-75).

3. ANSI N45.4-1972, "Leakage Rate Testing of Containment Structures for Nuclear Reactors", American Nuclear Society, (March 16, 1972).

4. Steam Tables, American Society of Mechanical Engineers, (1967).

5. CLERCAL, Computer Code, Gilbert Associates, Inc.

6. 1430-Y-23, "Reactor Building Integrated Leak Rate Test Instrumemt Calibrations", Metropolitan Edison Company Procedure.

APPENDICES

APPENDIX A

REDUCED LEAKAGE DATA

APPENDIX A

REDUCED TEST DATA

	Time	Average Containment Pressure (psia)	Partial Pressure of Containment Water Vapor (psia)	Partial Pressure of Containment Air (psia)	Average Containment Temperature (°R)	Weight of Containment Air (lbm)
4/18/77	0500	65.340	0.294	65.046	531.42	660,753.87
	0530	65.338	0.293	65.045	531.42	660,743.71
	0600	65.335	0.294	65.041	531.41	660,715.51
	0630	65.336	0.295	65.041	531.43	660,690.65
	0700	65.336	0.293	65.043	531.44	660,698.53
	0730	65.337	0.295	65.042	531.44	660,688.37
	0800	65.338	0.297	65.041	531.49	660,616.06
	0830	65.340	0.294	65.046	531.52	660,629.56
	0900	65.342	0.294	65.048	531.54	660,625.01
	0930	65.344	0.295	65.049	531.57	660,597.88
	1000	65.346	0.294	65.052	531.60	660,591.07
	1030	65.348	0.295	65.053	531.64	660,551.52
	1100	65.350	0.293	65.057	531.68	660,542.44
	1130	65.354	0.294	65.060	531.70	660,548.05
	1200	65.356	0.292	65.064	531.75	660,526.55
	1230	65.354	0.291	65.063	531.76	660,503.97
	1300	65.356	0.294	65.062	531.78	660,468.98
	1330	65.361	0.298	65.063	531.81	660,441.87
	1400	65.358	0.292	65.066	531.81	660,472.33
	1430	65.358	0.292	65.066	531.81	660,472.33
	1500	65.358	0.293	65.065	531.83	660,437.34
	1530	65.357	0.294	65.063	531.84	660,404.62
	1600	65.356	0.291	65.065	531.84	660,424.92
	1630	65.355	0.292	65.063	531.82	660,429.46
	1700	65.353	0.294	65.059	531.81	660,401.27
	1730	65.354	0.293	65.061	531.83	660,396.74
	1800	65.356	0.291	65.065	531.84	660,424.92
	1830	65.358	0.295	65.063	531.87	660,367.37
	1900	65.360	0.292	65.068	531.87	660,418.12

APPENDIX A (Cont'd)

REDUCED TEST DATA

	Time	Average Containment Pressure (psia)	Partial Pressure of Containment Water Vapor (psia)	Partial Pressure of Containment Air (psia)	Average Containment Temperature (°R)	Weight of Containment Air (lbm)
	1930	65.356	0.292	65.064	531.85	660,402.35
	2000	65.359	0.294	65.065	531.86	660,400.09
	2030	65.358	0.294	65.064	531.86	660,389.94
	2100	65.360	0.294	65.066	531.88	660,385.40
	2130	65.361	0.293	65.068	531.88	660,405.70
	2200	65.360	0.290	65.070	531.87	660,438.42
	2230	65.357	0.294	65.063	531.84	660,404.62
	2300	65.353	0.294	65.059	531.82	660,388.85
	2330	65.352	0.294	65.058	531.79	660,415.96
	2410	65.349	0.294	65.055	531.77	660,410.34
4/19/77	0030	65.350	0.293	65.057	531.74	660,467.90
	0100	65.346	0.292	65.054	531.73	660,449.87
	0130	65.348	0.292	65.056	531.73	660,470.17
	0200	65.346	0.291	65.055	531.72	660,472.44
	0230	65.346	0.291	65.055	531.74	660,447.60
	0300	65.346	0.292	65.054	531.73	660,449.87
	0330	65.348	0.294	65.054	531.75	660,425.03
	0400	65.348	0.292	65.056	531.77	660,420.49
	0430	65.344	0.292	65.052	531.71	660,454.40
	0500	65.342	0.287	65.055	531.70	660,497.29
	SUPERIMPOSED TEST					
	0700	65.330	0.294	65.036	531.68	660,329.22
	0800	65.333	0.289	65.044	531.73	660,348.34
	0830	65.334	0.291	65.043	531.76	660,300.94
	0900	65.333	0.292	65.041	531.76	660,280.63
	0930	65.330	0.288	65.042	531.73	660,328.04
	1000	65.328	0.291	65.037	531.73	660,277.28
	1030	65.327	0.291	65.036	531.72	660,279.54

APPENDIX A (Cont'd)

REDUCED TEST DATA

Time	Average Containment Pressure (psia)	Partial Pressure of Containment Water Vapor (psia)	Partial Pressure of Containment Air (psia)	Average Containment Temperature (^oR)	Weight of Containment Air (lbm)
1100	65.327	0.291	65.036	531.73	660,267.13
1130	65.330	0.290	65.040	531.74	660,295.32
1200	65.331	0.292	65.039	531.77	660,247.91
1230	65.327	0.289	65.038	531.77	660,237.76
1300	65.322	0.294	65.028	531.71	660,210.74
1330	65.320	0.293	65.027	531.70	660,213.00
1400	65.324	0.291	65.033	531.77	660,187.01
1430	65.328	0.293	65.035	531.84	660,120.41
1500	65.332	0.290	65.042	531.88	660,141.82
1530	65.332	0.291	65.041	531.90	660,106.84

APPENDIX B

WEIGHT OF CONTAINMENT AIR AND AVERAGE CONTAINMENT TEMPERATURE

TABLE OF CONTENTS

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

Howard
Hernandt

May 26, 1970

R. C. DeYoung, Assistant Director, PWRs
Division of Reactor Licensing
THRU: Charles G. Long, Chief, PWR Project Branch 2, DRL CGL

INITIAL MEETING WITH MET-ED ON THREE MILE ISLAND UNIT 1 POL (DOCKET 50-289)

The initial meeting with the Metropolitan Edison Company representatives concerning the Operating License application for Three Mile Island Unit No. 1 was held May 13, 1970. A list of attendees is attached. We discussed the proposed review schedule, which calls for our review completion in early 1971, and the major review items, as follows:

1. Sodium Thiosulfate
This is the first PWR-OL application for a plant specifically designed to use sodium thiosulfate as an additive; we stated that this would be a major review item. Two additional reports on thiosulfate are forthcoming. Supplemental information concerning stability and compatibility will be added to topical BAW-10017 by B&W, later this month. In addition, supplemental information on iodine removal efficiency will be filed as a Met-Ed amendment, around July 1.

2. Instrumentation
We said that we would use the results of the Oconee review where possible. Regarding prior Met-Ed commitments we noted ACRS comments on separation of control and safety, scram bus separation, failed fuel detector, and dilution system controls. We asked Met-Ed to show how the final design satisfied these points. B&W noted that the common mode failure topical was due in August.

3. Fan Coolers
We were informed that the fan cooler test report would be available in the 3rd quarter of 1970. We observed that the report was somewhat late as compared to earlier commitments by Met-Ed, and compared to actual procurement.

4. Environmental
Met-Ed was informed that some sort of environmental policy statement would be prepared and that an information request would be forthcoming.

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5. Site

a. Several meteorology data questions were brought out. A special meeting was set (for May 19, 1970).

b. Flooding was also considered as a major review item, and a spec al meeting set (for June 16, 1970).

c. Environmental Monitoring - We noted several points that were not considered in their program and said that this would be a significant review item.

6. Radwaste
The radwaste system, designed by GAI, is to be a major review item. Several inconsistencies in liquid radwaste isotopic discharge estimates, TMI #1 POL vs TMI #2 CP were noted.

7. Safety Analysis
We stated our intention to fully review the calculations on steam generator "residual" activity. We also wanted to be assured that the 72-hour cooling period assumed in the refueling accident is intended to be an operational limit.

8. Structures

a. We said all Class I structures, systems, and components would be reviewed, especially in consideration of the aircraft impact requirements.

b. We intend a comparative review on vessel thermal shock, material surveillance, seismic and other loads, and rod drives (all B&W topicals reviewed on Oconee).

9. Miscellaneous
I asked (as an audit or example) for the detailed analysis or calculations, to be discussed at our next technical meeting, on fuel rod swelling with burnup, on pressurizer stresses following a surge line rupture, fuel pool cooler design, HP pump capacity vs one stuck primary safety valve, and operation sequence of steam driven emergency feedwater pump.

Denwood F. Ross
Denwood F. Ross
PWR Project Branch 2, DRL

Enclosure: Attendance List

cc: See page 3

cc:
Docket
DRL Reading
PWR-2 Reading
P. A. Morris
F. Schroeder
T. R. Wilson
R. S. Boyd
R. C. DeYoung
D. Skovholt
E. G. Case, DRS
R. R. Maccary
Compliance (2)
DRL and DRS Branch Chiefs
D. F. Ross (2)
W. E. Nischan
T. M. Novak
F. W. Karas
R. W. Klecker

MET-ED MEETING
May 13, 1970

LIST OF ATTENDEES

Met-Ed
Kathy Matt, Project Administrator
J. L. Bachofer, Jr., Assistant Project Manager
George F. Bierman, Project Manager
G. Charnoff, Consultant Counsel
E. G. Roome, GPU

Gilbert
C. H. Bitting, Project Manager
F. W. Symons

B&W
E. G. Ward, Project Manager
J. M. Cutchin, Licensing

PLA
Keith Woodard

DRL
C. G. Long
D. F. Ross
T. M. Novak
W. E. Nischan
I. Van der Hoven (ESSA)

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

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June 19, 1970

R. C. DeYoung, Assistant Director, PWRs
Division of Reactor Licensing
THRU: Charles G. Long, Chief, PWR Project Branch 2, DRL

HYDROLOGY MEETING ON THREE MILE ISLAND UNIT 1, POL (DOCKET 50-289)

A meeting was held with representatives of Gilbert Associates, the A/E for Met-Ed on Three Mile Island Unit 1. The purpose of the meeting was to discuss the PMF calculations for the site. Attending were R. H. MacLemore and Joel Caves for Gilbert and D. Ross, W. Nischan, and D. Nunn for DRL. MacLemore demonstrated a plot of flood stage along the river for various discharges up to 1.75 million cubic feet per second. He also showed a map where flood contours had been calculated. He discussed the procedures used to calculate level versus discharge with the DRL hydrologist, D. Nunn. As a result of our discussions, we notified Gilbert that we contemplated four broad question areas to be included in our next formal list of questions, along the following lines:

1. We said that the applicant should provide a discharge hydrograph for the PMF, both regulated and unregulated.

2. We asked for the backwater analyses for the 1936 and 1964 floods and for the calculated discharges of 1.1 million and the PMF.

3. We requested a discussion of the procedures for calculation of backwater and the significance of overbank flow. This should include, we said, a table showing the Manning-n coefficients, and the discharge values and elevations at each cross section. We also want the comparison of measured versus computed elevations for the observed floods incorporating the known high water marks, and a map showing the location of the river cross sections used in the computer program. Finally, we want flood level contours.

4. We anticipate a need for a discussion by the applicant of the operating procedures in advance of and during extreme flood events. This should include the information that will be available to the operating staff and the decision levels that the staff must face in terms of river stage or precipitation.

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These comments were given to the Gilbert representatives. We cautioned them to await a formal transmission of these requests before submitting answers. We also notified them of our intention to visit the site next month.

Denwood F. Ross

Denwood F. Ross
PWR Project Branch 2
Division of Reactor Licensing

Distribution:
Docket
DRL Reading
PWR-2 Reading
P. A. Morris
F. Schroeder
T. R. Wilson
R. S. Boyd
R. C. DeYoung
D. Skovholt
E. C. Case, DRS
R. R. Maccary
R. W. Klecker
Compliance (2)
DRL & DRS Branch Chiefs
D. F. Ross
W. E. Nischan
D. Nunn
F. W. Karas

FEB

50-289

Metropolitan Edison Company
P. O. Box 542
Reading, Pennsylvania 19603

Attention: Mr. J. G. Miller, Vice President

Gentlemen:

This letter relates to the discussion Messrs. E. M. Howard and D. M. Hunnicutt of this office held with Mr. T. E. Hreczuch of your staff during the inspection of January 18 and 19, 1971, regarding the construction activities authorized by AEC Construction Permit No. CPPR-40.

As noted during the discussion, apparent deficiencies were identified involving items not in conformance with the Three Mile Island Unit 1 Final Safety Analysis Report or which may otherwise raise questions concerning the adequacy of construction. These items are as follows:

Volume II, Section 5 of the FSAR states in part; "The reactor building has been designed under the following codes: . . . Building Code Requirements for Reinforced Concrete, ACI 318-63 . . . Specification for Structural Concrete for Buildings, ACI 301-63, except as modified in the design and quality control of this building."

Building Code Requirements for Reinforced Concrete, ACI 318-63, paragraph 103, states in part; "(b) When the temperature falls below 40^oF . . . , a complete record of temperature shall be kept.

Paragraph 605 states in part; (a) Concrete shall be maintained above 50^oF and in a moist condition . . ."

Building Code Requirements for Reinforced Concrete, ACI 301-63, paragraph 1202, states in part; "(a) Cold Weather* - When the mean daily temperature of the atmosphere is less than 40^oF, the temperature of the concrete shall be maintained between 50 and 70^oF for the required curing period. When necessary, ar-

OFFICE ▶	CO					
SURNAME ▶	Hunnicutt/jd	Howard	Kirkman			Aly
DATE ▶	2/11/71					

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rangements for heating, covering, insulating, or housing the concrete work shall be made in advance of placement and shall be adequate to maintain the required temperature and moisture conditions without injury due to concentration of heat. *Detailed recommendations are given in 'Recommended Practice for Cold Weather Concreting (ACI 306)'.

ACI Standard 306-66, paragraph 3.1 states in part; Before any concrete is placed, all ice, snow, and frost should be completely removed and the temperature of all surfaces to be in contact with the new concrete should be raised to as close as may be practical to the temperature of the new concrete that is to be placed thereon.

The Three Mile Island Unit 1, Quality Assurance Procedure QC-30, Revision 2, dated February 16, 1970, states in part; If a condition arises wherein the UE&C Field Supervisor-Quality Control determines that project work or major portions thereof must be stopped in order to preserve the quality of the project, he shall so inform the UE&C General Superintendent and the Home Office Quality Control Engineer. . . . In the event that the General Superintendent, from the total project standpoint, does not agree with the recommendations of the Field Supervisor - Quality Control then he may decide to continue work. The Field Supervisor - Quality Control then will report the matter to the UE&C Project Manager and his recommendations to the Manager of Reliability and Quality Assurance in the Home Office for immediate resolution. However, the Met-Ed Project Manager and/or Met-Ed Site Quality Assurance Representative are authorized to initiate additional corrective action including the order to stop work.

Contrary to the above, site records indicate that approximately 230 cubic yards of concrete were poured in a fuel handling building wall (Gilbert Associates, Incorporated, Specification No. SP-5406) from elevation 331 feet to 346 feet, running north and south 17 feet west of the reactor centerline, at a time when three measured concrete surface temperatures were less than 32°F and the ambient temperature was 14°F.

Please provide us, within 30 days, with your comments concerning these items and any steps which have been or will be taken to correct them and to minimize recurrence, including any appropriate changes that have been or will be made to your quality assurance program.

Should you have any questions concerning the matters discussed in this letter, you may communicate directly with this office.

Very truly yours,

Robert W. Kirkman
Director

CO:I:DRH

METROPOLITAN EDISON COMPANY

P. O. Box 542
READING, PENNSYLVANIA 19603

JOHN G. MILLER
Vice President and Chief Engineer

March 8, 1971

U. S. Atomic Energy Commission
Division of Compliance
Region I
970 Broad Street
Newark, New Jersey 07102

Attention: Mr. Robert W. Kirkman, Director

Re: D/C letter dated February 12, 1971
Three Mile Island - Unit No. 1

Dear Sir:

This will acknowledge receipt of your letter of February 12, 1971 concerning the pouring of some concrete in the fuel handling building of TMI #1 not in accordance with the applicable codes.

Your letter suggests that this deficiency was identified by your inspectors during their visit of January 18 and 19, 1971. I wish to point out that this deficiency had been identified on the day of the pour (January 8, 1971) and that corrective action was under discussion and review prior to the visit of your inspectors. Since then the following corrective steps have been taken to (a) determine the acceptability of the concrete that was placed, and (b) prevent a repetition of such an occurrence:

1. Cores are being taken from the concrete joint at appropriate locations and these cores will be tested to determine whether the concrete meets specification strength requirements.

2. The Inspector's Concrete Check-Out Sheet has been modified to require the signature of the UE&C Q/C representative before any concrete placement is allowed to be made.

3. UE&C quality control procedure QC-30 covering work stoppage is being modified to clarify and emphasize that significant deficiencies noted by the UE&C Field Supervisor of Q.C. shall be brought immediately to the attention of the Manager of Reliability and Quality Assurance in the UE&C home office and the UE&C Project Superintendent for corrective action before proceeding with the work. The Met-Ed Project Manager and/or Met-Ed Site Quality Assurance representative will also be notified immediately. Furthermore, the Met-Ed Project Manager has delegated authority to stop work to the appropriate Met-Ed Resident Engineer until corrective action is taken.

Very truly yours,

J. G. Miller

J. G. Miller

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COMMISSION

MAR 1 1 1971

R. C. DeYoung, Assistant Director PWRs
THRU: C. G. Long, Chief, PWR Branch 2 GWL

MEETING WITH METROPOLITAN EDISON ON THREE MILE ISLAND NUCLEAR UNIT NO. 1

We met with representatives of Metropolitan Edison Company and their vendors and consultants on February 23 and 24, 1971 to discuss the Three Mile Island Nuclear Station Unit No. 1. A list of attendees is attached. The purpose of the meeting was to discuss Amendments 15 and 17 which contained the answers to our first question list. We also discussed the items that are at issue between us, and the proposed schedule for our first ACRS meeting. In many cases we informed the applicant of our final or near final positions, which were discussed at a recent Task Force meeting.

SITE

1. Meteorology

We informed Metropolitan Edison that the data made available so far did not substantiate the asserted diffusion values. In fact, we were not able to conclude that the two-hour accident meteorology of Pasquill-F and 1 meter-per-second was justified. On the very limited data made available so far, it may be that a wind speed of 0.5 meter per second is warranted. However, the applicant has installed a new meteorology tower which has Delta T instruments would provide more information on the Pasquill conditions actually present. We agreed to have a meteorology meeting sometime during the week of March 15, and further discuss the data that have been recently generated from the Delta T instrument. At that meeting, we expect to have Dr. Vanderhoven, our consultant from NOAA. Meteorology therefore remains an unresolved issue.

2. Flood

We reviewed the probable maximum flood calculations with the applicant. The representative from Gilbert Associates, Mr. MacLemore, and the DRL staff hydrologist, Dwight Nunn, discussed the issues in detail at a separate sub-meeting. They returned to the main meeting and summarized the additional information that we require, in order to complete our review. In general, we are satisfied with the procedures that Gilbert has used and the required information constitutes a

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written verification of our verbal understanding. The applicant agreed to furnish the required information on the basis of the understandings reached at the meeting. This item therefore can be considered as informally resolved, subject to satisfactory documentation.

REACTOR DESIGN

1. Fuel Design

We requested the applicant's reactor vendor, B&W, to discuss: the fuel design, in particular the high burnup tests that have been performed by B&W; the recently observed fuel pellet abnormalities; and the procedures that B&W uses in calculating fuel swelling and consequent clad strain. Neil Hooker of B&W gave a presentation on this subject. He stated that both the pellet vendor and B&W have QA procedures on the cladding and the fuel pellets. The observations to date show that the fuel pellets have been remaining well within the QA tolerances. He stated that the small amount of chipping and flaking that our reviewers had noticed during a tour of the fuel fabrication plant were only a minor aberration in the pellet design and that the pellet diameters were not becoming excessively large. In regard to the high burnup tests, B&W personnel stated that it was not their intent to prove current designs with the high burnup test; rather they were aimed at advanced designs. They stated that they could not get the same fluxes and enrichments at the B&W test reactor, therefore the commercial design and the high burnup test do not have a one-to-one correspondence. They have not completed their evaluation of the high burnup tests. They do intend to come to DRL with a presentation on this subject when they complete this work, sometime in 1971. At present, they do not plan a formal report.

Hooker stated that B&W calculates clad strain in the same manner that he believes the other vendors do. We asked, and he agreed, and Metropolitan Edison agreed, that the details of how clad strain is calculated be documented in a forthcoming amendment.

2. Burnable Poison Rod Assemblies (BPRA)

Hooker described the design criteria for the burnable poison rod assemblies. They were: (1) the zircaloy tube should be free standing; (2) there should be no clad strain due to diametral growth from thermal or radiation effects on the poison material and (3) there should be no clad strain due to axial thermal or irradiation swelling.

They provide a twelve mil diametral gap between the pellet and the clad. At end-of-life this gap will not be filled. For the axial strain they provide a design margin of 13 inches, using corrugated spacers. They predict only 7-1/2 inches of axial growth. The helium pressure from the B^{10} reaction will be approximately 600 lbs. at end of life. The clad thickness of the zircaloy tube is 32-1/2 mills.

We agree that B&W had properly assessed the safety aspects of the BPRA's and consider this item resolved.

3. Pressurized Fuel

The Metropolitan Edison Unit 1 will use pressurized fuel assemblies. This will be documented in the next amendment.

PRIMARY COOLANT SYSTEM

1. Flywheel Inspection

We asked B&W to summarize the flywheel inspection criteria for the primary coolant pumps. B&W told us that, at the time that increased flywheel inspection was becoming a regulatory requirement, the TMI-1 primary pump motors had already been fabricated, with the flywheels shrunk on. The motors are manufactured by Allis Chalmers. In order to provide inspection to the extent possible, AC performed an inspection on the upper face and outer rim of the upper flywheel on each pump, and took the flywheels completely off one pump. By drilling calibration holes, they determined that they could measure a flaw size approximately 3/4 of a 5/16 inch hole, 1/2 inch deep. Or, they estimate that a flaw in the general size of a 1/4 inch diameter by a half inch deep is detectable. They have computed the critical flaw size for the large flywheel (72-inch diameter); approximately an 8.4-inch radial crack from the bore out is required before critical stresses are reached. We asked that this information be documented and Metropolitan Edison agreed to furnish it. Based on our informal understanding we believe this item to be closed.

In a related discussion Metropolitan Edison informed us that due to problems that had developed with the Bingham pump, which they intended to use on unit 1, they have decided to switch to Westinghouse pumps. The changeover is not as severe as it was on Oconee primary system which was assembled; TMI-1 welding has not started. We told Metropolitan Edison that they should document the change and verify the stress calculations that might be affected by the switch in pump design.

2. Fracture Toughness of Primary System

We discussed with Metropolitan Edison the recent changes that have taken place regarding the determination of fracture toughness in the primary coolant system, including the vessel. We told them that our position on Oconee was, since certain brittle fracture data were not available, that we would use a conservative pressurization temperature. This temperature limit was 275°. Below that temperature the primary system pressure could not exceed 550 psi. Above that temperature the pressure could go to full system design pressure of 2200 psi.

When more information is realized through operation of the plant and from testing of the surveillance specimens, this temperature limit may be lowered. Metropolitan Edison understands our position. We expect Metropolitan Edison to adopt the same general temperature pressure limit in technical specifications.

3. Vibration Monitoring

Metropolitan Edison proposes a confirmatory vibration monitoring system. Neil Hooker of B&W discussed some preliminary vibration monitoring tests that had been performed at the B&W shop in Barberton, Ohio. The Three Mile Island internals, weighing some 300,000 lbs., were instrumented with accelerators and subjected to shaking action by a vibrator and impact action by a rubber mallet. In general, the measurements confirmed some preliminary design calculations and also confirmed the ability of the instrumentation to provide data during hot functional tests. We stated that it was our opinion that they should do either confirmatory vibration monitoring or that they should remove the internals for inspection for undue wear, galling, etc. after the hot functional tests. Note: At a subsequent meeting, we decided that confirmatory vibration monitoring would be sufficient and that the applicant is not required to remove the internals. However, we do intend to urge him and will so state at our technical specification meeting to visually inspect to the extent possible the core internals after the hot functional tests.

4. Feedwater Ring Header

We told Metropolitan Edison that during our Oconee review we required additional inspection of the welds of the primary system in the vicinity of the feedwater ring header on the steam generator, since it could not be established on Oconee that a failure of the primary system would not cause a subsequent failure of the secondary system. However, the Gilbert representative showed us detailed drawings and referred to the explicit design basis in the FSAR whereby restraints are provided on Three Mile Island that were not provided on Oconee. They have designed a primary system such that the piping cannot propagate a failure at the feedwater ring header area. This appears to be sufficient justification for not requiring increased primary system inspection.

5. In Service Inspection

We asked the applicant to describe the extent to which the ASME Section 11 code for inservice inspection could be utilized on the primary system. The applicant noted that there would be some areas that he could not inspect to Section 11 standards, due to access. We asked them to amend the FSAR and to be prepared to incorporate in the technical specification bases the extent they do not comply with Section 11 and why.

6. Decay Heat System Isolation Valve

The B&W design provides two isolation valves between the low pressure decay heat system and the high pressure primary system. Between the two isolation valves there is a small tell-tale relief valve which is sized on the basis of only a minute leakage from the high pressure side. One of the high pressure isolation valves is provided with an interlock to preclude inadvertent operation. Although this design is not strictly in accordance with our proposed new standard on isolation valves we shall accept it, due to the as-built nature of the design.

7. Once Through Steam Generator

We asked B&W if they had completed the vibratory measurements that they were taking on the as-built steam generator. They have completed the test; a supplemental report is in preparation.

STRUCTURES WERE COVERED AT THE SECOND DAY, WEDNESDAY, FEBRUARY 24th AND THEREFORE IS INCLUDED AT THE END OF THIS MEMORANDUM.

ENGINEERED SAFETY FEATURES

1. Thiosulfate

We told Metropolitan Edison in November 1970 we had listed four conditions relevant to the use of thiosulfate and we asked them to what extent they had been considered. Upon request, we reiterated the four items; they were: (a) that a pH monitor should be provided,

(b) that the ability to replenish the sodium hydroxide tank should be provided, (c) that the use of copper and aluminum should be kept at a minimum and (d) that the thiosulfate storage tanks should be monitored frequently. Metropolitan Edison agreed to document in the next amendment items (a) and (b). We said that from their design it appeared they had already complied with item (c), with the comment that they should not subsequently add copper or aluminum objects inside containment. As for item (d), we said that that item would be covered with the technical specifications.

In regard to the removal credit for thiosulfate, we said that our position had not changed. B&W and Metropolitan Edison are aware of how we calculate dose reductions. Gordon Burley described briefly our current model which shows doses slightly above Part 100 for the loss of coolant accident. Bill Nischan pointed out that the exact value was 328 rem at this site boundary. The potential for reduction in the meteorology to half-a-meter-per-second wind speed could double the dose. Burley said that he was considering, and it was under internal review, minor changes in the DRL evaluation model which could bring their dose from slightly over to slightly under Part 100. We notified Metropolitan Edison that we would communicate our final position with regard to the licensing of this plant at our meeting to be held during the week of March 15th.

This item remains unresolved.

2. Emergency Core Cooling System Report

B&W said that they expected to file this report on schedule next week. We said that we would try to have an initial evaluation in approximately six weeks. However, this evaluation would not be timely with respect to the ACRS meeting in May, and therefore a subsequent ACRS meeting on this and perhaps other subjects is foreseen. Since the report is not in hand and since we are presently committed to evaluating the ECCS report before final TMI-1 resolution with the ACRS, this item remains unresolved.

3. Engineered Safety Features Instrumentation

We inquired into the design of the level indicators for the borated water storage tank and core flooding tanks. Contrary to what the FSAR shows, two (not one) level instruments are provided on the borated water storage tank and the two core flooding tanks. Thus redundancy does exist, although it is not clear what independence exists. Don Sullivan asked if IEEE-279 was used as a design basis and B&W said that it was not. We said that we wanted Metropolitan Edison to document this information in the next amendment. We reserved decisions on the adequacy of the redundancy that is provided and the degree to which they don't meet IEEE-279. This item remains unresolved at this time.

4. <u>Spray System Actuation Set Point</u>

We asked why they set the spray system actuation pressure at 30 lbs. They responded that they saw no reason for turning on the spray system for pressures lower than that, and that inadvertent action of the spray would create a housekeeping problem inside containment, to say the least. They noted that some facilities even provide a time delay to preclude inadvertent actuation. We said that this is properly a technical specification discussion, but we thought that advance notice should be given so that B&W would have time for preparation. We suggested 10 psi as a lower value although we admitted that there is very little time difference between 10 and 30 psi for large breaks. This item remains unresolved in that is is a technical specification item and we do expect to resolve it at that time.

5. <u>Hydrogen Purge</u>

We discussed our present plans for facilities such as Three Mile Island Unit 1. Under certain conditions we expect to approve purging of the containment, when the purge dose should be less than 10% of Part 100 guidelines. We told Metropolitan Edison that we had not completed our dose calculations although it appeared that the thyroid purge dose would be within the 30 rem value. We expect to complete, we said, our calculations by next week using an estimated annual-average meteorology. We told Metropolitan Edison that we would communicate our calculations to them about March 15. If we can agree with the applicant that the whole body and thyroid doses at the site boundary due to purging are less than 10% of Part 100 guidelines then we do expect to accept the purging concept.

We have additional requests regarding the purge equipment for which Metropolitan Edison must provide additional information. We told Metropolitan Edison that we wanted them to document: (a) the purge procedure; (b) the meteorology instruments that would be available; (c) the long time utilization of the reactor building fan coolers; (d) the details on the hydrogen monitor qualification tests; (e) the ability to extract a grab sample; and (f) the effects of moisture on the hydrogen monitor. B&W understood the items and agreed that the next amendment would provide these details.

6. Long Term Cooling

We discussed the phenomena that might occur in the vessel following a cold leg break whereby boiling would occur as the principal mode of heat removal. Should this happen, the possibility exists for a long-term buildup of solids in the vessel, and subsequent interference with core cooling. We agreed to discuss by telephone the ground rules for this calculation. It was agreed that B&W would instigate the phone call and that Matt Taylor and D. Ross would be parties to the DRL end of the conversation.

7. ESF Pump Performance

We asked if Metropolitan Edison or Gilbert Associates had received the first four safety guides that had been published. They had. We asked if the engineered safety features pumps conform to the net positive suction head requirements of Safety Guide No. 1. Gilbert's answer was that they assumed a containment pressure to be in equilibrium with the sump water temperature. On inspection of the sump water temperature time relationship, it appeared that for some time after the accident, the sump was up to 220°F. They referred to figure 14-57 of the FSAR showing 220°F sump water at 15 minutes after the accident. Since the use of an equilibrium pressure equal to saturation pressure at this temperature implicitly assumes that the containment pressure is above atmosphere, then we can state that the design of the ESF pumps is not in conformance with the safety guide regarding NPSH. The Gilbert representative stated that should the containment pressure for some reason drop to a lower value than that assumed, then the operator would have to throttle back on the low pressure injection flow or would have to stop one pump. They noted that the low pressure system, nominally rated at 3,000 gpm, could well be delivering in excess of this value due to conservativism in the hydraulic design. We told the applicant that we had not made a final decision on this matter and therefore this is unresolved at this time.

8. Fan Cooler Design

We told Metropolitan Edison that the material submitted as Appendix 6A was generally satisfactory and that we had no further questions on the fan cooler design.

9. Core Flooding Tank Isolation Valve

B&W said that the isolation valve system for Three Mile Island was essentially the same as that provided on Oconee. We asked Metropolitan Edison to provide in the next amendment the following three items:

(a) that there are two independent means of determining valve position; (b) that the condition of not-full-open be alarmed in the control room; and (c) that the power to the motor-operated valve be locked out during normal operation without interrupting power to the valve position indicators or alarm. It appears that this design information can be supplied and this should not be an unresolved item.

10. Zinc

When asked, the applicant said that there was no exposed zinc inside the containment.

INSTRUMENTATION

1. Post-Accident Ranges

We asked how the range of the gamma instruments compared to the doses that might be expected after an accident. A Gilbert representative had some informal information. It appeared that this information is satisfactory, and we asked Metropolitan Edison to document it in the next amendment.

2. Diverse ECCS Signal for Reactor Trip

We told Metropolitan Edison that on Oconee we required a diverse reactor trip following the loss of coolant accident phenomenon and stated that on the Oconee reactor a high building pressure of 4 psig was added as a reactor trip input. The applicant understood the problem and it appears that the same reactor scram method will be provided for Three Mile Island.

3. Failed Fuel Detector

As on other plants a gamma monitor on the letdown line will be used. the sensitivity of the instrument was discussed. The upper limit of sensitivity corresponds to about 10% failed fuel. We asked and Metropolitan Edison had agreed to document the informal information presented at the meeting.

4. Use of Dummy Bistables

We discussed at some length how dummy bistables were used as a bypass apparatus on the TMI design. The specific designers were not present at the meeting, and information was not readily available. Our concern was that use of dummy bistables should be indicated in a manner readily visible to the control room operator. The B&W people and the Metropolitan Edison people were not sure whether that in fact was the case. We agreed that this information could be furnished by telephone and based on the telephone call, we would decide what additional information needed to be filed. B&W will instigate this telephone call and will call Don Sullivan directly.

5. Qualification of Equipment, Topical Report BAW-10003

B&W intends to file this topical report around April 1st. We stated that the report was essential to our review and it appeared that it might be part of a supplemental ACRS report. Sullivan asked the Gilbert people to what extent had tests been run on cables. His question concerned temperature, radiation, humidity. The Gilbert people did not have a ready answer and they will discuss this issue with Sullivan over the phone. At that time we will decide what additional information needs to be filed. In answer to Sullivan's question, the Gilbert people stated that now each diesel has a separate annunciator to indicate the out-of-service condition.

AUXILIARY

1. Fuel Pool Filters

Our soon-to-be-issued Safety Guide states that the fuel pool area should be exhausted through ducts and filters to the unit vent. The Metropolitan Edison design provides for isolating the exhaust system and the supply system and essentially bottling up the fuel pool auxiliary building. Metropolitan Edison feels that due to the airplane impact design that they have designed or provided a fairly leak-tight building. Therefore, they think that the dose to the public would be less if they simply turned off the fans than if they kept the fan running and have a high radiation signal. We had felt that filtration and ventilation should continue even if a high radiation signal existed downstream of the filters. As a result of discussions with the applicant, we are now not so sure. This item remains unresolved on our part.

2. Isotopic Analyses

We discussed our proposed Safety Guide which would require isotopic analyses on the general order of quarterly, and after startup or unusual changes in activity. We noted that this w... a suitable subject for our technical specification meeting. Bill Nischan also had some detailed questions on the Radwaste System. He asked how the applicant would detect and control iodine released through the condenser ejector. They stated that they planned to use Krypton 85 as an indicator.

Their plan is to establish a set-point on the meter on the basis that all of the activity (which should be Krypton 85) is hypothetically Iodine 131. If this set-point is reached during normal operation they will extract a sample and analyze it isotopically to determine the proportion of the activity that is Iodine 131. For example, if the ratio of Krypton 85 to Iodine 131 is 50, then they would readjust the set-point higher by a factor of 50. We asked if they intended to use an iodine monitor on their waste gas tank, in addition to isotopic analyses; they did not. Nischan had an additional question in response to the answer to our question 11.2 concerning concentrations downstream after a liquid release. He noted that there were four errors in a table 11-14 of the FSAR concerning MPC values. He also asked why no Cesium 134 was listed. Metropolitan Edison said that in a subsequent amendment they would correct table 11-14. Nischan noted that Molybdenum 99 was the primary isotope and wondered is there any procedure for further reducing liquid releases by concentrating on the most prolific emitter. There was no answer readily available. In regard to our question and their answer to 11.3 we asked about the use of the Radwaste Treatment equipment. Gilbert Associate representative pointed out that there was no way for high activity release to get to the effluent line without going through both an evaporator and a demineralizer. They provide for redundancy in equipment. Regarding our question 11.5 we noted that Yankee Rowe and Connecticut Yankee had experienced different values of release in that corrosion products constituted the principal items, in contrast to the table in the Metropolitan Edison FSAR where the corrosion products are only a minimum. The Metropolitan Edison people pointed out that the Yankee core is stainless steel and that the Three Mile Island unit postulates a certain failed fuel activity.

3. Fuel Cask

We asked if the fuel pool could withstand the effects of a dropped fuel cask. They said that for the portions of the fuel pool over which a cask might be moved (and there are interlocks on the crane to prevent any other movement) the fuel pool concrete is extended all the way to bedrock. Therefore the pool and its liner could withstand the effects of a dropped cask.

ACCIDENT ANALYSIS

1. LOCA Doses

We noted that in discussing the meteorology and thiosulfate we had already reasonably well defined our position on accident doses. As a review, the two hour thyroid dose following the loss of coolant accident is still above Part 100 Guidelines. If the meteorology gets down to 1/2 meter per second wind speed, then it is not impossible that the fuel pool handling accident would also approach Part 100 Guidelines.

2. ATWS

We notified Metropolitan Edison that the subject of anticipated transients without scram would not be a review item for their operating license.

MISCELLANEOUS

1. Staffing

Metropolitan Edison stated that they were in essential compliance with AMS-3 standard on training and staffing and that they would so document in the technical specifications.

2. Industrial Security

We notified Metropolitan Edison that we required a small amount of additional information on the record concerning industrial security. We referred them to our Oconee Safety Evaluation, page 75, and to the Duke Power Amendment No. 11 for on a guide as to the quantity and type of additional information. They agreed to furnish this information.

3. Technical Specifications

Metropolitan Edison plans to make the first draft available about the first day of May. We told them eight copies would be sufficient and to make them available informally.

4. Restricted Area

We asked Metropolitan Edison to define on a large scale map where the fence would be and what they considered a restricted area to be. They showed that essentially a 8-foot chain link fence topped by barbed wire would follow the dike around the island. No one can get on to the north end of the island without authorization, as there is a guard at the mainland side of the permanent bridge. The south end of the island which is accessible via a "temporary bridge" will be continuously available to the general public. However, the road access to the plant from the south will be barred to casual travelers. There will be double fencing from that side separated by open land area which will be useful for spotting interlopers.

5. <u>Startup Tests</u>

We notified Metropolitan Edison that we required additional information to be submitted with the FSAR concerning startup tests. As a beginning we referred them to what had been made available on Oconee We thought that the depth of the material could be increased in comparison to Oconee. The Oconee acceptance criteria were very short almost to the point of being meaningless. Metropolitan Edison agreed to file some additional information. We concluded the first day meeting and reconvened the following morning to discuss the structural design items.

6. <u>Structures</u>

Structures was category 4 on the agenda; agenda item 1 was the containment design in general. We had asked a number of questions in our September 1970 list. The answers which were made available in January 1971 were not fully acceptable. Most of our questions that we asked on structures revolved around the generally deficient area of their January 1971 response. Don Croneburger of Gilbert discussed their contentions that concrete strength under a biaxial stress condition has a higher ultimate strength value than in uniaxial compression. He referred to a November 1970 article in the journal of the American Concrete Institute proceeding V-67, Page 908. The article of the paper was "Strength of Plain Concrete Under Biaxial Stress". It appeared from that article that for the case where concrete was loaded biaxially in compression that the ultimate strength could be increased by approximately a factor of 2. If true, then the resistence of the structure to an aircraft impact would be considerably increased. Our consultant on aircraft impact design, Jim Proctor of Naval Ordinance Laboratory, was very interested in the utilization of that reference. It was sufficiently recent that it had not been noted by Metropolitan Edison in the January 1971 Amendment.

Mr. Proctor noted a number of deficiencies in the recent response to our question area regarding the calculation of the dynamic load factors, in particular, the utilization of a coarse approximation to the load time curve which does not preserve the momentum of the airplane. He also said that a factor of 20% increase that Gilbert assumed on the ultimate strength of concrete in compression would be difficult to approve, in that it asserted that the strain rate of the concrete was relevant in assessing the proper value of ultimate strength. Mr. Proctor pointed out that, at the time of maximum strain, the strain rate is zero, therefore there would be no attendant increase in concrete properties. He also noted some errors in some of the tables in Appendix 5A. In each case the Gilbert personnel agreed that mistakes have been made and they agreed to correct these values in the next amendment.

In discussion of the November 1970 paper in the ACI Journal, Doctor Gluckman said that what really exists in the dome of the containment is a triaxial field where there are two compression forces and one tensile. He thought that this might reduce the properties of ultimate strength, rather than increase. Mr. Chen Chang, a Gilbert employee, said that there would always be radical compression in the dome and there would not exist a tension field before impact. However, Doctor Gluckman pointed out that in the vicinity of the tendons there would be a tension field and that cracks would have grown. He said that if Gilbert could justify that due to impact there is radial compression, then perhaps we could agree that the ultimate strength properties of concrete could be increased above their nominal value in a multi-axial field. Doctor Gluckman reviewed the difficulties that have been encountered in a Turkey Point dome and said that these in part are responsible for our concern about the response of the Three Mile Island dome to an aircraft impact. We agreed with Metropolitan Edison and Gilbert to have an additional meeting during the week of March 15th on the subject of aircraft impact design. At that time, Gilbert expects to have corrected their tables and have drafts of the additional information which should be available for filing.

Our next question area concerned the calculations of thermal gradients and stresses in the vicinity of variable thickness zones, such as the transition from the base to the wall and in the vicinity of the ring girder. They use the finite element method for the calculation of stresses in the transition region. They had additional information on moments and shears that were not included in Figure 5B-18 of the application. They had calculated the temperature profiles for one-half day, one day, two days, six days, and twenty days after startup. They did not do stress analyses for all conditions. It appeared that we were satisfied with their informal response. We asked if they had considered slightly higher temperatures which would give slightly higher stresses should an event such as happened on Dresden 2 also occur at Three Mile Island.

Our next question area was on the use of $.85f'_c$ as a design basis. They stated that they use ultimate strength value of $.85'_c$ to design to dimensions, and to size the reinforcing. They did not use it in the final design however. We said that we would like to know the maximum compression in the structure and its relation to f'_c. On the next subject of bond and anchorage stresses, Doctor Gluckman said that the recent Los Angeles earthquake showed that rebars had been pulled out of the concrete rather than destroyed. He wanted to know therefore what are the critical bond and anchorage stresses and the relationship to the code. They stated that size 18 bars are provided in the base zone and at the ring girder zone. They are anchored in the wall on the inside face where compression exists.

We asked about the existence of shear stress and, had they been calculated? Their response was that the shear stress was about 24 psi near the ring girder. We asked if the shear stress influenced the allowable ultimate compression stress for the structure. That is, could there be everywhere enough prestressing to handle the slight tensile forces? We noted that they used load factors equal to 1.0 and asked if this was designing for rupture, in other words, would the stresses always be below $.85f'_c$ for the concrete, or .9 of the yield strength for the steel? We asked them to answer this by giving an example of the margin of safety. They agreed to provide the figure.

For the same type of information in the anchorage zone we asked what the safety factors would be. That is, would f'_c be reduced in the presence of tensile stresses? We said that they could answer this by giving an example of the high stress under the bearing plate and the relationship of tensile stresses at that value.

On the subject of reinforcing on the inside the concrete near the liner, we asked what the actual compression forces would be on the concrete. They said that the compression would be approximately 900 psi. Up near the ring girder they do get some tensile forces and they have provided steel there. They also get tensile forces on an aircraft impact.

We brought up the subject of surveillance of the structure and noted that when we discussed this item in the technical specification meetings, we would be discussing such items as the number of tendons, the location, frequency of the test, and how to pull out a sample wire. We said that we did not intend to accept an unstressed wire for surveillance. However, Croneburger of Gilbert said that stress corrosion has been proven not to be a problem. We asked if they had considered the number of tendons that should be inspected. They had used as a beginning the recommendations of the ACI 349 Committee for surveillance of the tendon anchorage zone. The Committee recommendations are 2% of the tendons which for Three Mile would be 13. Metropolitan Edison plans to inspect 15. For lift off tests the

Committee suggests 1/4% which for Metropolitan Edison would be 3. They intend to liftoff 6; they also intend that all 6 of these to be vertical tendons. Their justification for using all vertical tendons in the liftoff test was that the results would not be impeded by friction of the tendon wires in a curved conduit.

We noted that they had not provided the allowable bearing values for the structures adjacent to the bedrock. We asked what did they actually use, and to give the same information for dynamic conditions. We said that we had looked at their material on surveillance of the structure during the proof test. We noted that although they were taking three meridional measurements, we might prefer as many as six, with a smaller number of points per measurement. They said that this seemed to be excessive in comparison with what had been done recently on other prestressed containments.

We brought up the subject of the seismic instrument to be provided and said that we would like information in the FSAR concerning: how the instrument will be maintained; what will be done when the instrument has recorded a signal; and how the signals will be processed and interpreted. They said that the sensitivity of the instrument was .01G, and that they would inspect periodically. A local indicator would signify that a record had been made. At that time, the record would be played back. If the acceleration was greater than 1/2 of the design basis earthquake, that is, if the acceleration was greater than .03G then they would digitize the time history from the record to get a response spectra. This in turn would be compared to the design.

We brought up the fact that recent construction at their facility had used concrete which was poured in nonconformance with the specifications. They said that they were going to check the 28-day compression specimens on the concrete that was poured that day. There were some 200 yards involved where the pour was interfaced at a surface below 32°F. Based on the 28-day specimens, they will decide what to do next.

We next discussed the dynamic analysis of piping. We said that the AEC has not agreed with the Biggs and Roesset method for dynamic analyses. Chen of Gilbert said that it depends on how the Biggs and Roesset method is used. He said that Biggs did not use a single degree of freedom system and referred to the 1965 paper entitled "Earthquake Response of Appendage on a Multi-story Building", by J. Penzien and A. Chopra, given at the third world conference on earthquake engineering in 1965 at New Zealand, Volume 2. Chen discussed what had been calculated by Stone and Webster on their Beaver Valley calculation. Dave Lange of DRS asked how is resonance handled in the Biggs Method. The answer was that a

2-degree of freedom model was used to calculate the response in the resonance region. Lange felt that the time history can be conservative but the superposition of modes which is used in the Biggs method can possibly not be conservative. Lange said that the only way to demonstrate conservatism was to do a multimass time history analysis. Chen said that he does not think that the time history method is fully justified. Lange's response was that the time history envelopes the response spectra for this site. We noted that what DRL has come to refer to as the "Robinson Fix" could be employed at Three Mile Island also. This involves the application of pipe supports at a much more frequent interval. The unresolved items on structural design include the aircraft impact design, some elements of the static design involving the presence of tensile stresses, the dynamic analysis of piping, and certain aspects of the tendon and structure surveillance program.

The last structural subject discussed was the cavity design. Gilbert summarized the final calculations. They said that they had provided in their final design a vent area of 141.6 square feet. This corresponds to blowing out of the insulation around the primary pipe from the cavity to the steam generator area. For a 14.1 square foot pipe break the cavity pressure goes to 186 psi. At that point, they have an estimated 45,000 psi in their rebar of the outer fibers of the cavity. They had specified in procurement that the rebar should be at yield at 40,000 psi, however, the as-bought materials were somewhat higher. They do not expect therefore any significant deformation of the cavity. They do expect cracking, but no propagation of concrete missiles. The pipe tunnel for the primary piping is lined with a steel liner which served as a form for construction. We stated that that information was satisfactory and that no additional information on the cavity would be required.

D. F. Ross
Reactor Projects Branch 2
Division of Reactor Licensing

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D. Nunn
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J. Knight
D. Lange
D. Sullivan

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