MISSISSIPPI P⌐ VER & LIGHT COMPANY
He' r Build Mississippi
P. O. BOX 16⁴ JACKSON, MISSISSIPPI 39205

September 3, 1981

NUCLEAR PRODUCTION DEPARTMENT

U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D. C. 20555

Attention: Mr. Harold R. Denton, Director

Dear Mr. Denton:

SUBJECT: Grand Gulf Nuclear Station
Units 1 and 2
Docket Nos. 50-416 and 50-417
File 0260/0862/L-340.0/L-334.0
Transmittal of Proposed FSAR Changes and Responses to NRC Formal and Inf..mal Questions
AECM-81/345

In response to formal and informal requests for additional information, Mississippi Power & Light Company is submitting the enclosed information for your review. Be advised that Attachment 1, Question and Response to 260.1, and Attachment 5 on soil-structure interaction supersede our previous responses which were submitted via AECM-81/329 dated August 28, 1981, and AECM-81/333 dated August 28, 1981, respectively.

Proposed FSAR changes mentioned or provided in the attached will be incorporated into a forthcoming amendment to the FSAR. If you have any questions or require further information, please contact this office.

Yours truly,

John D Richardson for

L. F. Dale
Manager of Nuclear Services

DWF/JDR:cm

Attachments:
1. Question and Response 260.1
2. Revised Question and Response 362.7
3. HGEB SER Open Item - Ultimate Heat Sink
4. NRC Concern - Shared Systems
5. SEB SER Open Item - Soil-Structure Interaction

B001
1/1

cc: (See Next Page)

cc: Mr. N. '. Stampley
Mr. G. B. Taylor
Mr. R. B. McGehee
Mr. T. B. Conner
Mr. R. Scotti (ORI, Inc.)

Mr. Victor Stello, Jr., Director
Office of Inspection and Enforcement
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

SER Open Item - Q-List Review -(QAB)

260.1 (3.2) Section 17.1.2.2 of the standard format (Regulatory Guide 1.70) requires the identification of safety-related structures, systems, and components controlled by the QA program. You are requested to supplement and clarify Table 3.2-1 of the Grand Gulf FSAR in accordance with the following:

a. The following items do not appear on FSAR Table 3.2-1. Add the appropriate items to the table and provide a commitment that the remaining items are subject to the pertinent requirements of the FSAR operational quality assurance program or justify not doing so.

1. Safety-related masonry walls (see IE Bulletin No. 80-11).
2. Biological shielding within the containment structure, auxiliary building, and control building.
3. Missile barriers within the containment structure, auxiliary building, control building, diesel-generator building, and enclosure building.
4. Spent fuel pool and liner.
5. Radiation monitoring (fixed and portable).
6. Radioactivity monitoring (fixed and portable).
7. Radioactivity sampling (air, surfaces, liquids).
8. Radioactive contamination measurement and analysis.
9. Personnel monitoring internal (e.g., whole body counter) and external (e.g., TLD system).
10. Instrument storage, calibration, and maintenance.
11. Decontamination (facilities, personnel, and equipment).
12. Respiratory protection, including testing.
13. Contamination control.
14. Radiation shielding (permanently installed).
15. Ventilation exhaust radiation monitor for the fuel storage area.
16. Accident-related meteorological data collection equipment.
17. Expendable and consumable items necessary for the functional performance of safety-related structures, systems, and components (i.e., weld rod, fuel oil, boric acid, snubber oil, etc.
18. Drywell - wet well vacuum breakers.
19. Penetrations given in FSAR Table 6.2-44 which are not included in Table 3.2-1.
20. Seismic Category I electric manholes and ductbanks.

b. The following items from the FSAR Table 3.2-1 need expansion and/or clarification as noted. Revise the list as indicated or justify not doing so.

1. Identify the safety-related instrumentation and control systems to the same scope and level of detail as provided in Chapter 7 of the FSAR (this can be done by footnote). Verify that this includes I&C for

a) Hydrogen recombiner
b) Containment isolation system

2. For the systems shown below, expand the list in Table 3.2-1 to include the indicated components under the 10 CFR 50 Appendix B quality assurance requirements or verify that they are included as part of the components already listed.

IX. RHR System

a) Containment spray piping and nozzles
b) Conical stainless steel basket strainers
c) RHR heat exchanger, secondary side valves

XXVIII. Combustible Gas Control System

a) Drywell purge compressor piping, valves, and vacuum relief system
b) Containment purge piping, valves, and filtration system
c) Hydrogen analyzer

XXIX. Standby Gas Treatment System

a) Deep-bed charcoal filters

XXXIII. Auxiliary a-c Power System

a) Diesel generator packages including auxiliaries (e.g., lube system, jacket cooling, air start system, governor, voltage regulator, excitation system)
b) 4160 volt switchgear
c) 480 volt load centers
d) 480 volt motor control cent rs
e) Instrumentation, control, and power cables (including underground cable system, cable splices, connectors, and terminal blocks)
f) Conduit and cable trays and their supports for Class IE cables and those whose failure may damage other safety-related items.
g) Transformers
h) Valve operators
i) Protective relays and control panels
j) AC control power 'nverters
k) 120 AC vital bus distribution equipment
l) Containment electrical penetration assemblies

XXXIV. 125/250 Volt d-c Power System

a) Batteries, battery chargers, and distribution equipment
b) Cables
c) Conduit and cable trays and their supports for Class IE cables and those whose failure may damage other safety-related items
d) Battery racks
e) Protective relays and control panels

XLVI. Civil Structures

a) Pressure suppression containment
b) Enclosure building
c) Reactor pressure vessel shield wall annulus

d) Drywell head compartment
e) Main steam tunnel compartment
f) RWCU heat exchanger, holding pump, and valve nest compartments
g) RWCU pipe chase transfer
h) Filter/demineralizer compartment

1) Weir wall
2) Drywell-to-suppression pool vents

3. Feedwater spargers and other reactor internals (Table 3.2-1 item I-6) whose failure could reduce the functioning of the reactor system to an unacceptable safety level should have the pertinent QA requirements of 10 CFR 50 Appendix B applied per Regulatory Positions 2 & 4 of Regulatory Guide 1.29.

c. Enclosure 2 of NUREG-0737, "Clarification of TMI Action Plan Requirements" (November 1980) identified numerous items that are safety-related and appropriate for OL application and therefore should be on Table 3.2-1. These items are listed below. Add the appropriate items to Table 3.2-1 and provide a commitment that the remaining items are subject to the pertinent requirements of the FSAR operational QA program or justify not doing so.

		NUREG-0737 Enclosure 2 Clarification Item
1)	Plant-safety-parameter display console	I.D.2
2)	Reactor coolant system vents	II.B.1
3)	Plant shielding	II.B.2
4)	Post accident sampling capabilities	II.B.3
5)	Valve position indication	II.D.3
6)	Dedicated hydrogen penetrations	II.E.4.1
7)	Containment isolation dependability	II.E.4.2
8)	Accident monitoring instrumentation	II.F.1
9)	Instrumentation for detection of inadequate core-cooling	II.F.2
10)	HPCI & RCIC initiation levels	II.K.3(13)
11)	Isolation of HPCI & RCIC	II.K.3(15)
12)	Challenges to and failure of relief valves	II.K.3(16)
13)	ADS actuation	II.K.3(18)
14)	Restart of core spray and LPCI	II.K.3(21)
15)	RCIC suction	II.K.3(22)
16)	Space cooling for HPCI & RCIC	II.K.3(24)
17)	Power on pump seals	II.K.3(25)
18)	Common reference level	II.K.3(27)
19)	ADS valve, accumulators, and associated equipment and instrumentation	II.K.3(28)
20)	Emergency plans (and related equipment)	III.A.1.1/III.A.2
21)	Equipment and other items associated with the emergency support facilities	III.A.1.2
22)	Inplant I radiation monitoring	III.D.3.3
23)	Control-room habitability	III.D.3.4

RESPONSE

FSAR Table 3.2-1 is a tabulation of the major safety-related civil structures, mechanical systems and components and electrical systems and components which are part of the Grand Gulf Nuclear Station. This table is not intended to be a listing of every safety-related component in the station.

In many instances, various safety-related equipment are not included in this table. This is for one of several reasons; (1) the safety related aspect of the system is discussed in other sections of the FSAR, (2) system flow diagrams depict the safety-related status of the structure, system or component, (3) the level of detail in the FSAR does not address a particular component.

This response addresses all of the items in Quality Assurance Question 260.1.

Part a. Response

The following items will be either added to FSAR Table 3.2-1 in the next available amendment or justification will be given for their exclusion. Item numbers correspond to those in the request for information.)

1) This issue has been addressed in the response to I&E Bulletin 80-11.

2) Biological shielding is the same as radiation shielding. Radiation shielding is addressed in item a.14 of Question 260.1.

3) Missile barriers within the containment structure, auxiliary building, control building and diesel generator building are safety-related and are controlled by the QA program.

4) The spent fuel pool is safety related and controlled by the QA program. The spent fuel pool liner is not on the Q-List. See FSAR subsection 9.1.2.1.1.1(e).

5) The normal operation, fixed radiation monitoring system is discussed in FSAR subsection 12.3.4. This system is not safety-related and, therefore, is not controlled by the QA rogram.

   The post-accident high range radiation monitoring system for the drywell and containment is safety-related and the components are controlled by the QA program

   Portable radiation monitoring is not a "structure, system, or component" requiring entry in Table 3.2-1. Control of this activity is provided by the appropriate GGNS Administrative Procedures. These procedures are subject to pertinent requirements of the MP&L Operational QA Program.

6) The normal operation and post-accident fixed radioactivity monitoring systems are discussed in FSAR subsection 12.3.4 and Section 11.5. None of these systems are safety-related and, therefore, are not controlled by the QA program. The only components that are controlled by the QA program are the Flow Monitoring and Isokinetic Sampling sections which are installed in the SGTS A&B duct boundary. These components meet the same requirements as those for item XXIX.1.c in FSAR Table 3.2-1. As such, no revision to Table 3.2-1 is required.

   Portable radioactivity monitoring is not a "structure, system, or component" requiring entry in Table 3.2-1. Control of this activity is provided by the appropriate GGNS Administrative Procedures. These procedures are subject to pertinent requirements of the MP&L Operational QA Program.

   The GE-supplied process radiation equipment is safety class 2, meets the requirements of 10CFR50, Appendix B, and is included in FSAR Table 3.2-1, Section VIII item 1.

7) The normal operation and post-accident sampling systems are discussed in FSAR subsections 7.7.1.11, 9.3.2, and 12.3.4. These systems are not safety-related and are not controlled by the QA program. No revision to FSAR Table 3.2-1 is required.

8) Radioactivity contamination measurement and analysis is not a "structure, system, or component" requiring entry in Table 3.2-1. Control of this activity is provided by the appropriate GGNS Administrative Procedures. These procedures are subject to pertinent requirements of the MP&L Operational QA Program.

9) Personnel monitoring internal and external is not a "structure, system or component" requiring entry in Table 3.2-1. Control of this activity is provided by the appropriate GGNS Administritive Procedures. These procedures are subject to pertinent requirements of the MP&L Operational QA Program.

10) As required by the MP&L Operational Quality Assurance Program, GGNS has in-place measures to assure that measuring and testing equipment used in activities affecting quality are controlled, calibrated and adjusted to maintain accuracy within specified limits.

11) Decontamination is not a "structure, system, or component" requiring entry in Table 3.2-1. Control of this activity is provided by the appropriate GGNS Administrative Procedures. These procedures are subject to pertinent requirements of the MP&L Operational QA Program.

12) Respiratory protection including testing is not a "structure, system, or component" requiring entry in Table 3.2-1. Control of this activity is provided by the appropriate GGNS Administrative Procedures. These procedures are subject to pertinent requirements of the MP&L Operational QA Program.

13) Contamination control is not a "structure, system, or component" requiring entry in Table 3.2-1. Control of this activity is provided by the appropriate GGNS Administrative Procedures. These procedures are subject to pertinent requirements of the MP&L Operational QA Program.

14) Radiation shielding at Grand Gulf may be classified as 1) shielding required to limit offsite radiation doses to allowable limits and 2) shielding required to limit in-plant doses for personnel access to various plant areas.

Radiation shielding to limit offsite doses is considered safety-related and is provided by the containment and auxiliary buildings. These structures are fully designed as safety-related structures and are capable of withstanding all postulated natural phenomena, and dynamic events.

Radiation shielding for personnel access to various plant areas is not considered safety-related. Radiation shielding for this purpose is provided in the auxiliary building, containment, radwaste building, turbine building and control building. Reinforced concrete and masonry walls are used to provide the necessary shielding. The in-plant radiation shielding walls in the control building, containment and auxiliary building are considered safety-related <u>only</u> to the extent that they must maintain structural integrity, i.e., the radiation shielding capability is not safety-related.

The radiation shielding walls in the turbine building and radwaste building have no safety-related function.

The quality assurance requirements for the containment, auxiliary building, and control building are given in FSAR Table 3.2-1, Section XLVI, items 1, 2 and 6.

15) The Fuel Handling Area ventillation exhaust system is discussed in FSAR subsections 11.5.1 and 12.3.4. This system is not safety-related and, therefore, is not controlled by the QA program. No revision to FSAR Table 3.2-1 is required.

16) Accident-related meteorological data collection equipment is not a "structure, system, or component" requiring entry in Table 3.2-1. Control of this activity is provided by the appropriate GGNS Administrative Procedures. These procedures are subject to pertinent requirements of the MP&L Operational QA Program.

17) This item is not a "structure, system, or component" requiring entry in Table 3.2-1. Control of this activity is provided by the appropriate GGNS Administrative Procedures. These procedures are subject to pertinent requirements of the MP&L Operational QA Program.

Welding electrodes used on safety-related components are subject to the appropriate Quality Assurance requirements.

18) The vacuum breakers for Grand Gulf are drywell-containment vacuum breakers which are safety-related and controlled by the QA program.

19) Penetrations given in FSAR Table 6.2-44 which are not included in FSAR Table 3.2-1 are safety-re ted and controlled by the QA program.

20) Seismic Category I electric manholes and ductbanks are safety-related and controlled by the QA program.

Part b. Response

1) FSAR Table 3.2-1 and the various Sections of FSAR Chapter 7 describe the instrumentation and control system in sufficient detail. Grand Gulf drawings, procedures and equipment listings provide the detail necessary for determining the safety-related status of individual components. A footnote will be added to Table 3.2-1 indicating that the instrumentation described in FSAR Sections 7.2 through 7.6 is subject to pertinent requirements of the QA program.

2) IX

Containment spray piping and nozzles and conical stainless steel basket strainers are safety-related and controlled by the QA program. The secondary side valves of the RHR Heat Exchanger are safety-related and have already been included in FSAR Table 3.2-1, Section XXIV - Standby Service Water System (#5, valves other). The SSW System is the secondary side of the RHR Heat Exchanger.

2) XXVIII

a. Drywell purge compressor piping, valves and vacuum relief system are safety-related and controlled by the QA program.

b. Containment purge piping, valves and filtration system are not safety related except for the portion associated with containment penetration (see Section XXVIII, item 1).

c. The hydrogen analyzer is safety-related and controlled by the QA program.

2) XXIX

a. The deep-bed charcoal filters of the Standby Gas Treatment System are safety-related and controlled by the QA program.

2) XXXIII

All of the items in Section XXXIII of Question 260.1 are safety-related and controlled by the QA program except for the following clarifications.

a. Diesel generator packages, including auxiliaries, are safety-related to the extent as defined in FSAR Table 3.2-1, Section XLI, Items 1 through 5.

h. Valve operators are considered with the valves where they are installed.

i. Control room panels are mentioned under Section XXI and Q-list protective relays are considered as part of the switchgear or panels.

2) XXXIV

All of the items in Section XXXIV of Question 260.1 are safety-related and controlled by the QA program with the following clarifications.

b. Cables are considered under Section XXXIII, item 2(c) and 4.

c. Conduit and cable trays and their supports for class IE cables and those whose failure may damage other safety-related items are safety-related and controlled by the QA program.

d. Battery racks are considered with the batteries.

e. Protective relays and control panels are considered with the equipment panel they are servicing.

2) XLVI

All of the items in Section XLVI of Question 260.1 are safety-related and controlled by the QA program with the following clarification.

a. The containment is covered in Section XLVI, item 1.

b. The enclosure building is covered in Section XLVI, item 7.

3) Item I.6 of FSAR Table 3.2-1 includes those non-safety class internal structures such as feedwater spargers, steam dryers, shroud head and steam separator assembly, in-core guide tubes and stabilizers, and surveillance sample holders. These structures do not perform a safety function and are not required to prevent or mitigate the consequences of accidents. A failure of the feedwater sparger will not prevent transmission of cooling water to the core affecting the safety of the reactor system. Although these structures are not safety-related, they are so designed that they will not adversely affect the safety function of the safety related structures. Therefore, the classifications shown in FSAR Table 3.2-1 for this item are consistent with the safety requirements, and the QA requirements of 10CFR50 Appendix B do not apply.

Part c. Response

1) The Plant-safety-parameter display console is not safety related. Justification is contained in NUREG-0696 Paragraph 2.5 and 4.2 (Table and Footnotes). Emergency Facilities are neither 1) required for safe shutdown or immediate or long term operation following a LOCA or, 2) will not cause the release of radioactivity in excess of 10CFR100 limits or increase severity of a DBA if they should fail. Therefore, Emergency Support Facilities will not be added to Table 3.2-1.

2) Reactor Coolant System Vents are safety-related and controlled by the QA program. This item is already addressed in Section II, item 16 of FSAR Table 3.2-1.

3) The plant shielding item requires a review of the accessibility of various station areas under post-accident conditions. This review is not considered safety-related.

4) See the response to item a.7 above.

5) As stated in FSAR subsection 18.1.24, a safety/relief valve position monitoring system is being added to Grand Gulf to indicate the open/closed condition of each safety/relief valve. The system will meet the same quality requirements as stated for Section II, item 13 of FSAR Table 3.2-1.

6) Dedicated hydrogen penetrations as described in subsection 18.1.25 of the FSAR and in Section II.E.4.1 of NUREG-0737 are not applicable to the Grand Gulf Nuclear Station.

7) The containment isolation valves and their associated circuits are safety related and controlled by the QA program. The isolation valves are listed in FSAR Table 3.2-1 under each individual applicable system. The isolation instrumentation is described in FSAR subsection 7.3.2.2. See also the response to item b.1 above.

8) See the response to item b.2.XXVIII.c above.

9) As stated in FSAR subsection 18.1.28, no additional instrumentation is needed to monitor inadequate cooling at Grand Gulf. Therefore, no changes to FSAR Table 3.2-1 are required.

10) FSAR subsection 18.1.30.2 indicates that modification of the initiation logic for automatic restart of the RCIC system on low water level is being incorporated into the Grand Gulf design. These modifications will meet the same requirements as given in FSAR Table 3.2-1, Section XII, item 8.

11) As stated in FSAR subsection 18.1.30.3, a time delay to the HPCI/RCIC break detection circuitry will be added. This addition will meet the same quality requirements as given in FSAR Table 3.2-1, Section XII, item 8.

12) FSAR subsection 18.1.30.4 indicates that modification to the Grand Gulf design would not significantly reduce the frequency of safety/relief valve events. Therefore, no changes to FSAR Table 3.2-1 are required.

13) Modifications referred to in FSAR subsection 18.1.30.5 to the ADS actuation logic will meet the same requirements as stated for the Nuclear Boiler System in FSAR Table 3.2-1, item 13.

14) Grand Gulf will provide an automatic HPCS reset feature as described in FSAR subsection 18.1.30.7. This feature will meet the same quality requirements as given in FSAR Table 3.2-1, Section XI, item 9.

15) The automatic switchover of the RCIC suction from the condensate storage tank to the suppression pool is considered safety-related and is subject to the pertinent QA requirements for class IE electrical systems.

16) Grand Gulf does not have an HPCI system, rather an HPCS system. Both the HPCS and RCIC room cooling systems are safety-related and are covered by the QA Program. See Section XXX in FSAR Table 3.2-1.

17) As stated in FSAR subsection 18.1.30.10, no change in the Grand Gulf design is required. Therefore, no addition to FSAR Table 3.2-1 is required.

18) The changes referred to in FSAR subsection 18.1.30.11 will meet the same quality requirements as stated in FSAR Table 3.2-1, Section XXI.

19) All equipment associated with the ADS system is safety-related and controlled by the QA program. Major components are listed in FSAR Table 3.2-1, Section II. The review of the ADS system to ensure operability was conducted in accordance with Bechtel Power Corporation design procedure.

20) Emergency Plans involve administrative controls to assure that various actions are taken in the event plant conditions warrant these actions. The GGNS Emergency Plan is a controlled document and the implementing procedures for the Emergency Plan are safety-related procedures. These procedures and the Emergency Plan are subject to audit by MP&L QA in accordance with the NRC accepted Topical Report MPL-TOP-1A.

21) Equipment and other items associated with the Emergency Support Facilities are not safety related. Justification is contained in NUREG-0696 Paragraph 2.5 and 4.2 (Table and Footnotes). Emergency Facilities are neither 1) required for safe shutdown or immediate or long term operation following a LOCA or, 2) will not cause the release of radioactivity in excess of 10CFR100 limits or increase severity of DBA if they should fail. Therefore, Emergency Support Facilities will not be added to Table 3.2-1.

22) Inplant I_2 radiation monitoring would be performed under post-accident conditions using portable survey instruments and silica gel cartridges as discussed in the GGNS Emergency Plan. Calibration and use of these devices under post-accident conditions will be controlled by procedures subject to the pertinent requirements of the MP&L Operational Quality Assurance Program.

23) The control room HVAC system is safety-related and controlled by the QA program. This system is addressed in FSAR Table 3.2-1, Section XXXV.

362.7 Your response to NRC Question 362.5 (page Q&R 2.5-14, Amendment 30) is not complete. Indicate what measured value of differential settlement would cause code allowable stresses to be exceeded for individual buildings and buried piping and require notification of NRC. Provide a table of these values and the limiting stress criteria. Update the plots which show settlement recorded to date versus time; this is particularly important since construction of Unit 2 was still underway when your response to Question 362.5 was being developed.

RESPONSE

The response to this question is given in revised subsection 2.5.4.13.1, revised Table 2.5-10, and revised Figures 2.5-75a through 2.5-75h.

deflectometers. The type of foundation instrumentation and locations are presented on Figure 2.5-75.

2.5.4.13.1 Foundation Rebound and Settlement

Rebound extensometers, which were installed prior to construction to monitor foundation heave, are discussed in subsection 2.5.4.5.4. Only half of the sensing elements are presently operating. These elements are read on a regular basis to monitor the recompression under loading of the structures.

A settlement monitoring program, based on conventional surveying techniques, was developed and is being implemented to determine the magnitude and rate of settlement for the power block structures. A minimum of two settlement markers is established on all Category I buildings except the control building which has one marker. A settlement marker just inside the turbine building at the junction of the control and turbine buildings serves as a settlement monitoring point for both buildings. Major structures which are monitored for settlement include the turbine-generator pedestals, radwaste building, containment buildings, auxiliary buildings, diesel generator buildings, and standby service water cooling towers. The approximate location of these markers is shown on Figure 2.5-75. Any marker which is destroyed or becomes inaccessible is replaced in the same vicinity. After initial settlement, marker elevations are established and surveys are scheduled at intervals of approximately 30 days until the structures are complete. The survey data are compiled and settlement graphs are prepared. A graphical plot of settlement versus time for all major structures is shown in Figures 2.5-75a through 2.5-75h. Graphical plots of time versus excavation and rebound, as measured by the rebound extensometers, are shown in Figures 2.5-76 through 2.5-80. The total measured settlement is compared to the estimated settlement.

Excavation to reach the top of the Catahoula-bearing stratum resulted in removal of approximately 11 ksf of overburden and, based on the rebound extensometers, resulted in about 2 inches of measured rebound at the top of the Catahoula Formation. As discussed in subsection 2.5.4.10.2, the predicted settlement was based on elastic theory and a modulus of elasticity of the Catahoula Formation computed from the results of the rebound extensometers. Table 2.5-10 shows the comparison of the predicted settlements with the measured values to the end of March, 1981 |
along with the percent of structure dead loads.

The settlements predicted, based on measured rebound modulus of elasticity, agree well with the measured values, but since construction of Unit 2 is still under way, some settlement will continue to occur. Settlements measured to date are all within the measured rebound value of the Catahoula Formation.

The settlement markers will be read at one-month intervals until construction is complete and all pools are filled with water, at which time the results of the settlement monitoring program will be evaluated and long-term monitoring of the structures will be assessed. Total settlement will not be a Geotechnical concern until the following measured values are observed:

Turbine building	3"
Radwaste building	3"
Control building	3"
Containment building	3.5"
Auxiliary building	3.5"
Diesel generator building	3"

The elastic properties of the Catahoula formation and Category I structural backfill (as applicable), have been used to model the supporting media for the foundation mats of all Category I structures. In this manner, stresses in the foundation mats due to non-uniform (differential) settlement (i.e., unequal column loadings, wall loadings, etc.) are accounted for in the mat foundation designs. As can be seen from Table 2.5-10, the total settlement of the Category I structures has been small and differential settlement has not been a concern for foundation mat design. However, differential settlement is important with regard to maintaining proper isolation gaps between adjacent buildings and has been considered in the design. The maximum allowable differential settlements of the containment structure and auxiliary building are 0.6 inches and 1.15 inches, respectively, in order to prevent these buildings touching during an OBE.

Differential vertical settlement between buildings and foundation rotation also affect piping passing between these buildings and to surrounding soil and are considered in the design of piping. Piping allowable stresses resulting from differential settlement between buildings are limited to the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division I, Paragraph NC 3652.3, Equation 10a. The predicted and measured settlements on Table 2.5-10 are less than $1\frac{1}{2}$ inches, with differential settlements between adjacent buildings with piping passing between them being less than 1/2 inch. Since the structures with the largest measured settlements are essentially completed from a loading aspect, these values should remain at this level in the future. It is anticipated that differential settlements between adjacent edges of the buildings will be even less than the recorded settlements, due to overlapping pressure influence. In addition, a substantial portion of the building settlement has usually occurred before the installation of the penetrations is complete. Thus, any differential settlement experienced by pipes passing between buildings will be very small. In the case of the Category I standby service water piping which runs between the auxiliary building and the standby service water cooling tower basin, there will be no significant settlement between the building and the surrounding soil, since movement of the soil adjacent to the building will take place as building movement occurs. In addition, any differential

settlement between the building and the soil will be slow enough to ensure that stresses built up in the soil due to penetration movement will be redistributed with time, thus reducing the level of stress in the pipes and anchors.

2.5.4.13.2 Tieback Wall Instrumentation

A soldier pile and lagging tieback wall was installed for the construction of the power block. The wall had a perimeter of 4200 linear ft with a surface area of 180,000 sq ft. The average height of the wall was 43 ft. The wall was retained by approximately 2400 earth tiebacks, each with a capacity of 50 to 60 tons. In addition to the ordinary surveying done to monitor the movement of the wall, load cells to measure the load in each tieback at various elevations and to compare the actual loading with the design loading were installed. Load cell data indicated that the actual load ranged from 50 percent to 90 percent of the design load. Wall deflection and ground movement were continually monitored during and after wall construction. Borehole deflectometers were installed behind soldier piles. Multiple position borehole extensometers were installed between tiebacks at approximately 20 degrees from the horizontal. This electronic instrumentation compared favorably with the regular surveying measurements for horizontal and vertical movement. Surveying methods were the primary source of monitoring the wall. Lateral and vertical movements were generally less than 1 inch.

TABLE 2.5-10

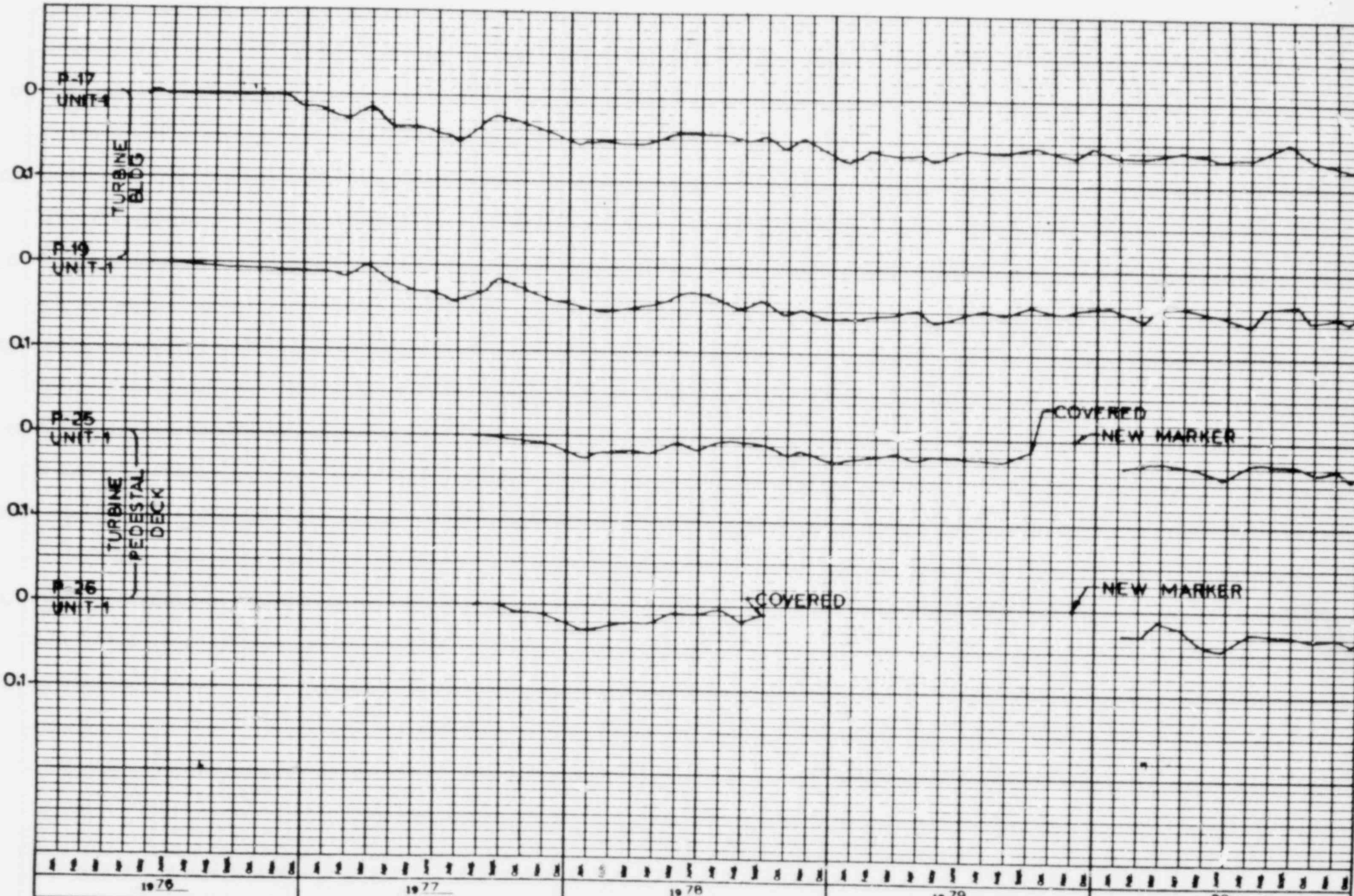
PREDICTED VS MEASURED TOTAL SETTLEMENT VALUES

(Based on Data at the end of March, 1981)

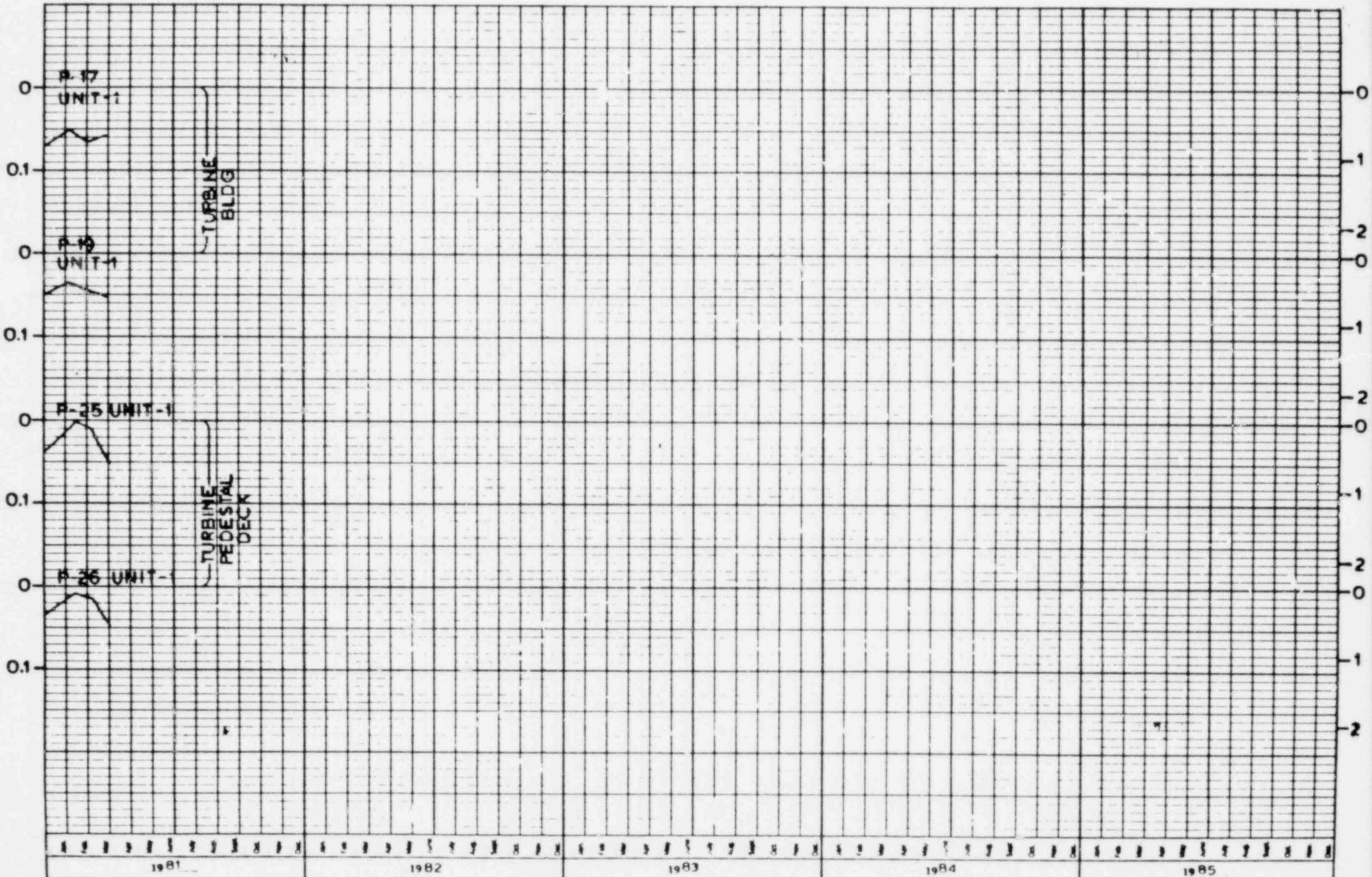
	Settlement Maximum *Predicted (in)	**Measured (in.)	Percent of Dead Load Completed
Containment - Unit 1	0.8	1.1 (1.2)	96
Containment - Unit 2	0.8	0.8 (0.8)	40
Auxiliary building - Unit 1	1.0	1.1 (1.4)	96
Auxiliary building - Unit 2	1.0	0.4 (0.4)	10
Radwaste building	0.8	0.7 (0.8)	96
Control building	0.5	0.9	96
SSW basin - Unit 1	0.7	0.4 (0.4)	100% water
SSW basin - Unit 2	0.7	0.5 (0.7)	100% water
Diesel gen building - Unit 1	0.8	0.4 (0.4)	98

* Based on elastic modulus values as determined from rebound measurements. Refer to Figure 2.5-90 for predicted total settlements for different assumed groundwater levels. (These values are 40 year predictions).

** Values given are average of two settlement markers except for control building where there is only one marker. Values in parentheses are for the marker with greater settlement.

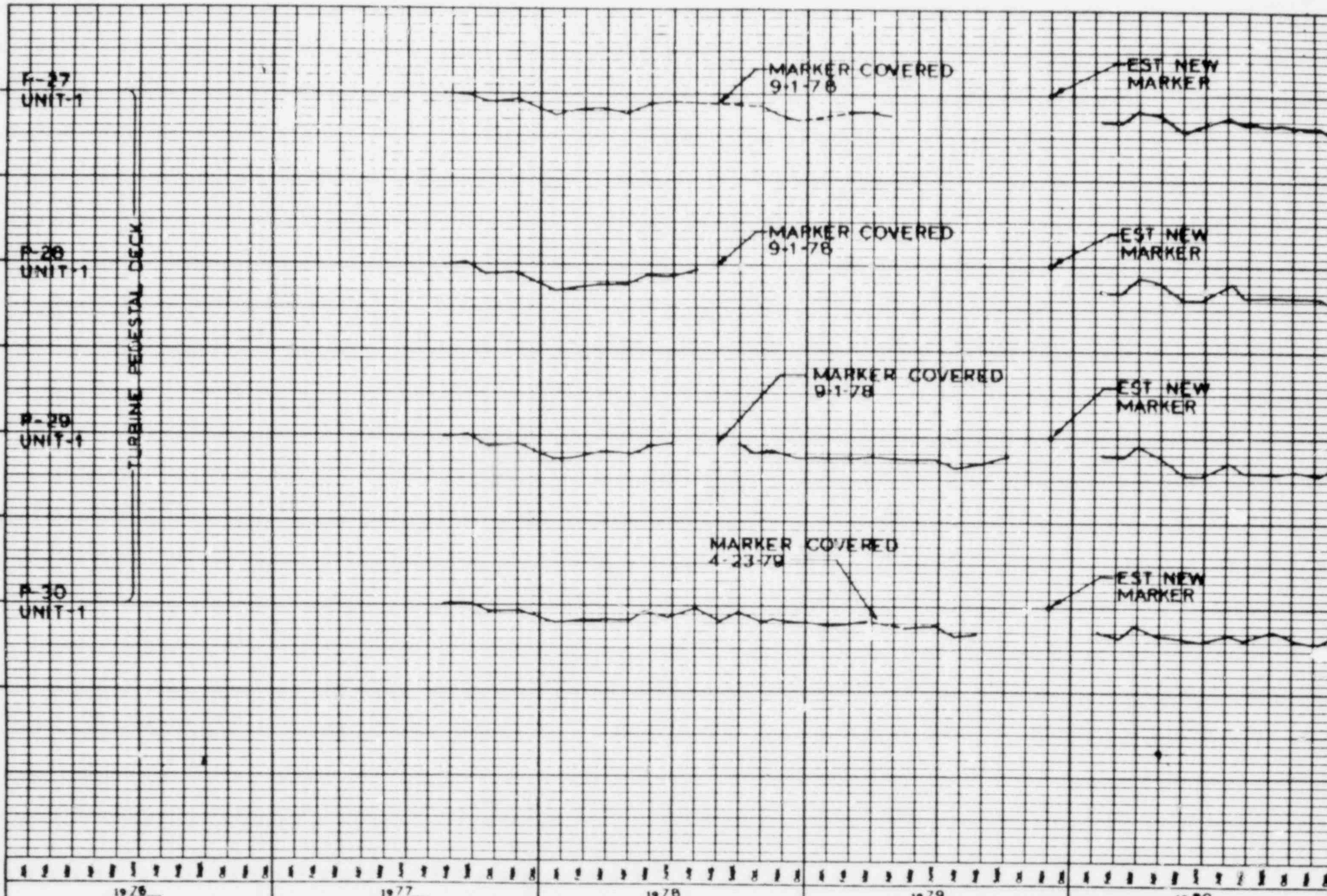


MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT	SETTLEMENT VS TIME UNIT 1 TURBINE BUILDING FIGURE 2.5-75a SHEET 1 OF 2

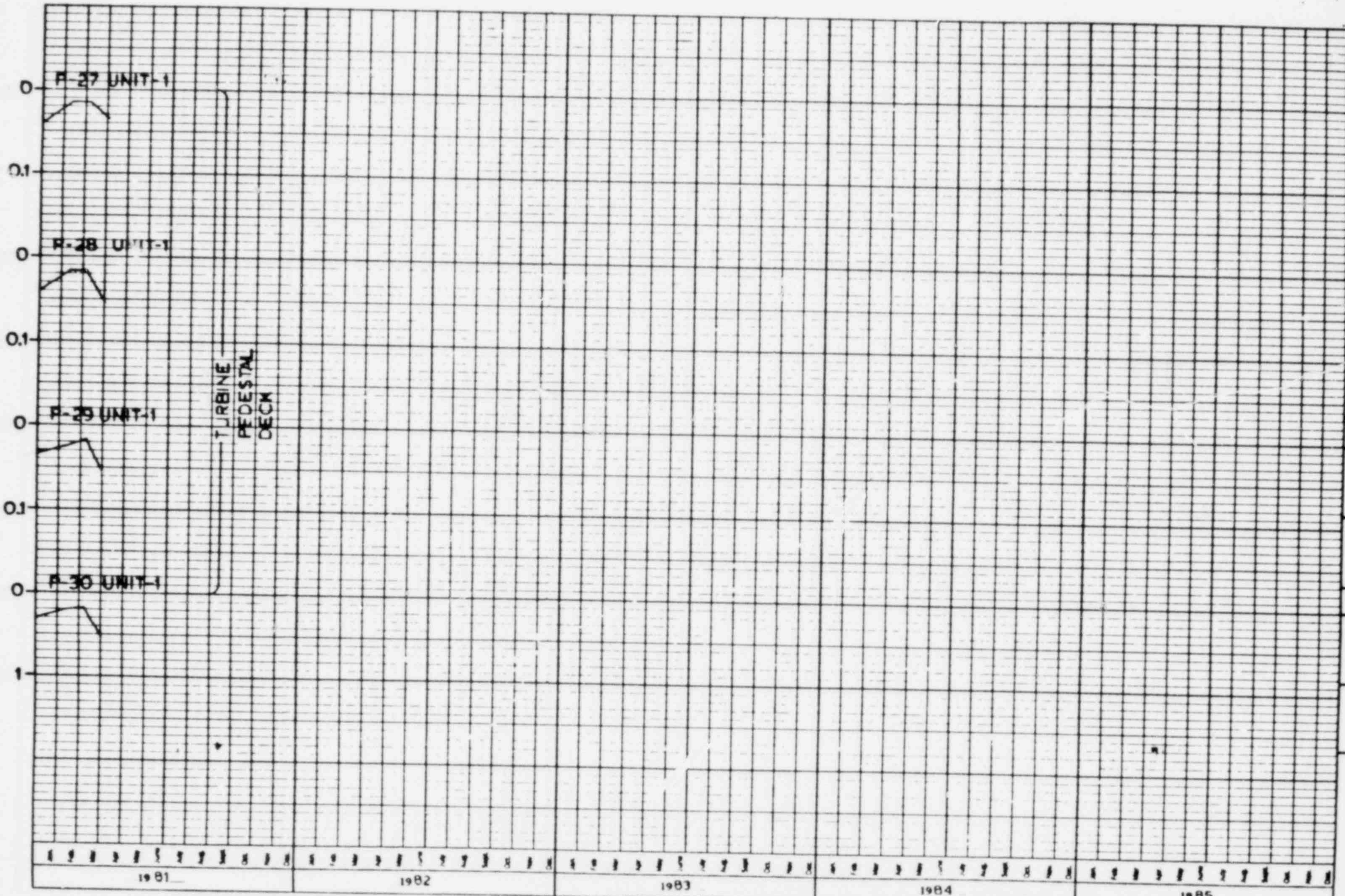


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2

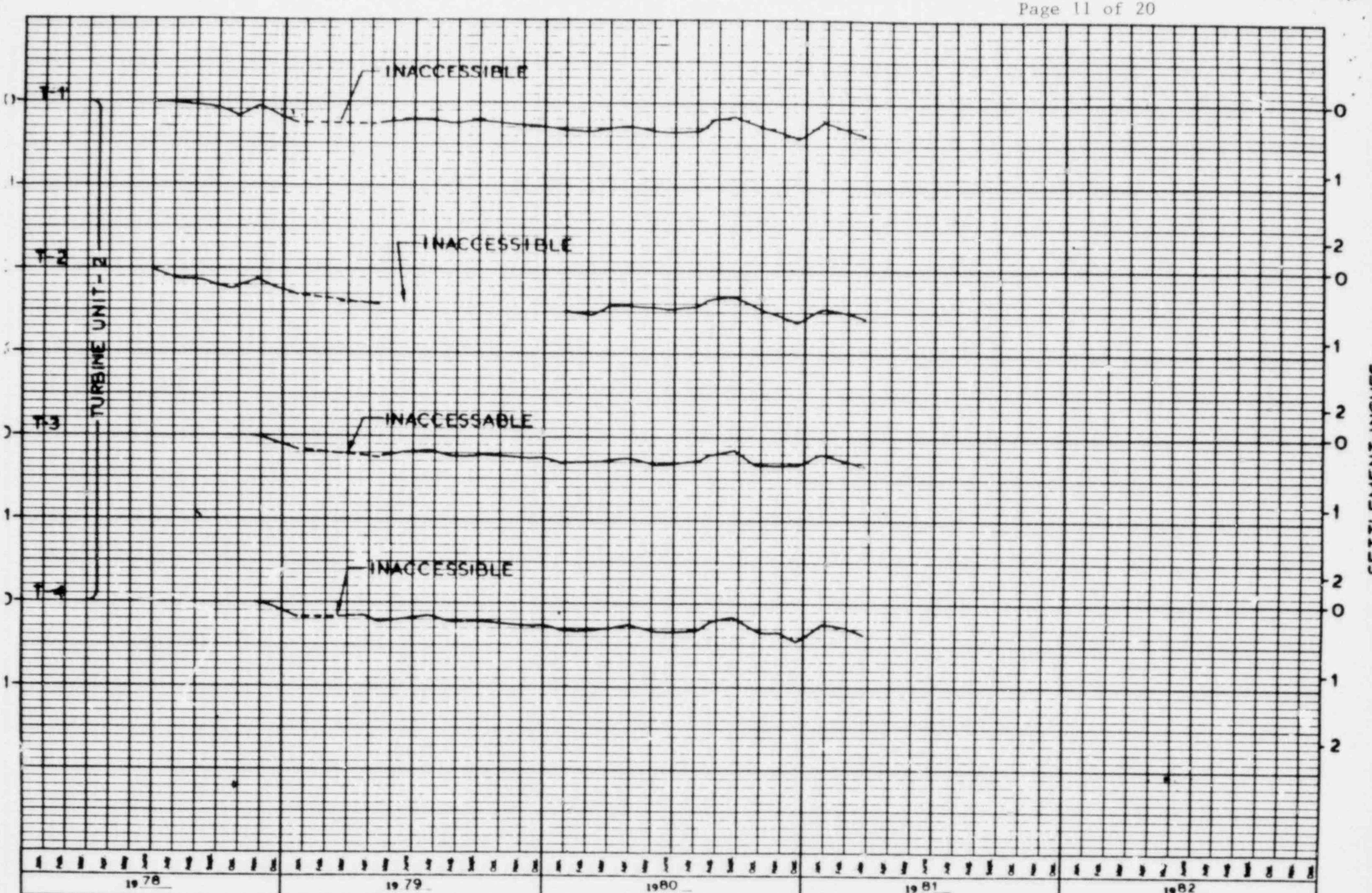
SETTLEMENT VS TIME
UNIT 1
TURBINE BUILDING



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT	SETTLEMENT VS TIME UNIT 1 TURBINE BUILDING FIGURE 2.5-75b SHEET 1 OF 2

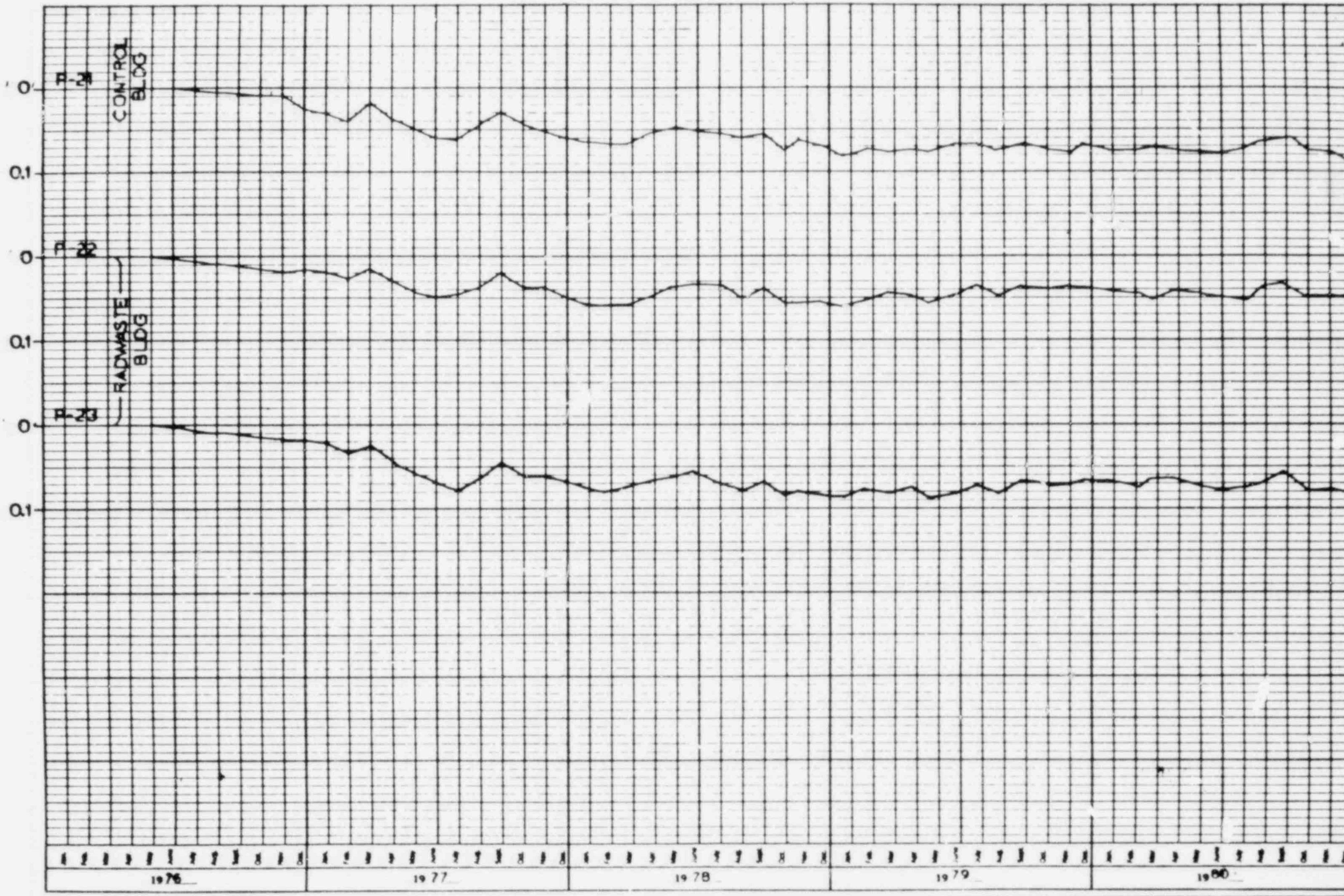


MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT	SETTLEMENT VS TIME UNIT 1 TURBINE BUILDING FIGURE 2.5-75b SHEET 2 OF 2



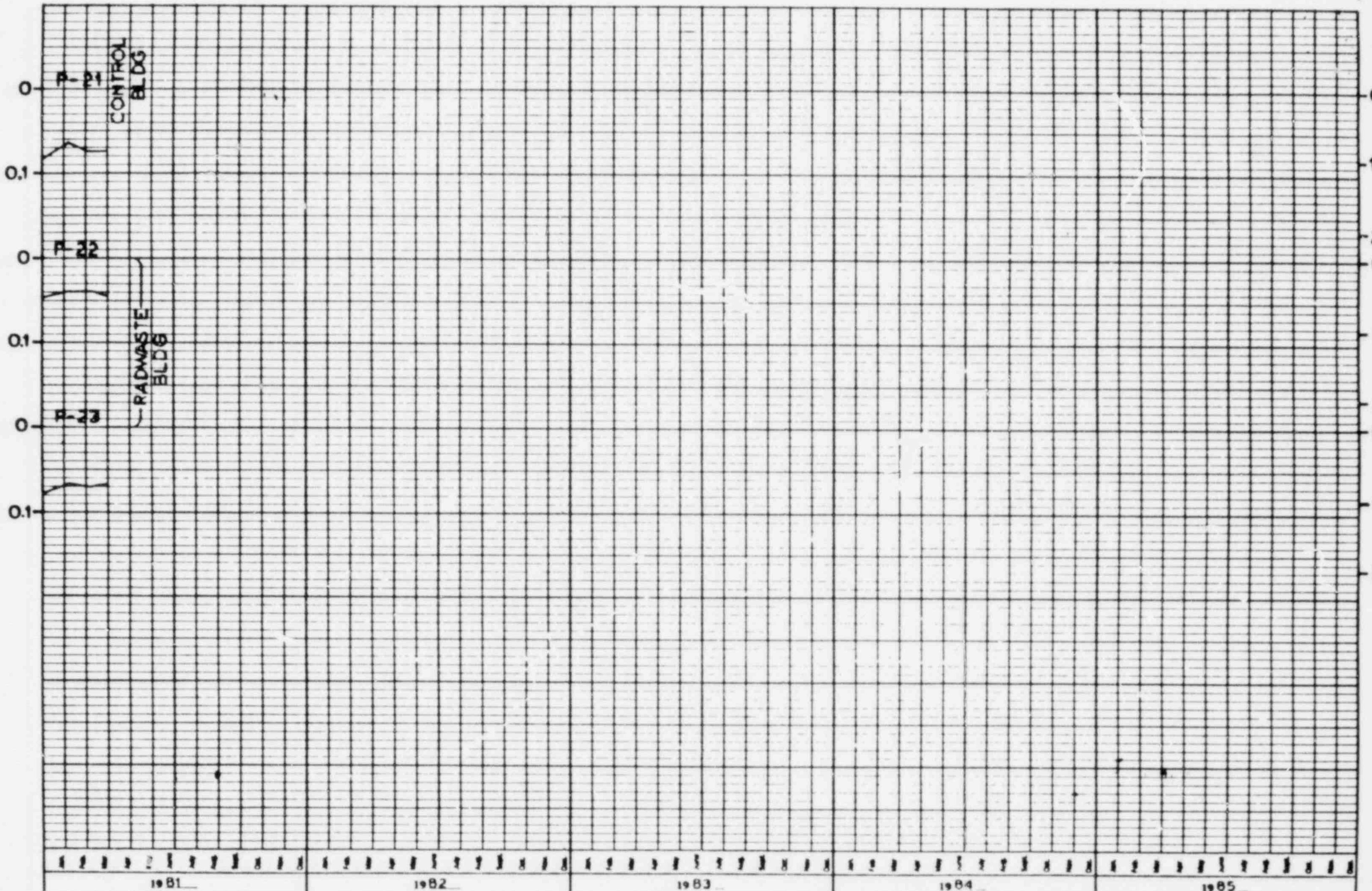
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT	SET.LEMENT VS TIME UNIT 2 TURBINE BUILDING FIGURE 2.5-75c

Amend. 50 8/81



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT	SETTLEMENT VS TIME RADWASTE BLDG/CONTROL BLDG FIGURE 2.5-75d SHEET 1 OF 2

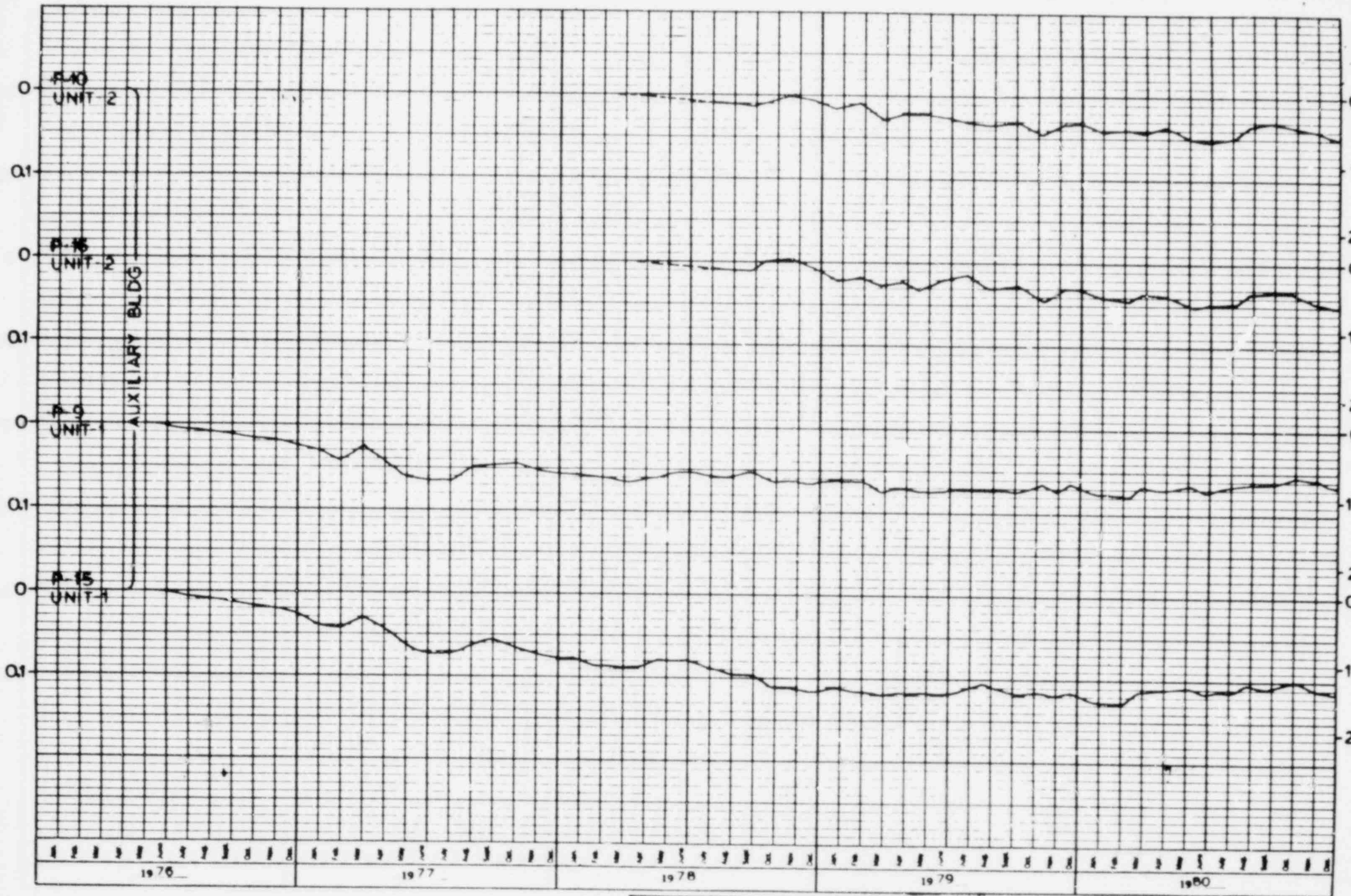
Amend. 50 8/81



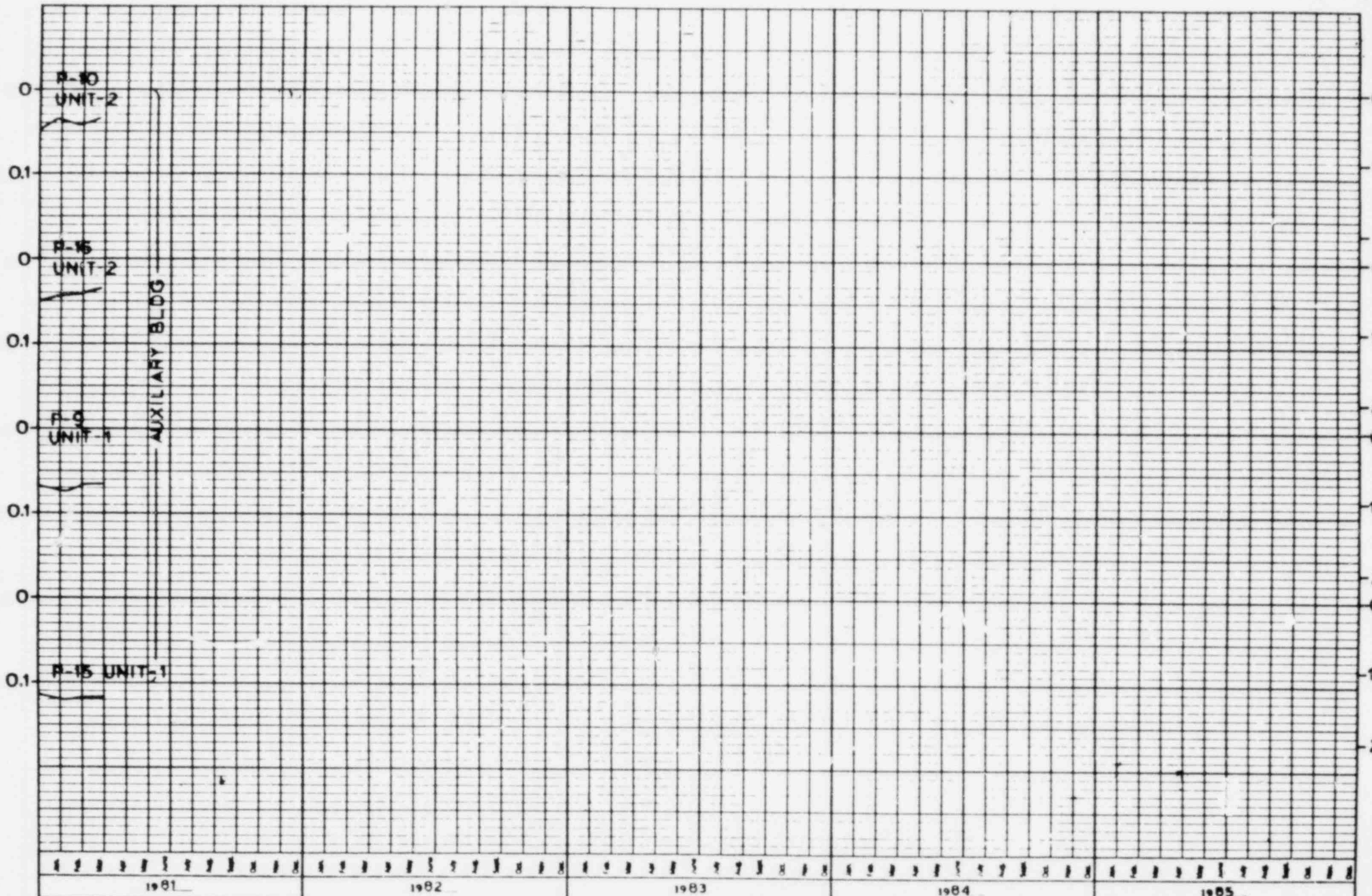
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

SETTLEMENT VS TIME
RADWASTE BLDG/CONTROL BLDG

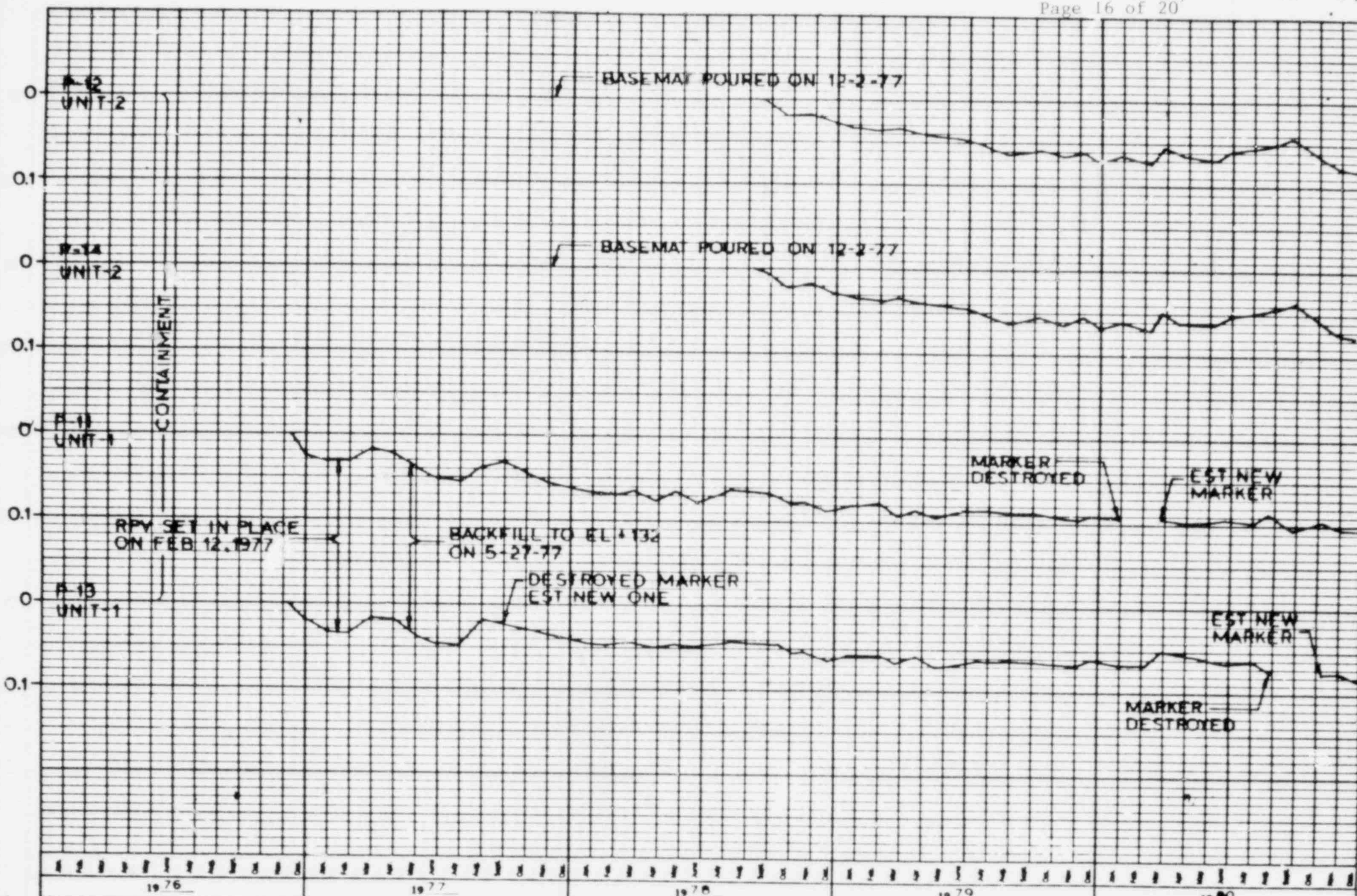
FIGURE 2.5-75d SHEET 2 OF 2



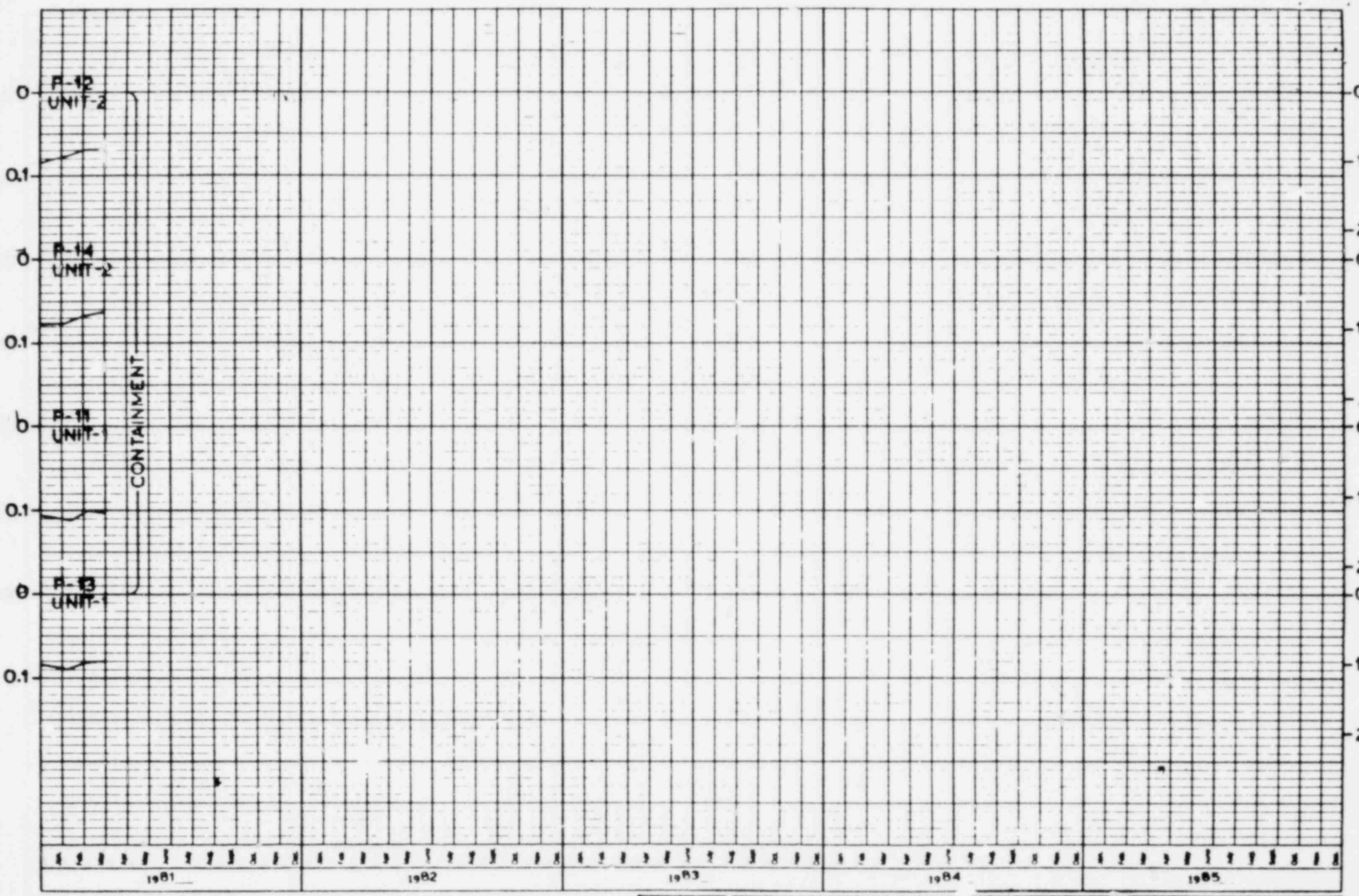
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT	SETTLEMENT VS TIME UNITS 1 & 2 AUXILIARY BUILDING FIGURE 2.5-75e SHEET 1 OF 2



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT	SETTLEMENT VS TIME UNITS 1 & 2 AUXILIARY BUILDING FIGURE 2.5-75e SHEET 2 OF 2



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT	SETTLEMENT VS TIME UNITS 1 & 2 CONTAINMENT BUILDING FIGURE 2.5-75f SHEET 1 OF 2



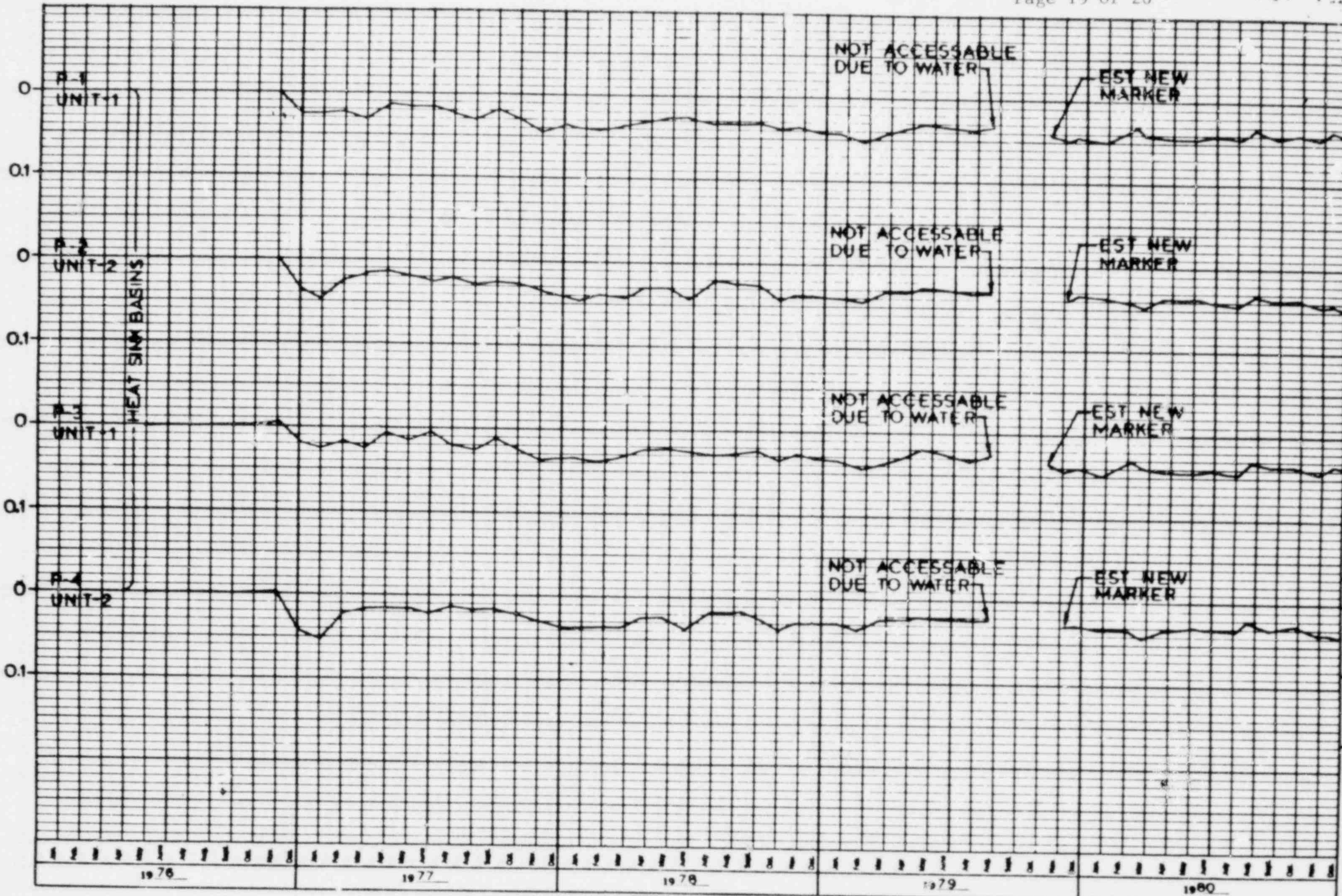
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT	SETTLEMENT VS TIME UNITS 1 & 2 CONTAINMENT BUILDING FIGURE 2.5-75f SHEET 2 OF 2

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

SETTLEMENT VS TIME
UNITS 1 & 2
DIESEL GENERATOR BUILDING

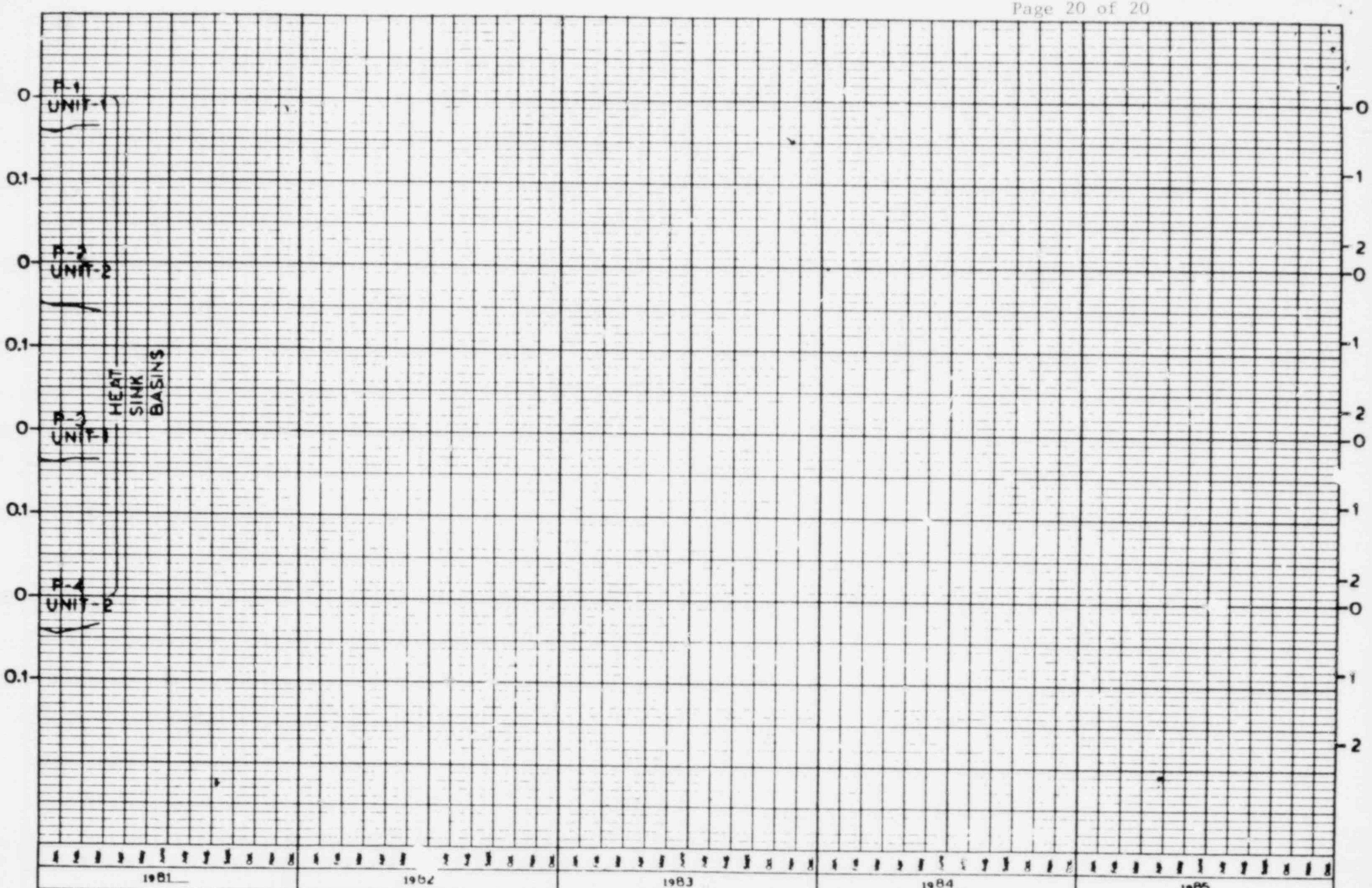
FIGURE 2.5-75g

Amend. 50 8/81



MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

SETTLEMENT VS TIME
UNITS 1 & 2
STANDBY SERVICE WATER
COOLING TOWER BASINS
FIGURE 2.5-75h SHEET 1 OF 2



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT	SETTLEMENT VS TIME UNITS 1 & 2 STANDBY SERVICE WATER COOLING TOWER BASINS FIGURE 2.5-75h SHEET 2 OF 2

SER Open Item

Ultimate Heat Sink - Effect of Flooding

1. Available water inventory after 30 days?

   a. There will be at least 3 days of usable water in the UHS after 30 days.

   b. Each of the two circulating water basins contains 6,700,000 gal. of water when 100% full. Assuming that both are 80% full (a conservative minimum) they will provide 30 days of cooling (for both units) based on the water usage rate at day 30 which is specified in the response to Item 2.

2. What is the water demand after 30 days?

   Water demand after 30 days for both units is 347,000 gal/day. This is a conservative estimate based on the usage rate at day 30. This breaks down to 241 gpm.

3. How much water would be needed for 90 days?

   Total water use for 90 days is 33,235,900 gal. where the same conservatism as above applies.

4. How much hose is available on-site?

   There are 7 hose houses that can be used without compromising fire protection. No hose will be taken from inside any buildings. The seven hose houses each contain 200 ft of hose and the pumper truck contains 600 ft of hose. This gives a total of 2000 ft of hose which is the required length for a connection between the circ water basins and the UHS.

5. Justify the initial water temperature assumption of 75°F in the UHS.

   A temperature gradient was provided by Bechtel Geotechnical Staff, it showed that from elevation 80'-110' the temperature range is 68-70°F, and from elevation 110'-132'6" (grade) the temperature range is 65-66°F. These temperatures are expected to remain relatively constant throughout the year.

   The calculation further states that since the basins extend ~60 ft below the surface, 37.5% of the basin is exposed to the 65-66°F range and 62.5% is exposed to the 68-70°F range.

   The make up water is from the plant service water radial wells and has a design temperature of 75°F maximum (an actual test in August of 1978 measured a water temperature of 68°F).

Based on the above information, the calculation conservatively assumes the 75°F as the maximum initial temperature in the basin. There will be some heat input from atmospheric heat at the basin cover, however, this cover consists of several feet of concrete.

There is perhaps a misconception of the term cold water temperature (CWT) in relation to the cooling towers. CWT is the temperature of the water that has been cooled in the towers, i.e., the temperature of the water returning to the basin. CWT is independent of the temperature of the water in the basin. CWT is a function of heat load, flow rate, and wet bulb temperature (WBT). A calculation was performed by Bechtel Environmental Staff to determine what the actual CWT would be assuming worst meteorlogical conditions, maximum flow rate, and maximum heat load. This temperature was 89.5°F and occurred between the 15-18 hour period after initiation (See Tables 1-2)

6. Provide a summary of results and assumptions on vendor tests of the mechanical draft cooling towers on the UHS.

   The Ceramic Cooling Tower Company Engineering Report (Figure 1) shows performance curves (Figure 2) for 3 different conditions. Case 1 is closest to our conditions. The actual conditions that we could expect are 11,385 gpm and 130.9°F hot water temperature.

7. Discuss conservatisms used in calculating the heat load and dissipations in the UHS.

   The 75°F initial cold water temperature in the basin is based on the maximum temperature of the make-up water (PSW). This in itself is conservative in that it is based on the maximum temperature of the river water. In actual practice, this water will normally be much lower than 75°. Due to the basin's location below ground level and the fact that the water is not exposed to direct sunlight, the actual temperature in the basin will be less than 75°. If the make-up water were to exceed 75°, it would require a very large input of make-up water to raise the mean temperature in the basin. This would not normally be the case, as make-up is usually added in relatively small amounts.

   Additional Safety Margins:

   a) The water drift losses are based on .020% (design) of the flow rate. An actual test performed on the cooling tower indicated that the drift losses were less than .000018% of the flow rate. This would decrease daily water loss due to drift from ~6,500 gal. to ~6 gal.

   b) All calculations assume worst possible case and maximum design heat loads. Many of the loads are assumed to remain constant over the 30 days. Actually some of the loads would probably be removed and some would not be generating the maximum heat the entire duration of 30 days.

TABLE I

Day 1 UHS Cooling Tower Analysis

For 24-Hr Worst Meteorological Conditions

Hour	DBT (°F)	WBT (°F)	Cold Water (°F)	Hot Water (°F)
3	78	77	86.9	110.5
6	77	76	87.5	117.5
9	77	76	87.2	115.4
12	86	79	88.5	114.4
15	92	81	89.5	113.9
18	94	81	89.2	112.4
21	90	82	89.3	109.2
24	84	78	84.9	99.1

TABLE II

UHS Cooling Tower Analysis

For 30-Day Worst Meteorological Conditions

Day	DBT (°F)	WBT (°F)	Cold Water (°F)	Hot Water (°F)
1	78.1	74.4	87.3	120.9
2	81.0	75.7	86.4	111.4
3	81.0	74.8	85	107.1
4	81.0	76.1	85.4	106
5	79.6	76.0	85	104.1
6	81.7	75.7	84.5	102.4
7	83.4	77.5	85.3	102.2
8	82.0	77.1	84.9	101.4
9	83.1	78.0	85.4	101.4
10	82.7	77.9	85.2	100.8
11	82.5	78.5	85.6	100.8
12	81.5	78.4	86	101
13	84.7	78.7	85.5	100.2
14	81.4	78.1	85.2	99.8
15	81.9	77.6	84.6	98.9
16	83.2	77.5	84.6	98.9
17	80.1	77.1	84.2	98.3
18	79.4	75.2	82.8	96.8
19	77.0	73.1	81.3	95.1
20	79.2	74.5	82.3	96.1
21	80.9	75.0	82.5	96.2
22	81.0	75.7	83.0	96.5
23	81.7	77.2	84.1	97.6
24	79.9	76.7	83.7	97.2
25	78.6	75.7	83.3	96.8
26	81.1	76.7	83.6	97.0
27	80.6	76.5	83.45	96.85
28	78.5	76.4	83.4	96.8
29	81.9	77.7	84.3	97.7
30	80.0	77.2	84.0	97.4

FIGURE 1

Prepared by
Checked by

CERAMIC COOLING TOWER COMPANY
CT-047
BECHTEL POWER CORPORATION
AGENTS FOR
MISSISSIPPI POWER AND LIGHT
GRAND GULF NUCLEAR STATION
PORT GIBSON, MISSISSIPPI
DESIGN SPECIFICATION NO. 9645-M-015.1

BECHTEL RECEIVED AUG 0 . 1977 JOB NO. 9645

ENGINEERING REPORT

11576 0463

1. **PURPOSE** - This report is in response to Bechtel letter dated July 14, 1977, asking for cooling tower performance characteristics in terms of cold water temperature and heat removal duty.

2. **PERFORMANCE CONDITIONS** - Per two (2) cells.

	Case 1	Case 2	Case 3
Flow (GPM)	12,435	11,655	13,049
Hot Water Temperature (°F)	128	123	120
Wet Bulb Range (°F)	73-82	73-82	73-82

3. **PERFORMANCE CURVE** - The curves included in this report represent predicted performance of the tower when subjected to the conditions described in paragraph 2. above.

RECORD DOCUMENT

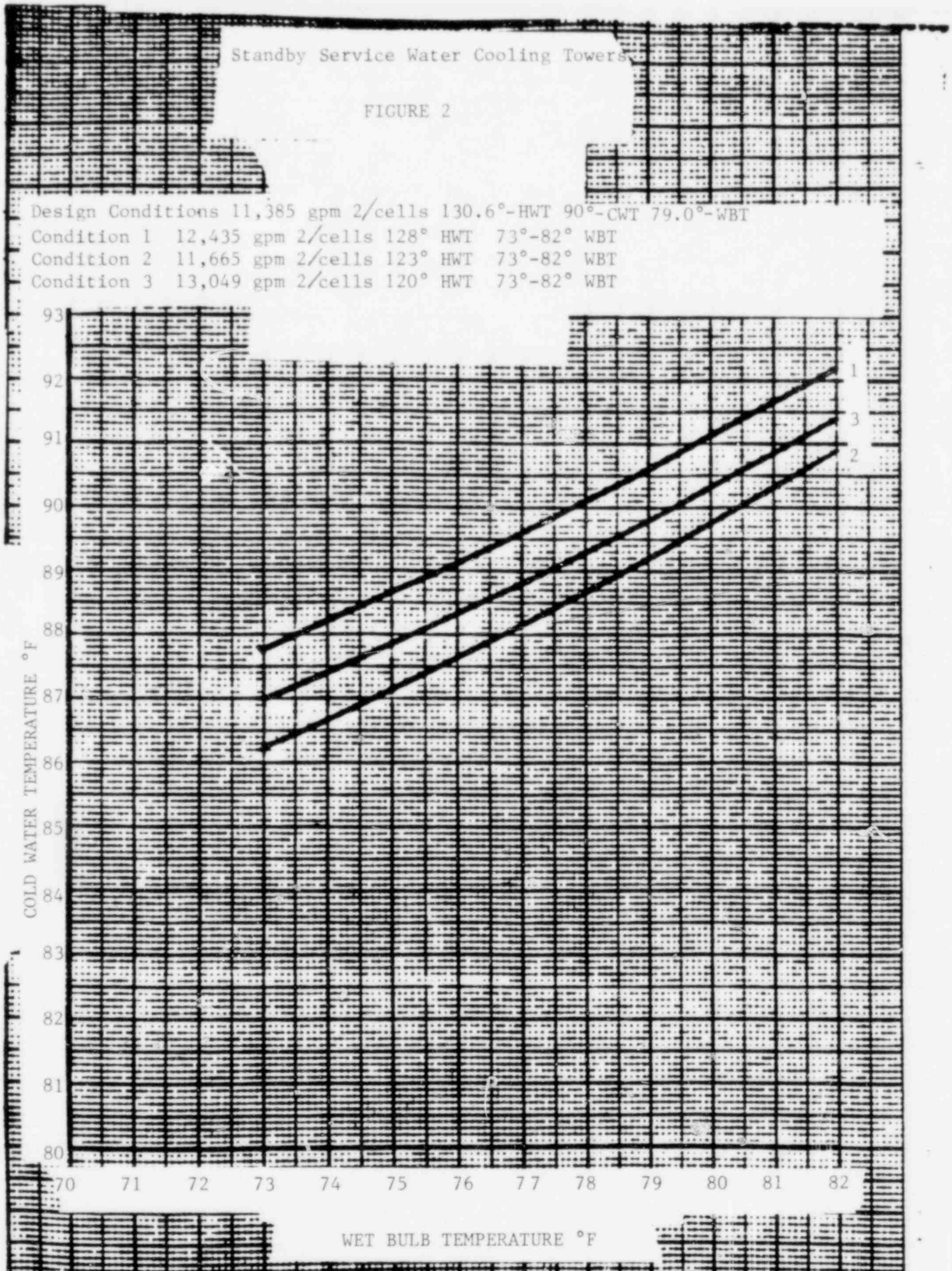
VENDOR'S DOCUMENT REVIEW
J. Wernial 8/16/77
BECHTEL
JOB NO 9645 BECHTEL CORPORATION POWER & INDUSTRIAL DIVISION P.O. BOX 607 GAITHERSBURG, MD

9645-M-015.1

QSP41B001A-290-2-0

ALSO APPLIES TO QSP41B001B

253268


Standby Service Water Cooling Towers
FIGURE 2
Design Conditions 11,385 gpm 2/cells 130.6°-HWT 90°-CWT 79.0°-WBT
Condition 1 12,435 gpm 2/cells 128° HWT 73°-82° WBT
Condition 2 11,665 gpm 2/cells 123° HWT 73°-82° WBT
Condition 3 13,049 gpm 2/cells 120° HWT 73°-82° WBT
COLD WATER TEMPERATURE °F
93
92
91
90
89
88
87
86
85
84
83
82
81
80
1
3
2
70 71 72 73 74 75 76 77 78 79 80 81 82
WET BULB TEMPERATURE °F

Shared Systems/Facilities/Buildings Between Units 1 and 2

This information augments information in FSAR subsection 3.1.2.1.5 which will be updated at the next available amendment.

System/Facility/Building

Security and Fire Protection Systems Computer System
Meteorological Monitoring System
Seismic Instrumentation System
Microwave System
Process Radiation Monitoring System - parts shared
Area Radiation Monitoring System - parts shared
Liquid Radwaste System
Solid Radwaste System
Auxiliary Steam System
Hydrogen and Carbon Dioxide System
Chlorination System
Make-Up Water Treatment System
Floor and Equipment Drain System
Radial Well System
Service Air System
Fire Protection System
Fire Detection System
Domestic Water System
Radwaste Building Ventilation System
Hot Lab Vacuum System
Emergency Operations Facility HVAC System
Water Treatment Building Ventilation System
Water Treatment Building Clean Lab System
Radial Well Pump/Switchgear House Ventilation System
Fire Water Pump House Ventilation System
Sewage Treatment Plant System
Control Building HVAC System
Control Building Sanitary Sump System
Control Room HVAC System
Control Building
Standby Service Water Basins - crossover piping and redundant pumps allow either basin to serve as the UHS for either unit.
Radwaste Building
Water Treatment Building
Fire Water Pump House
Radial Well Switchgear House
ESF Transformers
Standby Service Water Cooling Towers
Fuel Handling Equipment
Storage Equipment
Administration Building HVAC System
Instrument Air System - parts shared
Control Building and Radwaste Building Secondary Loop of the Chilled Water System

Administration Building
Warehouse
Was:e Water Basin
Tu:bine Building Cranes
4.16 kV Power Transformers (including the 34.5 kV Grounding Transformers)
6.9 kV Power Transformers
34.5 kV Service Power Transformer
480 Volt Load Centers, or Motor Control Centers
4.16 kV Switchgear & Load Shedding & Sequencing Panels
6.9 kV Swithcgear
13.8 kV Switchgear
34.5 kV Switchgear
500 kV Circuit Breakers (and 115 kV Motor Operated Disconnects)
120/208 Volt Distribution or Lighting Panels
125 V Switchgear and Distr. Panels
145 V Battery Chargers
Inverters

SER Open Items

Seismic Analysis - Soil Structure Interaction - (SEB)

Response

In a meeting with NRC's Structural Engineering Branch on August 27, 1981, MP&L made the following commitment to resolve NRC concerns in the area of seismic analysis soil-structure interaction.

A finite element seismic (FLUSH) analysis (FEM) will be performed for the containment, auxiliary, control and diesel generator buildings. Appropriate soil properties necessary as input to the analysis will be determined based on existing soils data presented in the FSAR. No further subsurface exploratory work is necessary; however, the basis for seismic soil property determination used in the analysis will be provided. Free field input motions will be in accordance with Regulatory Guide 1.60 with damping values provided in Regulatory Guide 1.61. Ground motion will be applied in the free field at the foundation level of the structures.

After completion of the analyses for each building, acceleration response spectra at key levels will be developed and compared with existing EHS lumped mass response spectra. The SEB position will then be applied to these results in order to assess the impact of the use of both methods of analyses to piping, equipment and components.

The FEM/EHS comparison of ARS will be used as a basis for design qualification of structures, systems and components at Grand Gulf.

NRC-SEB stated that if the FEM/EHS envelope exceeded the Grand Gulf EHS ARS envelope by a considerable amount, modifications to equipment or strengthening of structures may be required. If the EHS envelope is exceeded by less than a considerable amount, a discussion of conservatisms in the analyses will be provided. An explanation of how major differences, if observed, will be disposed will be provided to NRC.

The above stated analyses and comparisons will be completed and submitted to NRC by March 1, 1982. General statements regarding any modifications and associated schedules will also be provided. Specific modifications necessary will be completed before plant restart after the first regularly scheduled refueling outage.