QUESTION 110.1

GE indicated that a dynamic analysis method from the currently acceptable seismic analysis procedures will be used for calculating piping and component response to the pool swelling loads. The use of time history force-response calculations is acceptable. However, further justification is needed if either of the following methods are used:

(1) Static analysis using dynamic load factor,

(2) Modal superposition using response spectra.

During discussions at the Mark II owners meeting it was concluded that the method of using response spectra definitely shall not be applied to piping and components under hydraulic loads in areas directly effected by pool dynamics. Clarify circumstances under which various mentioned methods will be applied in calculating piping and component response and specify types of information needed for each case. If response spectra will be used for calculating response due to support movement only, provide justification for such application.

RESPONSE:

Please refer to the generic response to NRC Question MEB 1 provided by GE in a May 5, 1978 letter from L. J. Solson (GE) to J. T. Knight (NRC).

QUESTION 110.2

GE indicated the OBE damping of R.G. 1.61 will be used for the upset plant condition and SSE damping for the emergency and faulted plant conditions. The level of damping used should generally be associated with the piping and component service stress limit. Verify that SSE damping will only be used when Service level D stresses are designated, and that OBE damping will be used for all the other cases.

RESPONSE:

Please refer to the generic response to NRC Question MEB 2 provided by GE in a May 5, 1978 letter from L. J. Solson (GE) to J. T. Knight (NRC).

QUESTION 110.3

GE has indicated that for piping the SRSS method is expected to be used to combine the primary and secondary stresses resulting from a given dynamic event (i.e. inertia effects and relative displacement effects of a seismic event) because the response are sufficiently different in frequency content to be so combined. Since this procedure is not specifically covered by the typical example in NB-3600 of the ASME code, and an inquiry to the ASME Boiler & Pressure Committee did not produce conclusive results, verify your position to conform with future resolution of SRSS application, which is currently under separate generic review. For expediting the licensing process verify your intention to provide justification on a case by case basis when absolute sum is not used.

RESPONSE:

Please refer to the generic response to NRC Question MEB 3 provided by GE in a May 5, 1978 letter from L. J. Solson (GE) to J. T. Knight (NRC).

QUESTION 110.4

GE has indicated that seismic slushing loads were also considered in the OBE and SSE effects on the piping and components. However, such loads are not addressed in the DFFR. Provide definitions of such slushing loads and their appropriate combinations with other loadings or verify that such information will be provided for your plant.

RESPONSE:

Refer to the response to question 21.73.

QUESTION 110.5

Since LOCA + SSE is a required load set specified in the DFFR and annulus pressurization is a part of LOCA induced loading, verify that the annulus pressurization loading will be treated consistently for combination with SSE loading or if not provide justification.

RESPONSE:

Please refer to the generic response to NRC Question MEB 7(a) provided by GE in a May 5, 1978 letter from L. J. Solson (GE) to J. T. Knight (NRC).

QUESTION 110.6

Investigate "OBE + SRV to level B limits" and compare with the proposed "OBE + SRV to level C limits" to determine the more controlling event for piping and component design. The SRV would be typical of the number of valves needed following a turbine trip resulting from an OBE.

RESPONSE:

Please refer to the generic response to NRC Question MEB 7(b) provided by GE in a May 5, 1978 letter from L. J. Solson (GE) to J. T. Knight (NRC).

QUESTION 110.7

Provide function verification when stress exceeds Service level C for Class 1 and level B for Class 2 and 3 piping components especially at tees, elbows and areas of structural discontinuity. We feel that meeting code limits does not necessarily assure functioning. For ASME Class 1 piping, the B indices, when used in Code Equation (9) and with the 1.5 Sm limit, are intended to restrict combinations of loading to those that are less than two-thirds of the limit load combinations. Therefore, when Equation (9) is used with the Service Limit C and D the limit load may be exceeded. Furthermore, the ASME Code Class 2 and 3 stress indices are based upon fatigue considerations, rather than limit loading.

You also referred to Paragraph NA 2142.2 of the ASME Code that discusses large deformations which are possible in areas of structural discontinuity stressed to Service Limit C and gross general deformations which are possible at Service Limit D. Although this does not imply that large deformations will occur in every case where Service Limit B is exceeded, it is our position that an approach such as the following be used:

You should examine areas of structural discontinuity, in the context of the geometry and stresses in the system in which they exist, to insure that structural collapse cannot occur at either the equipment nozzles or in the piping. Examples of possible collapse modes are situations, such as:

(1) A piping system with a cantilevered length of straight pipe where the formation of one hinge would lead to gross plastic deformation, and

(2) A piping system with two anchors, where three points stressed to Service Limits C or D could form hinges and lead to gross plastic deformation.

If a possible collapse mode is identified a sufficiently detailed analysis should be performed to insure that functional capability is not impaired.

For further explanation of our position on Service Limits, operability assurance, and functional capability, see Attachment 1.

The November 10, 1977 report by Sargent & Lundy is inadequate to verify functional capability of ASME piping components due to the following reasons:

(1) The report does not adequately address the differences in material properties between carbon steel and stainless steel, which is substantially weaker in load bearing capability.

(2) The report relies heavily on the limited test data of reference 5, which did not provide any measurement on actual yield strength of the piping material. The use of yield stresses specified by code, the lower bound valves, may overly estimate the margins and cause non-conservative results.

<u>RESPONSE</u>:

Please refer to the generic response to NRC Question MEB 8 provided by GE in a May 5, 1978 letter from L. J. Solson (GE) to J. T. Knight (NRC).

QUESTION 110.8

Expand Sections 3.6.2.1.1.1a through d (page 3.6-10) and 3.6.2.1.1.b 1 through 3 (pages 3.6-11a and 12) to indicate the plant operating conditions to be considered in the evaluation of equations (10), (12), and (13) of NB-3653.

RESPONSE:

For response, see Subsections 3.6.2.1.1. a, b, c, and d; and 3.6.2.1.1.b. 1 through 4).

QUESTION 110.9

Expand Sections 3.6.2.1.1.1e (Page 3.6-10) and 3.6.2.1.1.b 4 (page 3.6-12) to indicate that the maximum stress range is calculated by the sum of equations (9) and (10) of NC-3652.

RESPONSE:

For response, refer to Subsections 3.6.2.1.1.1., part e and 3.6.2.1.1.b, parts 5 and 6).

QUESTION 110.10

It is the Staff's position that piping between the containment isolation valves for which no breaks are postulated shall receive a 100 percent volumetric examination of all circumferential, longitudinal, and branch to main run welds during each inspection interval (IWA-2400 of the ASME Code).

Expand Sections 3.6.2.1.1.7 (page 3.6-11), 3.6.2.5.1.2 (page 3.6-11), 5.2.4.7 (page 5.2-40) and 6.6.8 (page 6.6-4) to provide a commitment to such an augmented inservice inspection program.

RESPONSE:

For response see revised Subsections 3.6.2.1.1, 5.2.4.7 and 6.6.8.

QUESTION 110.11

Expand Sections 3.6.2.1.1.b1 through 3 (pages 3.6-11a and 12) and 3.6.2.1.1.b4 (page 3.6-12) to indicate how postulated pipe break locations are chosen when less than two intermediate locations are required by the stress and usage factor criteria.

RESPONSE:

For response, see Subsection 3.6.2.1.1.b, parts 4, 6 and 8.

QUESTION 110.12

Complete Section 3.6.2.1.1.b4 (page 3.6-12) by indicating the subscripts in the expression at the end of the section.

RESPONSE:

For response, see Subsection 3.6.2.1.1.b, Part 5.

QUESTION 110.13

Expand Sections 3.6.2.1.1.a and b (pages 3.6-11 through 12) to indicate the criteria used for postulating break locations in high energy piping not designed to seismic Category I standards.

RESPONSE:

For response, refer to Subsection 3.6.2.1.1.b, parts 7 and 8.

QUESTION 110.14

Expand Section 3.6.2.1.2.2 (pages 3.6-12a and 13) to indicate the criteria used for postulating cracks in moderate energy ASME Class 1 piping.

RESPONSE:

Refer to Subsection 3.6.2.1.2, parts 2 and 4 for response.

QUESTION 110.15

Expand Sections 3.6.2.1.2.2 (pages 3.6-12a and 13) and 3.6.2.1.2.4 (page 3.6-13) to include a definition of maximum stress range.

RESPONSE:

Refer to Subsection 3.6.2.1.2, parts 2) and 4) for response.

QUESTION 110.16

Expand Sections 3.6.2.1.2 (pages 3.6-12a and 13) to indicate the criteria used for postulating cracks in moderate energy piping not designed to seismic Category I standards.

RESPONSE:

For response, see Subsection 3.6.2.1.2, part 5.

QUESTION 110.17

Expand Sections 3.6.2.1.3 (pages 3.6-13 and 14) and 3.6.2.1.4.6.4c (pages 3.6-17 and 18) to indicate how consideration of the maximum stress range is used to exempt certain break orientations when the postulated break location is due to a usage factor in excess of 0.1.

RESPONSE:

Both circumferential and longitudinal breaks are postulated in fluid system piping other than the Recirculation piping system when the postulated break location is due to a usage factor in excess of 0.1 as discussed in Subsection 3.6.2.1.1.

QUESTION 110.18

Section 3.6.2.1.4.3 (page 3.6-15) states that a "...pipe break or crack outside the containment (is not) postulated concurrently with a postulated pipe break inside containment."

Section 3.6.1.1 (first paragraph on page 3.6-2) states that "A design basis for Susquehanna SES is that a postulated pipe break inside containment (up to and including a rupture of the recirculation piping), in conjunction with the SSE,...will not prevent the plant from..."being able"...to shut down the reactor safely and maintain it is a safe shutdown condition."

Resolve the conflict between these two sections recognizing that an SSE is assumed to cause the failure of piping which is not designed to seismic Category I standards.

RESPONSE:

BTP APCSB 3-1 paragraph B.3.a., titled Analysis and Effects of Postulated Piping Failures states that "...each...break in high-energy fluid system piping or leakage crack in moderate-energy fluid system piping should be considered separately as a single postulated initial event occurring during normal plant conditions." For purposes of piping failure analysis only one initial event is postulated during normal plant conditions. That is, for any single postulated seismic Category I or II pipe break, the plant is assumed to be in a normal plant condition (no SSE) with only that single postulated pipe break.

For conservatism and defense-in-depth, SSES has committed to the design basis stated in Subsection 3.6.1.1 of the FSAR that for inside the containment pipe breaks would be postulated in conjunction with an SSE. But once again for this particular case, only a single pipe break (the LOCA) was considered in the evaluation of the piping failure.

It should be pointed out that the combination of a LOCA and an SSE is used as a loading combination for the design of systems, components and structures required to bring about a safe shutdown. But except as noted above, this combination is not used to analyze the effects of postulated piping breaks. In all cases the design basis includes the requirement to be able to bring the reactor to a safe shutdown and maintain it in a safe shutdown condition without taking credit for operation of non-seismic Category I equipment.

QUESTION 110.19

Section 3.6.1.4.4 (page 3.6-15) identifies the criteria used to exempt certain postulated pipe breaks from consideration of pipe whip.

Verify that the other affects (such as jet impingement, pressure, temperature, humidity, wetting of all exposed equipment, flooding) of such breaks are considered.

RESPONSE:

See revised Subsection 3.6.1.4.4.

QUESTION 110.20

Expand Section 3.6.2.1.4.6.(5) (page 3.6-18) to describe the "mechanistic approach" used to justify longitudinal breaks with a break area less than the flow area of the pipe.

RESPONSE:

The "mechanistic approach" was not used in the analysis of longitudinal breaks postulated in recirculation system piping. In all cases the equivalent longitudinal break area was taken to be the flow area of the pipe.

QUESTION 110.21

Expand Section 3.6.2.2.2.1.3 (page 3.6-20) to describe the "experimental data or analytical theory" used to justify crack opening times exceeding one millisecond.

RESPONSE:

For recirculation system piping postulated breaks, crack opening times were assumed to be not more than one millisecond.

QUESTION 110.22

Expand Section 3.6.2.2.2.a.6 (page 3.6-24) and 3.6.2.3.2.2.1a(ii) (page 3.6-27) to indicate what limits will be used, and how they will be used, to ensure operability of essential components.

RESPONSE:

One of the motor operated gate valves in the recirculation piping system may be required to operate during accident conditions. To ensure operability, combination of analysis and testing of the discharge gate valve is described below:

In the recirculation loop, only recirculation discharge gate valves are required to be operable for the safe shutdown of the plant in the case of a recirculation line suction nozzle break. Analysis results of the valve body, bonnet and yoke under the limiting loading conditions indicate that the deformations do not exceed the elastic limit of the materials. Hence, this assures that the components will return to their original position after the loads are removed. Since these discharge valves are required to operate after the LOCA induced loads are not present, the operability of the valve is assured. In addition, the representative Susquehanna SES motor operators designated by the vendor as one SMB family have been qualified for operability under expected environments and loading conditions. Therefore the above analysis and testing adequately assures that the recirculation loop discharge gate valve will be operable when required to operate.

During the LOCA loading on the valve, there is no binding that prevents valve operation when the LOCA load is released.

See also response to Question 110.45.

QUESTION 110.23

Expand Sections 3.6.2.2.2 (pages 3.6-20 through 25), 3.6.2.3 (pages 3.6-25 through 30) and 3.6.2.5.1.4 (page 3.6-31) to describe the protection criteria for the effects due to jet impingement.

RESPONSE:

Subsection 3.6.1.1 provides a description of the protection criteria for the effects due to jet impingement; specifically that portion that reads:

The failure of piping containing high energy fluid may lead to damage of surrounding systems and equipment. The effects of such a failure including pipe whip, fluid jet impingement, flooding, compartment pressurization, and environmental effects require special consideration to ensure the following:

a) The ability to shut down the reactor safely and maintain it in a safe shutdown condition.

b) Containment integrity.

c) A pipe break which is not a loss of reactor coolant must not cause a loss of reactor coolant.

d) Resultant doses are below the guideline values of 10CFR50.67.

This criteria is applicable to those piping systems described in Subsection 3.6.2.2, 3.6.2.3, and 3.6.2.5 as well as all other high or moderate energy piping systems.

QUESTION 110.24

Expand Section 3.6.2.3.1 (pages 3.6-25 and 26) to indicate how maintaining stress below the yield strength of the material ensures operability of the valve.

RESPONSE:

For response, see Subsection 3.6.2.3.1, Dynamic Analysis Methods to Verify Integrity and Operability.

QUESTION 110.25

Expand Section 3.6.2.3.2.2.(2) (page 3.6-28) to indicate what and how the displacement effects on structures and other systems and components are analyzed.

RESPONSE:

See revised Subsection 3.6.2.3.2.2.

Pipe displacements for the postulated ruptures of the Recirculation Piping System after the postulated break locations have been provided.

Pipe displacement effects on components (nozzles, valves, tees, etc.) which are in the same piping run that the break occurred in are addressed in section 3.6.2.3.2.2 (see response to Q 110.22).

QUESTION 110.26

Sections 3.6.2.5.2.1 (page 3.6-31) and 3.6.2.5.2.3 (page 3.6-32) are indicated as "Later." Provide a schedule for their inclusion in the FSAR.

RESPONSE:

See revised Subsection 3.6.1 and 3.6.2.

QUESTION 110.27

It is the Staff's position that a branch connection to a main run need not be considered as a terminal end when all of the following are met:

(1) The branch and main runs are of comparable size and fixity (i.e., the nominal size of the branch is at least one half of that of the main);

(2) The intersection is not rigidly constrained to the building structure; and

(3) The branch and main runs are modeled as a common piping system during the piping stress analysis.

Expand the definition of "Terminal Ends" in Section 3.6.3 (page 3.6-33) to correspond with this definition.

RESPONSE:

For response, see Subsection 3.6.3.

The identification of branch runs and terminal ends is consistent with the intent of Regulatory Guide 1.46 and AND Standard 58.2. For the GE scope of supply recirculation piping system, the criteria of items 1, 2 and 3 in Question 110.27, are satisfied, without exceptions.

QUESTION 110.28

Expand Table 3.6-1 to include the following systems:

(1) Main stream drains
(2) Head vent
(3) Head spray

RESPONSE:

Systems (1) and (2) have been added to revised Table 3.6-1. System (3) is listed in the table under "Residual Heat Removal" system.

QUESTION 110.29

Figures 3.6-1, 3.6-1A, 3.6-1B, 3.6-1C, 3.6-1D, 3.6-2, 3.6-3, 3.6-4, 3.6-5, 3.6-6, 3.6-7, 3.6-8, 3.6-8a.1, 3.6-8a.2, 3.6-8a.3, 3.6-8b, 3.6-8c, 3.6-8d, 3.6-8e, 3.6-9 and 3.6-14 are indicated as "Later." Provide a schedule for their inclusion in the FSAR.

RESPONSE:

See revised Figures 3.6-1, 3.6-1A, 3.6-1B, 3.6-1C, 3.6-1D, 3.6-2, 3.6-3, 3.6-4, 3.6-5, 3.6-6, 3.6-7, 3.6-8, 3.6-8a.1, 3.6-8a.2, 3.6-8a.3, 3.6-8b, 3.6-8c, 3.6-8d and 3.6-8e.

Figure 3.6-9 has been intentionally left blank.

Figure 3.6-14 has been provided.

QUESTION 110.30

Provide a listing of moderate energy lines as required by Section 3.6.1.2 of the Standard Format (Regulatory Guide 1.70, Revision 2).

RESPONSE:

See revised Subsection 3.6.1.2.

QUESTION 110.31

Appendix 3.6A (last paragraph on page 3.6A-10) cites a draft ANSI Standard as the method of calculating break flow rates. As such draft documents are subject to frequent changes during their development, they are not acceptable as a reference. Therefore, expand the appropriate section to specifically identify the criteria used.

RESPONSE:

For this information refer to Section 3.6A.

QUESTION 110.32

This question replaces item 112.6 which is deleted. Previous analyses for other nuclear plants have shown that certain reactor system components and their supports may be subjected to previously underestimated asymmetric loads under the conditions that result from the postulation of ruptures of the reactor coolant piping at various locations. It is therefore necessary to reassess the capability of these reactor system components to assure that the calculated dynamic asymmetric loads resulting from these postulated pipe ruptures will be within the bounds necessary to provide high assurance that the reactor can be brought safely to a cold shutdown condition. The reactor system components that require reassessment shall include:

a. Reactor pressure vessel
b. Core supports and other reactor internals
c. Control rod drives
d. ECCS piping that is attached to the primary coolant piping
e. Primary coolant piping
f. Reactor vessel supports

The following information should be included in the FSAR about the effects of postulated asymmetric LOCA loads on the above-mentioned reactor system components and the various cavity structures.

1. Provide arrangement drawings of the reactor vessel support systems in sufficient detail to show the geometry of all principal elements and materials of construction.

2. If a plant-specific analysis will not be submitted for your plant, provide supporting information to demonstrate that the generic plant analysis under consideration adequately bounds the postulated accidents at your facility. Include a comparison of the geometric, structural, mechanical, and thermal-hydraulic similarities between your facility and the case analyzed. Discuss the effects of any differences.

3. Consider all postulated breaks in the reactor coolant piping system, including the following locations:

    a. Steam line nozzles to piping terminal ends.
    b. Feedwater nozzle to piping terminal ends.
    c. Recirculation inlet and outlet nozzles to recirculation piping terminal ends.

Provide an assessment of the effects of asymmetric pressure differentials[1] on the systems and components listed above in combination with all external loadings including safe shutdown earthquake loads and other faulted condition loads for the postulated breaks described above. This assessment may utilize the following mechanistic effects as applicable.

a. limited displacement break areas

b. fluid-structure interaction

c. actual time-dependent forcing function

d. reactor support stiffness

e. break opening times

4. If the results of the assessment in item 3 above indicate loads leading to inelastic action in these systems or displacement exceeding previous design limits, provide an evaluation of the inelastic behavior (including strain hardening) of the material used in the system design and the effect on the load transmitted to the backup structures to which these systems are attached.

5. For all analyses performed, include the method of analysis, the structural and hydraulic computer codes employed, drawings of the models employed, and comparisons of the calculated to allowable stresses and strains or deflections with a basis for the allowable values.

6. Demonstrate that active components will perform their safety function when subjected to the combined loads resulting from the loss-of-coolant accident and the safe shutdown earthquake.

7. Demonstrate the functional capability of any essential piping when subjected to the combined loads resulting from the loss-of-coolant accident and the safe shutdown earthquake.

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[1] Blowdown jet forces at the location of the rupture (reaction forces), transient differential pressures in the annular region between the component and the wall, and transient differential pressures across the core barrel within the reactor vessel.

RESPONSE:

Section 3.9 of the FSAR has been revised to reassess the capability of specific reactor system components to withstand calculated dynamic asymmetric loads resultant to postulated ruptures of the reactor coolant piping. The Susquehanna SES analyses is plant specific for all General Electric scope reactor components. The submittal is patterned after the LaSalle submittal (Docket Nos. 50-373 and 374) and addresses the areas of concern identified in the question.

In addition, see revised Appendix 6A, Tables 6A-1(a), 6A-1(b), 6A-1cc), and 6A-1(d), and Figures 6A-1(a), 6A-1(b), 6A-2, 6A-3g, 6A-3h, 6A-3i, 6A-3j, 6A-3k, 6A-31, 6A-9 and 6A-10.

QUESTION 110.33

Your response to 110.2 referenced the generic Mark II response of May 5, 1978, to NRC question MEB-2. This response is not completely acceptable to the staff. We feel that the level of damping used should be associated with the piping and component service stress limit. The staff position was originally included in Enclosure 5 of the NRC Mark II Generic Acceptance Criteria transmitted by letter in September, 1978 to the three Lead Plants, Zimmer, LaSalle, and Shoreham. The staff position is repeated below:

(1) Use OBE damping when Service Limits A or B are designated.

(2) Use SSE damping only when Service Limits C or D are designated.

RESPONSE:

The level of damping used in the piping analysis is in accordance with the stated staff position, i.e.,

1) OBE damping valves as given in Regulatory Guide 1.61 are used for load cases with acceptance criteria service limits A or B.

2) SSE damping valves as give in Regulatory Guide 1.61 are used for load cases with acceptance criteria service limits C or D.

The damping values used in the piping analysis for OBE and SSE are 0.5 and 1.0 percentage of critical damping, respectively, as given in FSAR Section 3.7b, Table 3.7b-2.

For NSSS systems Regulatory Guide 1.61 establishes two levels of damping which are to be used in dynamic analysis. These two levels are related to conditions of design where the maximum nominal stresses are either 1/2 yield or beyond yield. The R.G. illustrates that OBE seismic loading is required to be no higher than normal or upset (Service Limit B) plant conditions; therefore, the use of 1/2 yield level of damping is appropriate. The SSE seismic loading is at faulted (Service Limit D) plant conditions where stresses are allowed to be at or exceeding the yield; therefore, the use of the higher damping level is appropriate.

To extend this concept to other combinations of dynamic loadings, the important unifying concept is the allowable stress levels. The ASME code has specified the component allowable stress levels for emergency and faulted conditions to

be at or above the material yield stress. Upset condition allowables are intended to correspond to below material yield stress. Thus the OBE and other dynamic events defined as an upset (Service Limit B) plant condition should use the lower damping value, while the SSE and other dynamic events defined as emergency (Service Limit C) or faulted (Service Limit D) conditions should use the higher damping value.

QUESTION 110.34

Your response to 110.6 referenced the generic Mark II response of May 5, 1978, to NRC question MEB 7(b). This response is not entirely acceptable to the staff. The staff position was originally included in Enclosure 3 of the NRC Mark II Generic Acceptance Criteria for Lead Plants. The staff position is repeated below:

The requirement in 10 CFR 100, Appendix A, paragraph VI (2) is that structures, systems and components of the nuclear power plant necessary for continued operation shall be designed to remain functional and within applicable stress and deformation limits when subjected to the effects of the vibratory motion of the Operating Basis Earthquake in combination with normal operating loads. Current staff review requirements to meet this section of the Regulations are that such structures, systems and components be designed within the Service Level B limits (formerly termed upset) of Section III, Division 1, of the ASME Code when subjected to the "OBE plus SRV" loading condition. This loading condition represents an anticipated operational occurrence; i.e., a condition of normal operation expected to occur during the life of the plant resulting from the following scenario: OBE, loss of offsite power, turbine trip, and actuation of an undetermined number of safety relief valves. As requested in question 110.42 following, provide a commitment to include this load combination in your design of safety related components.

RESPONSE:

The load combinations N+OBE+SRV will be included in the design of safety-related components as an upset condition.

Safety-related NSSS components are designed to Service Level B limits when subjected to the "OBE plus SRV" loading combination, without fatigue considerations.

QUESTION 110.35

Your response to 110.7 referenced the generic Mark II response of May 5, 1978, to NRC question MEB-8. This response is not completely acceptable to the staff. A staff position regarding functional capability of piping was included as Attachment B to Enclosure 5 of the NRC Mark II Generic Acceptance Criteria for Lead Plants. Subsequent discussions with the Mark II Owners Group has resulted in further refinement of the staff position.

We will require that PP&L provide assurance of the functional capability of safety related piping for the Susquehanna plant. Enclosure 110-1 provides one method acceptable to the staff for providing such assurance. If you choose to employ other criteria, sufficient information should be provided to demonstrate the conservatism of the proposed criteria.

RESPONSE:

For response, see revised Subsection 3.9.3.1.1.6.

The subject of functional capability of safety-related piping needed for the safe shutdown of the plant is a generic issue. This issue is addressed by GE and the Mark II containment group on the basis of the report:

E. Rodabaugh, "Functional Capability of Essential Mark II Piping."

This report has been submitted to the NRC for review.

QUESTION 110.36

The response to question 112.5 is not completely acceptable.

For reactor coolant pressure boundary components and supports, we have accepted the use of the square root of sum of squares methodology for combining dynamic responses resulting from LOCA and SSE. This acceptance is documented in NUREG-0484 "Methodology for Combining Dynamic Responses." At this time, we have not accepted the use of SRSS for combining responses from other combinations of dynamic loads and for other components and supports. Our review of the SRSS methodology is continuing and we are concentrating on the proposed Kennedy-Newmark criteria. The eventual outcome is expected to establish our position and criteria for general acceptance of response combination using SRSS methods.

We request that you provide in the FSAR a specific listing of all combinations of dynamic loads and all components for which combination of dynamic responses by the SRSS method is proposed. The listing should specifically include such loads as OBE inertia loads, OBE anchor point movement loads, SRV loads, turbine stop valve closure loads, Mark II containment hydrodynamic vibratory loads, SSE loads, and LOCA loads (including annulus pressurization).

RESPONSE:

All dynamic responses from such dynamic load events as LOCA and SRV when required to be combined with OBE or SSE, are combined in accordance with the Square Root of the Sum of the Squares (SRS method).

The FSAR documents the results of the hydrodynamic load evaluation on NSSS equipment. This includes a listing of all load combinations.

QUESTION 110.37

The FSAR contains several apparently conflicting statements regarding the consideration of OBE loads in the NSSS ASME Class 1 fatigue calculations.

In FSAR Section 3.9.1.1 it is stated that Table 3.9-4 lists the transients used in the fatigue analyses of Class 1 components and supports. Page 21 of Table 3.9-4 states that 60 maximum load cycles due to the OBE were considered for GE Class 1 piping. This is consistent with commitments in FSAR Section 3.7a.3.2. However, in 3.9.1.1 it is stated that the OBE was not considered in the fatigue analyses of some components such as control rod drives, CRD housings, incore housings, hydraulic control units, core supports, other reactor internals, reactor vessel, support skirt, shroud support, shroud plate, MSIV's, SRV's, recirculation pumps, and recirculation gate valves. These apparent exceptions conflict with FSAR statement in FSAR Section 3.7a.3.2 that "the OBE is an upset condition and therefore must be included in fatigue evaluations according to ASME Section III.

Provide clarification of the consideration of OBE loads for the NSSS ASME Class 1 components to resolve the apparent conflicts between the FSAR sections. As noted in Enclosure 110-2 OBE loads are to be evaluated against service level B requirements which include fatigue analyses.

RESPONSE:

For NSSS Safety Class 1 piping and equipment where applicable codes require it, the OBE is considered as an upset condition and -- as per NRC Enclosure 110.2 -- OBE loads are evaluated against Service Level B requirements which include fatigue analysis. One OBE intensity earthquake with 10 peak stress cycles is postulated for the fatigue evaluation. Necessary amendments have been made to Subsection 3.7a.3.2 and to the transients listed in Subsection 3.9.1.1 reflecting the above bases.

For main steam piping isolation valves, main steam piping safety/relief valves and reactor recirculation piping gate valves, seismic loads were used as a design basis, but the cycles associated with the OBE were not considered for the fatigue evaluation. This is in compliance with the requirements of ASME, Section III, NB-3500 for the normal-duty fatigue analysis of these valves. The applicable thermal cycles considered are listed in Subsection 3.9.1.1.

For the recirculation pumps, seismic loads were used as a design basis, but the applicable code requires no fatigue evaluation.

Tables 3.9-1 and 3.9-4 have also been amended for clarification.

QUESTION 110.38

There are apparent conflicting statements in the FSAR regarding the consideration of the OBE in the BOP ASME Class 1 component fatigue calculations. Page 20 of FSAR Table 3.9-15 states that 2 OBE's are assumed to occur, resulting in 600 maximum load cycles for the BOP ASME Class 1 piping fatigue calculations. However, it is stated in FSAR Section 3.7b.3.2 that 5 OBE's with a corresponding total of 50 maximum stress cycles are used for the BOP ASME Class 1 piping fatigue calculations.

Provide clarification of the consideration of the OBE loads for BOP ASME Class 1 components to resolve the apparent conflicts between the FSAR Sections.

RESPONSE:

Table 3.9-15 has been revised to clarify the consideration of the OBE Loads for BOP ASME Class I components and to resolve the conflict with Section 3.7b.3.2.

QUESTION 110.39

Provide confirmation that Mark II containment SRV discharge and suppression pool vibratory loads have been taken into account, i.e. load cases 1 and 2 of Enclosure 110-2, for determination of postulated pipe break locations in ASME Class 1, 2 and 3 piping using the stress and usage factor criteria specified in 3.6 of the FSAR.

RESPONSE:

Mark II containment SRV discharge and suppression pool vibratory loads will be considered in the determination of postulated pipe break locations in ASME Class 1, 2 and 3 high-energy piping using the stress and usage factor criteria specified in Section 3.6 of the FSAR.

Containment SRV discharge and suppression pool vibratory loads have been considered in determining postulated break locations in the ASME Class I Recirculation piping using the stress and usage factor criteria of Section 3.6. One exception is the combined loading of OBE and SRV loads where the fatigue usage is not included.

QUESTION 110.40

We have identified several portions of your vibration, thermal, and dynamic effects testing program for NSSS and BOP piping which deviate from the criteria of SRP section 3.9.2. we require certain additional information to more fully define your program. Modify FSAR sections 3.9.2.1a and 3.9.2.1b to provide this information for both NSSS and BOP systems.

(1) Expand your program to include the following piping systems, including their supports and restraints.

(a) All ASME Class 1, 2 and 3 systems,
(b) Other high energy piping systems inside seismic Category I structures,
(c) High energy portions of systems whose failure could reduce the functioning of any seismic Category I plant feature to an unacceptable safety level, and
(d) Seismic Category I portions of moderate energy piping systems located outside containment.

A visual check of many of these systems is acceptable.

(2) Describe how your program will verify that no restraint of normal thermal movement occurs in the systems listed in (1).

(3) Describe in more detail how your program will verify the adequate performance of snubbers for the systems listed in (1).

(4) You provide various references to "Code limits" and "endurance limits" for the allowable values against which measurements will be compared. Indicate how the acceptance criteria of your test program will be related to such limits.

(5) Provide a cross reference between FSAR section 3.9.2.1 and the appropriate test descriptions in FSAR Chapter 14.

RESPONSE:

The following response applies to piping in the NSSS scope of supply (recirculation and main steam).

1) No expansion of Subsection 3.9.2 is necessary for NSSS piping since recirculation and main steam piping are already addressed.

2) A thermal expansion preoperational and startup testing program, performed through the use of potentiometer sensors, has been established to verify that normal thermal movement occurs in the systems within the NSSS piping scope of supply. The main considerations of this program are as follows:

   a) The piping system during heatup and cooldown is free to expand and move without planned obstruction or restraint in x, y, and z directions.
   b) The piping system does "shakedown" after a few thermal expansion cycles.
   c) The piping system is working in a manner consistent with the assumption of the NSSS stress analysis.
   d) There is adequate agreement between calculated values of displacements and measured value of displacement.
   e) Assure consistency and repeatability in thermal displacements during heatup and cooldown of the NSSS systems.

Limits of thermal expansion displacements have been established prior to start of piping testing to which the actual measured displacements can be compared to determine acceptability of the actual motion.

If the measured displacement does not vary from the acceptance limits values by more than the specified tolerance, the piping system is responding in a manner consistent with predictions and is therefore acceptable. Two levels of limits of displacements have been established to check the systems. These are:

- Level 1 which is the maximum limits that specify level of pipe motion which if exceeded, make a test hold or termination mandatory.

  If a Level 1 limit is exceeded the plant will be placed in a satisfactory hold condition, and the responsible Piping Design Engineer will be advised. Following resolution applicable tests must be repeated to verify that the requirements of the Level 1 limits are satisfied.

- Level 2 is that specified level of pipe motion which is exceeded requires that the responsible Piping Design Engineer be advised.

If a Level 2 limit is not satisfied, plant operating and startup testing plans would not necessarily be altered. Investigations of the measurements and of the criteria and calculations used to generate the pipe motion limits would be initiated. An acceptable resolution must be reached by all appropriate and involved parties, including the responsible Piping Design Engineer of the affected piping system. Depending upon the nature of such resolution the applicable tests may or may not have to be repeated.

A walkdown of the piping and suspension shall be made to identify any obstruction or improperly operating suspension components. The instrumentation installation and calibration shall be checked and any discrepancies corrected. Snubbers shall be in their operating range about the midpoint of the total travel range at operating temperature. Hangers shall be in the operating range between the hot and cold settings.

3) See Subsection 3.9.2.1b.

4) The criteria for vibration displacements shall be based on assumed linear relationship between displacements, snubber loads and magnitude of applied loads for any function and response of system.

Thus the magnitude of limits of displacements, snubber loads, nozzle loads, are all proportional. Maximum displacements (Level 1 limits) are established to prevent the maximum stress in the piping systems from exceeding the normal and <u>upset primary stress limits</u> and/or the maximum snubber load from exceeding the maximum load to which the snubber has been tested.

Based on the above criteria, Level 1 displacement limits are established for all instrumented points in the piping system. These limits shall be compared with the field measured piping displacements. The method of acceptance is defined in the response to Question 2.

5) See Table 3.9-33.

QUESTION 110.41

Provide the following additional information regarding the dynamic analysis of reactor internals under faulted conditions:

a) Provide response time histories at one key location (having either the maximum stress combination or the most critical deflection combination, whichever is governing the design) for each of the following internal components:

(1) Jet pump
(2) Shroud wall
(3) Shroud head
(4) Control rod
(5) Instrumentation guide tube
(6) Core plate

For each location, separate response time histories for the various load effects associated with the SSE and most severe pipe break event should be provided.

b) If the method of response combination other than absolute sum is used for the combination of responses due to the various seismic and pipe break load effects in a. above, provide justification. This justification should address whether a particular response time history is considered static or dynamic, and if dynamic, should describe the predominate frequencies.

RESPONSE:

a) The following additional response time information for selected internal components under the SSE and the most severe pipe break accidents is provided.

Table 110.41-1 - Internals component vs. location list.

Figures 110.41-1 & 110.41-2 - Horizontal and vertical mathematical model figures.

Figures 110.41-3 thru 110.41-10 - Response time history plots due to horizontal SSE.

Figures 110.41-11 thru 110.11-16 - Response time history plots due to vertical SSE.

Figures 110.41-17 thru 110.41-24 - Response time history plots due to recirculation pipe break.

Figures 110.41-25 thru 110.41-32 - Response time history plots due to feedwater pipe break.

There are two sets of SSE responses due to either cracked or uncracked primary containment building. Judging from the results, the uncracked case is the more severe one.

As for the pipe break events, we have considered both the feedwater line break and the recirculation line break. Judging from the response time histories, it is not conclusive as to which one is governing. Hence, we have included both sets.

b) A finite element beam model was used to perform the structural analysis for the RPV pedestal, shield and RPV and internals to account dynamic loads due to a LOCA (pipe break effect). The loads considered include annulus pressurization (AP), jet impingement loads on vessel and shield wall, jet reaction loads, pipe restraint reaction loads and FW line reaction load (applied to containment). All the loads are applied as a force-time history at relevant nodes of the model simultaneously and a dynamic analysis performed to generate a dynamic response time history of various components due to a LOCA. Similarly, a seismic dynamic analysis on the same beam model is performed to generate seismic response time history of various components. Then the peak value of dynamic response due to LOCA and due to seismic load are combined based on the following:

Current practice in BWR design is to combine response time histories of two or more dynamic loads by SRSS or by absolute sum of peak magnitudes. SRSS is technically justified based on the facts that (i) the maximum peaks of individual responses are highly unlikely to coincide in time, (ii) the probability of significant exceedance of SRSS value is very small, and (iii) the dynamic reserve margin inherent in nuclear power plant structures (related to energy absorption capability of component) designed to meet ASME code stress limits is significantly greater than the static code design margin to protect against failure.

Extensive documentation which substantiates the above technical bases for the use of SRSS for Mark II applications has been submitted to the NRC in the form of reports generated by the Mark II Owner's group.

TABLE 110.41-1

COMPONENT VERSUS TIME HISTORY TABLE

Components	Horizontal Model* Time History Type(1)	Horizontal Model* Location(2)	Vertical Model** Time History Type(1)	Vertical Model** Location(2)
Core Plate	F	N-7 OF E-7 (C-2 OF E-7)	A	N-7
Shroud	 F M	N-58 OF E-31 (C-8 OF E-31) (C-12 OF E-31)	F	N-7 OF E-5 (C-7 OF E-5)
Shroud Head	 F M	N-22 OF E-22 (C-2 OF E-22) (C-6 OF E-22)	F	N-5 OF E-3 (C-7 OF E-3)
CRD Guide Tube and Housings Jet Pump	A A	N-7, N-55 N-47	A A	N-2, N-3 N-8

(1) A = Acceleration Time History
F = Force Time History
M = Moment Time History

(2) N = Node
E = Element
C = Load Component***

* Pertaining both to horizontal seismic and pipe break events.

** Pertaining to vertical seismic event only. The vertical pipe break events are judged to be insignificant.

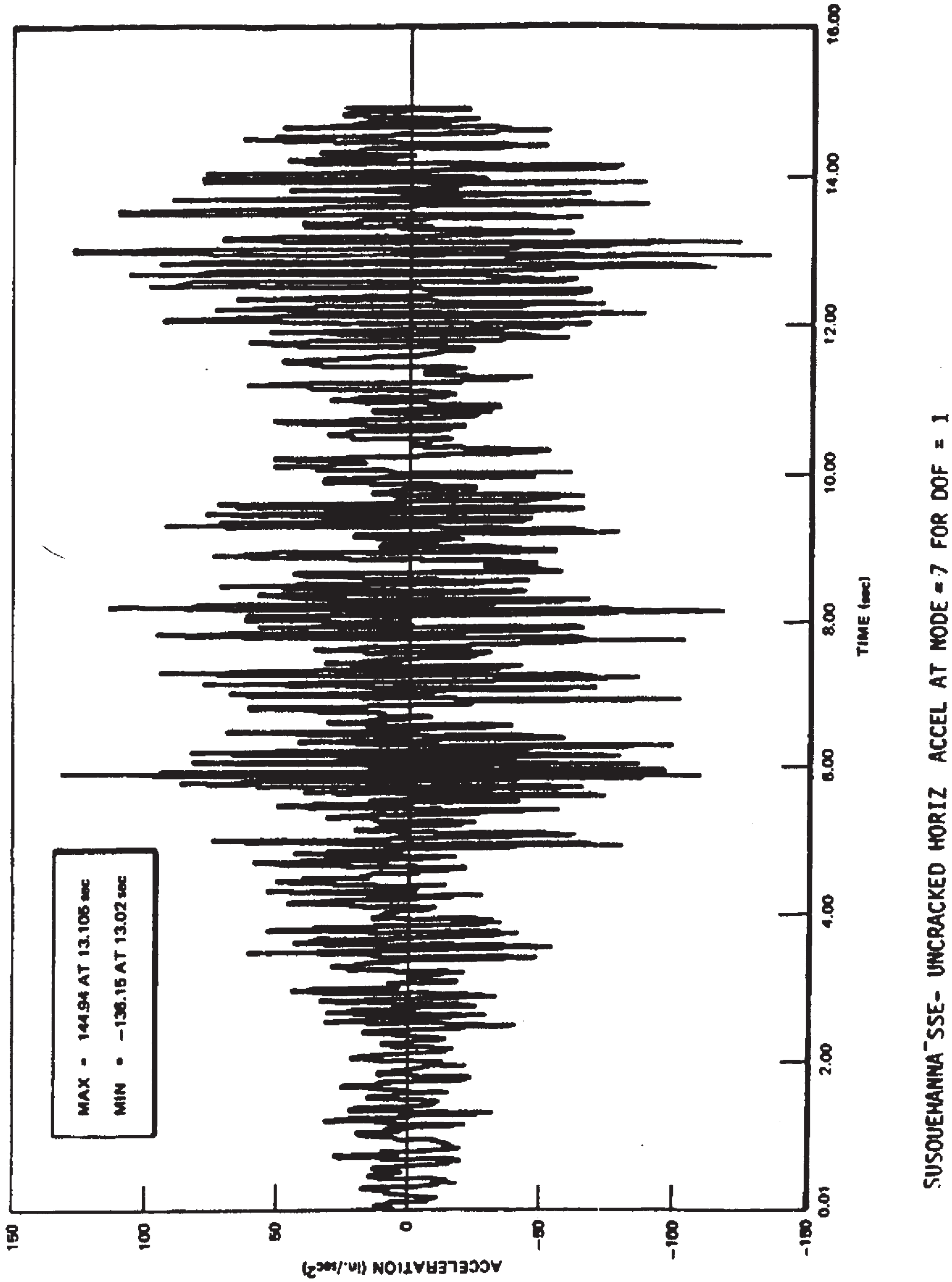
*** C-1 = Force at End I
C-2 = Shear at End I
C-3 = Shear at End I
C-4 = Torque at End I
C-5 = Moment at End I
C-6 = Moment at End I
C-7 = Force at End J
C-8 = Shear at End J
C-9 = Shear at End J
C-10 = Torque at End J
C-11 = Moment at End J
C-12 = Moment at End J

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
REACTOR PRESSURE VESSEL AND INTERNALS HORIZONTAL MATHEMATICAL MODEL FOR EARTHQUAKE AND ANNULUS PRESSURIZATION LOADING
FIGURE 110.41-1

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
REACTOR PRESSURE VESSEL AND INTERNALS VERTICAL DYNAMIC MODEL FOR EARTHQUAKE LOADING FIGURE 110.41-2



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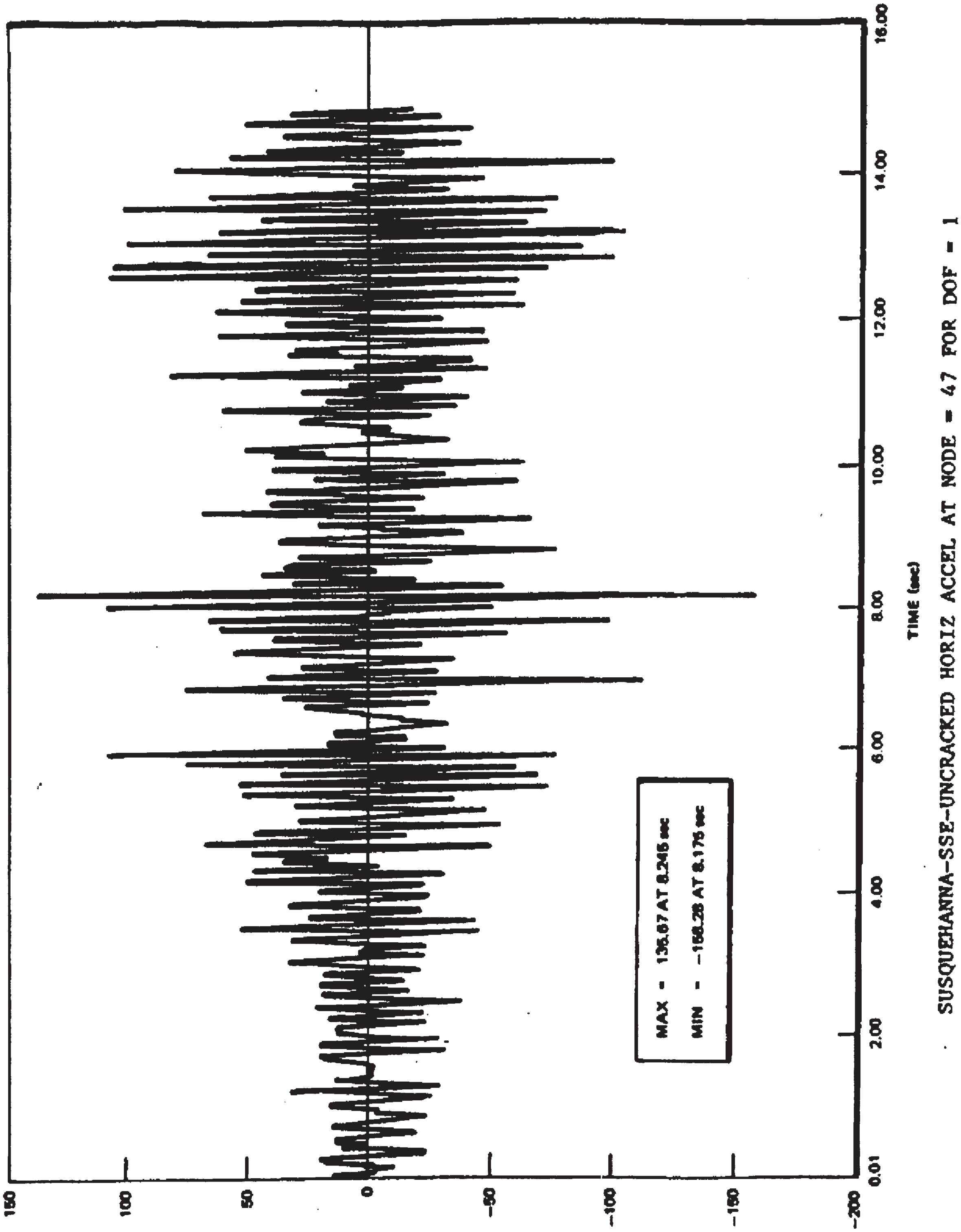
RESPONSE TIME HISTORY PLOT
DUE TO HORIZONTAL SSE

FSAR FIGURE 110.41-3

PP&L DRAWING



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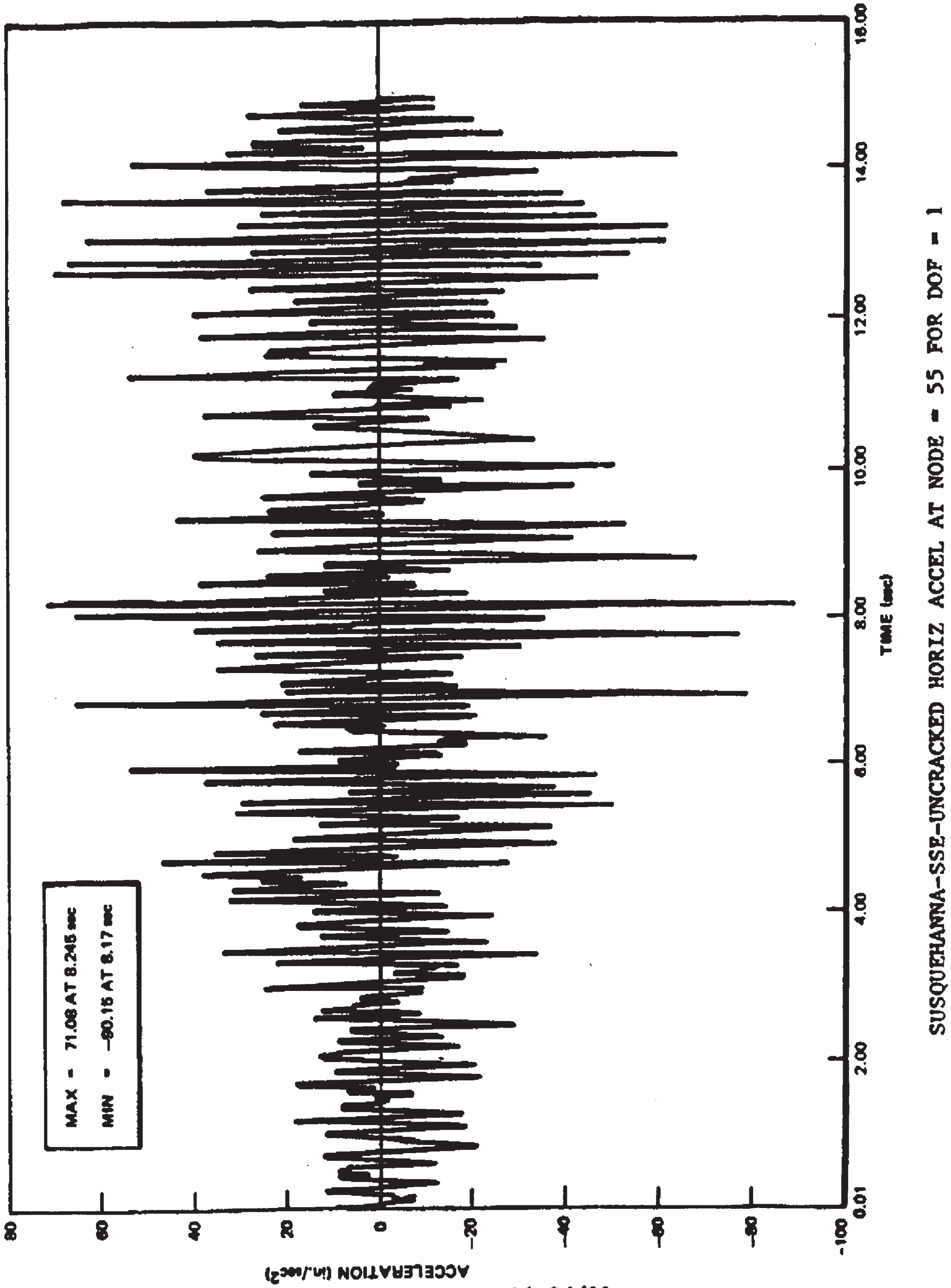
RESPONSE TIME HISTORY PLOT DUE TO HORIZONTAL SSE FSAR FIGURE 110.41-4

PP&L DRAWING



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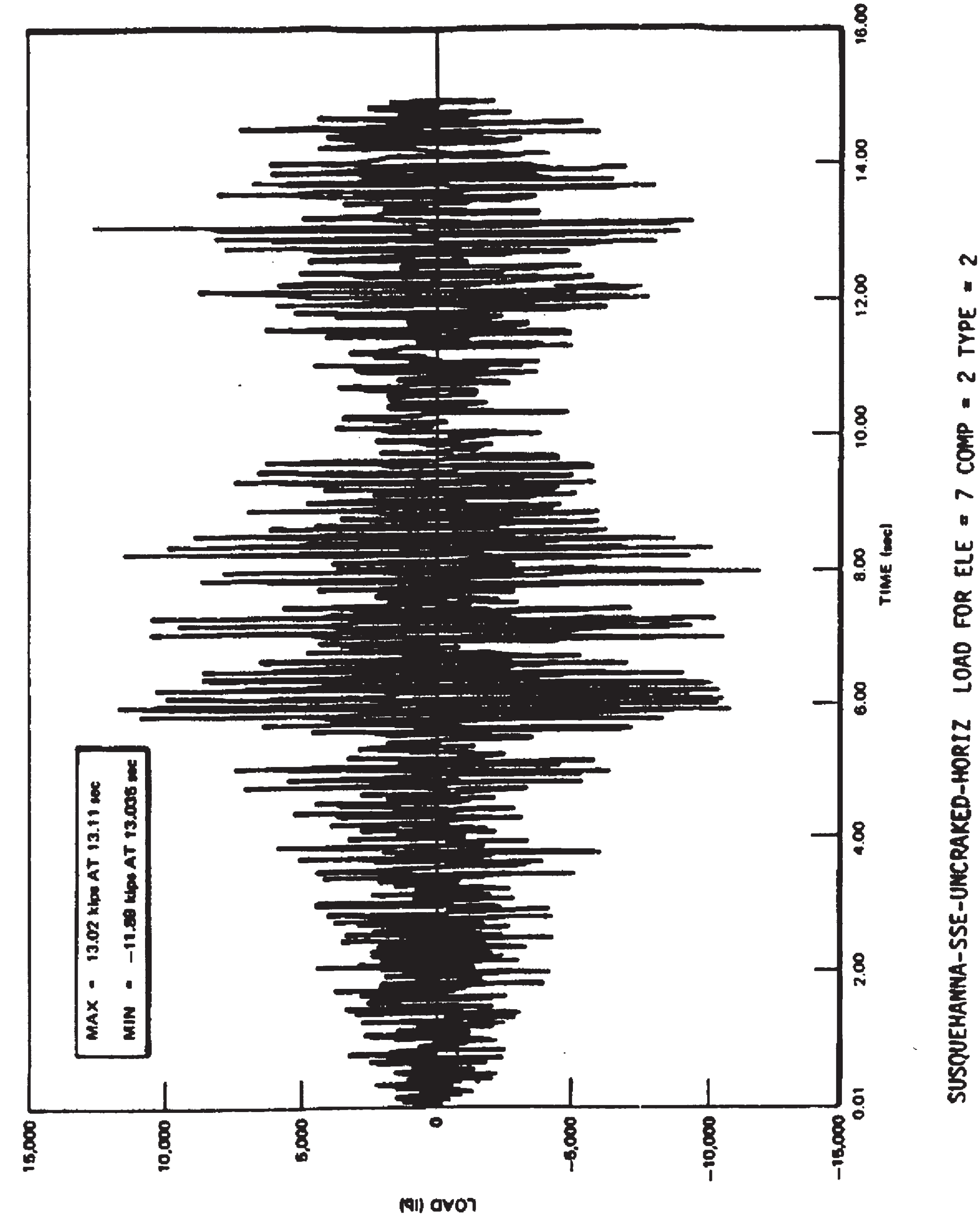
RESPONSE TIME HISTORY PLOT
DUE TO HORIZONTAL SSE

FSAR FIGURE 110.41-5

PP&L DRAWING



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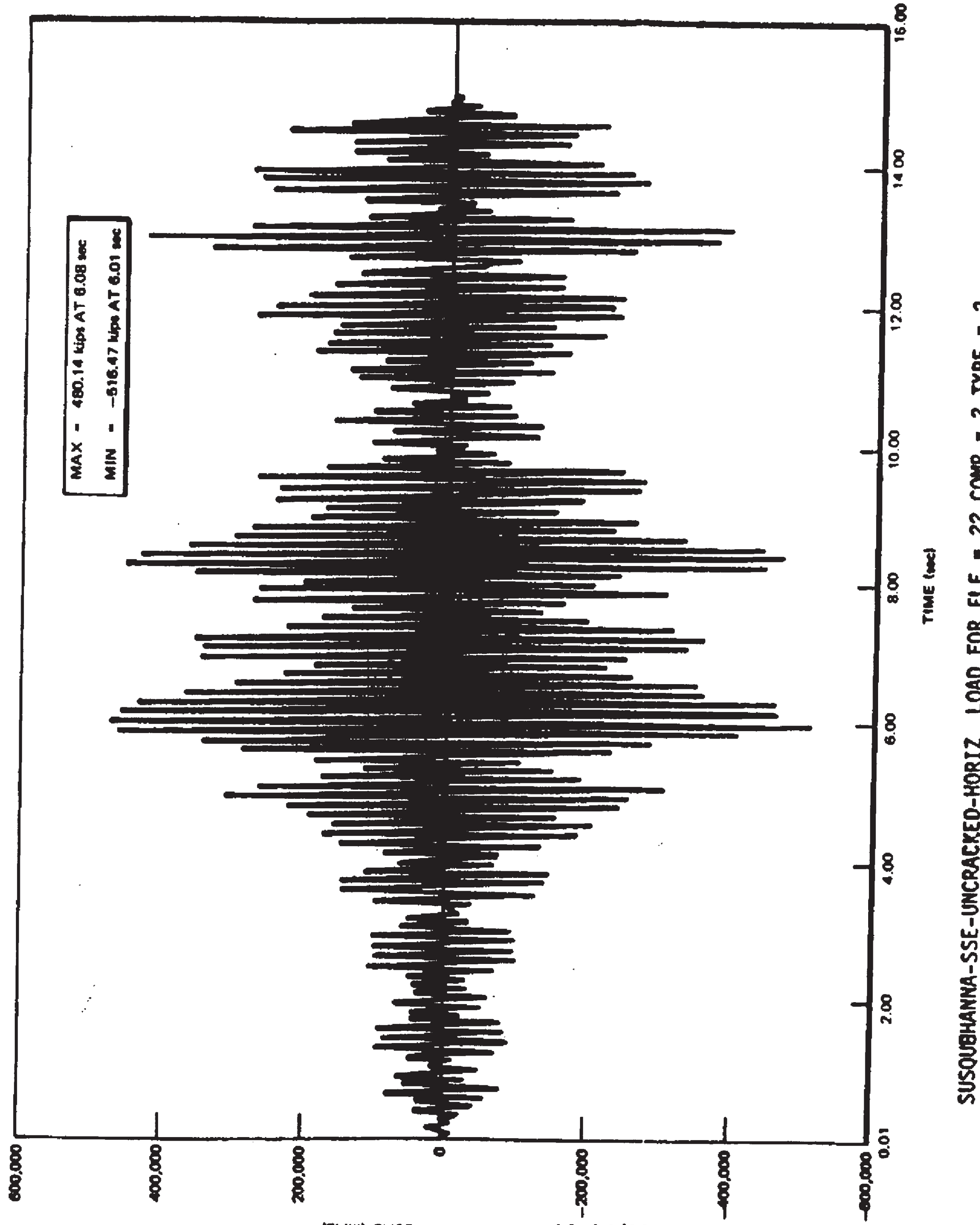
RESPONSE TIME HISTORY PLOT
DUE TO HORIZONTAL SSE

FSAR FIGURE 110.41-6

PP&L DRAWING



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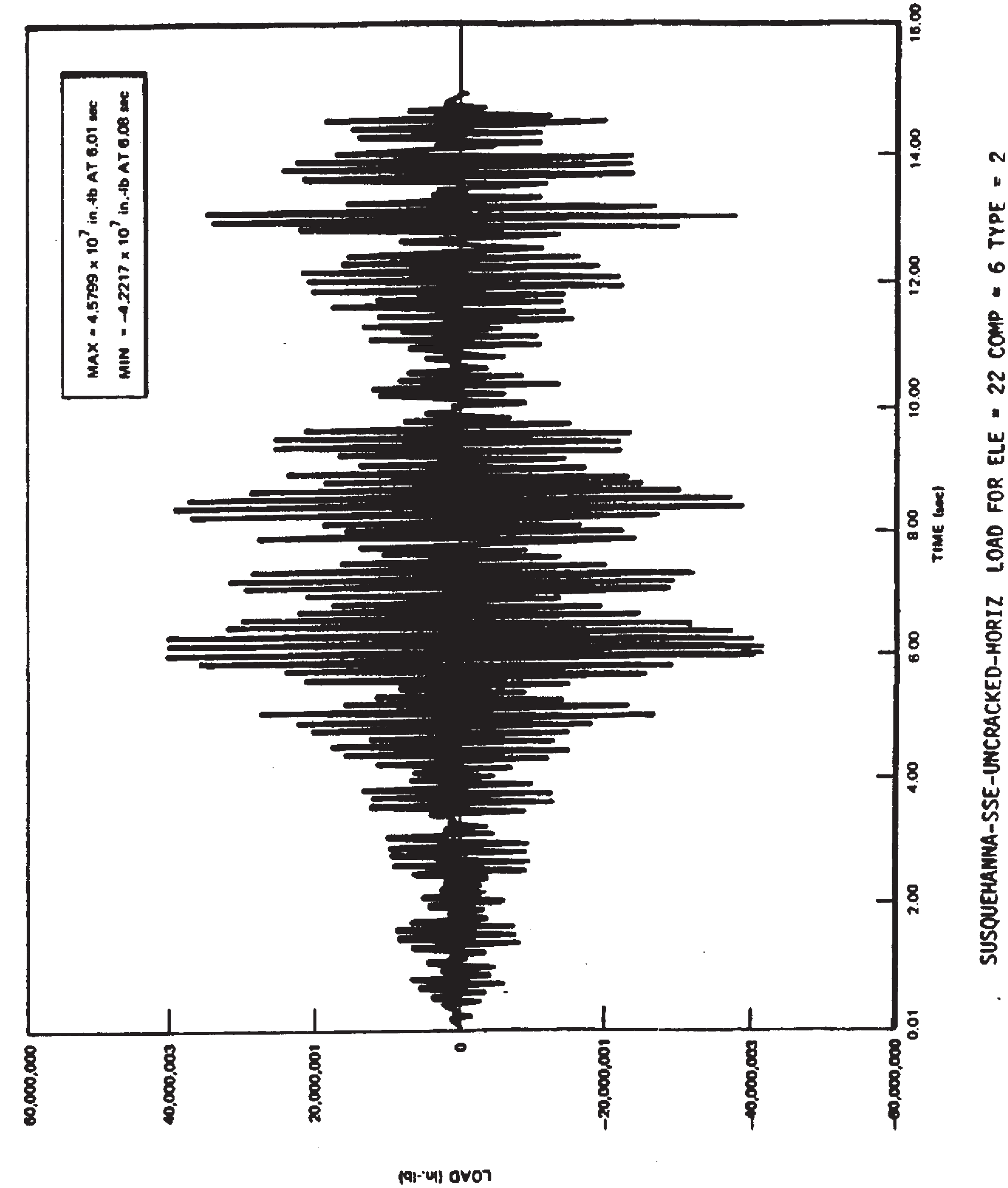
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT DUE TO HORIZONTAL SSE

FSAR FIGURE 110.41-7

PP&L DRAWING



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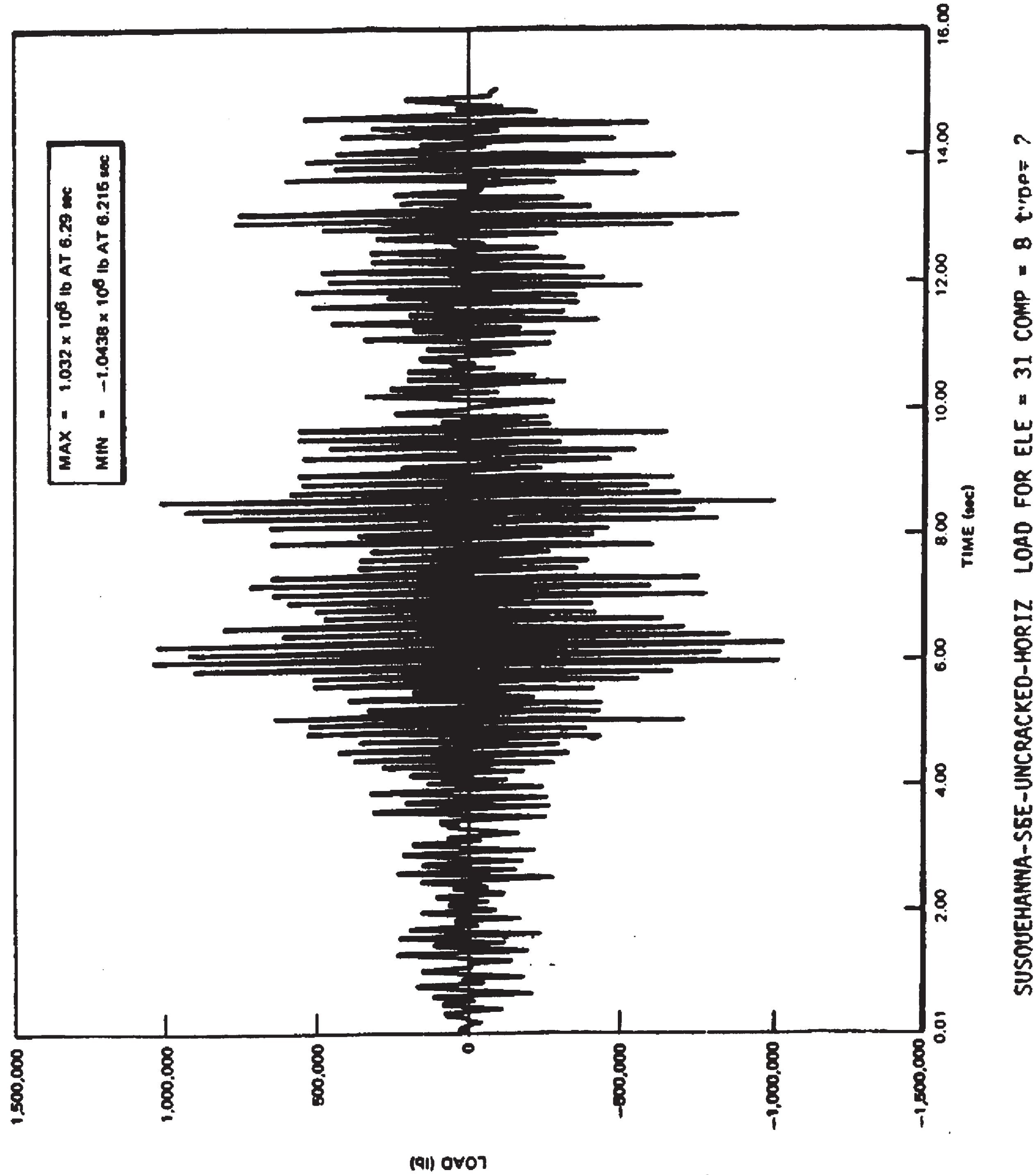
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO HORIZONTAL SSE

FSAR FIGURE 110.41-8

PP&L DRAWING



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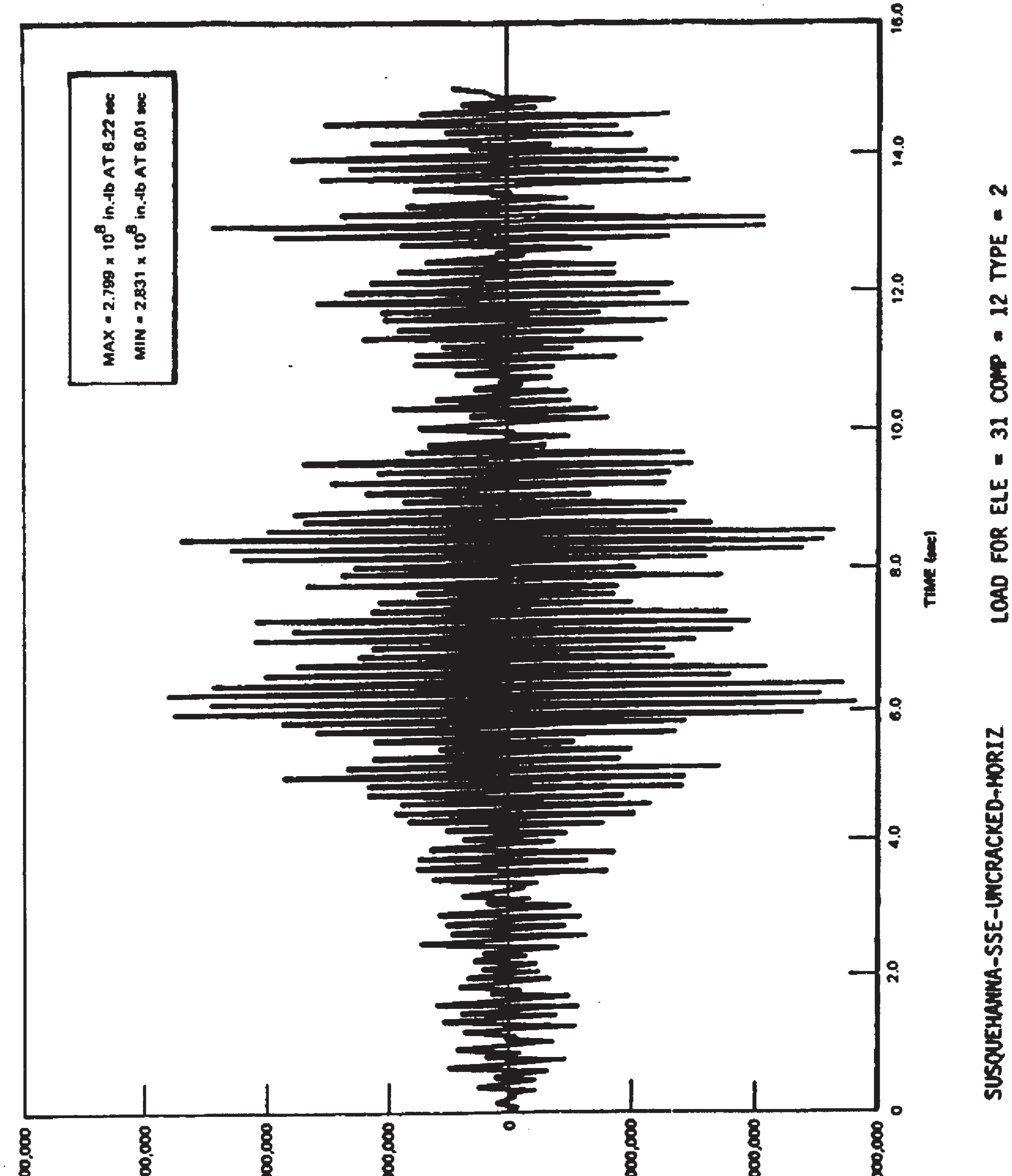
RESPONSE TIME HISTORY PLOT
DUE TO HORIZONTAL SSE

FSAR FIGURE 110.41-9

PP&L DRAWING



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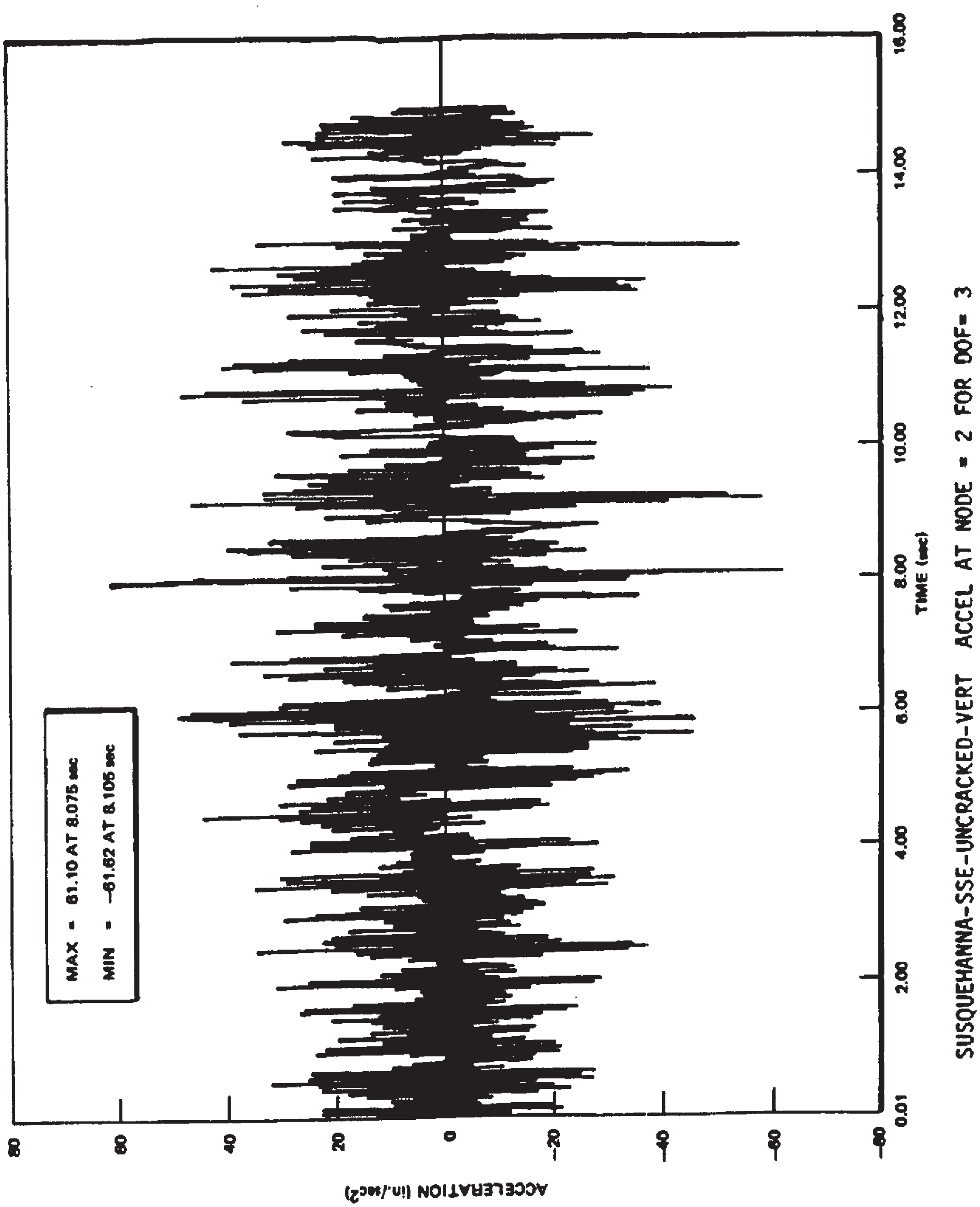
SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 AND 2 FINAL SAFETY ANALYSIS REPORT
RESPONSE TIME HISTORY PLOT DUE TO HORIZONTAL SSE
FSAR FIGURE 110.41-10

PP&L DRAWING



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SUSQUEHANNA STEAM ELECTRIC STATION
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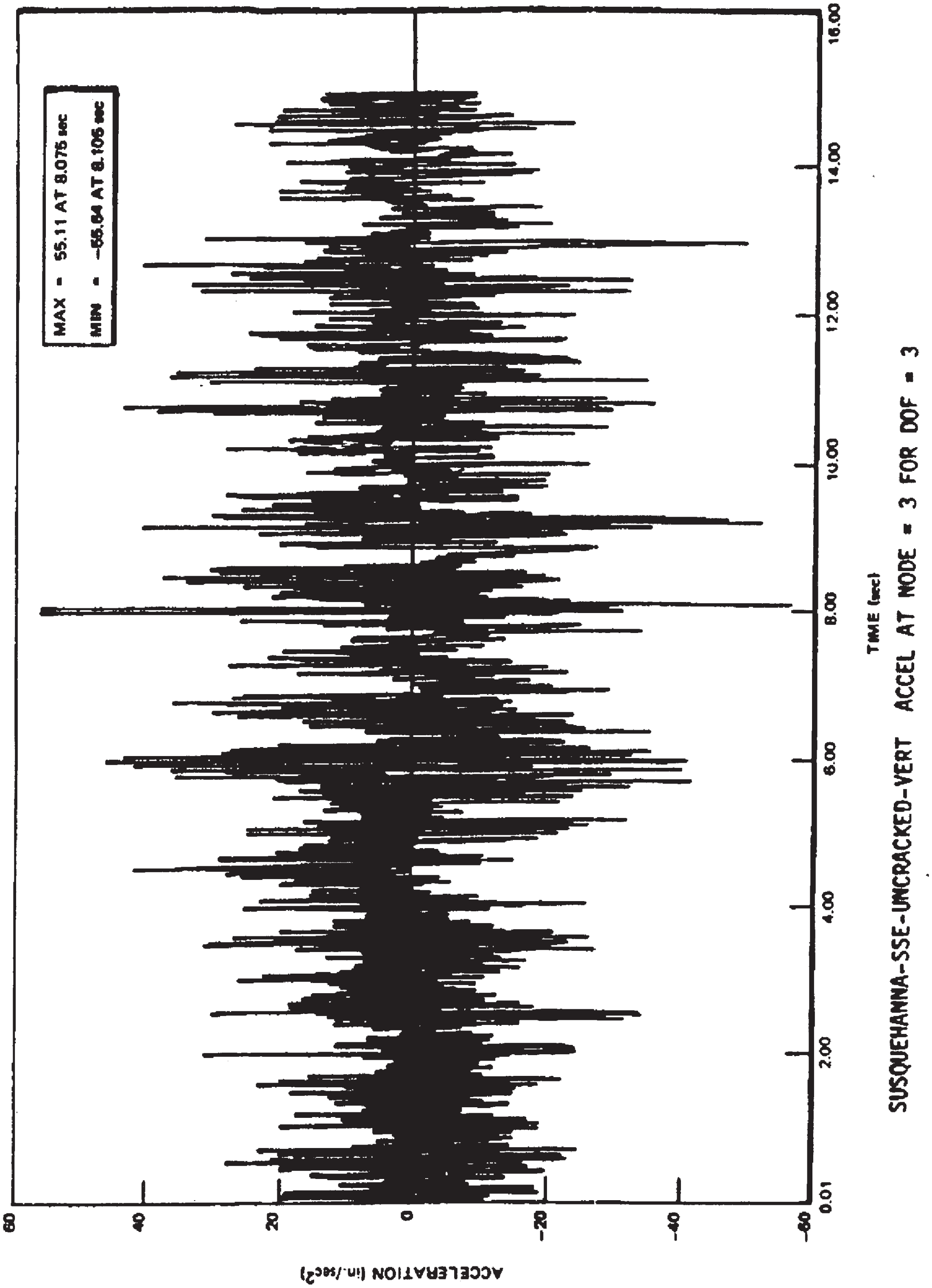
RESPONSE TIME HISTORY PLOT
DUE TO VERTICAL SSE

FSAR FIGURE 110.41-11

PP&L DRAWING



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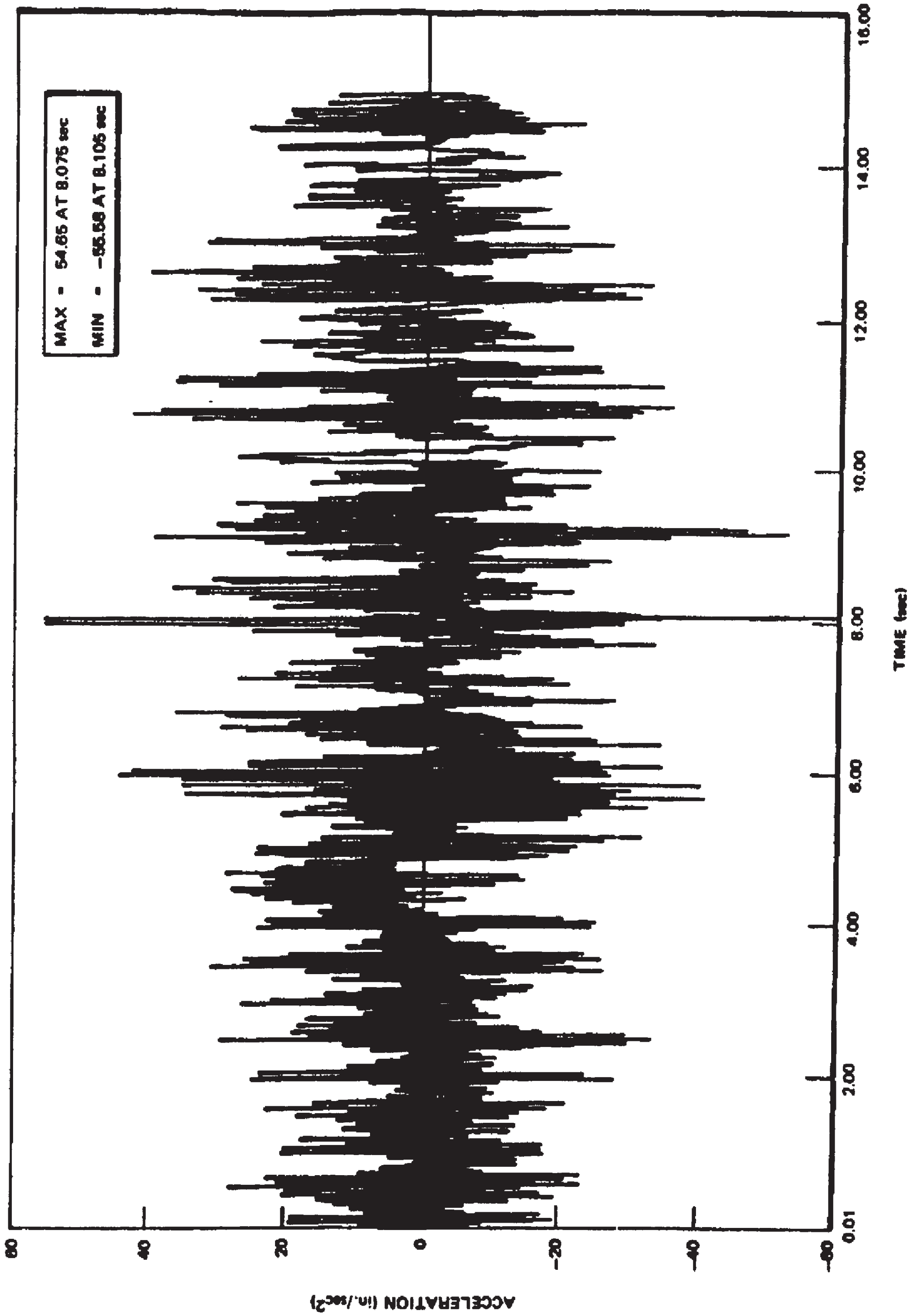
RESPONSE TIME HISTORY PLOT
DUE TO VERTICAL SSE

FSAR FIGURE 110.41-12

PP&L DRAWING



SUSQUEHANNA-SSE-UNCRACKED-VERT ACCEL AT NODE = 7 FOR DOF = 3

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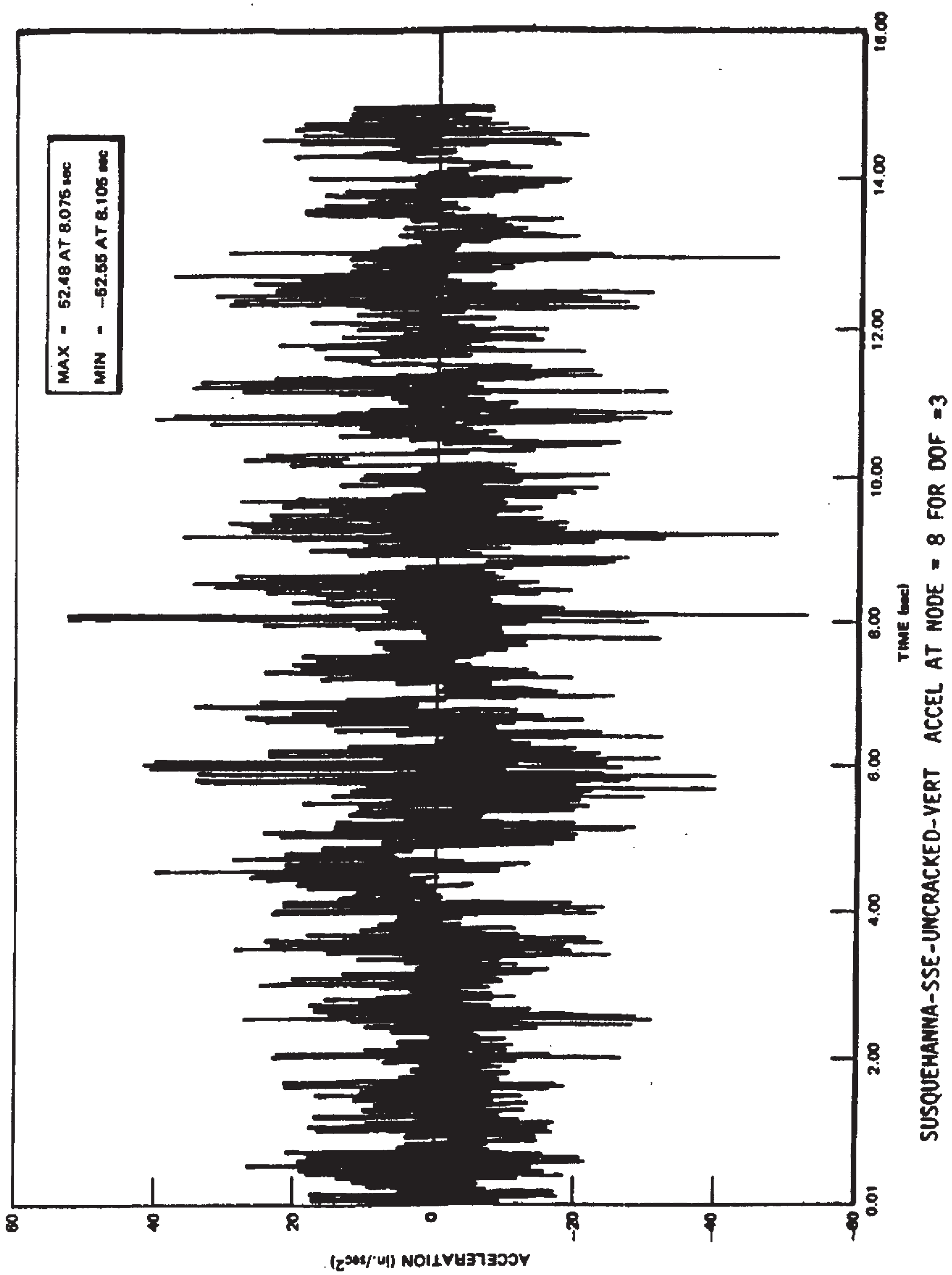
SUSQUEHANNA STEAM ELECTRIC STATION **UNITS 1 AND 2** **FINAL SAFETY ANALYSIS REPORT**
RESPONSE TIME HISTORY PLOT DUE TO VERTICAL SSE **FSAR FIGURE** 110.41-13

PP&L DRAWING



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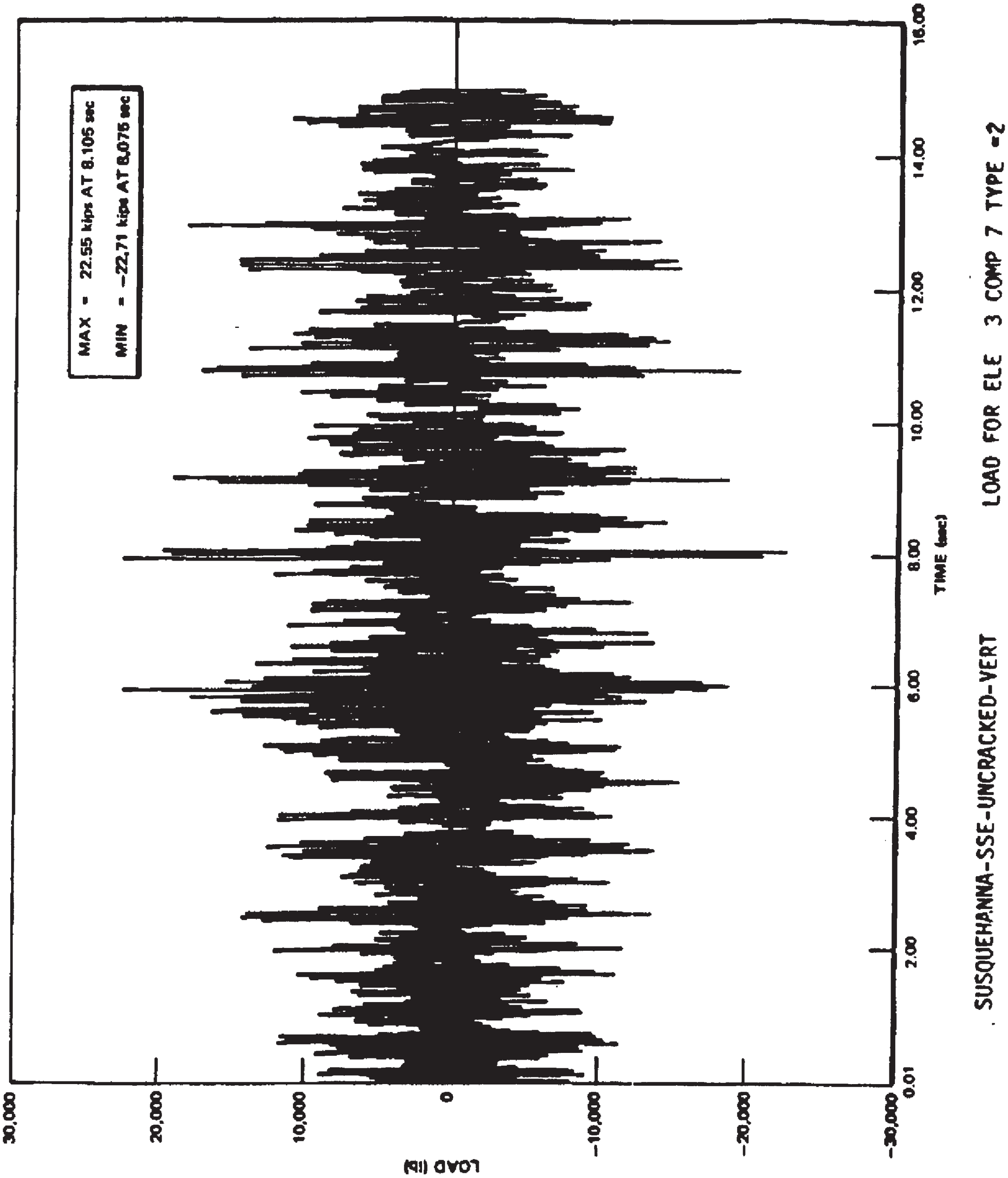
RESPONSE TIME HISTORY PLOT
DUE TO VERTICAL SSE

FSAR FIGURE 110.41-14

PP&L DRAWING



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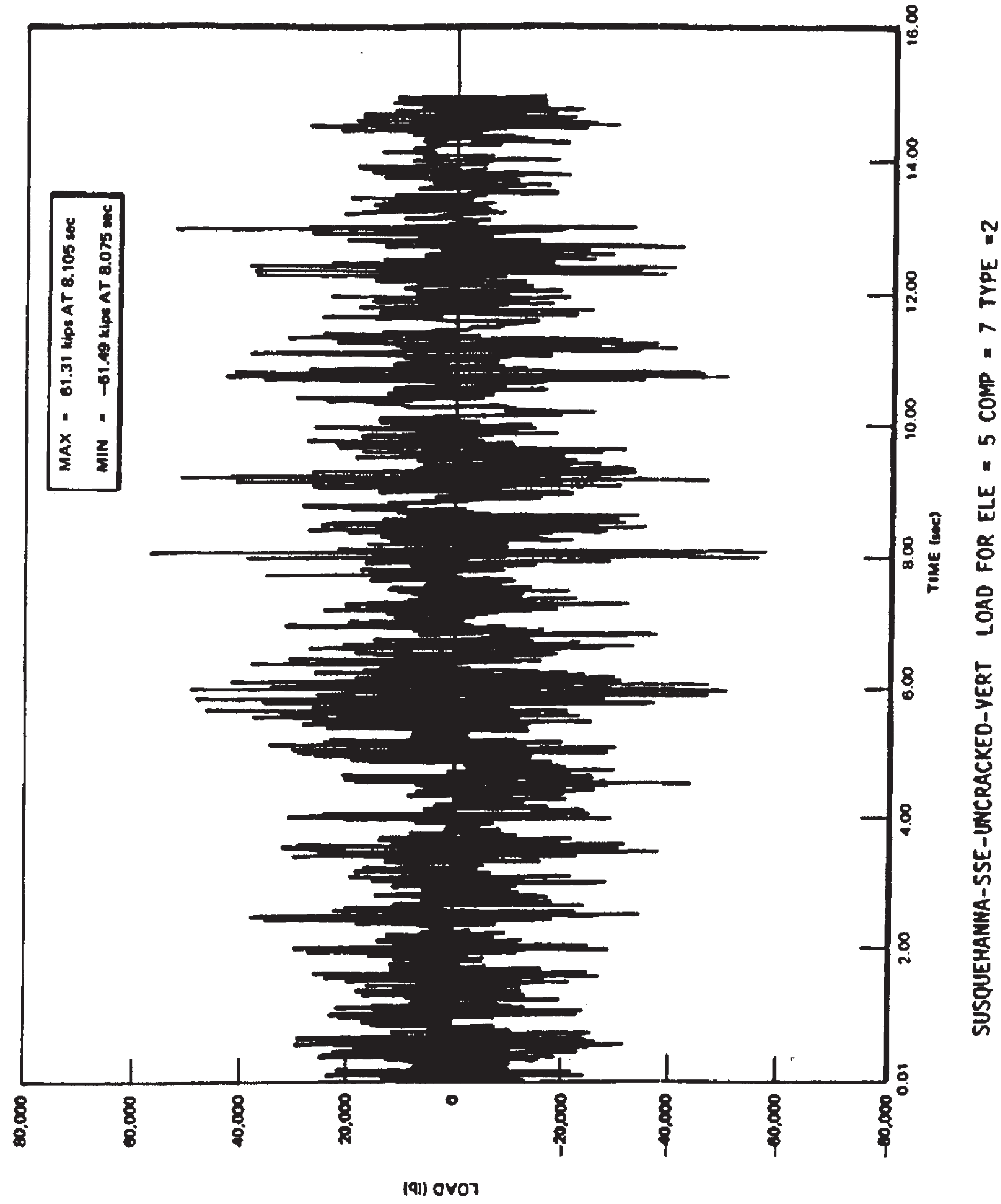
RESPONSE TIME HISTORY PLOT
DUE TO VERTICAL SSE

FSAR FIGURE 110.41-15

PP&L DRAWING



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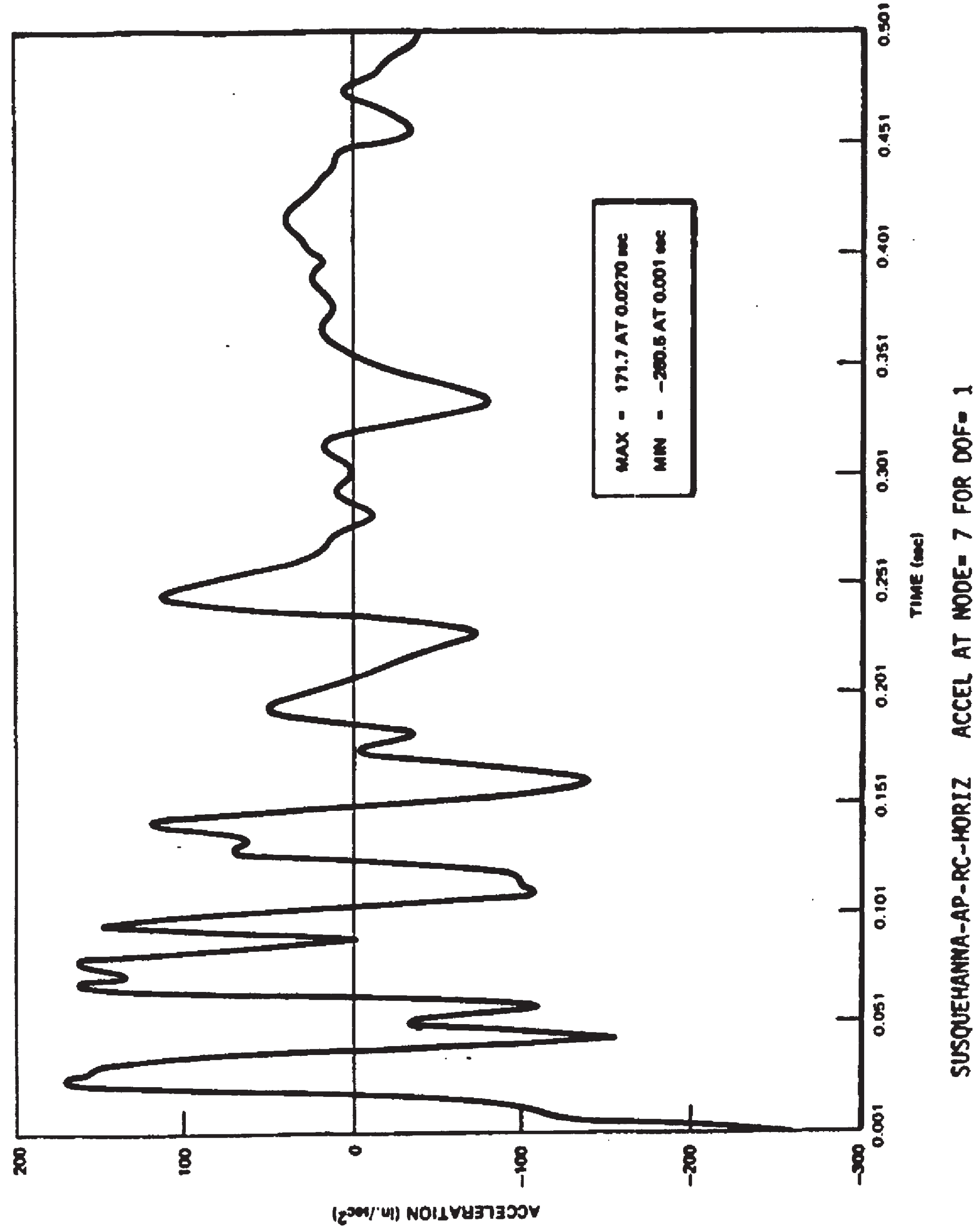
RESPONSE TIME HISTORY PLOT
DUE TO VERTICAL SSE

FSAR FIGURE 110.41-16

PP&L DRAWING



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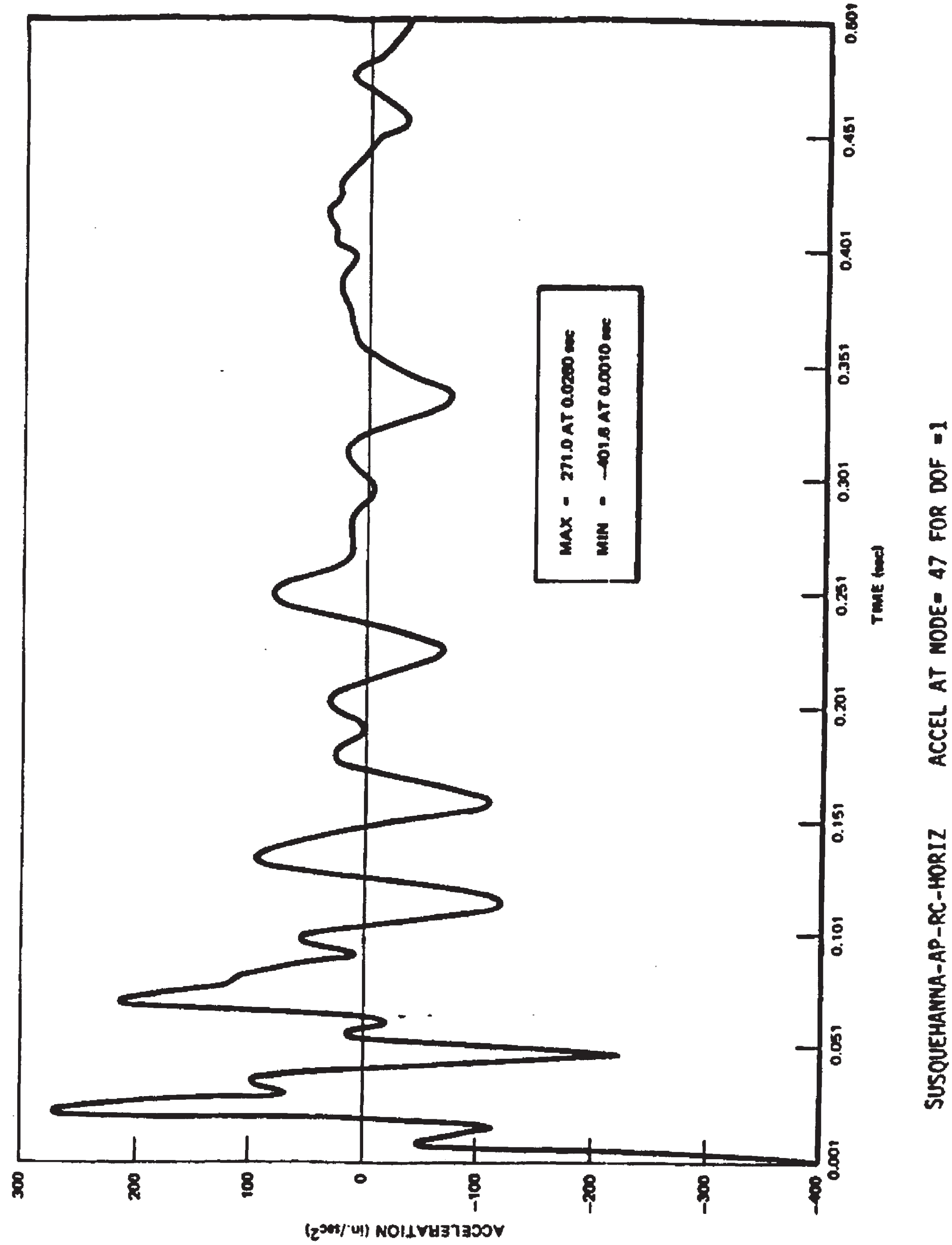
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO
RECIRCULATION PIPE BREAK

FSAR FIGURE 110.41-17

PP&L DRAWING



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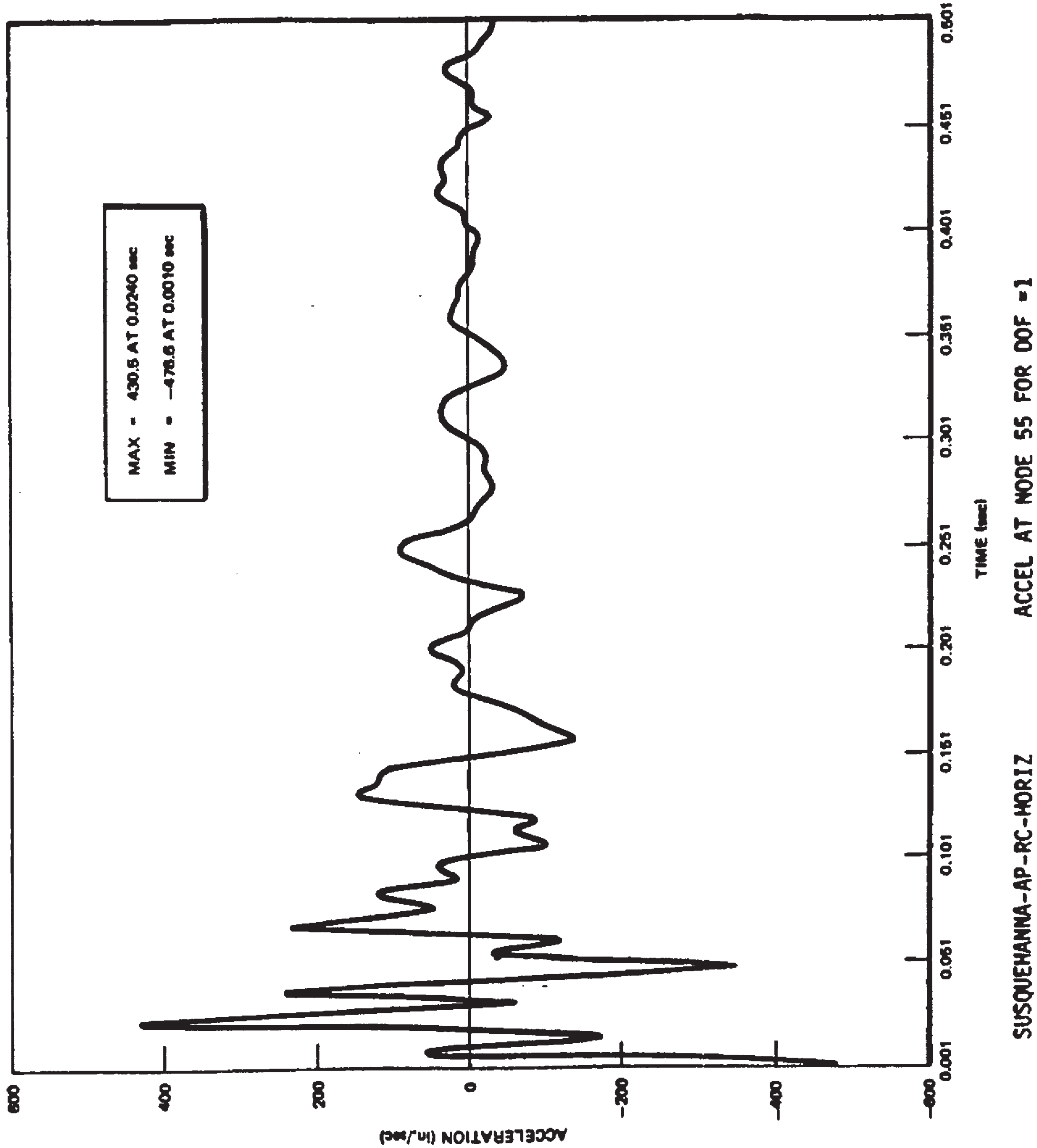
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO
RECIRCULATION PIPE BREAK

FSAR FIGURE 110.41-18

PP&L DRAWING



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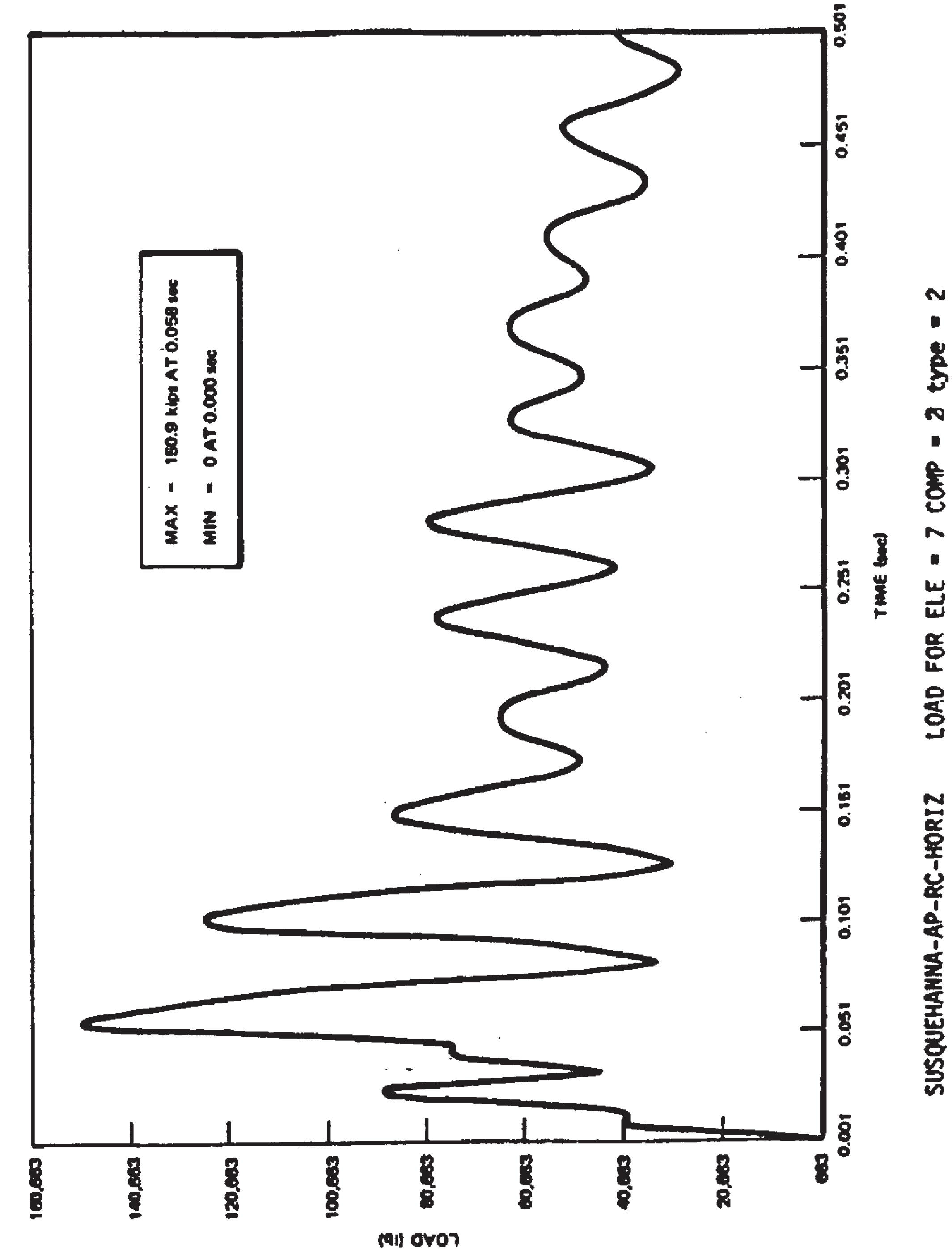
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO
RECIRCULATION PIPE BREAK

FSAR FIGURE 110.41-19

PP&L DRAWING



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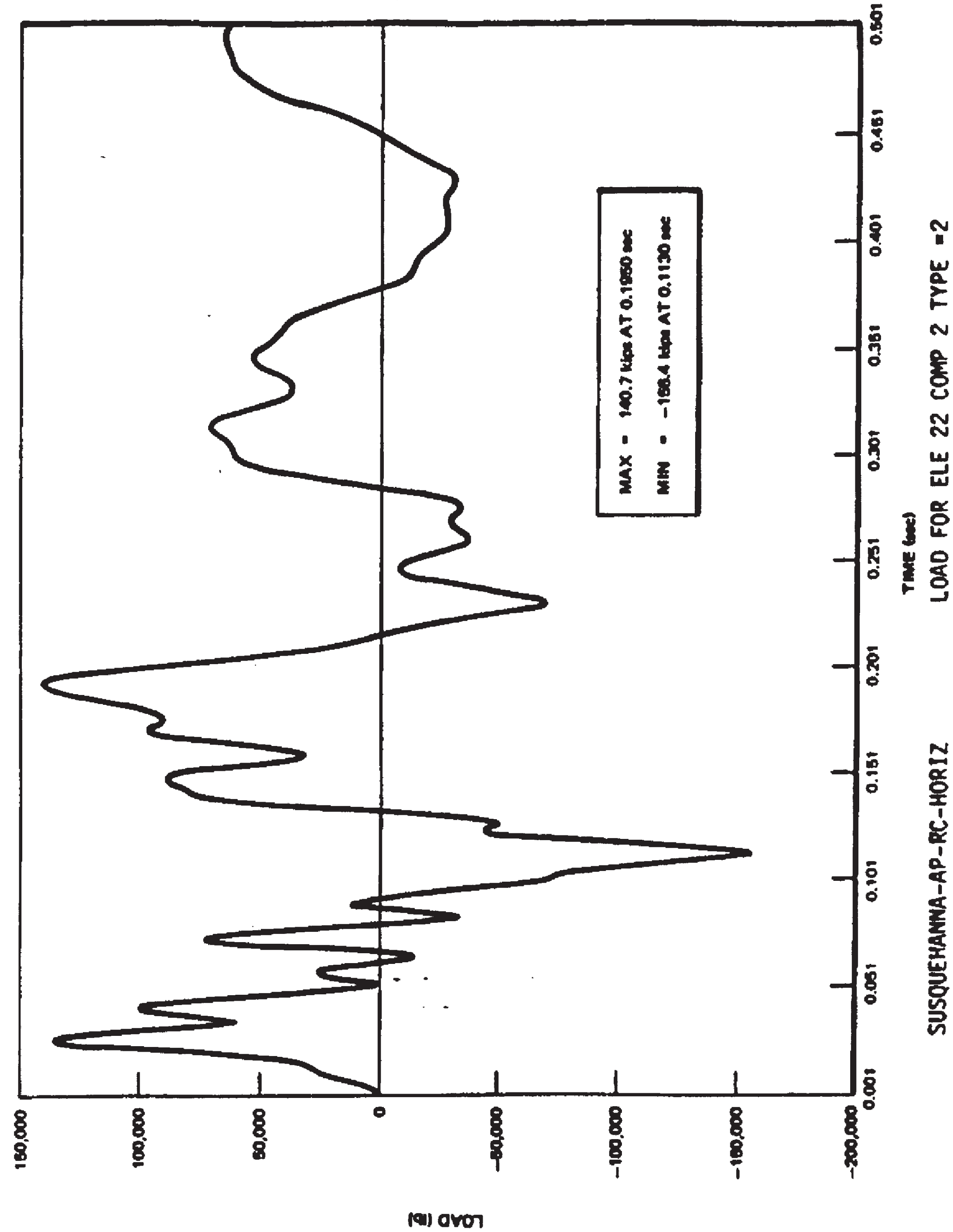
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO
RECIRCULATION PIPE BREAK

FSAR FIGURE 110.41-20

PP&L DRAWING



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SUSQUEHANNA STEAM ELECTRIC STATION
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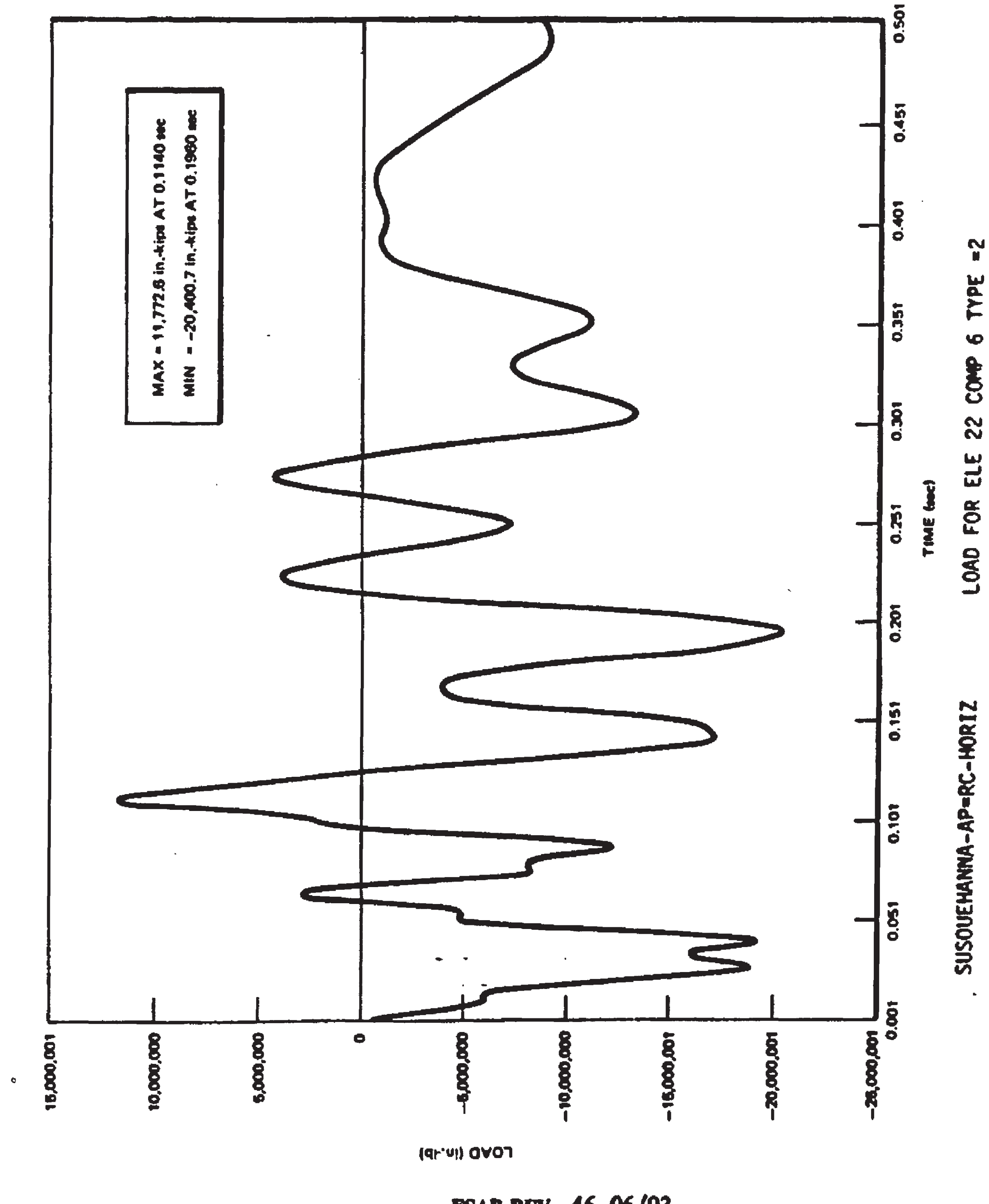
RESPONSE TIME HISTORY PLOT
DUE TO
RECIRCULATION PIPE BREAK

FSAR FIGURE 110.41-21

PP&L DRAWING



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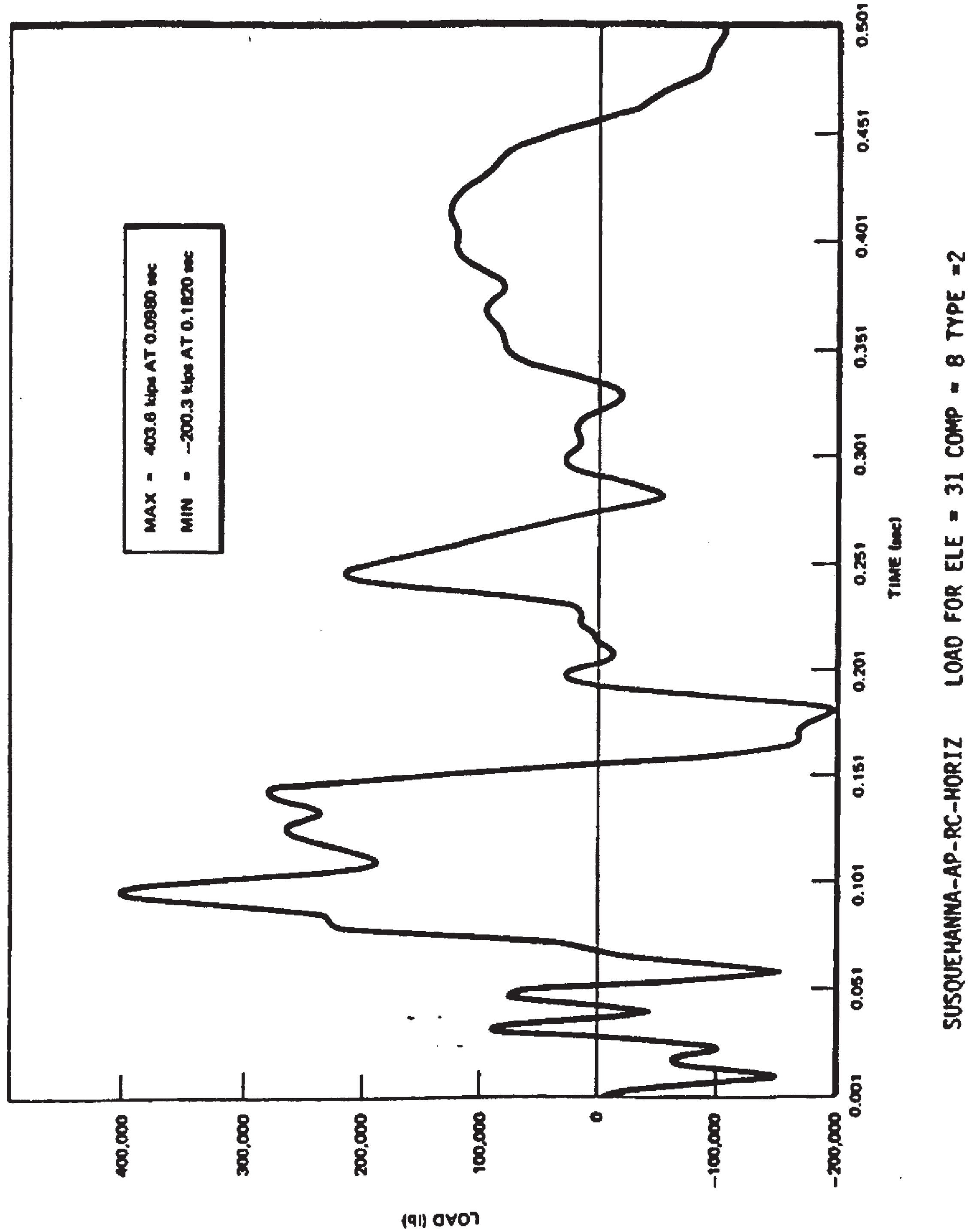
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO
RECIRCULATION PIPE BREAK

FSAR FIGURE 110.41-22

PP&L DRAWING



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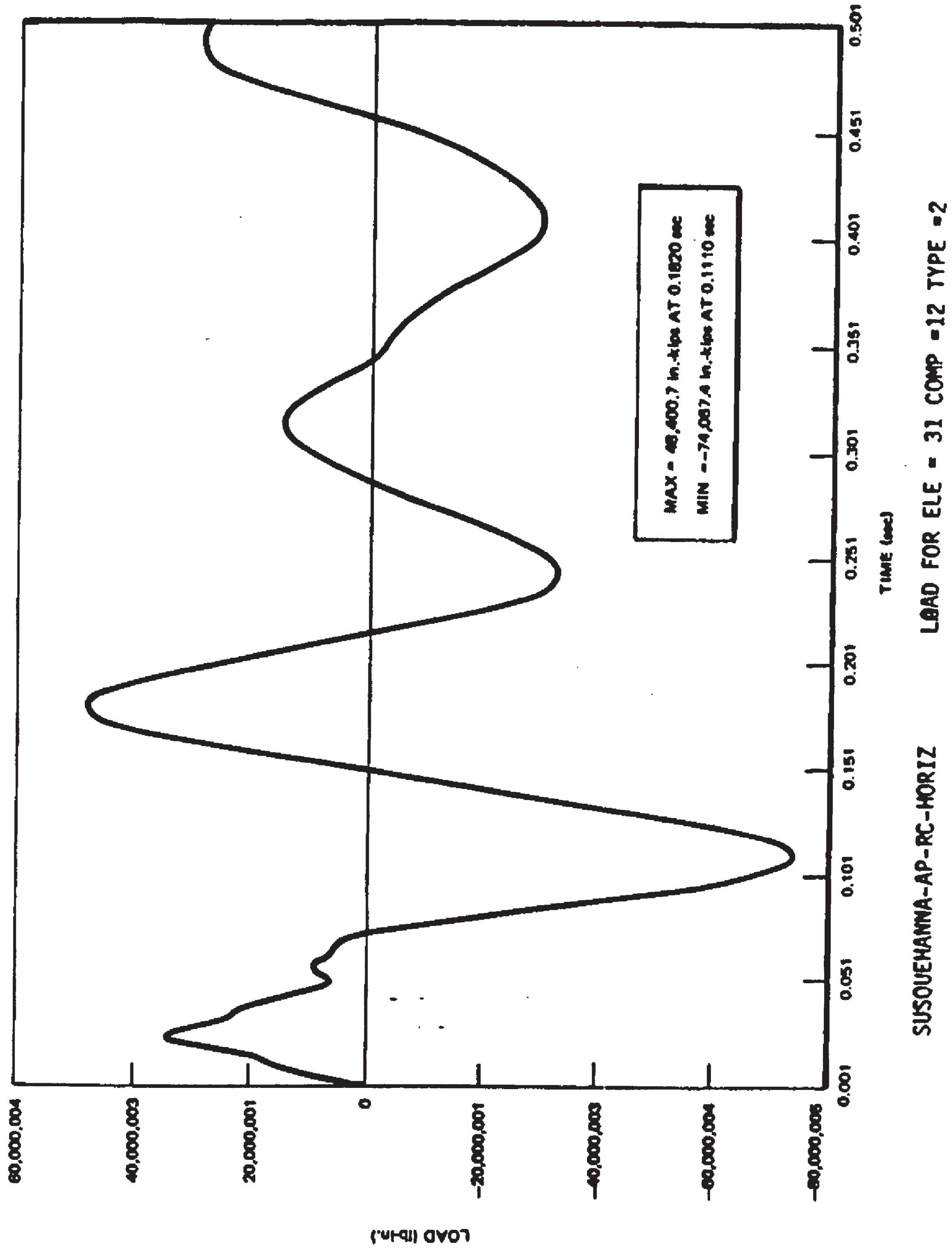
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO
RECIRCULATION PIPE BREAK

FSAR FIGURE 110.41-23

PP&L DRAWING



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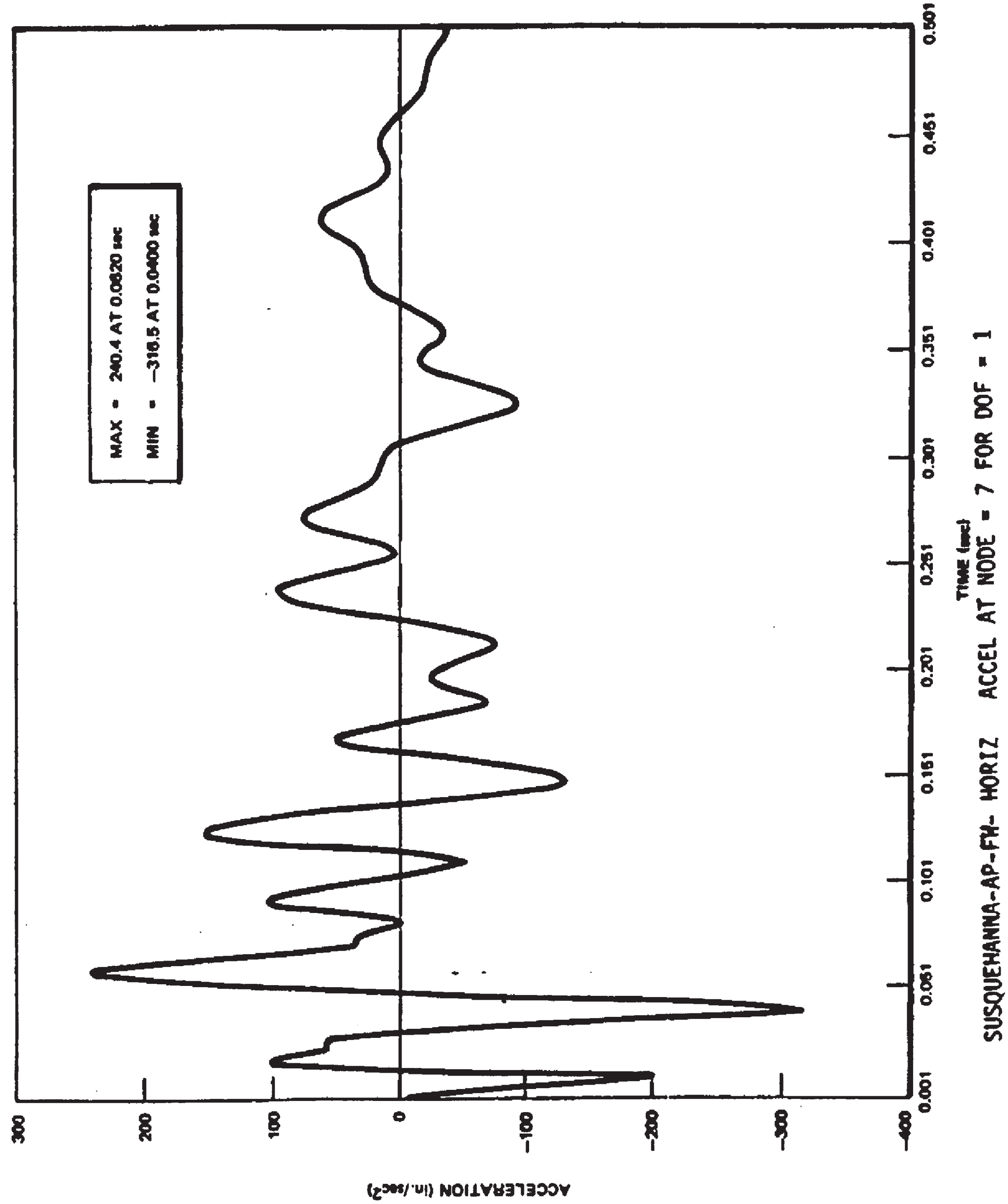
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO
RECIRCULATION PIPE BREAK

FSAR FIGURE 110.41-24

PP&L DRAWING



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SUSQUEHANNA STEAM ELECTRIC STATION
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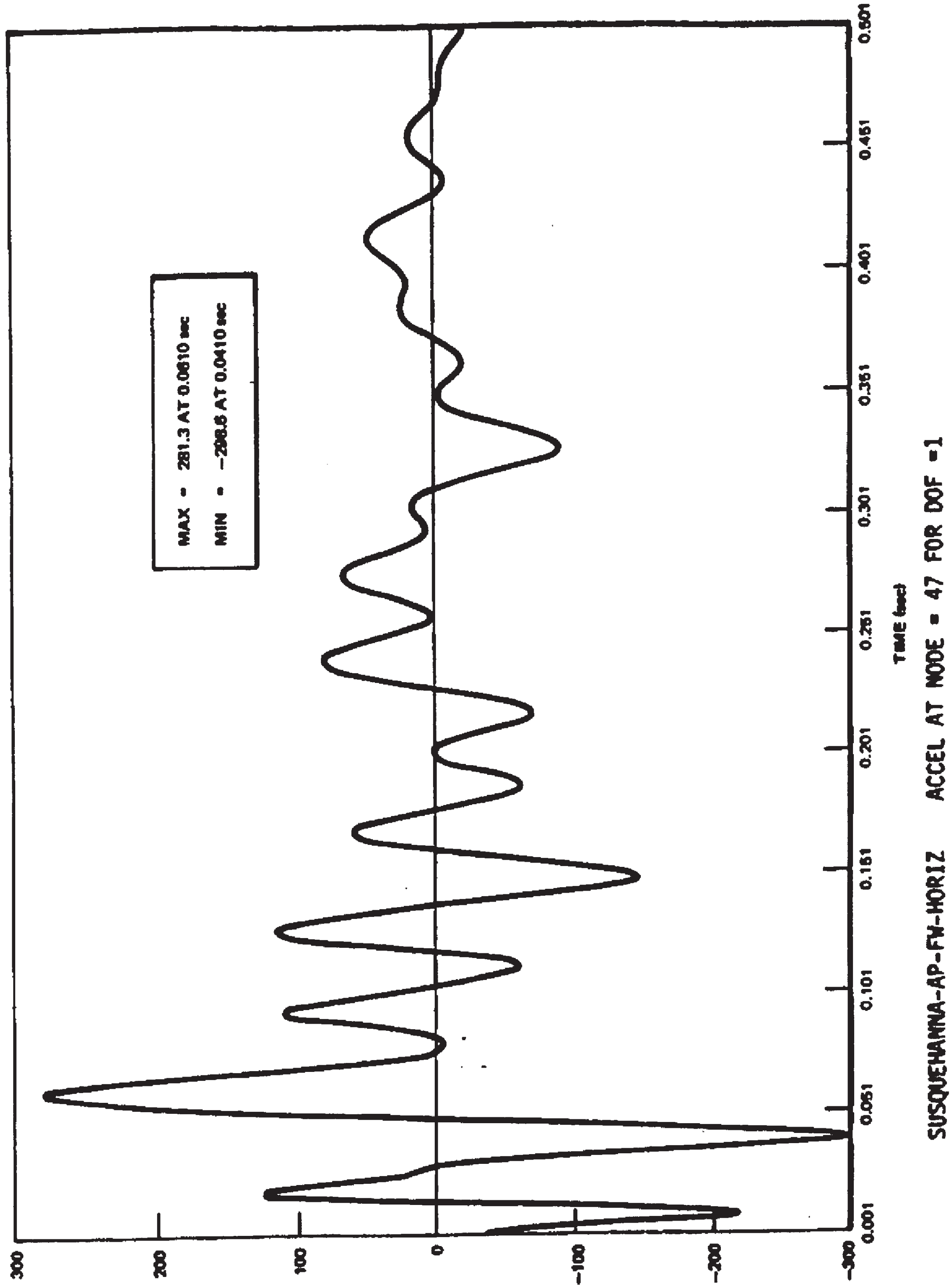
ANNULUS PRESSURIZATION
(HORIZONTAL)
FEEDWATER LINE BREAK

FSAR FIGURE 110.41-25

PP&L DRAWING



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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
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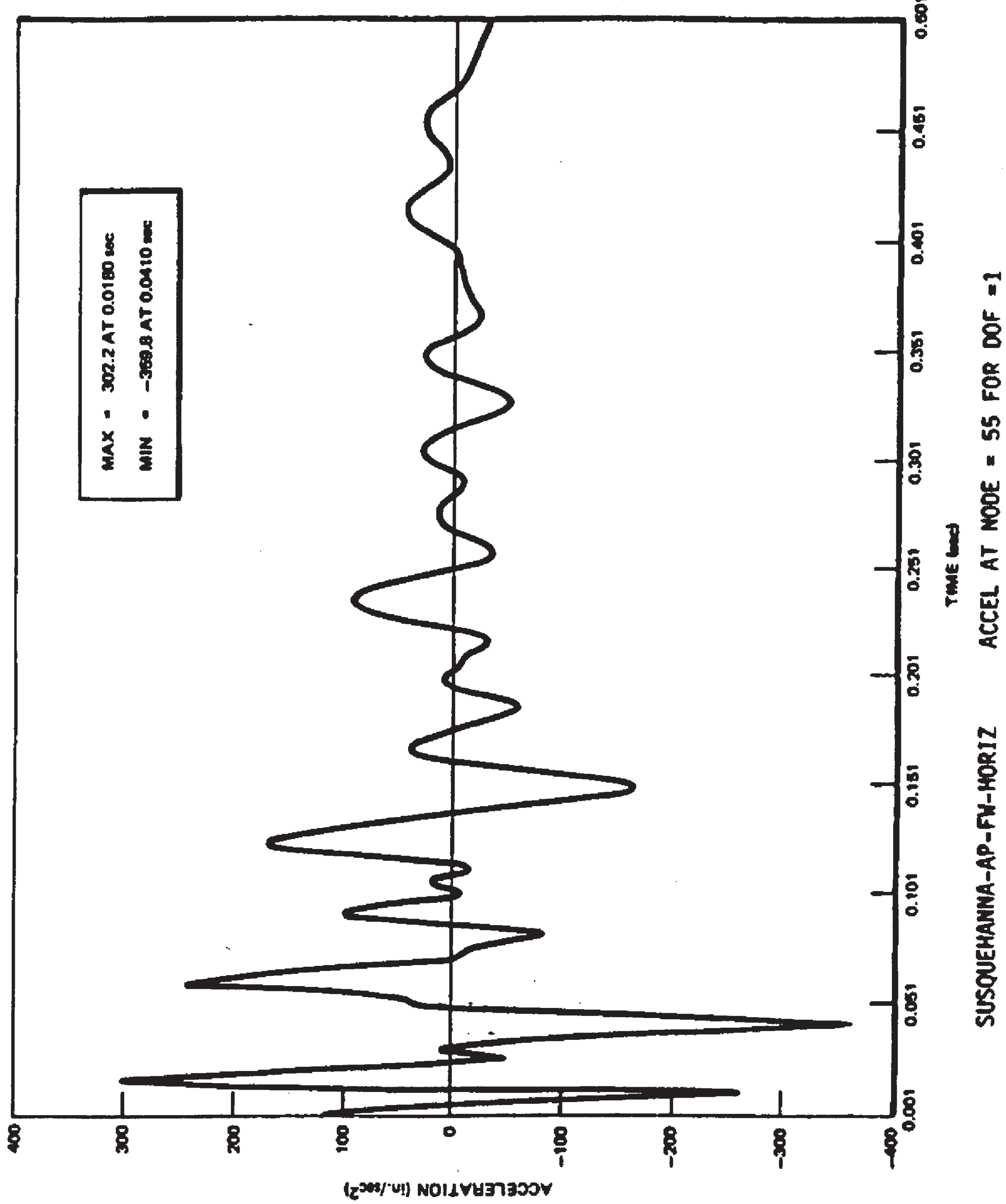
ANNULUS PRESSURIZATION
(HORIZONTAL)
FEEDWATER LINE BREAK

FSAR FIGURE 110.41-26

PP&L DRAWING



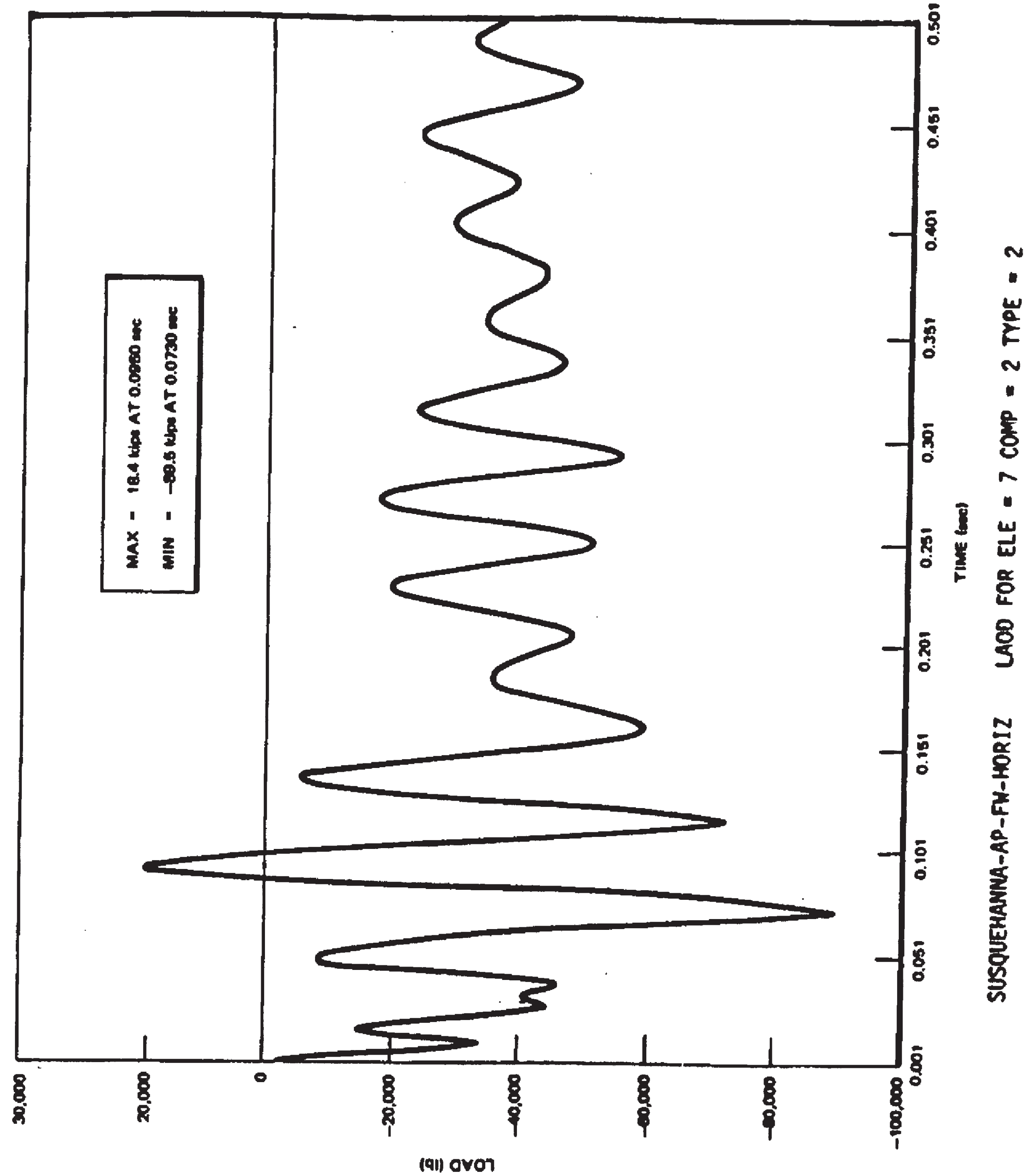
FSAR REV. 46, 06/93

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

ANNULUS PRESSURIZATION
(HORIZONTAL)
FEEDWATER LINE BREAK

FSAR FIGURE 110.41-27

PP&L DRAWING



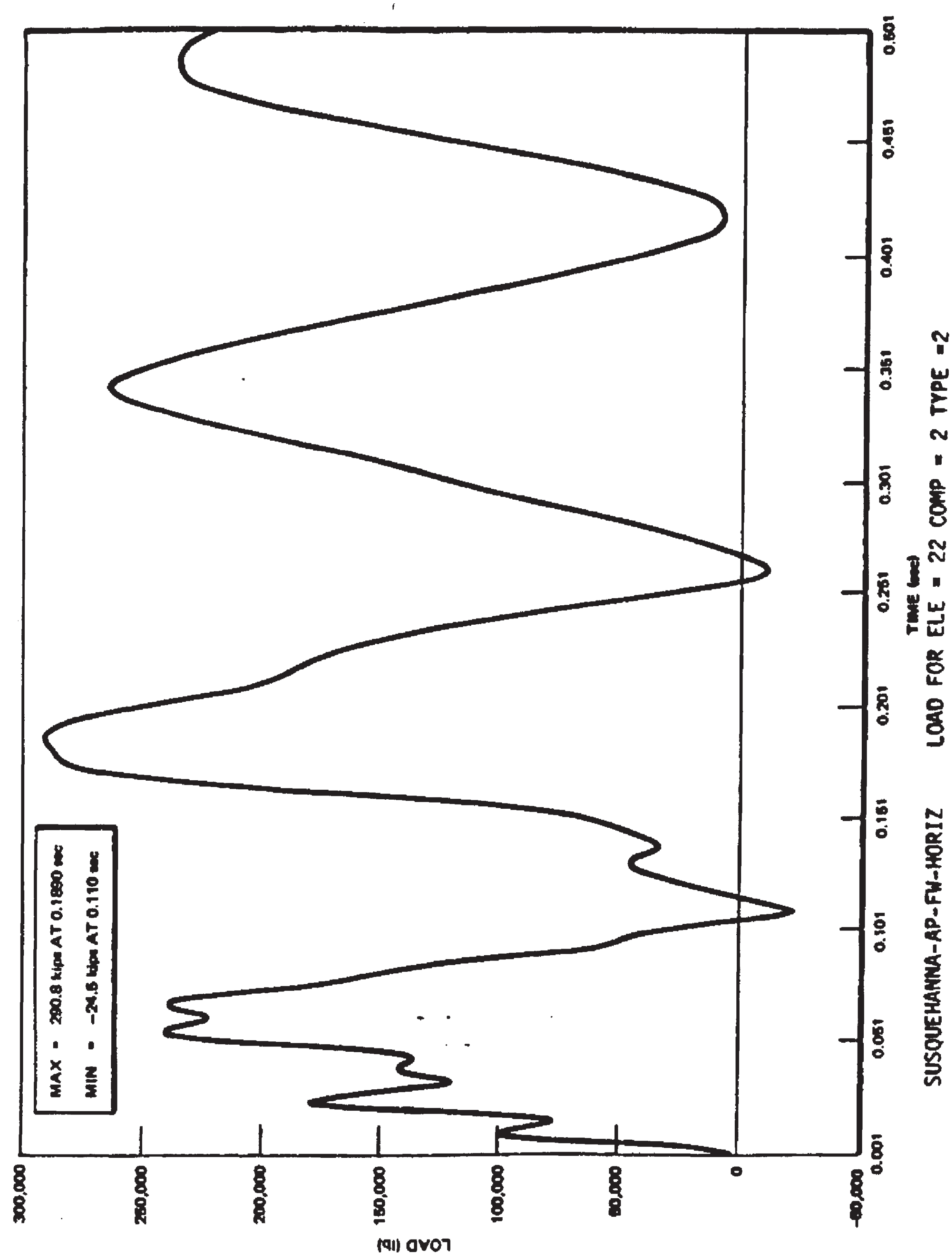
FSAR REV. 46, 06/93

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

ANNULUS PRESSURIZATION
(HORIZONTAL)
FEEDWATER LINE BREAK

FSAR FIGURE 110.41-28

PP&L DRAWING



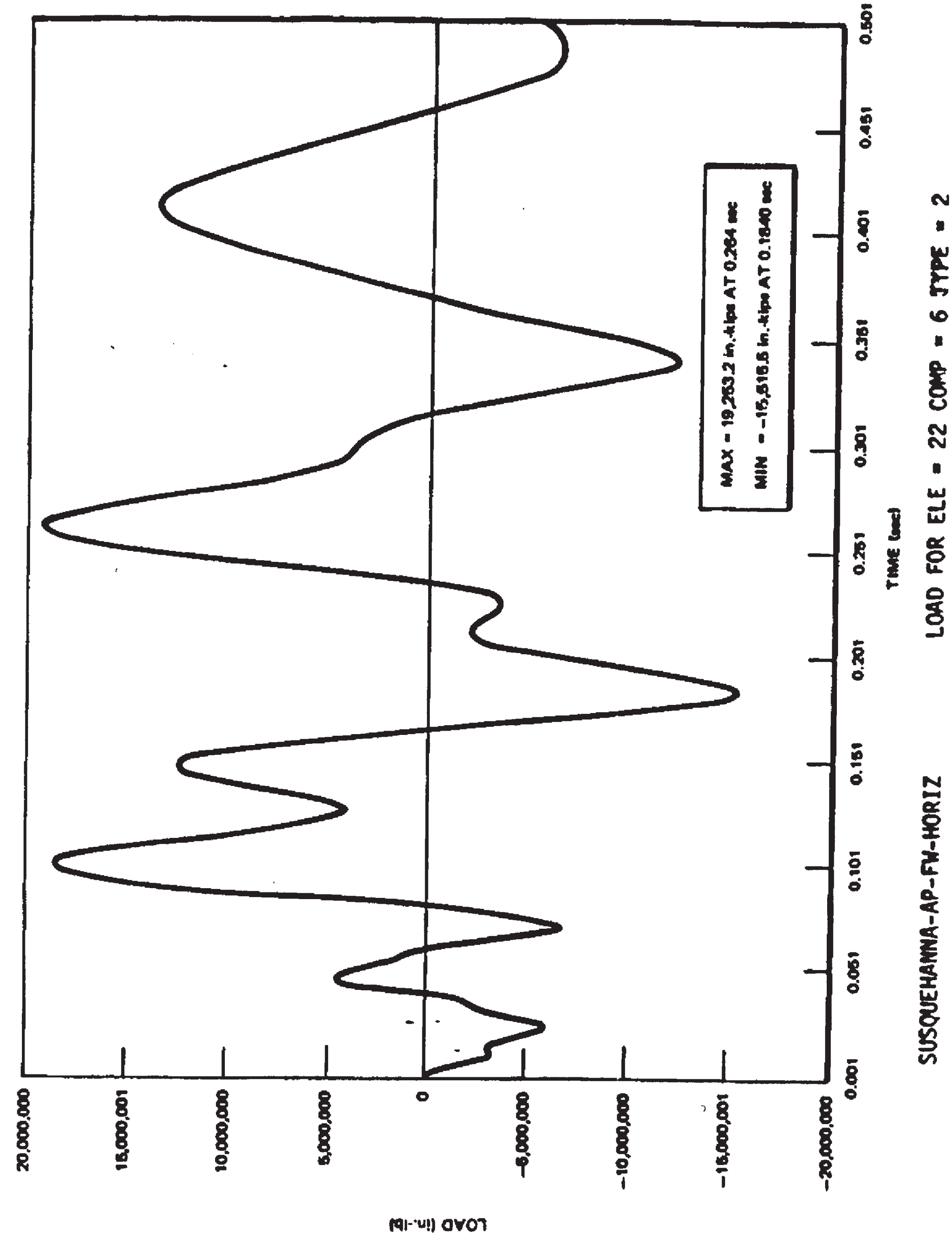
FSAR REV. 46, 06/93

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

ANNULUS PRESSURIZATION
(HORIZONTAL)
FEEDWATER LINE BREAK

FSAR FIGURE 110.41-29

PP&L DRAWING



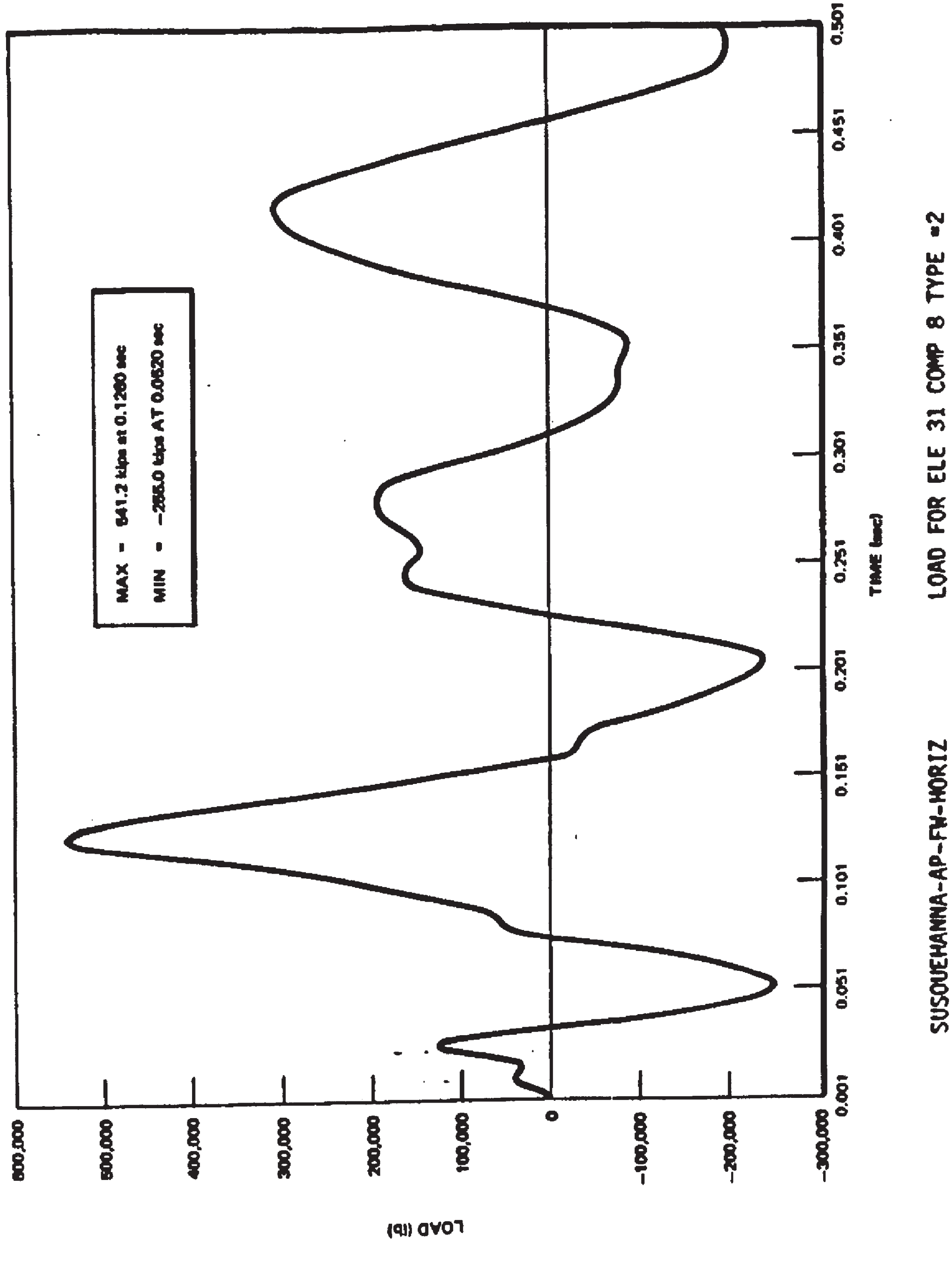
FSAR REV. 46, 06/93

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

ANNULUS PRESSURIZATION
(HORIZONTAL)
FEEDWATER LINE BREAK

FSAR FIGURE 110.41-30

PP&L DRAWING



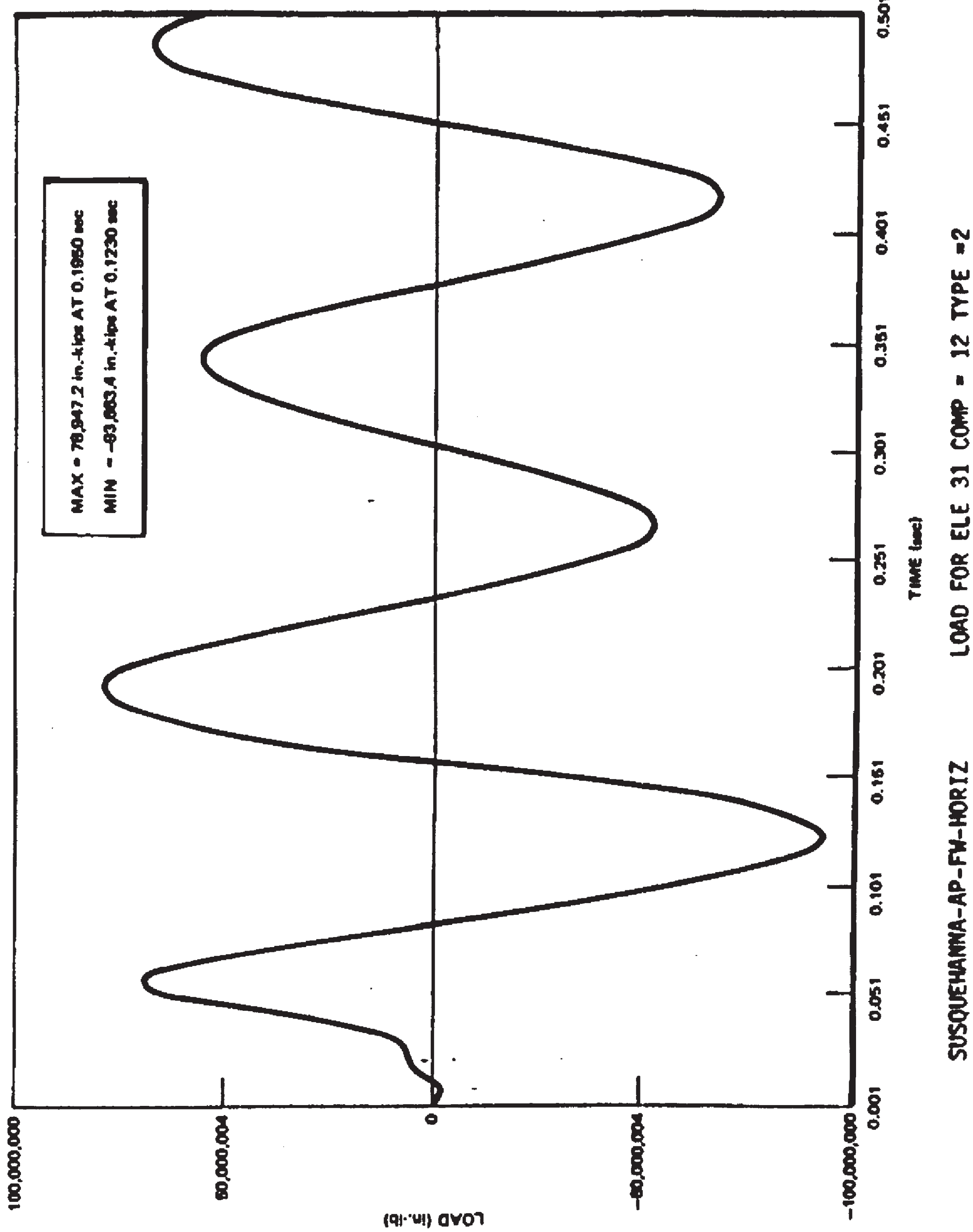
FSAR REV. 46, 06/93

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

ANNULUS PRESSURIZATION
(HORIZONTAL)
FEEDWATER LINE BREAK

FSAR FIGURE 110.41-31

PP&L DRAWING



FSAR REV. 46, 06/93

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

ANNULUS PRESSURIZATION
(HORIZONTAL)
FEEDWATER LINE BREAK

FSAR FIGURE 110.41-32

PP&L DRAWING

TABLE 110.41-1

COMPONENT VERSUS TIME HISTORY TABLE

	Horizontal Model*		Vertical Model**	
Components	Time History Type(1)	Location(2)	Time History Type(1)	Location(2)
Core Plate	F	N-7 OF E-7 (C-2 OF E-7)	A	N-7
Shroud	 F M	N-58 OF E-31 (C-8 OF E-31) (C-12 OF E-31)	F	N-7 OF E-5 (C-7 OF E-5)
Shroud Head	 F M	N-22 OF E-22 (C-2 OF E-22) (C-6 OF E-22)	F	N-5 OF E-3 (C-7 OF E-3)
CRD Guide Tube and Housings Jet Pump	A A	N-7, N-55 N-47	A A	N-2, N-3 N-8

(1) A = Acceleration Time History
F = Force Time History
M = Moment Time History

(2) N = Node
E = Element
C = Load Component***

* Pertaining both to horizontal seismic and pipe break events.

** Pertaining to vertical seismic event only. The vertical pipe break events are judged to be insignificant.

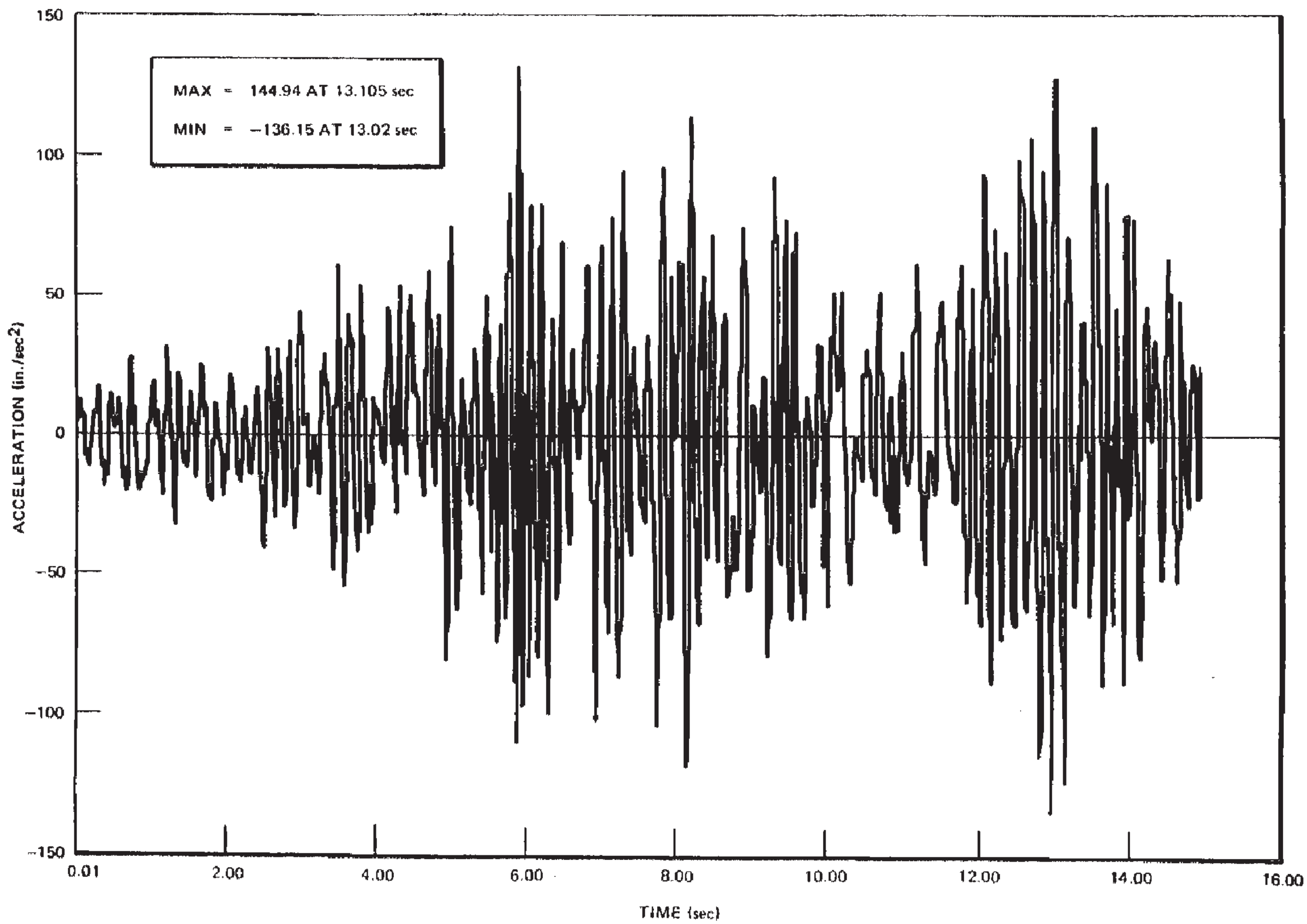
*** C-1 = Force at End I
C-2 = Shear at End I
C-3 = Shear at End I
C-4 = Torque at End I
C-5 = Moment at End I
C-6 = Moment at End I
C-7 = Force at End J
C-8 = Shear at End J
C-9 = Shear at End J
C-10 = Torque at End J
C-11 = Moment at End J
C-12 = Moment at End J

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
REACTOR PRESSURE VESSEL AND INTERNALS HORIZONTAL MATHEMATICAL MODEL FOR EARTHQUAKE AND ANNULUS PRESSURIZATION LOADING FIGURE 110.41-1

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
REACTOR PRESSURE VESSEL AND INTERNALS VERTICAL DYNAMIC MODEL FOR EARTHQUAKE LOADING FIGURE 110.41-2



SUSQUEHANNA SSE- UNCRACKED HORIZ ACCEL AT NODE = 7 FOR DOF = 1

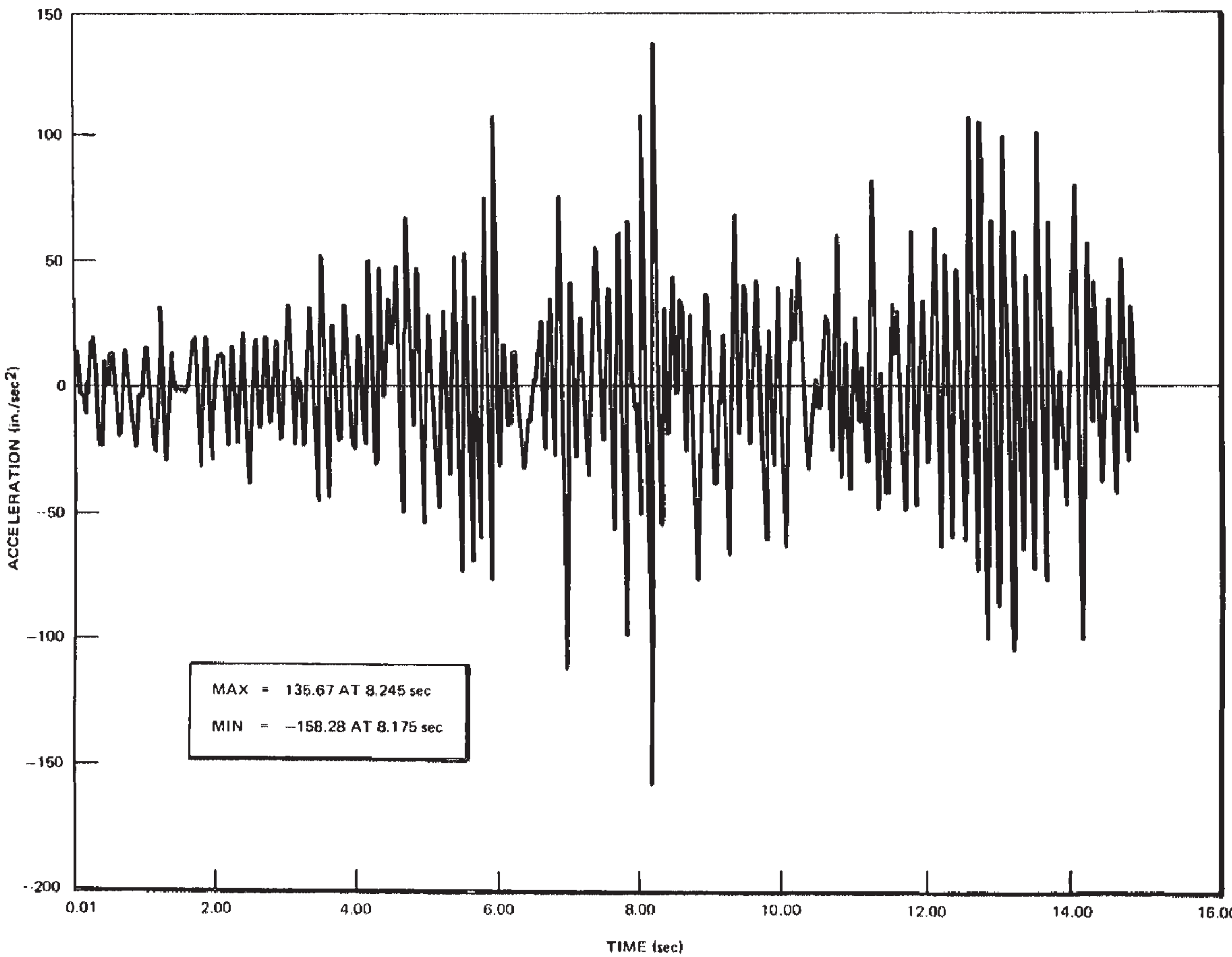
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO HORIZONTAL SSE

FIGURE 110.41-3, Rev 47

AutoCAD: Figure Fsar 110_41_3.dwg



SUSQUEHANNA-SSE-UNCRACKED HORIZ ACCEL AT NODE = 47 FOR DOF = 1

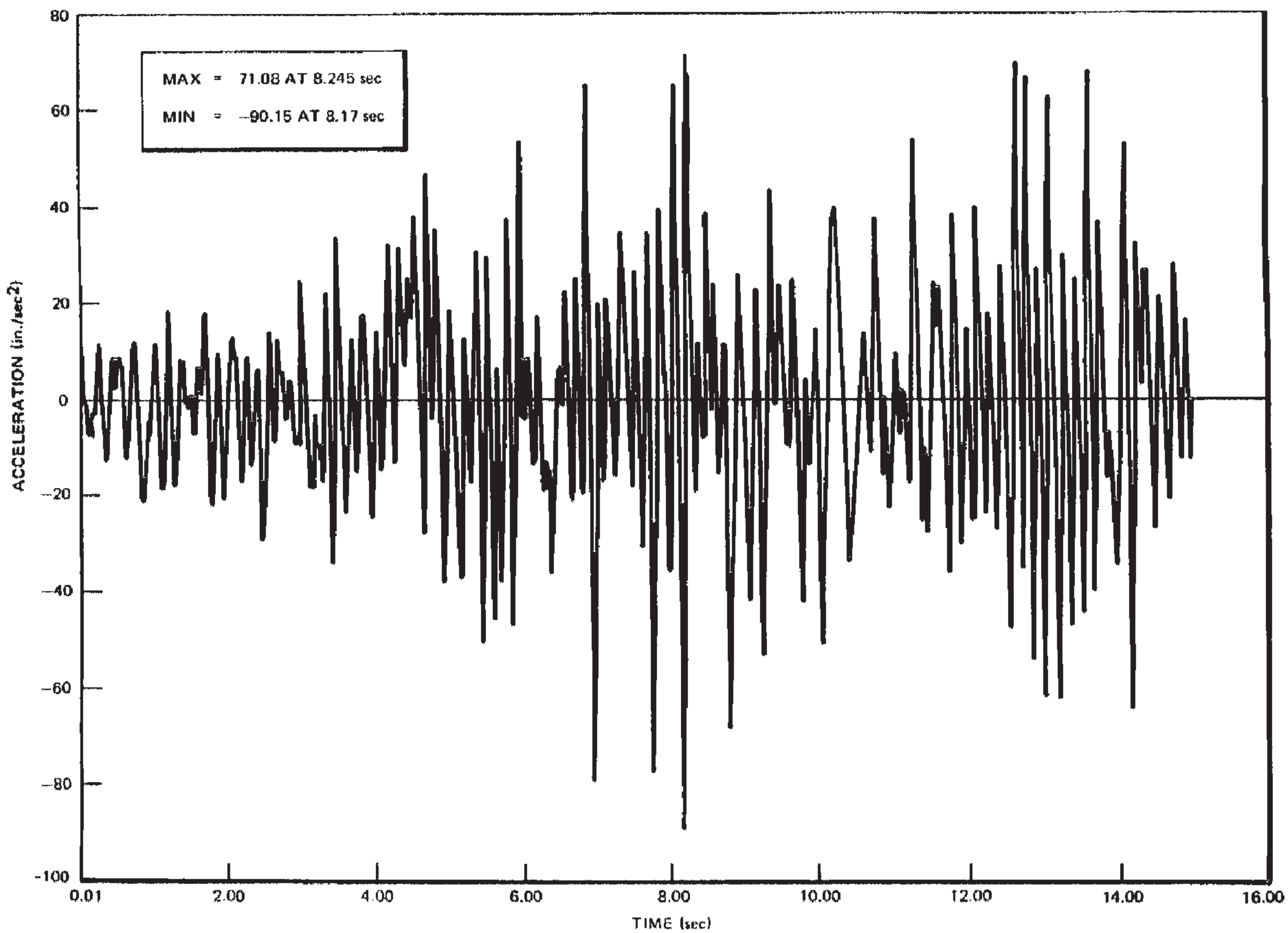
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO HORIZONTAL SSE

FIGURE 110.41-4, Rev 47

AutoCAD: Figure Fsar 110_41_4.dwg



SUSQUEHANNA–SSE–UNCRACKED HORIZ ACCEL AT NODE = 55 FOR DOF = 1

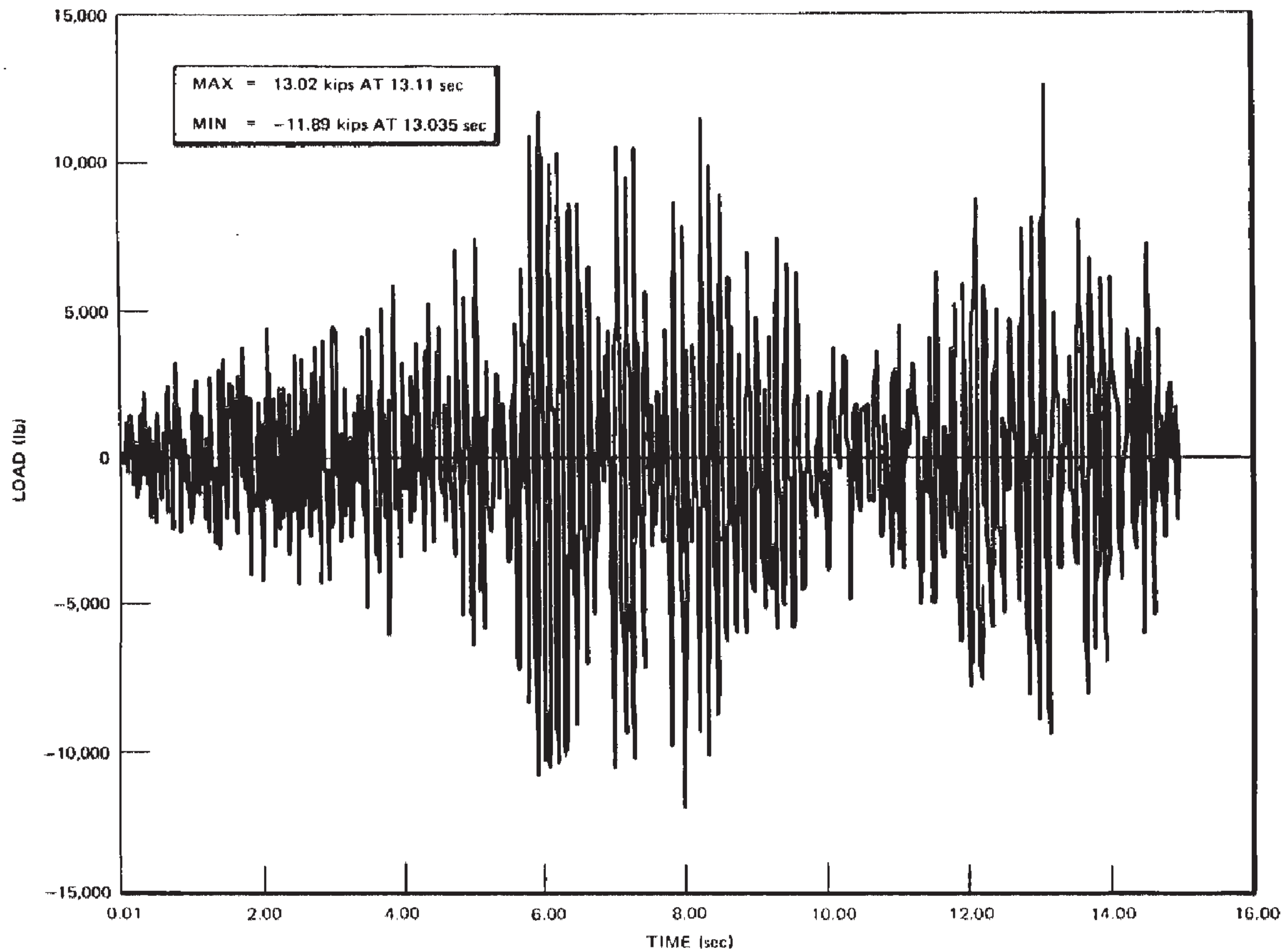
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO HORIZONTAL SSE

FIGURE 110.41-5, Rev 47

AutoCAD: Figure Fsar 110_41_5.dwg



SUSQUEHANNA-SSE-UNCRAKED-HORIZ LOAD FOR ELE = 7 COMP = 2 TYPE = 2

FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO HORIZONTAL SSE

FIGURE 110.41-6, Rev 47

AutoCAD: Figure Fsar 110_41_6.dwg

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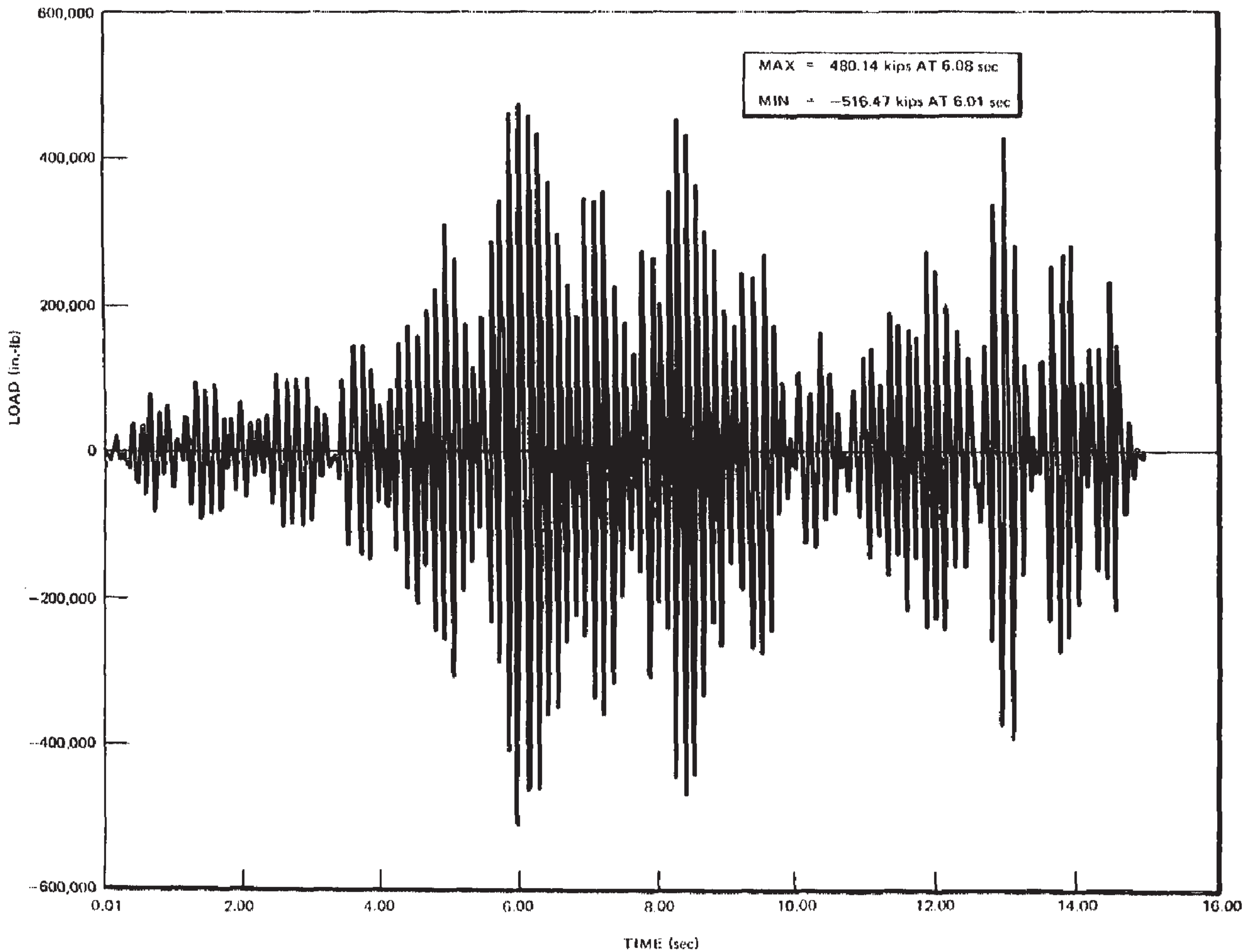
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

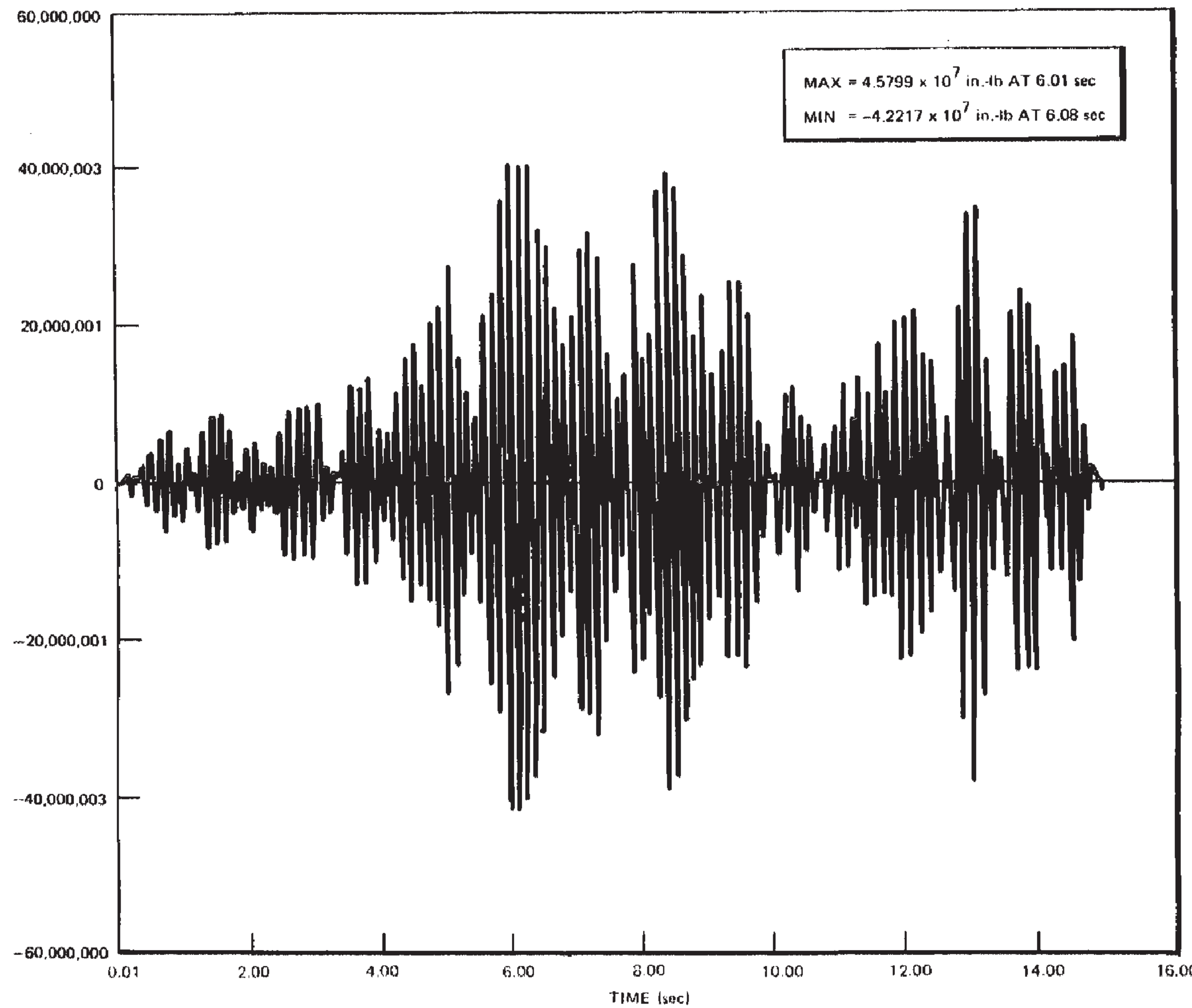
RESPONSE TIME HISTORY PLOT
DUE TO HORIZONTAL SSE

FIGURE 110.41-7, Rev 47

AutoCAD: Figure Fsar 110_41_7.dwg



SUSQUBHANNA-SSE-UNCRACKED-HORIZ LOAD FOR ELE = 22 COMP = 2 TYPE = 2



SUSQUEHANNA-SSE-UNCRACKED-HORIZ LOAD FOR ELE = 22 COMP = 6 TYPE = 2

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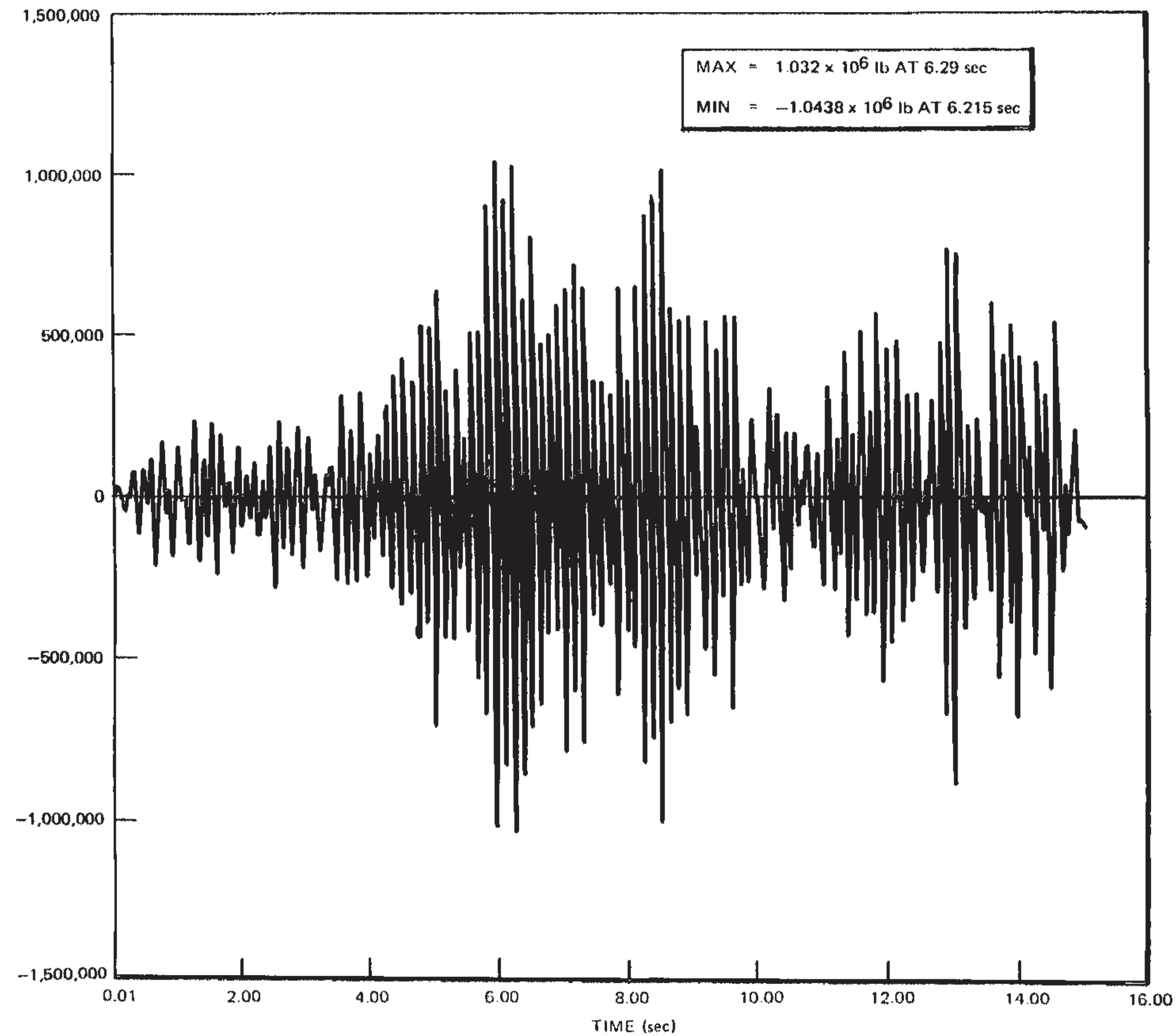
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO HORIZONTAL SSE

FIGURE 110.41-8, Rev 47

AutoCAD: Figure Fsar 110_41_8.dwg



SUSQUEHANNA-SSE-UNCRACKED-HORIZ LOAD FOR ELE = 31 COMP = B type= 2

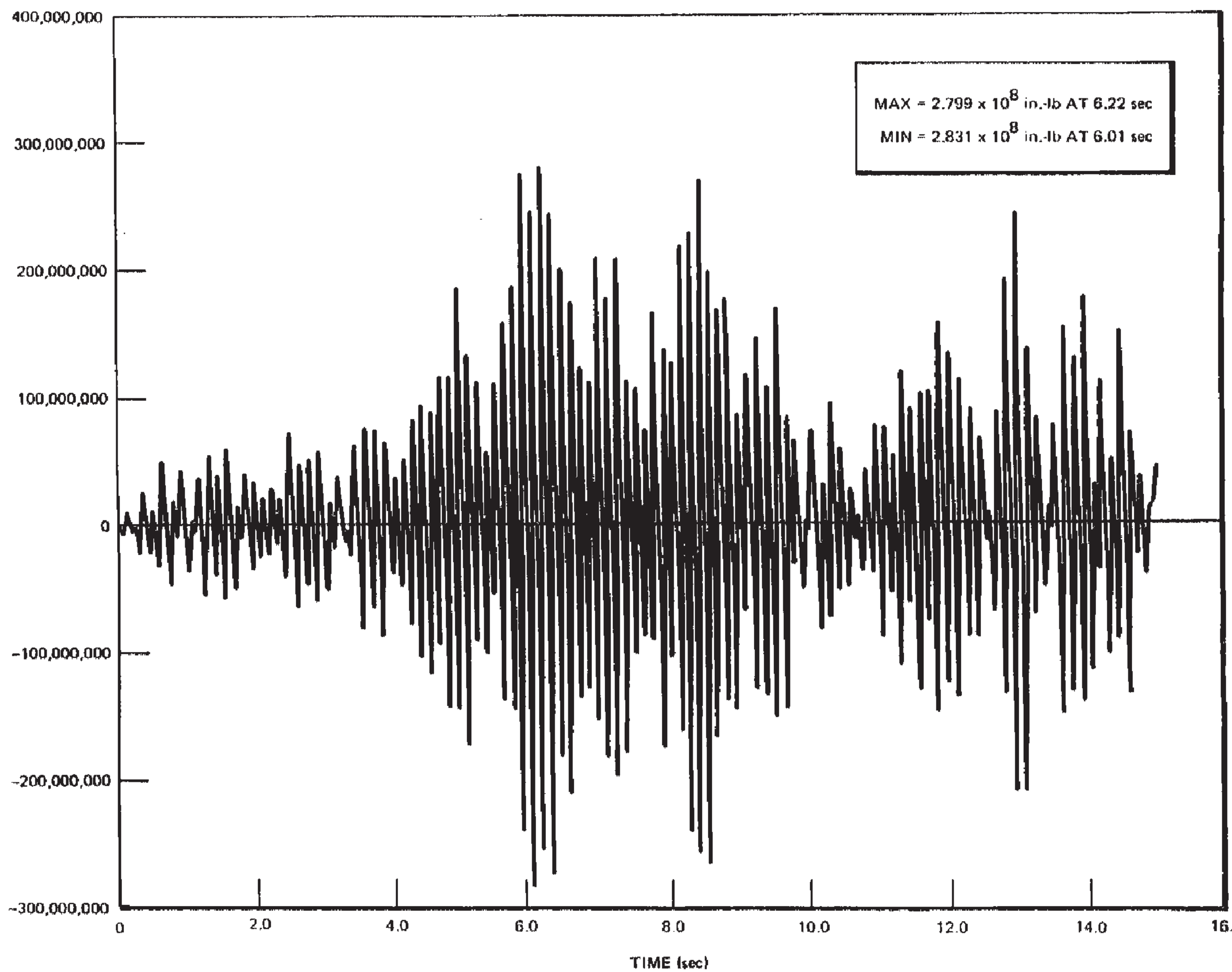
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO HORIZONTAL SSE

FIGURE 110.41-9, Rev 47

AutoCAD: Figure Fsar 110_41_9.dwg



SUSQUEHANNA-SSE-UNCRACKED-HORIZ

LOAD FOR ELE = 31 COMP = 12 TYPE = 2

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO HORIZONTAL SSE

FIGURE 110.41-10, Rev 47

AutoCAD: Figure Fsar 110_41_10.dwg

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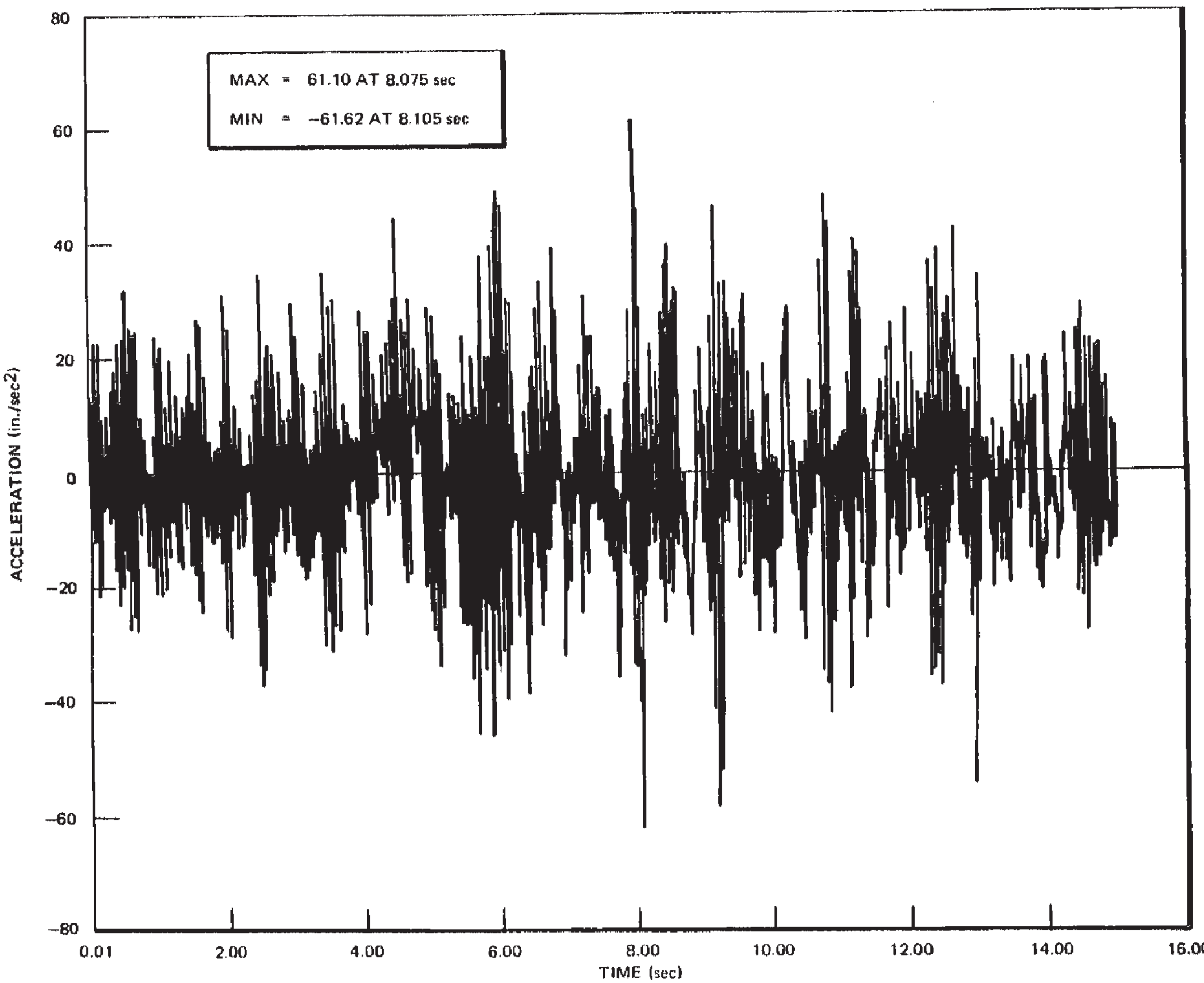
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO VERTICAL SSE

FIGURE 110.41-11, Rev 47

AutoCAD: Figure Fsar 110_41_11.dwg



SUSQUEHANNA-SSE-UNCRACKED-VERT ACCEL AT NODE = 2 FOR DOF= 3

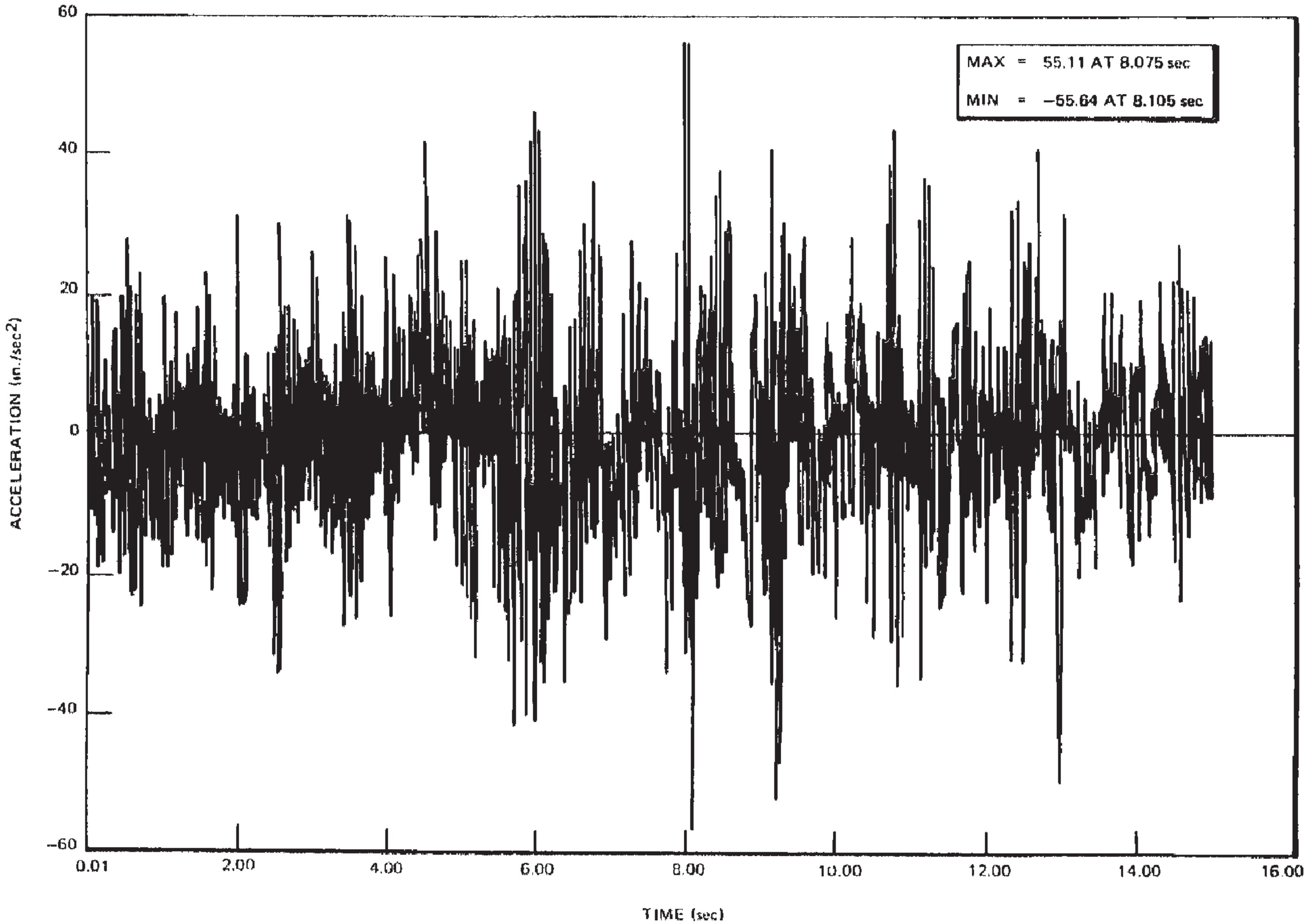
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO VERTICAL SSE

FIGURE 110.41-12, Rev 47

AutoCAD: Figure Fsar 110_41_12.dwg



SUSQUEHANNA-SSE-UNCRACKED-VERT ACCEL AT NODE = 3 FOR DOF = 3

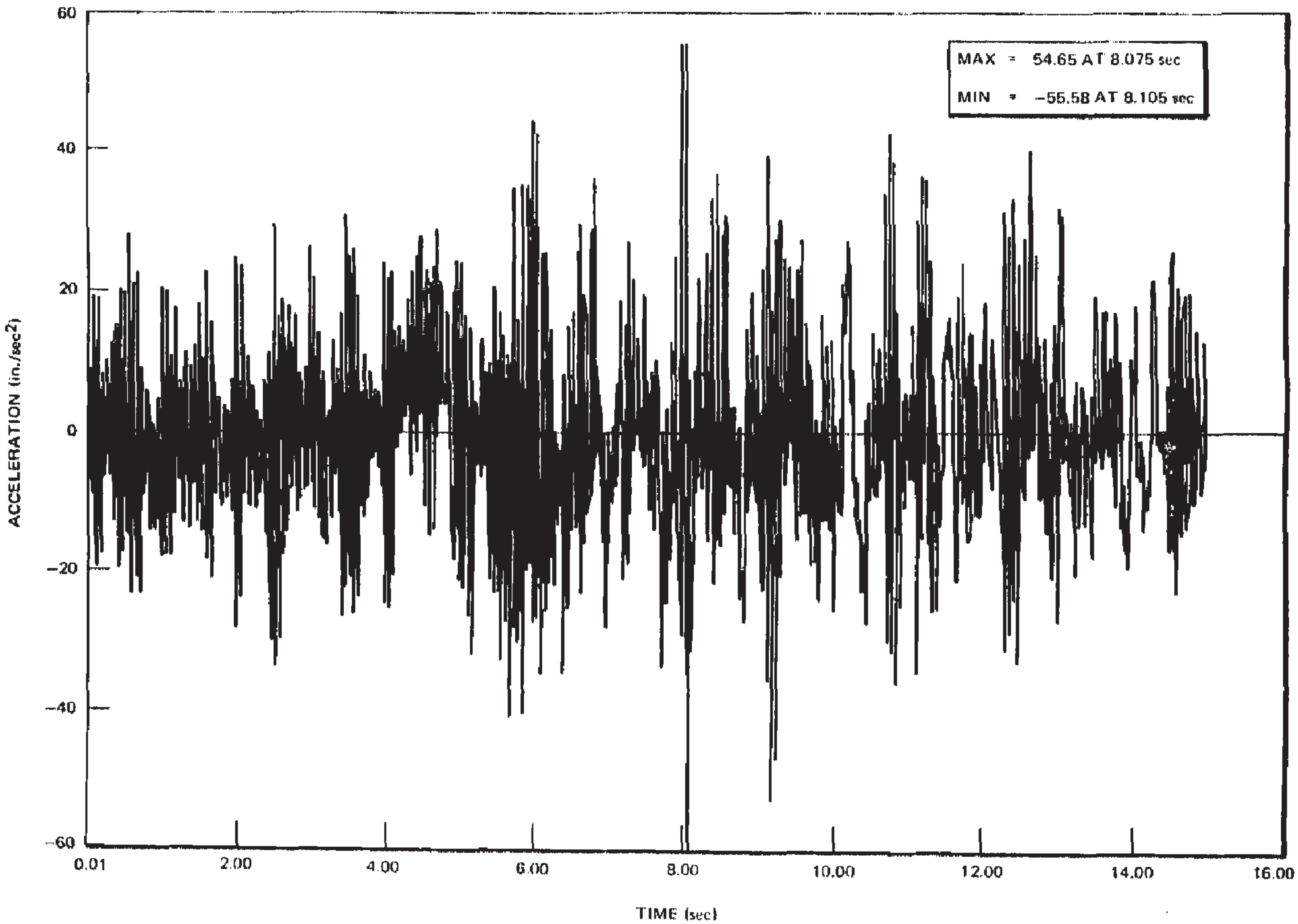
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

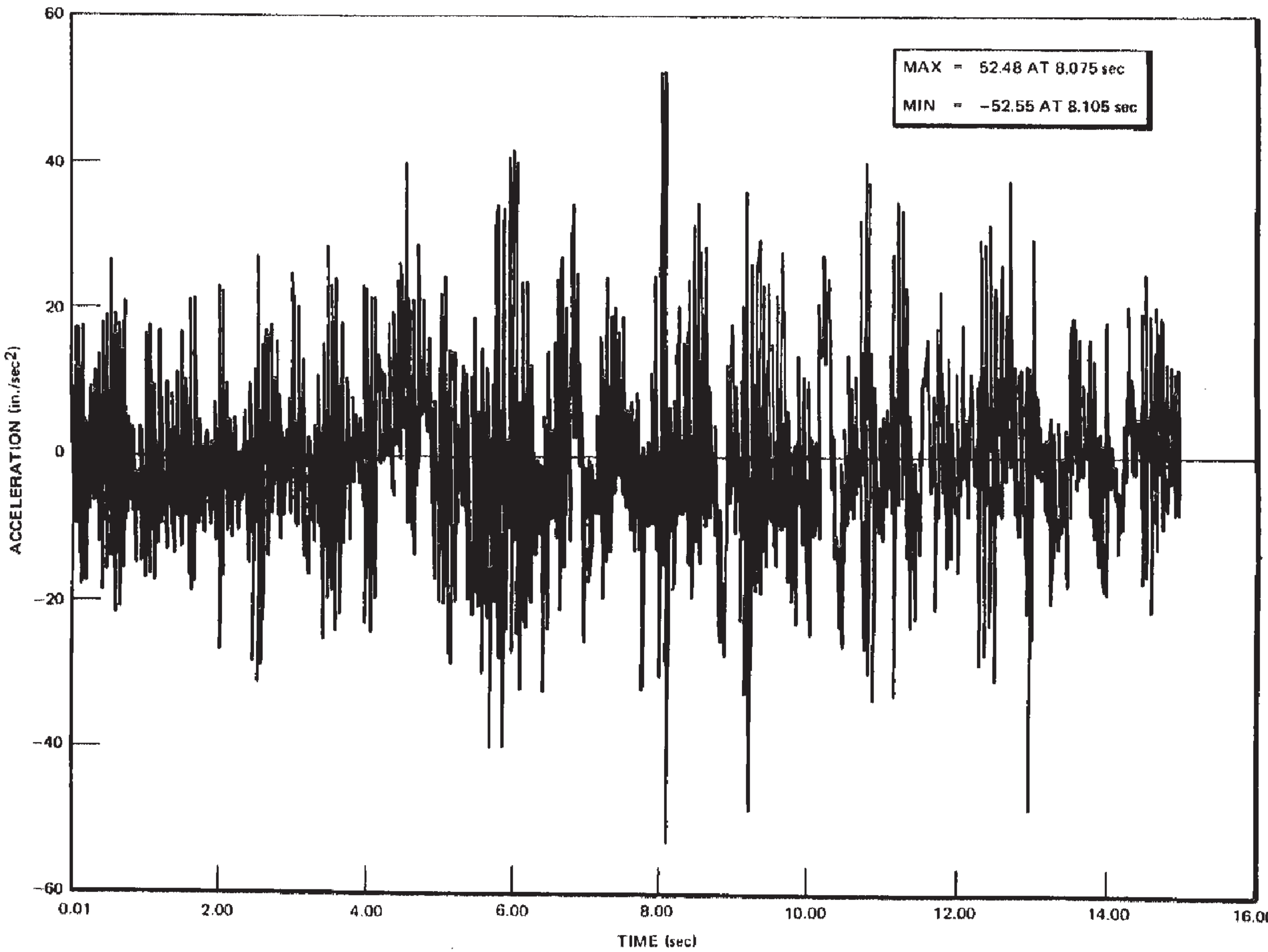
RESPONSE TIME HISTORY PLOT
DUE TO VERTICAL SSE

FIGURE 110.41-13, Rev 47

AutoCAD: Figure Fsar 110_41_13.dwg



SUSQUEHANNA-SSE-UNCRACKED-VERT ACCEL AT NODE = 7 FOR DOF = 3



SUSQUEHANNA-SSE-UNCRACKED-VERT ACCEL AT NODE = 8 FOR DOF =3

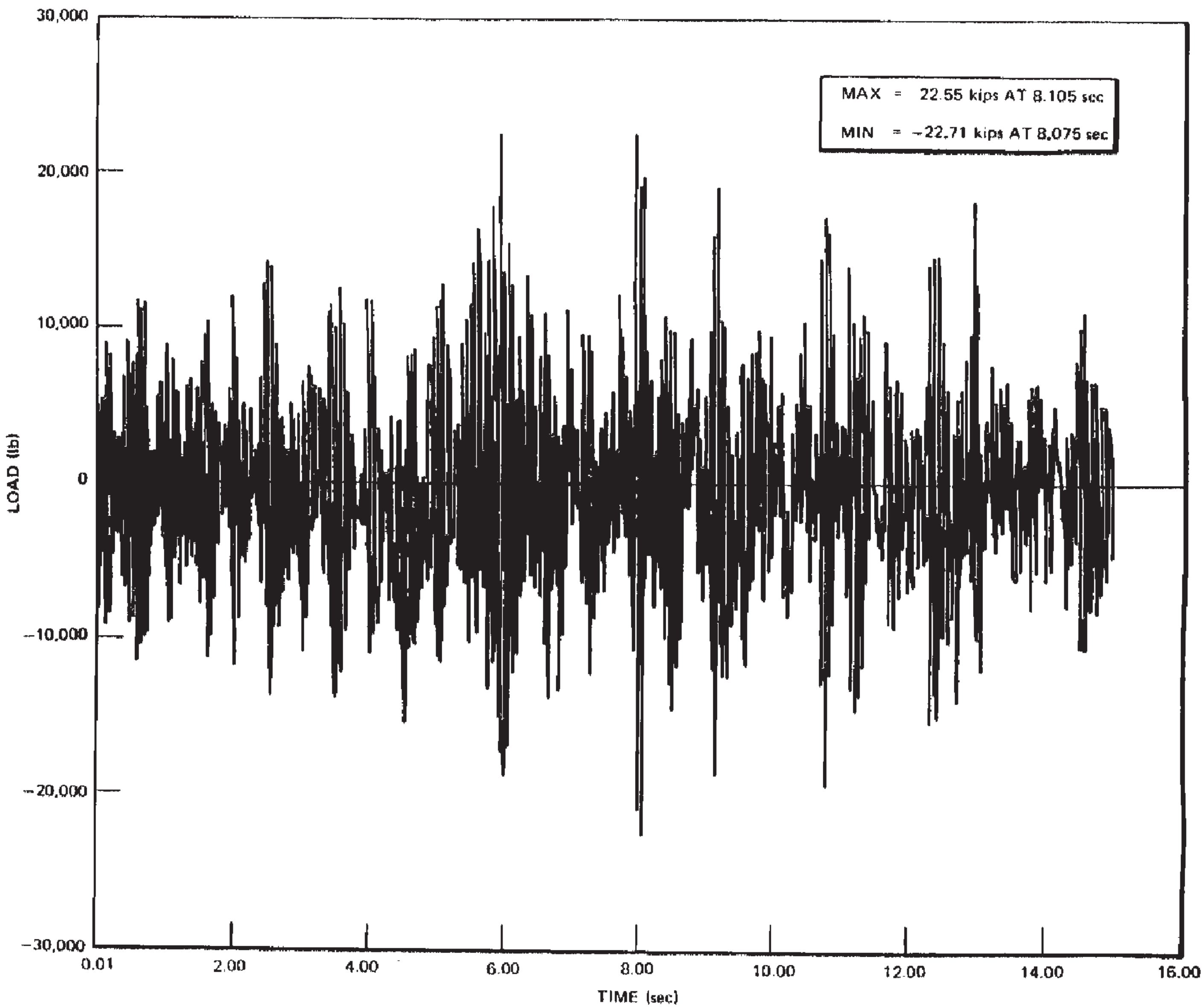
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO VERTICAL SSE

FIGURE 110.41-14, Rev 47

AutoCAD: Figure Fsar 110_41_14.dwg



SUSQUEHANNA-SSE-UNCRACKED-VERT LOAD FOR ELE 3 COMP 7 TYPE =2

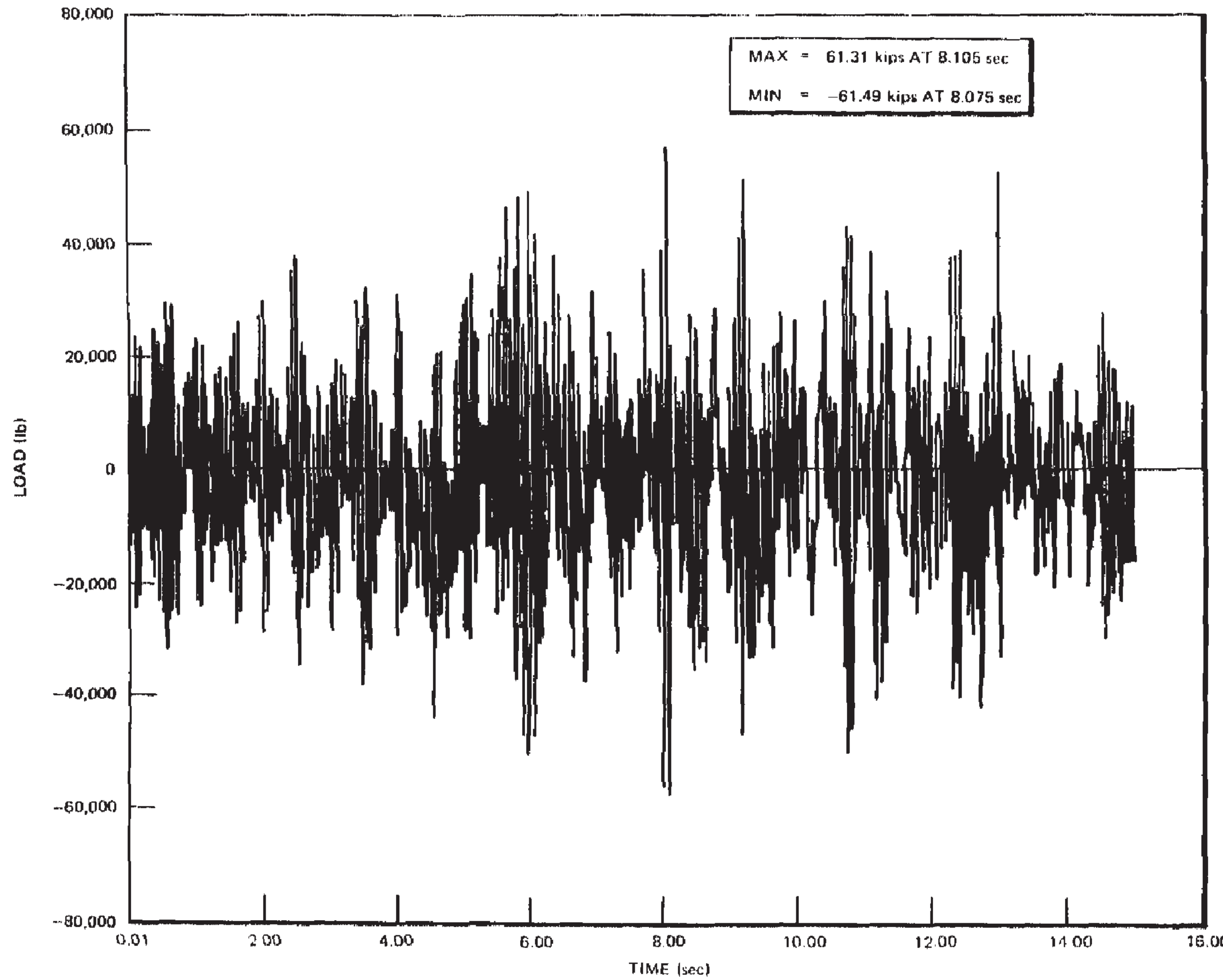
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO VERTICAL SSE

FIGURE 110.41-15, Rev 47

AutoCAD: Figure Fsar 110_41_15.dwg



SUSQUEHANNA-SSE-UNCRACKED-VERT LOAD FOR ELE = 5 COMP = 7 TYPE =2

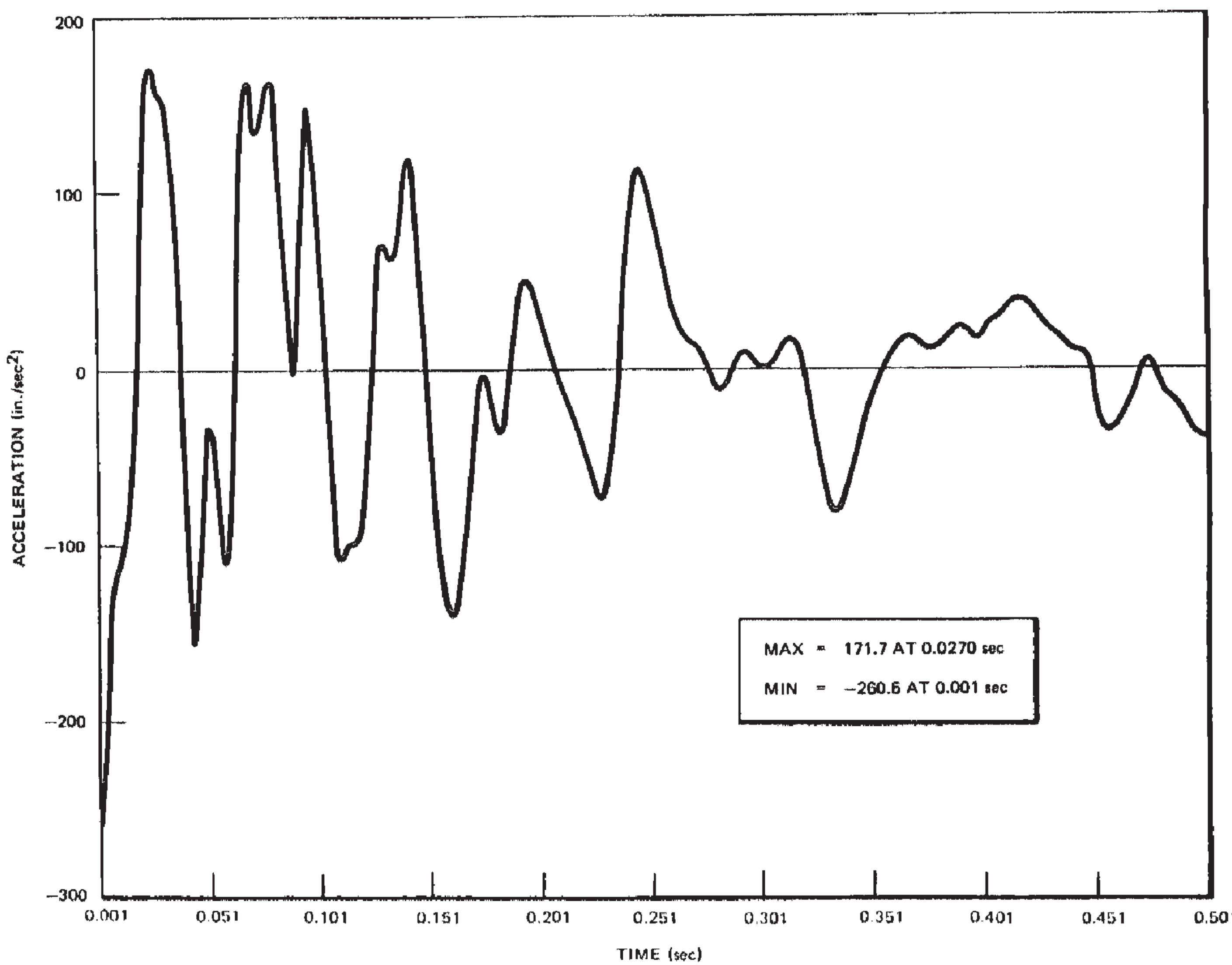
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO VERTICAL SSE

FIGURE 110.41-16, Rev 47

AutoCAD: Figure Fsar 110_41_16.dwg



SUSQUEHANNA-AP-RC-HORIZ ACCEL AT NODE= 7 FOR DOF= 1

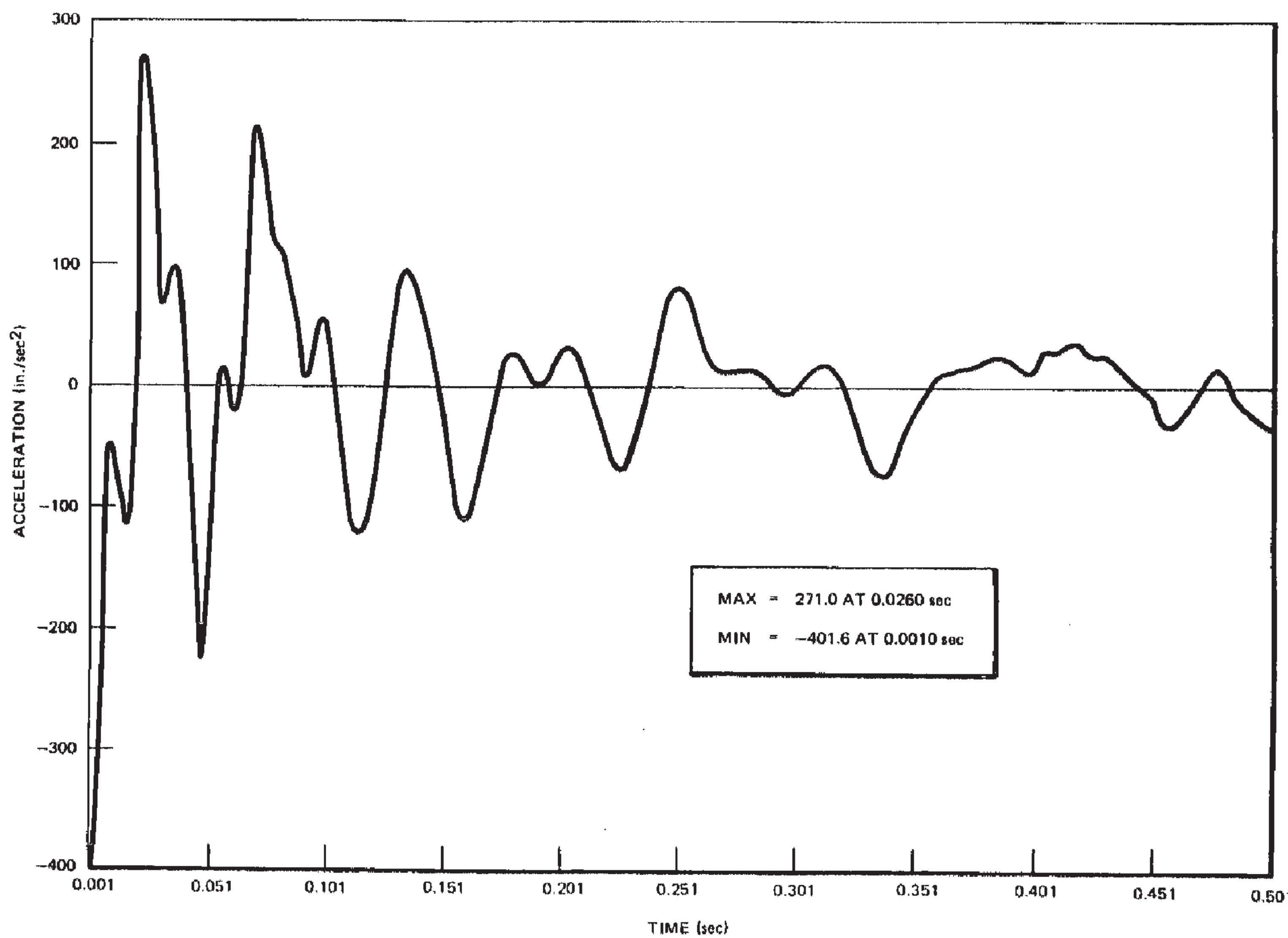
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO
RECIRCULATION PIPE BREAK

FIGURE 110.41-17, Rev 47

AutoCAD: Figure Fsar 110_41_17.dwg



SUSQUEHANNA-AP-RC-HORIZ ACCEL AT NODE= 47 FOR DOF =1

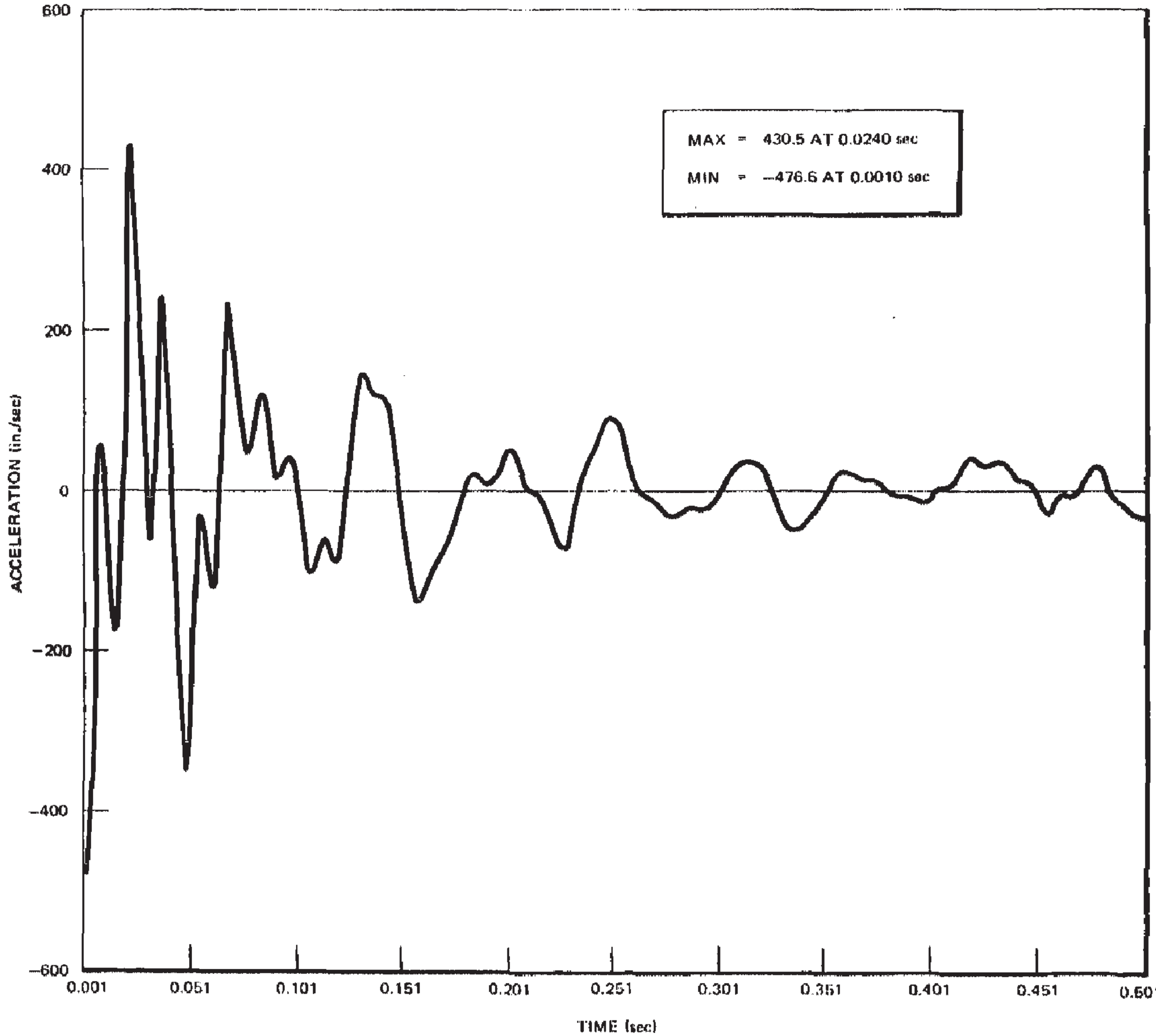
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO
RECIRCULATION PIPE BREAK

FIGURE 110.41-18, Rev 47

AutoCAD: Figure Fsar 110_41_18.dwg



SUSQUEHANNA-AP-RC-HORIZ ACCEL AT NODE 55 FOR DOF =1

FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO
RECIRCULATION PIPE BREAK

FIGURE 110.41-19, Rev 47

AutoCAD: Figure Fsar 110_41_19.dwg

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SUSQUEHANNA STEAM ELECTRIC STATION
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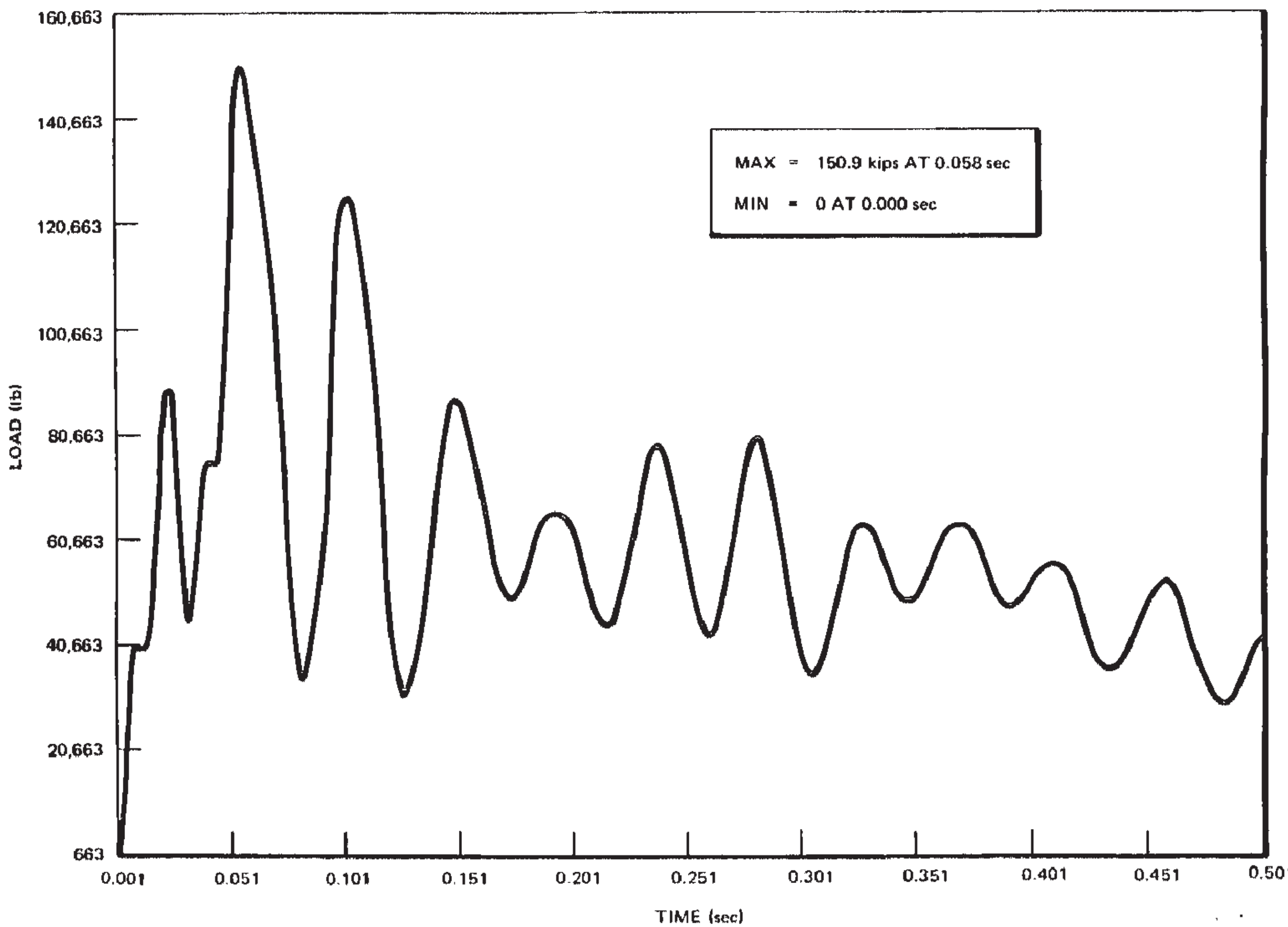
FINAL SAFETY ANALYSIS REPORT

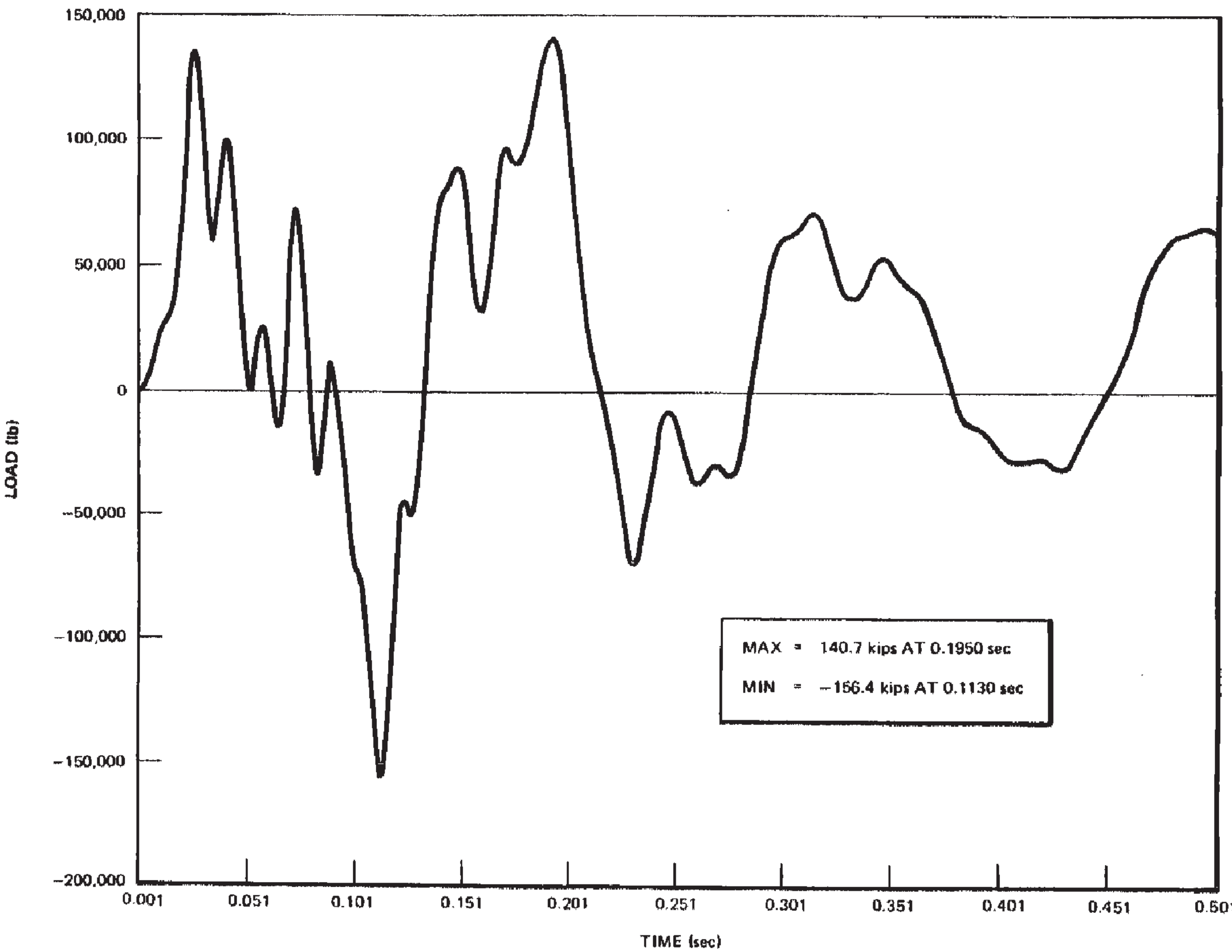
RESPONSE TIME HISTORY PLOT
DUE TO
RECIRCULATION PIPE BREAK

FIGURE 110.41-20, Rev 47

AutoCAD: Figure Fsar 110_41_20.dwg



SUSQUEHANNA-AP-RC-HORIZ LOAD FOR ELE = 7 COMP = 2 type = 2



SUSQUEHANNA-AP-RC-HORIZ LOAD FOR ELE 22 COMP 2 TYPE =2

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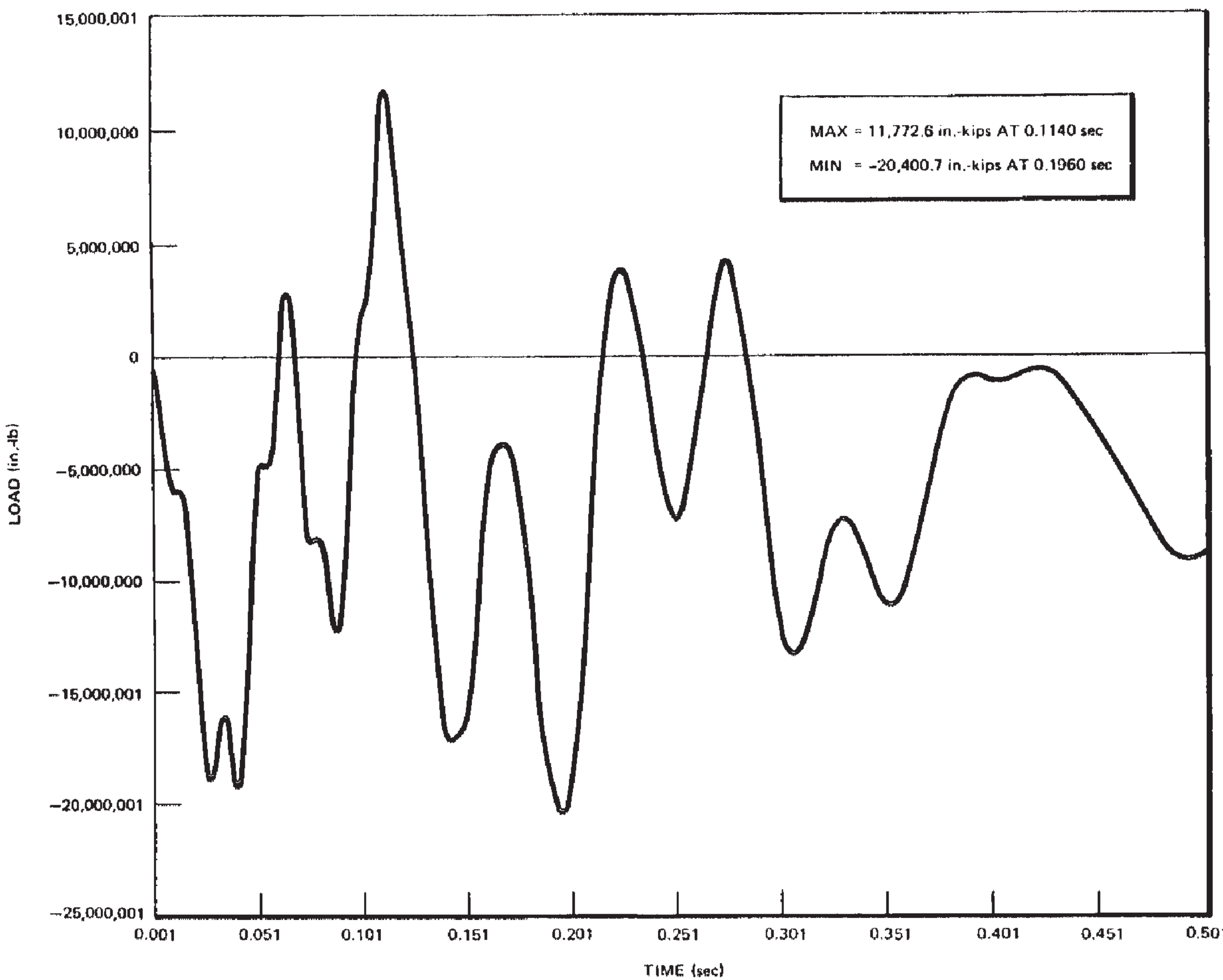
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO
RECIRCULATION PIPE BREAK

FIGURE 110.41-21, Rev 47

AutoCAD: Figure Fsar 110_41_21.dwg



SUSQUEHANNA-AP=RC-HORIZ LOAD FOR ELE 22 COMP 6 TYPE =2

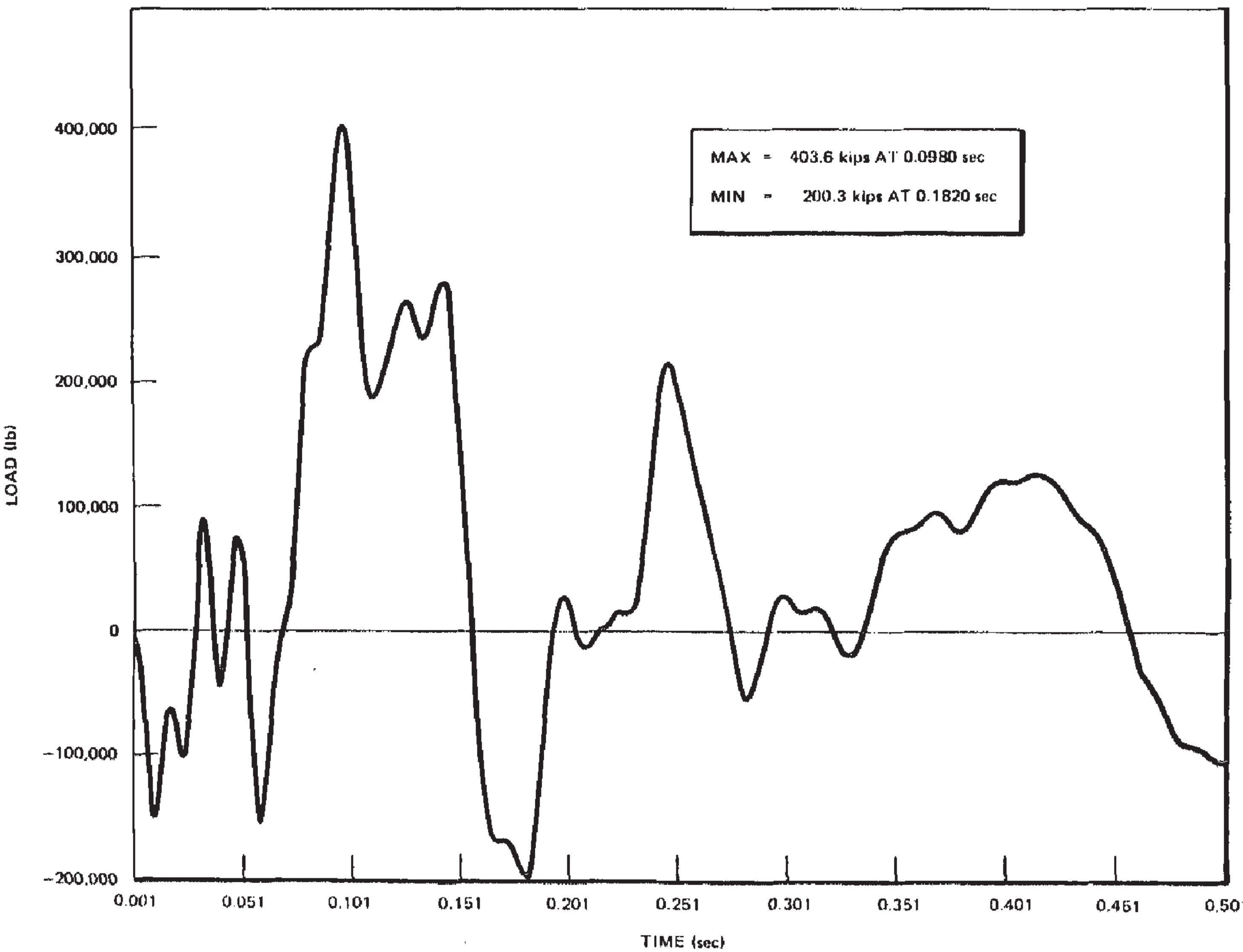
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO
RECIRCULATION PIPE BREAK

FIGURE 110.41-22, Rev 47

AutoCAD: Figure Fsar 110_41_22.dwg



SUSQUEHANNA-AP-RC-HORIZ LOAD FOR ELE = 31 COMP = 8 TYPE =2

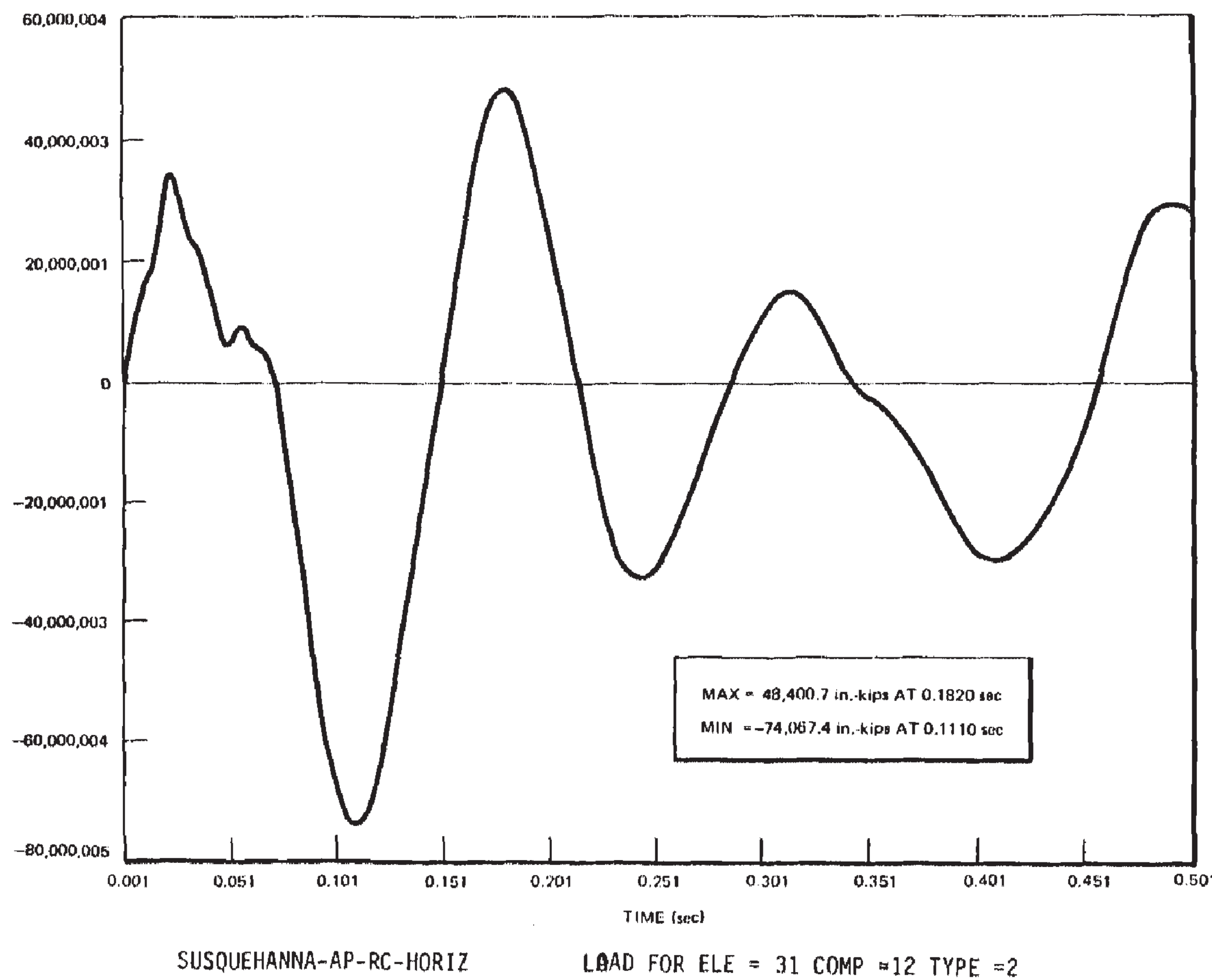
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO
RECIRCULATION PIPE BREAK

FIGURE 110.41-23, Rev 47

AutoCAD: Figure Fsar 110_41_23.dwg



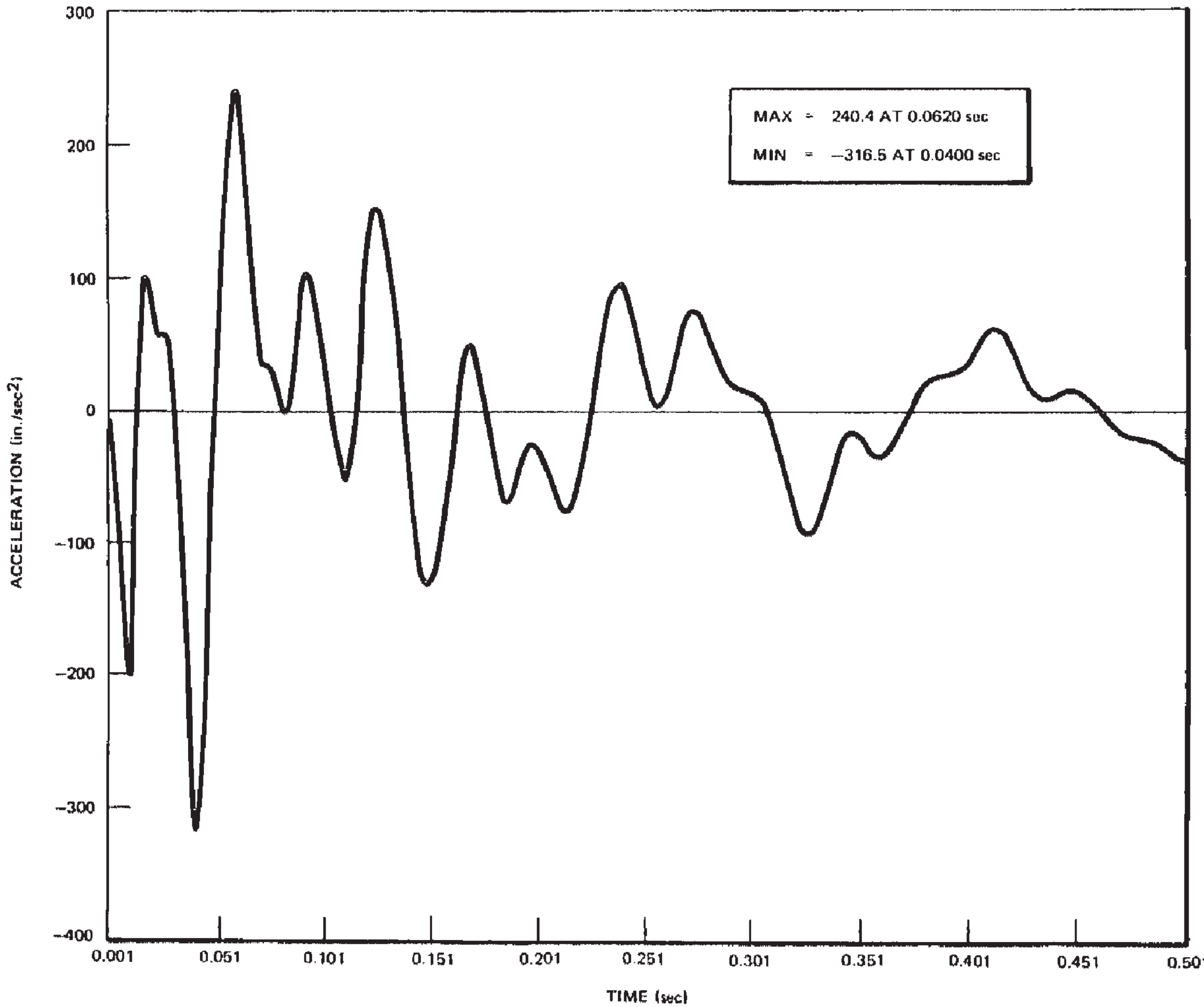
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO
RECIRCULATION PIPE BREAK

FIGURE 110.41-24, Rev 47

AutoCAD: Figure Fsar 110_41_24.dwg



SUSQUEHANNA-AP-FW- HORIZ ACCEL AT NODE = 7 FOR DOF = 1

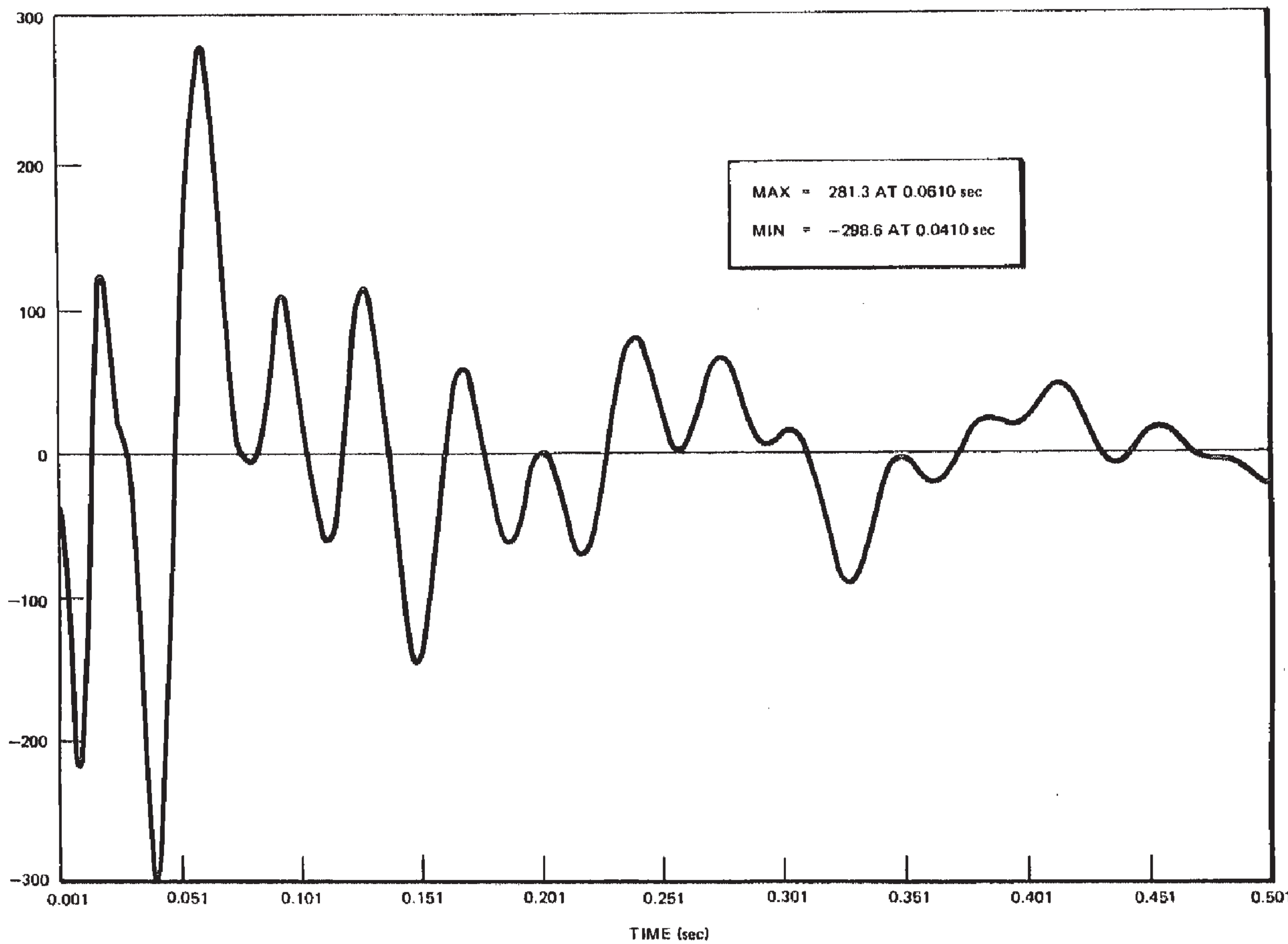
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

ANNULUS PRESSURIZATION
(HORIZONTAL)
FEEDWATER LINE BREAK

FIGURE 110.41-25, Rev 47

AutoCAD: Figure Fsar 110_41_25.dwg



SUSQUEHANNA-AP-FW-HORIZ   ACCEL AT NODE = 47 FOR DOF =1

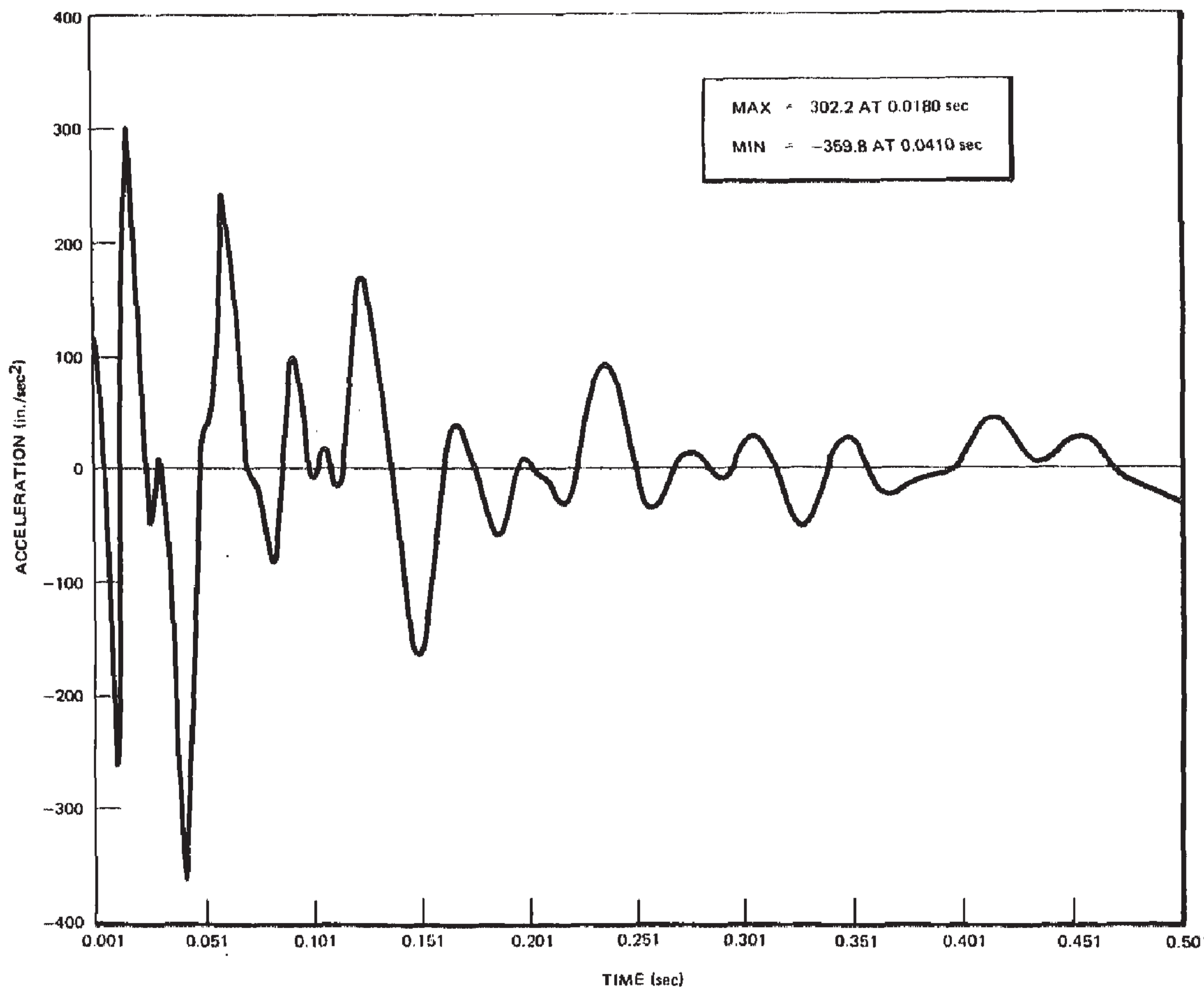
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

ANNULUS PRESSURIZATION
(HORIZONTAL)
FEEDWATER LINE BREAK

FIGURE 110.41-26, Rev 47

AutoCAD: Figure Fsar 110_41_26.dwg



SUSQUEHANNA-AP-FW-HORIZ ACCEL AT NODE = 55 FOR DOF =1

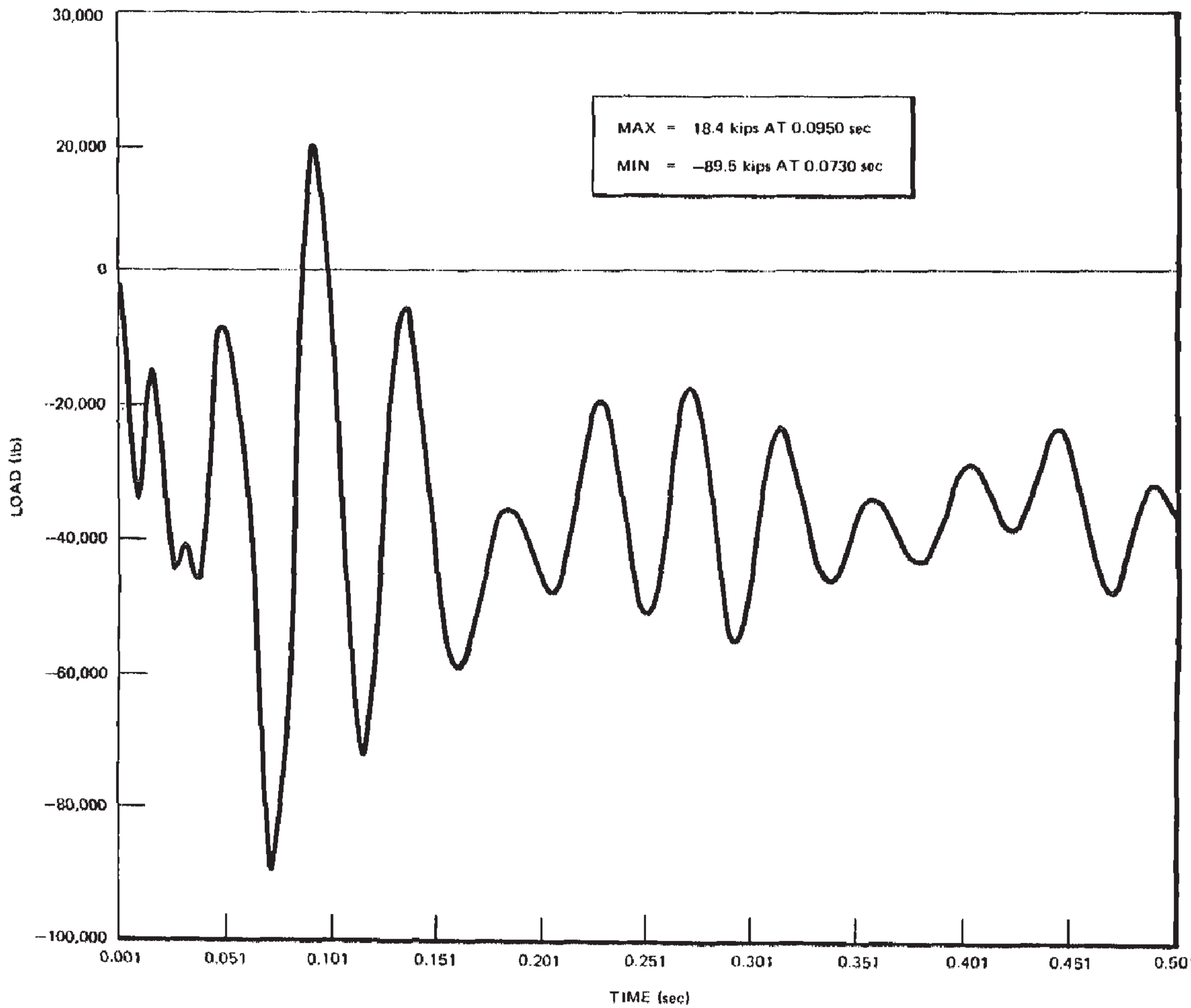
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

ANNULUS PRESSURIZATION
(HORIZONTAL)
FEEDWATER LINE BREAK

FIGURE 110.41-27, Rev 47

AutoCAD: Figure Fsar 110_41_27.dwg



SUSQUEHANNA-AP-FW-HORIZ LAOD FOR ELE = 7 COMP = 2 TYPE = 2

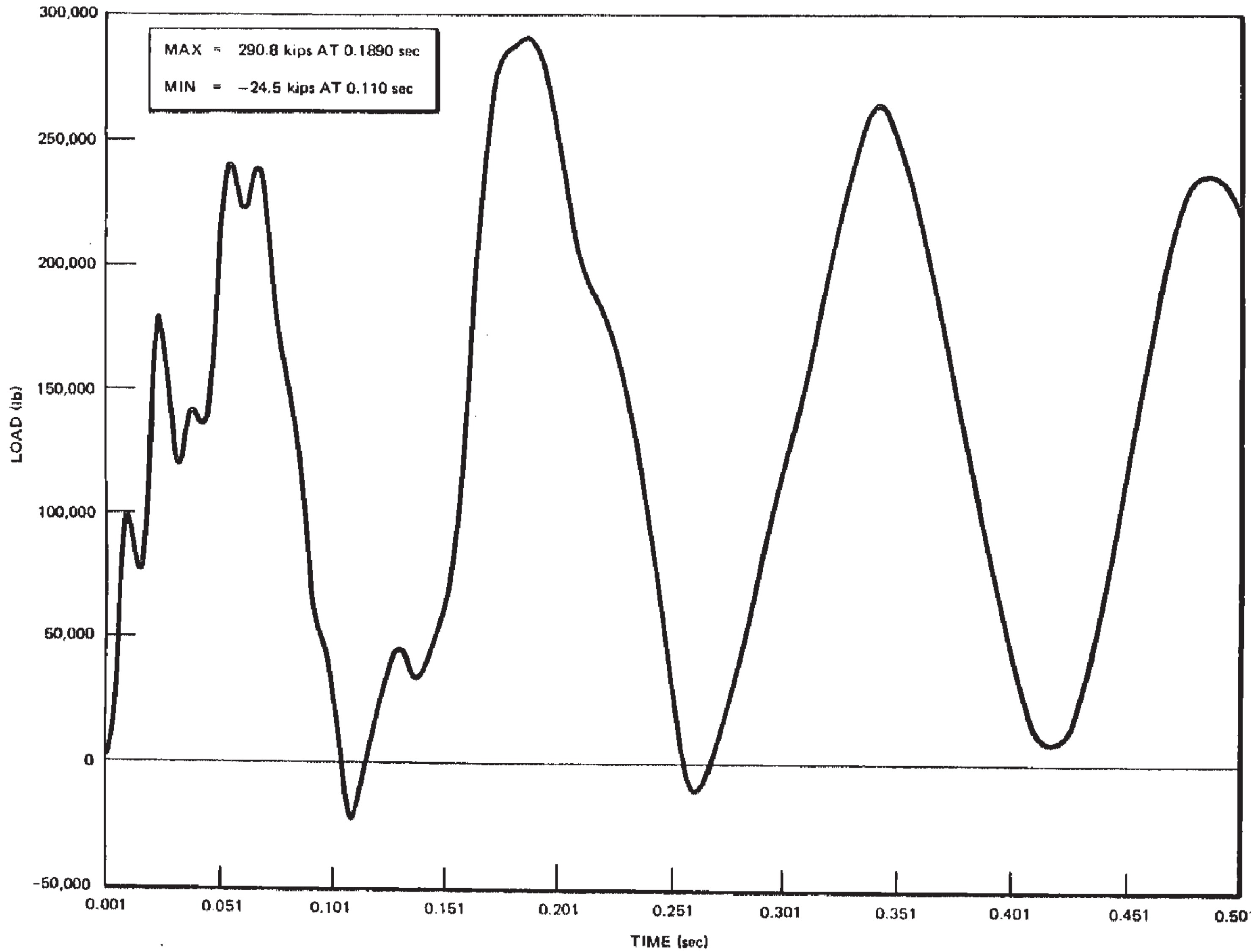
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

ANNULUS PRESSURIZATION
(HORIZONTAL)
FEEDWATER LINE BREAK

FIGURE 110.41-28, Rev 47

AutoCAD: Figure Fsar 110_41_28.dwg



SUSQUEHANNA-AP-FW-HORIZ LOAD FOR ELE = 22 COMP = 2 TYPE =2

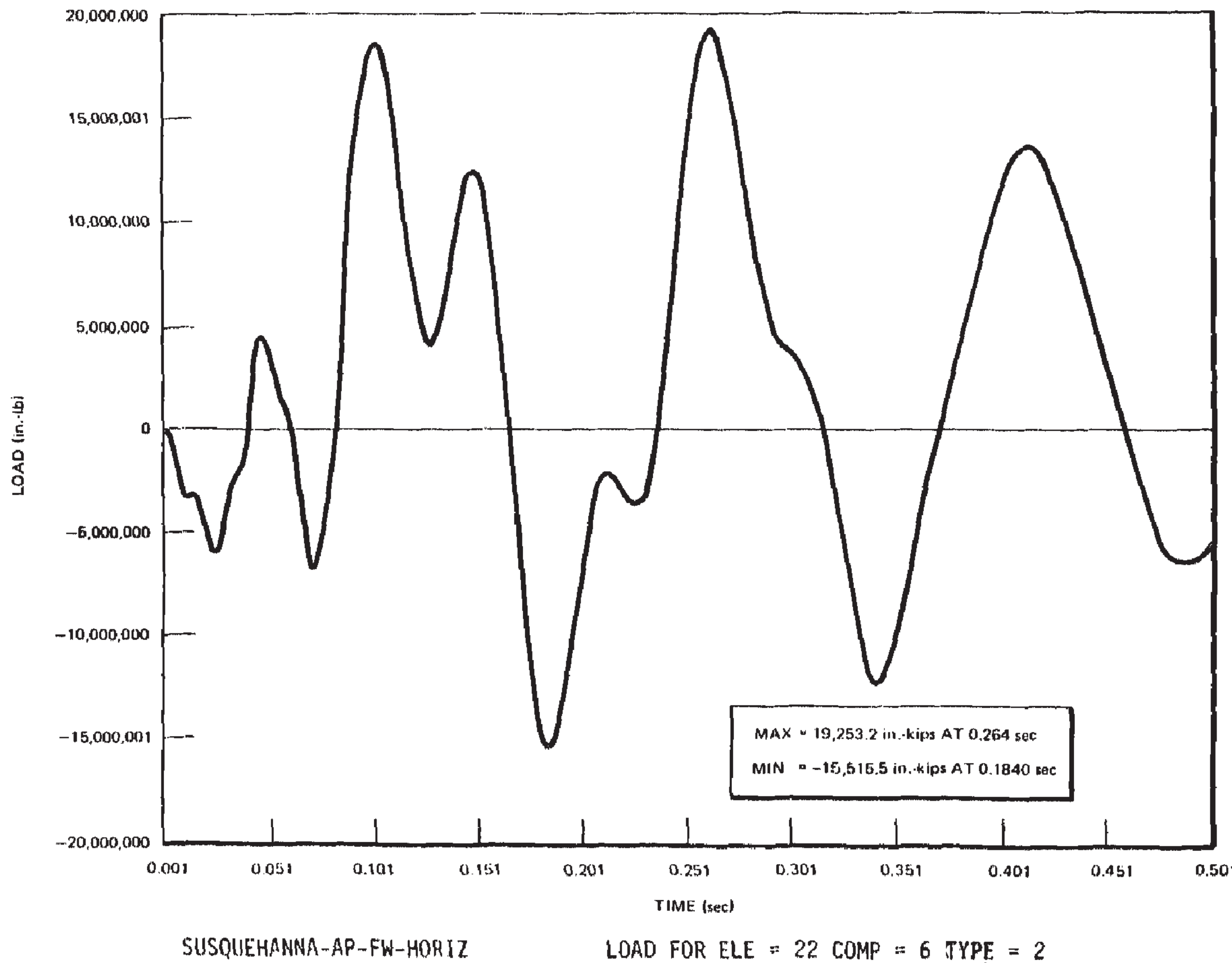
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

ANNULUS PRESSURIZATION
(HORIZONTAL)
FEEDWATER LINE BREAK

FIGURE 110.41-29, Rev 47

AutoCAD: Figure Fsar 110_41_29.dwg



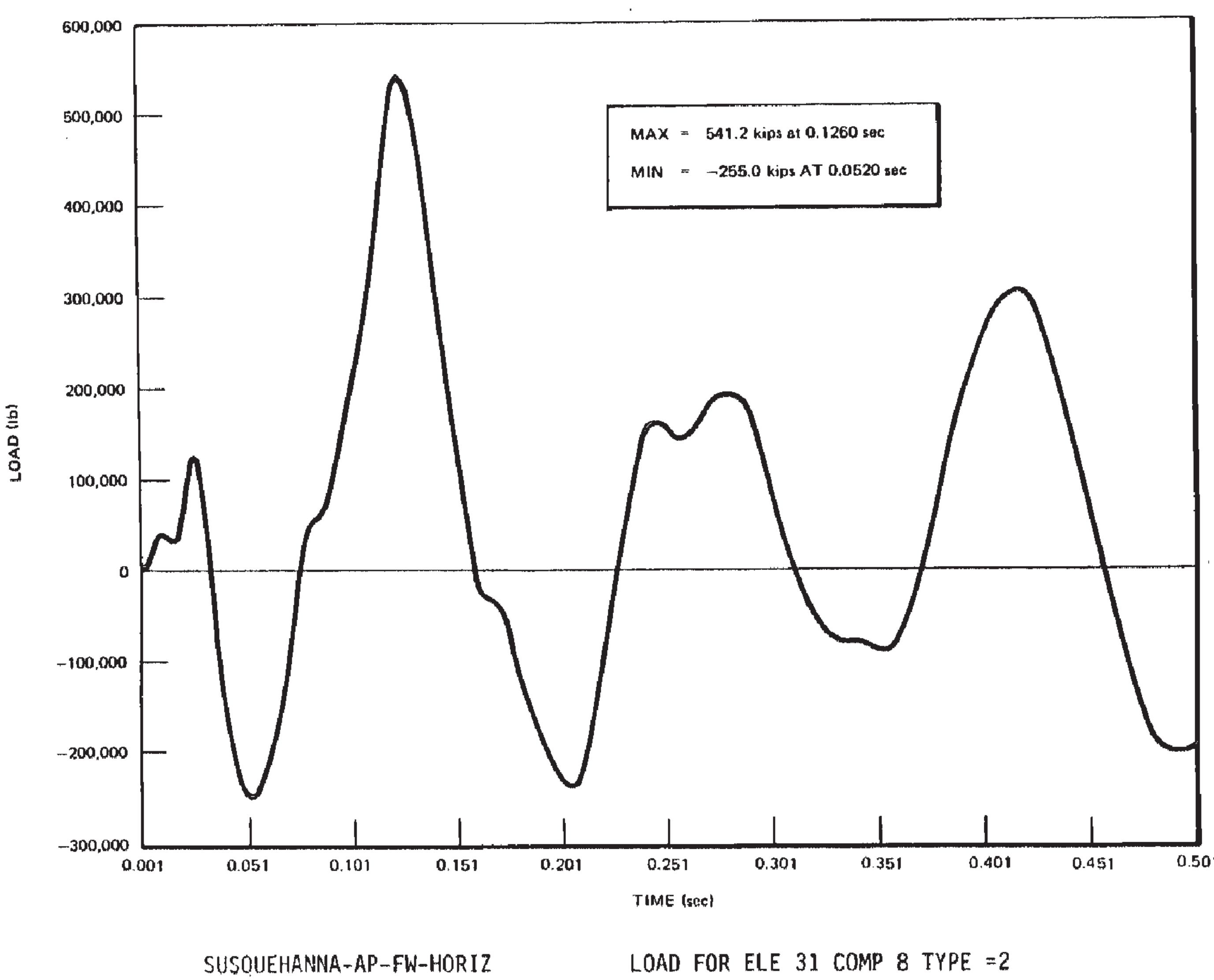
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

ANNULUS PRESSURIZATION
(HORIZONTAL)
FEEDWATER LINE BREAK

FIGURE 110.41-30, Rev 47

AutoCAD: Figure Fsar 110_41_30.dwg



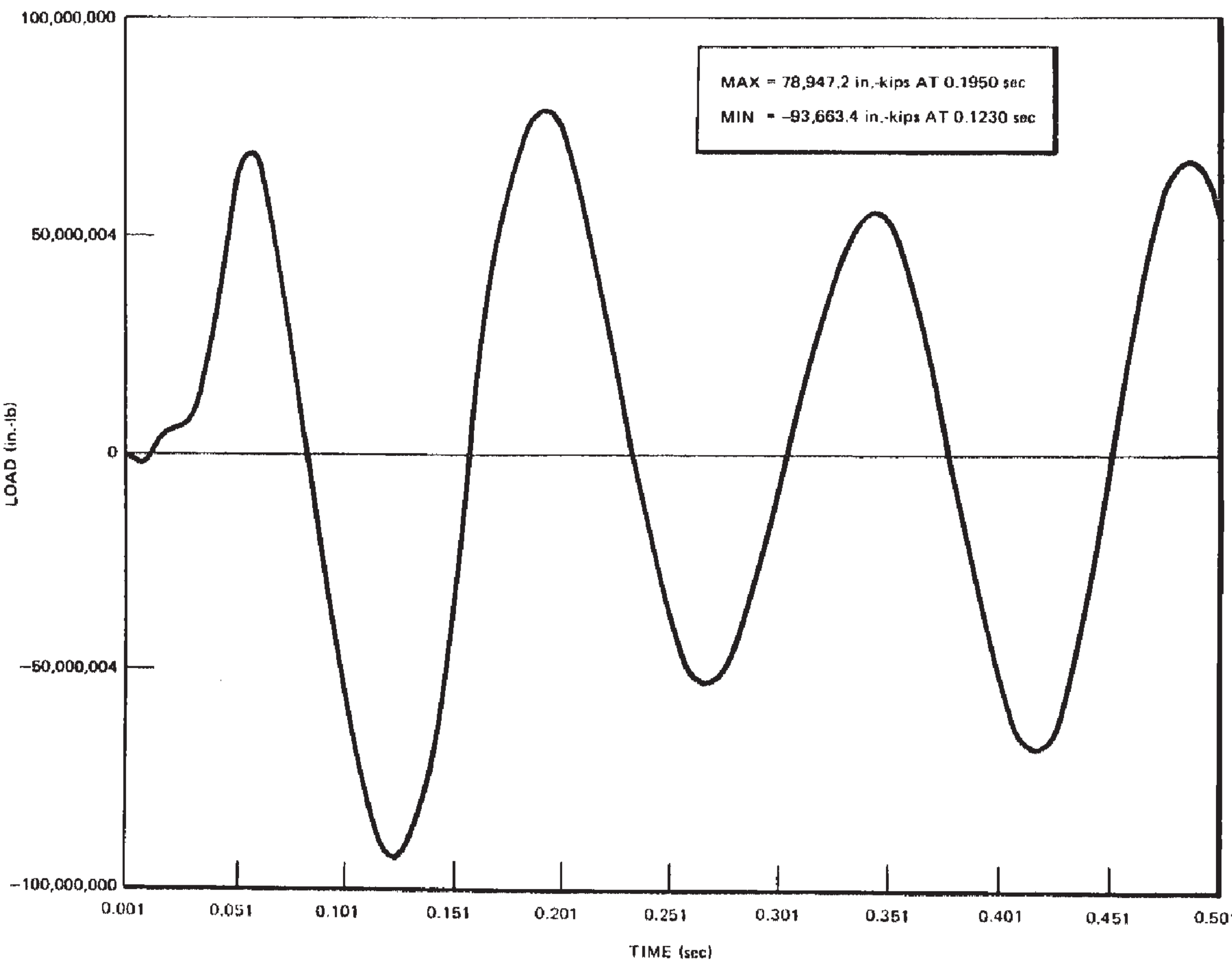
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

ANNULUS PRESSURIZATION
(HORIZONTAL)
FEEDWATER LINE BREAK

FIGURE 110.41-31, Rev 47

AutoCAD: Figure Fsar 110_41_31.dwg



SUSQUEHANNA-AP-FW-HORIZ LOAD FOR ELE 31 COMP = 12 TYPE =2

FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

ANNULUS PRESSURIZATION
(HORIZONTAL)
FEEDWATER LINE BREAK

FIGURE 110.41-32, Rev 47

AutoCAD: Figure Fsar 110_41_32.dwg

QUESTION 110.42

FSAR Sections 3.9.3, 3.9.4 and 3.9.5 reference several tables (3.9-2, 3.9-6, 3.9-14, etc.) that describe the various loading combination considered in the design of ASME Class 1, 2, and 3 components, component supports, core support structures, control rod drive components, and other reactor internals.

We have had discussions with the Mark II Owner's Group concerning the load combinations appropriate for the design of BWR Mark II plants. Our position with respect to load combinations has been documented as Attachment A to Enclosure 5 of the NRC Mark II Generic Acceptance Criteria for Lead Plants. This staff position is repeated here as Enclosure 110-2. These loading combinations are applicable to the Susquehanna plant.

Therefore, provide a commitment that all ASME Class 1, 2, and 3 components, component supports, core support structures, control rod drive components, and other reactor internals have been or will be analyzed or otherwise qualified in accordance with Enclosure 110-2, as modified by the following two clarifications:

(a) For load cases 1 and 2 all ASME Code Service Level B requirements are to be met, including fatigue usage factor requirements, and should take into account all SRV discharge load effects (initial actuation and continuous suppression pool vibratory) taken for the number of cycles consistent with the 40 yr. design life of the plant.

(b) For load case 10, SRV_1 should be assumed to be one SRV.

RESPONSE:

I. Non-NSSS

For load cases 1 and 2 as identified in Question 110.42, enclosure 110-2, all ASME Code Service Level B requirements, including fatigue consideration for Class 1 components, are met for piping in non-NSSS's scope.

For load case 10, SRV_1 (one SRV) is not considered in combination with DBA induced loads. However, the loads resulting from condensation oscillation and chugging are considered in combination with the effects of SRV_{ADS}.

FSAR Tables 3.9-6, 3.9-10 and 3.9-14 are revised to reflect the loading combinations and acceptance criteria that are used for ASME Code Class 1, 2 and 3 components and their supports.

II. NSSS

Load Case 1 combinations meet the cited staff position.

General Electric believes that the loading combination OBE + SRV_{All} (Load Case 2) ought to be considered as an Emergency condition. The classification of this low probability combination of loads as Emergency (Service Level C requirements) is consistent with 1) the encounter frequency of the OBE, 2) the number of combined stress cycles expected over the plant lifetime, and 3) the intent of the ASME code. However, response to continued regulatory staff inquiry, GE agreed to meet Upset limits (Service Level B requirements) without fatigue analysis. The considerations for not conducting the fatigue analysis involve the same technical justifications enumerated above.

For load case 10, SRV_s (one SRV) is not considered in combination with DBA-induced loads.

TABLE 110.42-1

ACCEPTANCE CRITERIA FOR NSSS PIPING & EQUIPMENT

Load Case	N	SRV_x	SRV_{ADS}	OBE	SSE	IBA[5]	DBA[5]	Acceptance Criteria
1	X	X						B[1]
2	X	X		X				B[6]
3	X	X			X			D[3]
4	X		X			X[2]		C
5	X		X		X	X[2]		D[3]
6	X				X		X[2]	D[3]
7	X							A
8	X			X				B
9	X		X	X		X		C[3]

NOTES:

(1) For load case 1, all ASME Code Service Level B requirements are to be met, including fatigue usage. All SRV discharge load effects will be combined with mechanistically associated loads and taken for the number of cycles consistent with the 40 year design life of the plant.

(2) Loading due to DBA/SBA/IBA is determined from rated steady-state conditions.

(3) Piping functional capability will be assured for essential piping per Enclosure 110-2 or NEDE 21985.

(4) Not used.

(5) IBA and DBA includes all associated loads such as annulus pressurization, pool swell, chugging, etc.

(6) For load case 2, all ASME Code service requirements are to be met, excluding fatigue usage.

(7) For specific load combinations refer to Table 3.9-2.

TABLE 110.42-1

ACCEPTANCE CRITERIA FOR NSSS PIPING & EQUIPMENT

Load Case	N	SRV_x	SRV_{ADS}	OBE	SSE	IBA(5)	DBA(5)	Acceptance Criteria
1	X	X						B(1)
2	X	X		X				B(6)
3	X	X			X			D(3)
4	X		X			X(2)		C
5	X		X		X	X(2)		D(3)
6	X				X		X(2)	D(3)
7	X							A
8	X			X				B
9	X		X	X				C(3)

NOTES:

(1) For load case 1, all ASME Code Service Level B requirements are to be met, including fatigue usage. All SRV discharge load effects will be combined with mechanistically associated loads and taken for the number of cycles consistent with the 40 year design life of the plant. These cycles also bound those expected for the period of extended operation of the plant.

(2) Loading due to DBA/SBA/IBA is determined from rated steady-state conditions.

(3) Piping functional capability will be assured for essential piping per Enclosure 110-2 or NEDE 21985.

(4) Not used.

(5) IBA and DBA includes all associated loads such as annulus pressurization, pool swell, chugging, etc.

(6) For load case 2, all ASME Code service requirements are to be met, excluding fatigue usage.

(7) For specific load combinations refer to Table 3.9-2.

QUESTION 110.43

Provide the bases for the allowable buckling loads, including the buckling allowable stress limit, under faulted conditions for all NSSS and BOP ASME Class 1 component supports. For the reactor vessel support skirt, provide a comparison of the calculated buckling loads against the critical buckling loads of the skirt under the most limiting faulted loading condition. Describe the analytical techniques used in determining both the calculated buckling loads under faulted conditions and the critical buckling loads of the Susquehanna support skirt. Provide the most limiting load combination considered in the buckling analyses for the reactor vessel support skirt.

RESPONSE:

Structural elements used in the support of ASME Class 1 components and piping are evaluated for buckling using a stress criteria that limits the allowable stresses for supports to 2/3 of the critical buckling stress.

Commercially available struts and snubbers have been specified in accordance with vendor catalogue data. The capacity of these items, stated in the catalogue, relative to buckling is limited by the maximum permissible pin to pin dimension. Calculations and tests substantiating these data are on file at the vendors engineering office.

Per GE design specification the permissible compressive load on the reactor vessel support skirt cylinder modeled as plate and shelltype component support was limited to 90 percent of the load which produces yield stress, divided by the safety factor for the condition being evaluated. The effects of fabrication and operational eccentricity was included. The safety factor for faulted conditions was 1.125.

An analysis of reactor pressure vessel support skirt buckling for faulted conditions shows that the support skirt has the capability to meet ASME Code Section III, Paragraph F-1370(c) faulted condition limits of 0.67 times the critical buckling strength of the support at temperature assuming that the critical buckling stress limit corresponds to the material yield stress at temperature. The faulted condition analyzed included the compressive loads due to the design bases maximum earthquake, the overturning moments and shears due to the jet reaction load resulting from a severed pipe, and the compressive effects on the support skirt due to the thermal and pressure expansion of the reactor vessel. The expected maximum earthquake loads for the Susquehanna 1 & 2 reactor vessel support skirts are less than 60% of the maximum design bases loads used in the buckling analysis described; therefore, the

expected faulted loads are well below the critical buckling limits of Paragraph F-1370(c) for this reactor vessel support skirt. The expected earthquake loads for this reactor were determined using the seismic dynamic analysis methods described in Section 3.7 of the Susquehanna 1&2 Safety Analysis Report.

QUESTION 110.44

For ASME Class 1, 2, and 3 components that could be exposed to jet impingement or pipe whip impact loads resulting from postulated pipe breaks in adjacent high energy piping, describe the procedure used to determine the stress levels in the components and all other components in the target system resulting from exposure to such loads in combination with those resulting from other applicable loads. Provide specific assurance that the calculated stress levels are kept below ASME Service Level D limits or, if applicable, more conservative limits for active components or where piping functional capability must be assured.

RESPONSE:

For a discussion of the methods used for determining the jet impingement or pipe whip impact loads resulting from postulated pipe breaks in adjacent high energy piping, see BN-TOP-2, Rev. 2. Because of the high degree of separation in Susquehanna SES, we have not identified any postulated pipe break location in which loads on an adjacent pipe would require a stress analysis. This situation is a result of applying the criteria in BTP MEB 3-1 and APCSB 3-1 for those systems and equipment that must remain operational after a pipe break.

QUESTION 110.45

Your FSAR indicates that active valves will be qualified for operability under seismic loading on a prototype basis. We agree that a prototypical test can qualify a limited range of similar valves. Your FSAR does not sufficiently describe the characteristics you consider in determining that a valve is similar to the tested prototype valve, and therefore can be qualified by analysis only.

Provide a discussion of how you establish the "similarity" of valves to a tested prototype. This discussion should include, but not be limited to, characteristics such as valve type, size, geometry, pressure rating, stress level, manufacturer, actuator type, and actuator load rating.

RESPONSE:

For response see revised Subsection 3.9.3.2b.2.

Additional information regarding bases for demonstrating operability under seismic loading used for qualifying active values in the GE scope at supply is provided below:

Recirculation System Gate Valves

Operability of recirculation valves was demonstrated by tests involving the value/actuator combination and the valve singly. Similarly designed valves that by analysis exhibited greater stress-to-acceleration relationships in the extended structure than those used in SSES have been successfully seismically tested. All recirculation gate valves for SSES are provided by one vendor (LUNKENHEIMER) and are of the same configuration (i.e., valve body with weld end preps, bonnet, yoke and motor operator). All electrical actuators used are of similar design (LIMITORQUE-SMB). A complete range of sizes of actuators, covering the SSES installed sizes, have been tested and successfully qualified.

Main Steam Isolation Valves

Operability of the MSIV's was demonstrated by analysis and by testing. A dynamic loading test and a static deflection test have been completed on the representative SSES valve actuator assembly. Both tests demonstrated operability of the valve to the specified limits.

Safety/Relief Valves and Standby Liquid Control Valves

Operability of S/R valves and SBLC valves was demonstrated by successfully testing representative SSES valves (production units).

QUESTION 110.46

Describe in more detail the dynamic testing performed to demonstrate the operability of safety related NSSS and BOP snubbers under upset, emergency and faulted load combinations. Describe the magnitudes of the applied loads, the frequency content, and the number of load cycles at each applied load level in these tests.

RESPONSE:

For response see Subsection 3.9.3.4.

QUESTION 110.47

As required by 10 CFR 50.55a(g) we request that you submit your preservice and initial 20 month inservice testing program for pumps and valves. Enclosure 110-3 provides a suggested format for this submittal and a discussion of information we require to justify any relief requests.

RESPONSE:

The preservice and initial 20-month inservice testing program for pumps and valves has been submitted under separate cover.

QUESTION 110.48

It is not clear from the FSAR how the seismic analyses of seismic Category I electrical and mechanical equipment have taken into consideration all three seismic accelerations (i.e., x, y and z directions) acting on the equipment.

Regulatory Guide 1.92 provides methods acceptable to the staff for combining the responses to the three spatial components of seismic excitation.

Describe how your analyses have considered the three spatial components of seismic excitation.

RESPONSE:

Consideration of the three spatial components of the earthquake motion for the equipment has been addressed in Subsection 3.7b.3.6. It is considered in the same manner as for the structures in Subsection 3.7b.2.6 which states that "----the response value used is the maximum value obtained by adding the response due to the vertical earthquake with the larger value of the responses due to one of the horizontal earthquakes by the absolute sum method."

The use of three components of earthquake motion, as described by Regulatory Guide 1.92, Revision 1, was not a requirement for the issuance of the Susquehanna SES construction permit. Therefore, the majority of NSSS equipment analyzed used the methods described in Section 3.7(a).3.6 for combining the responses to the three spatial components of seismic excitation. However, for all current analyses of the NSSS ASME safety Class I piping the total seismic response in a given direction is predicted by combining the responses calculated from the two horizontal and the vertical directional inputs. Both time history and response spectra methods are used to compute the responses.

Where the time history analysis method is used to compute the responses from the three directional earthquakes, the vector sum at every time step is used to calculate the maximum combined response.

Where response spectra method is used, the structural responses to each of the three components of earthquake motion are combined by the method of square root of the sum of squares (SRSS).

QUESTION 110.49

Hydrodynamic vibratory loadings result from the flow of a steam-water-air mixture into the suppression pool. This flow may result from SRV actuation or from a postulated pipe break. In either case the resultant vibration of the suppression pool may affect components in other portions of the reactor building. Therefore, hydrodynamic vibratory loadings of various magnitude any frequency content can be associated with the following cases: SRV_1, SRV_x, SRV_{ADS}, SRV_{ALL}, IBA and DBA.

The staff will require that electrical and mechanical equipment required for cold shutdown be demonstrated capable of performing their safety function under the most severe of the following combinations of seismic and hydrodynamic vibratory loadings:

(1) SRV_x or SRV_{ALL} (whichever is controlling) + OBE
(2) SRV_x or SRV_{ALL} (whichever is controlling) + SSE
(3) SRV_{ADS} + OBE + IBA
(4) SRV_{ADS} + SSE + IBA
(5) SSE + DBA
(6) SRV_1 + SSE + DBA

Provide a commitment that all NSSS and BOP seismic Category I mechanical and electrical equipment will be qualified for the most severe combined seismic and hydrodynamic vibratory loadings. The LaSalle (docket 05000373) and Zimmer (docket 05000358) plants have stated that, in general, the SRV_{ALL} case imposes the most severe hydrodynamic vibratory loadings on safety-related equipment. However, this does not preclude the possibility that other hydrodynamic loads might be limiting for particular components at your plant. As noted above, you should consider the most limiting case.

RESPONSE:

Procedures for the assessment and requalification of NSSS and BOP Category I mechanical and electrical equipment for the additional hydrodynamic loads has been described in Sections 7.1.6 and 7.1.7 respectively of the Design Assessment Report (DAR). Basically, the hydrodynamic loads which are comprised of SRV loads and LOCA related loads are added to OBE or SSE by the absolute sum method. SRV loads consider the enveloping case that includes all appropriate pressure traces for axisymmetric and asymmetric discharges. LOCA related loads consider the enveloping case that includes pool swell/large bubble loading, condensation-oscillation loading and chugging loading. The load combinations and the capability assessment criteria for the equipment assessment are described in Sections 5.7 and 6.7 respectively of the DAR. These load combinations

indicate that the effects of the seismic and the hydrodynamic loads are combined by the absolute sum method. If an equipment marginally fails to qualify for the dynamic loads by the absolute sum method, such equipment is qualified by combining the dynamic loads by the SRSS method.

However, if an equipment did not qualify by the SRSS method, it will be redesigned for the dynamic loads combined by the absolute sum method.

Subsections 3.9.2.2a and 3.9.2.2b and Section 3.10 have been revised to include this information.

All NSSS Seismic Category I equipment required for cold shutdown will be demonstrated to be capable of performing their safety function under the most severe of the following combinations as appropriate:

(1) SRV_x or SRV_{ALL} (whichever is controlling) + OBE
(2) SRV_x or SRV_{ALL} (whichever is controlling) + SSE
(4) SRV_{ADS} + SSE + IBA (this case envelopes case (3) above)
(5) SSE & DBA
(6) Exception is taken to this load combination consistent with the position of the Mark II owners group.

QUESTION 110.50

A review of the design adequacy of your safety-related electrical and mechanical equipment under seismic and hydrodynamic loadings will be performed by our Seismic Qualification Review Team (SQRT). A site visit at some future date will be necessary to inspect and otherwise evaluate selected equipment after our review of the following requested information.

The SQRT effort will be primarily focused on two subjects. The first is the adequacy of the original single-axis, single-frequency tests or analyses of equipment qualified per the criteria of IEEE Std. 344-1971.

The second subject is the qualification of equipment for the combined seismic and hydrodynamic vibratory loadings. The frequency of this vibration may exceed 33 hertz and negate the original assumption of a components rigidity in some cases.

Attached Enclosure 110-4 describes the SQRT and its procedures. Section V.2.A requires information which you should submit so that SQRT can perform its review.

Several of the BWR Mark II OL applicants have stated in their Closure Reports that equipment will be qualified for the SRSS combination of the hydrodynamic and seismic required response spectra (RRS). Similarly, when qualified by analysis, the peak dynamic responses of the equipment to the hydrodynamic and seismic loads will be combined by SRSS. The combining of SRSS of either the RRS or peak dynamic responses for hydrodynamic and seismic loadings is not acceptable at this time.

To aid the staff in its review, provide a compilation of the required response spectra listed below for each floor of the seismic Category I buildings at your plant.

(1) The RRS for the OBE or SSE, whichever is controlling. If the OBE is controlling, explain why.
(2) The controlling hydrodynamic RRS
(3) Items (1) and (2) combined by SRSS
(4) Items (1) and (2) combined by absolute sum.

RESPONSE:

The concerns raised by this question are addressed in the SRQT submittals of December, 1980, January, 1981, and February, 1981.

QUESTION 110.51

FSAR Sections 3.10b.1.2.6.2(g) states that "the above tests are used to validate an analytical method which is subsequently used without the actual test data." Describe this analytical method with emphasis on the following:

(1) Describe how this method considers the multifrequency, triaxial response of the panel.

(2) Describe how this method predicts resonant response of the panel and the corresponding amplified motion at the instrument mounting locations.

(3) Describe how this method has been verified by test or analysis and any limitations of the method.

RESPONSE:

See Subsections 3.10b.1 and 3.10b.2 for response.

QUESTION 110.52

In FSAR Sections 3.10b.2.4 and 3.10b.2.5 you state that a panel is "required to withstand the seismic level of 1.5 g units." Then you state that "the application seismic environment is established by the particular floor spectra." Clarify these statements to indicate their interrelationship, to define the term "application seismic environment," and to describe more clearly how the required response spectra for the component was derived.

RESPONSE:

See Subsections 3.10b-1 and 3.10b.2 for response.

QUESTION 110.53

Your discussion of Regulatory Guide 1.84 implies that ASME Code Cases not approved by this guide were used in the design of NSSS Class 2 and 3 components. Provide a list of each unapproved Code Case used in the Susquehanna design. This information is needed so that we can complete our review of the design criteria that has been used for NSSS ASME Class 2 and 3 components.

RESPONSE:

In the case of NSSS Class 2 and 3 components, no Code Cases were used in the Susquehanna design.

QUESTION 110.54

The second paragraph of Section 3.7a.3.6.1 (first full paragraph on page 3.7a-15) indicates that the seismic design is based on the largest absolute value of the algebraic sum of one horizontal response and the vertical response. (i.e., the larger of [x + y] or [y + z].

It is the Staff's position that this may lead to non-conservative results because of opposite signs tending to cancel each other when not justified on a time history basis. Therefore, the appropriate method should be to use the largest value of the sum of the absolute values of one horizontal response and the vertical response. (i.e., the larger of [x] + [y] or [Y] + [z]. Provide a commitment to meet this position.

RESPONSE:

The seismic design of NSSS piping and equipment for combining the responses to the spatial components of seismic excitation will comply with the staff position of the largest value of the sum of the absolute values of one horizontal response and the vertical response (i.e., the larger of [x] + [y] or [y] + [z]. However it should be noted that the "SSES" test in Section 3.7a.3.6.1 is correct and is also consistent with the staff position because in the response spectrum method of modal analysis the modal responses are first combined by "SRSS" method and then the responses to the spatial components of seismic excitation which are devoid of sign are combined by the absolute sum method. Therefore, it is immaterial if the responses are combined as [x + y] or [y + z] or [x] + [y] or [y] + [z].

See also the response to 110.48.

QUESTION 110.55

Section 3.7a.3.7 (page 3.7a-15) indicates that all modes are combined using the SRSS method when the response spectra method of modal analysis is used.

It is the Staff's position that this may lead to non-conservative results when dealing with closely spaced modes. Several acceptable procedures for combining modal responses, when closely spaced modes are present, are contained in Regulatory Guide 1.92. Provide a commitment to meet the procedures in Regulatory Guide 1.92.

RESPONSE:

The seismic design of Susquehanna SES was established prior to the issuance of Regulatory Guide 1.92 and therefore the consideration of closely spaced modes in the response spectrum method of seismic analysis as described in this Regulatory Guide was not a licensing requirement for Susquehanna SES. However, the seismic analysis of the NSSS ASME Safety Class 1 piping for SSES was updated by employing the double sum method of Regulatory Guide 1.92 for combining closely spaced modal responses. For other NSSS equipment where the response spectrum method of seismic analysis was used, the square root of the sum of the squares method (SRSS) was used to combine the closely spaced modes.

QUESTION 110.56

The second paragraph of Section 3.7b.3.7 indicates that modal responses will be combined in accordance with Section 5.2 of BP-TOP-1, Rev. 2. This referenced topical report has not been accepted by the Staff and a later revision (Rev. 3), which revises Section 5.2, is currently under review by the Staff. Therefore, to eliminate any possible confusion as to the method(s) used for this plant, it is requested that you clearly indicate the method(s) of combining modal responses which were used.

Your attention is also directed to Question 110.55.

RESPONSE:

Refer to revised Subsection 3.7b.3.7 for response to this question.

QUESTION 110.57

It is the staff's position that all essential safety-related instrumentation lines should be included in the vibration monitoring program during pre-operational or startup testing. We require that either a visual or instrumented inspection (as appropriate) be conducted to identify any excessive vibration that will result in fatigue failure.

The essential instrumentation lines to be inspected should include the following:

a) Reactor pressure vessel level indicator instrumentation lines (used for monitoring both steam and water levels)
b) Main steam instrumentation lines for monitoring main steam flow (used to actuate main steam isolation valves during high steam flow)
c) Reactor core isolation cooling (RCIC) instrumentation lines on the RCIC steam line outside containment (used to monitor high steam flow and actuate isolation)
d) Control rod drive lines inside containment (not normally pressurized but required for scram)

RESPONSE:

The essential instrumentation lines will be inspected as part of the pre-operational or startup testing for excessive vibration levels although typically, these lines do not experience high vibration levels.

a) Reactor pressure vessel level indicator instrumentation lines will be walked down after installation by cognizant design personnel to assure that the piping and constraints are such that the steady state vibratory effects of RPV induced vibration are minimized. The instrumentation lines and constraints in the cold condition will exhibit like vibratory behavior as in the hot condition. Therefore, a visual inspection of the RPV level instrumentation lines will be made during pre-operation or startup testing during a recirc pump flow. The acceptance criteria for the testing is an in FSAR Section 3.9.2.1b.2.

b) Main steam instrument lines for monitoring main steam flow will be walked down after initial operation by personnel to assure that the steady state vibratory effects will be minimized since the source of any vibration would be main steam flow, a visual inspection is impractical.

Inspection using remote instrumentation and evaluation as outlined in FSAR Section 3.9.2.1b.2 will be performed or, if it can be demonstrated that the first mode vibration of the instrument line has a frequency greater than main steam line significant vibration mode freq., the instrumentation lines will be considered dynamically isolated from the main lines and, as such, require no vibration monitoring.

c) HPCI and RCIC instrument lines (to monitor high steam flow) will be walked down by cognizant design personnel after installation and prior to startup to assure that steady state vibratory effects will be minimized. Any vibration in these lines will result from steady state vibrations of the large bore HPCI and RCIC turbines steam supply lines. These steam supply lines are included in the steady state power escalation testing. Remote instrumentation will be placed on these steam supply lines to assure that excessive vibration levels do not exist. The remote instrumentation would identify any large bore piping line. The test data will be evaluated as described in FSAR Subsection 3.9.2.1b.2.

d) For Unit 1, the control rod drive inside containment will be visually inspected during cold recirc. pre-operational or startup flow testing for steady state vibratory effects and during pre-operation rod insertion/withdrawal testing for dynamic transient effects.

For Unit 2, the control rod drive inside containment will be visually inspected only during preoperational rod insertion/withdrawal testing for dynamic transient effect (Ref. PLA-2142, dated March 29, 1984). The acceptance criteria for the testing is in FSAR Subsection 3.9.2.1b.2.