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.

RESPONSE:

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.

[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 upset primary stress limits 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

	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

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REACTOR PRESSURE VESSEL AND INTERNALS HORIZONTAL MATHEMATICAL MODEL FOR EARTHQUAKE AND ANNULUS PRESSURIZATION LOADING

FSAR FIGURE 110.41-1

PP&L DRAWING

Security-Related Information
Figure Withheld Under 10 CFR 2.390

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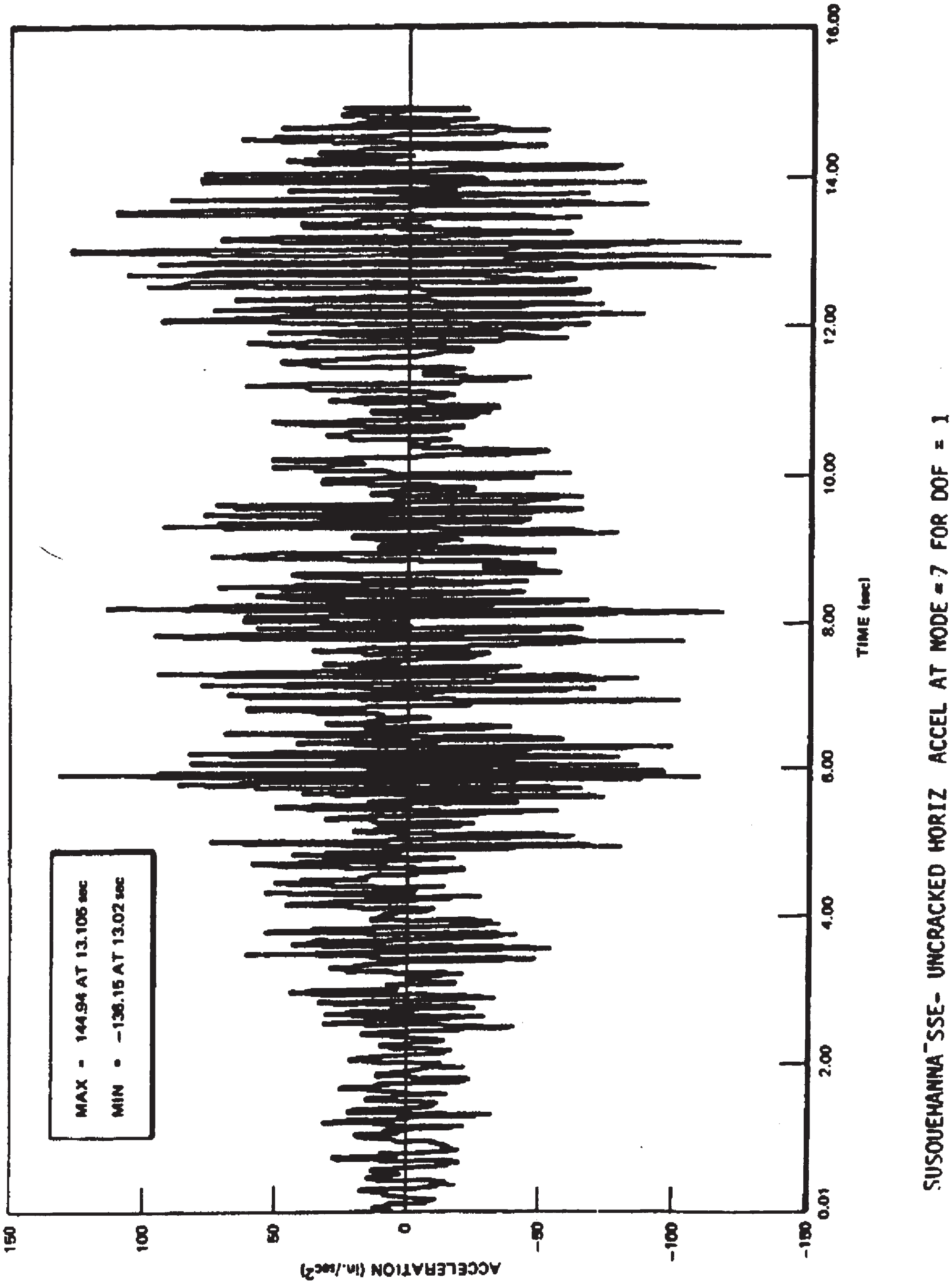
REACTOR PRESSURE VESSEL AND INTERNALS VERTICAL DYNAMIC MODEL FOR EARTHQUAKE LOADING FSAR FIGURE 110.41-2

PP&L DRAWING



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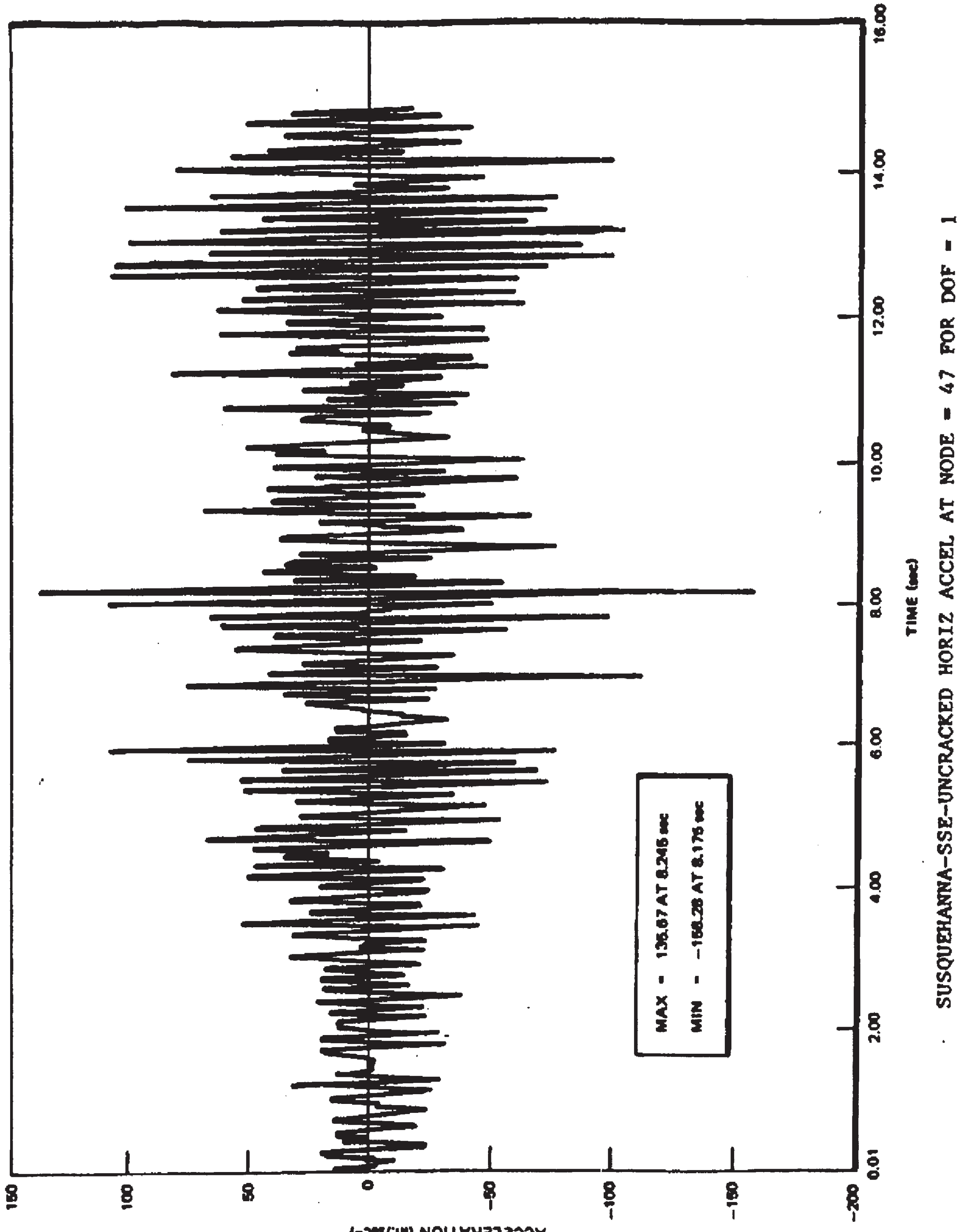
RESPONSE TIME HISTORY PLOT
DUE TO HORIZONTAL SSE

FSAR FIGURE 110.41-3

PP&L DRAWING



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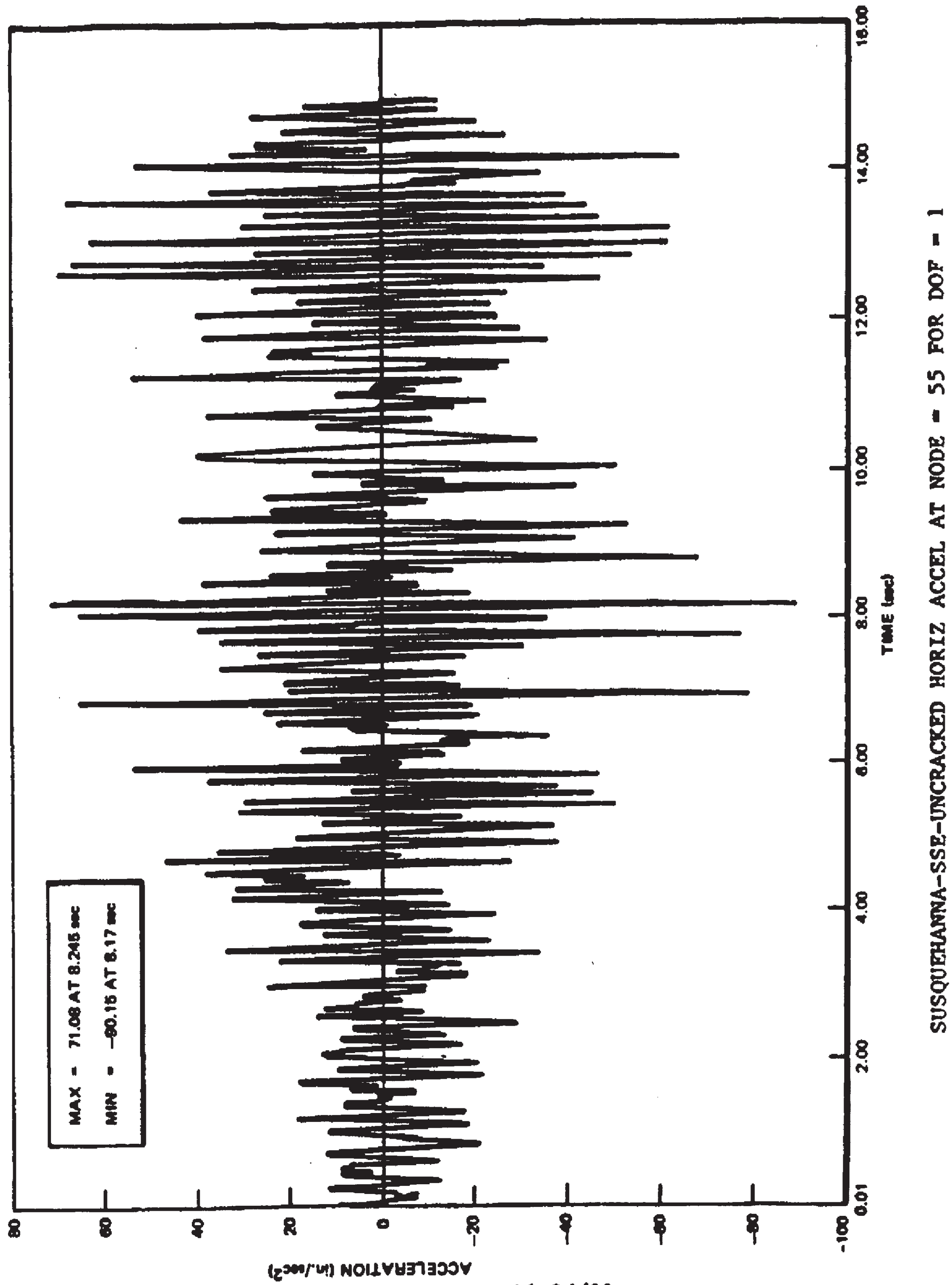
RESPONSE TIME HISTORY PLOT
DUE TO HORIZONTAL SSE

FSAR FIGURE 110.41-4

PP&L DRAWING



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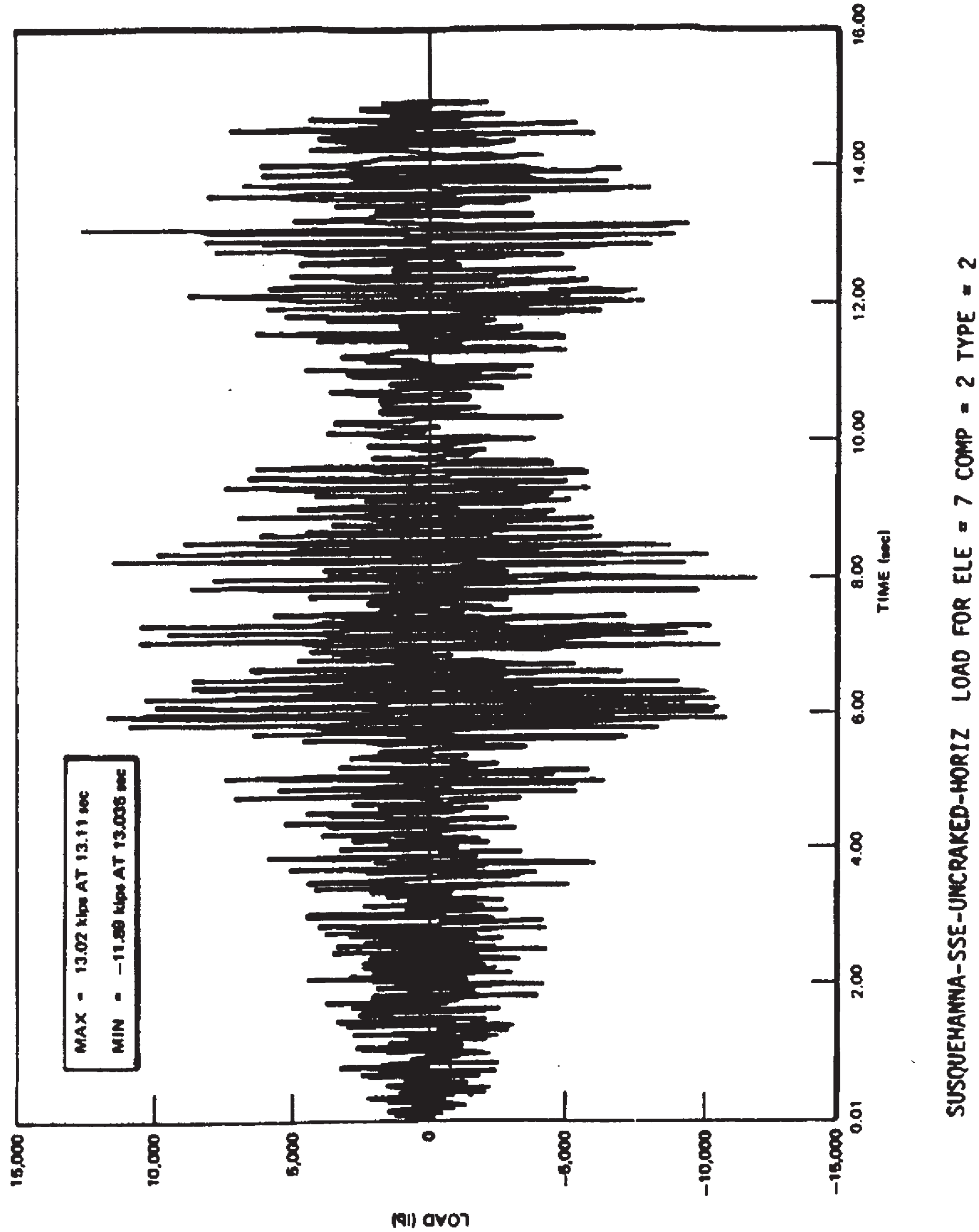
RESPONSE TIME HISTORY PLOT
DUE TO HORIZONTAL SSE

FSAR FIGURE 110.41-5

PP&L DRAWING



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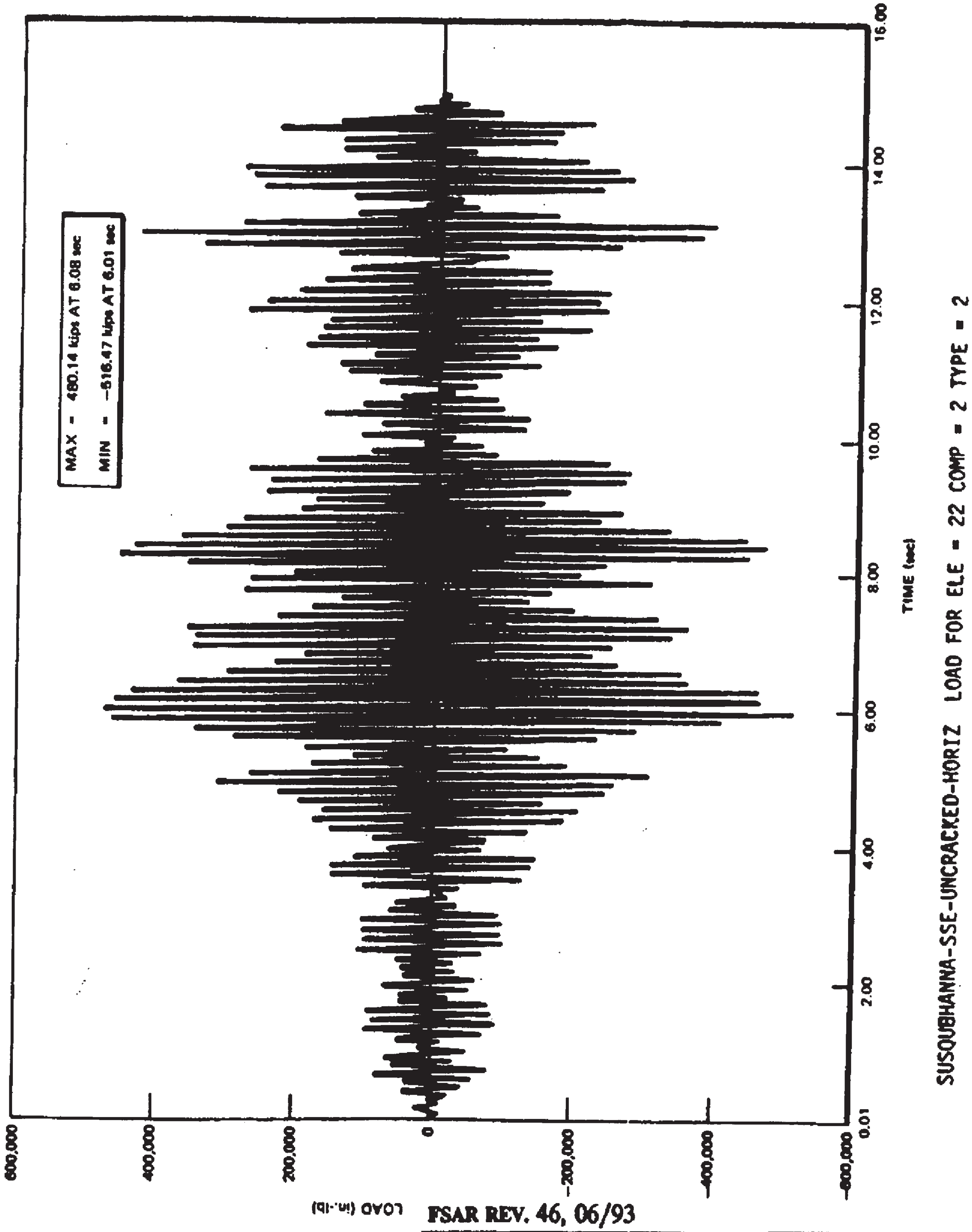
RESPONSE TIME HISTORY PLOT
DUE TO HORIZONTAL SSE

FSAR FIGURE 110.41-6

PP&L DRAWING



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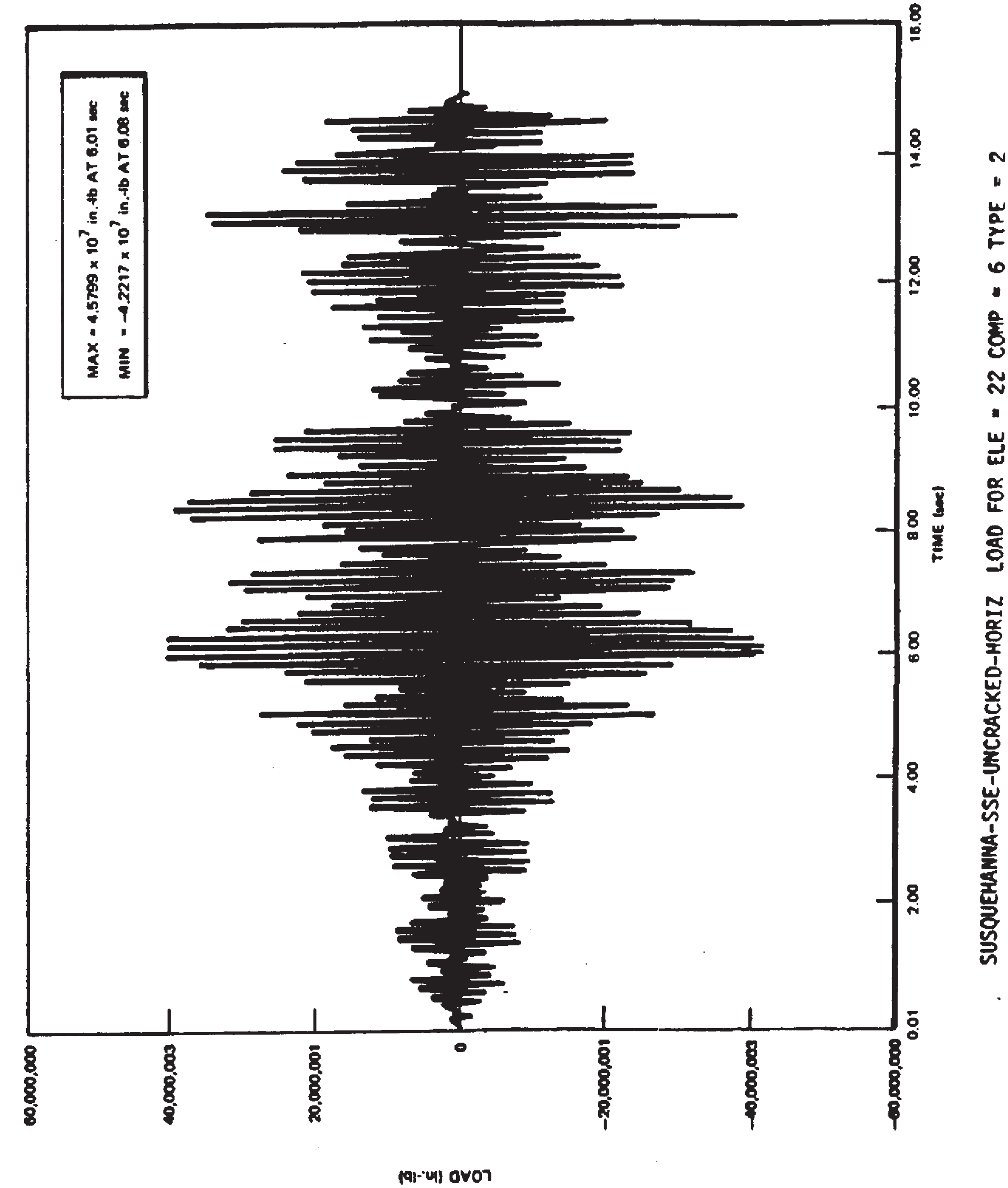
RESPONSE TIME HISTORY PLOT
DUE TO HORIZONTAL SSE

FSAR FIGURE 110.41-7

PP&L DRAWING



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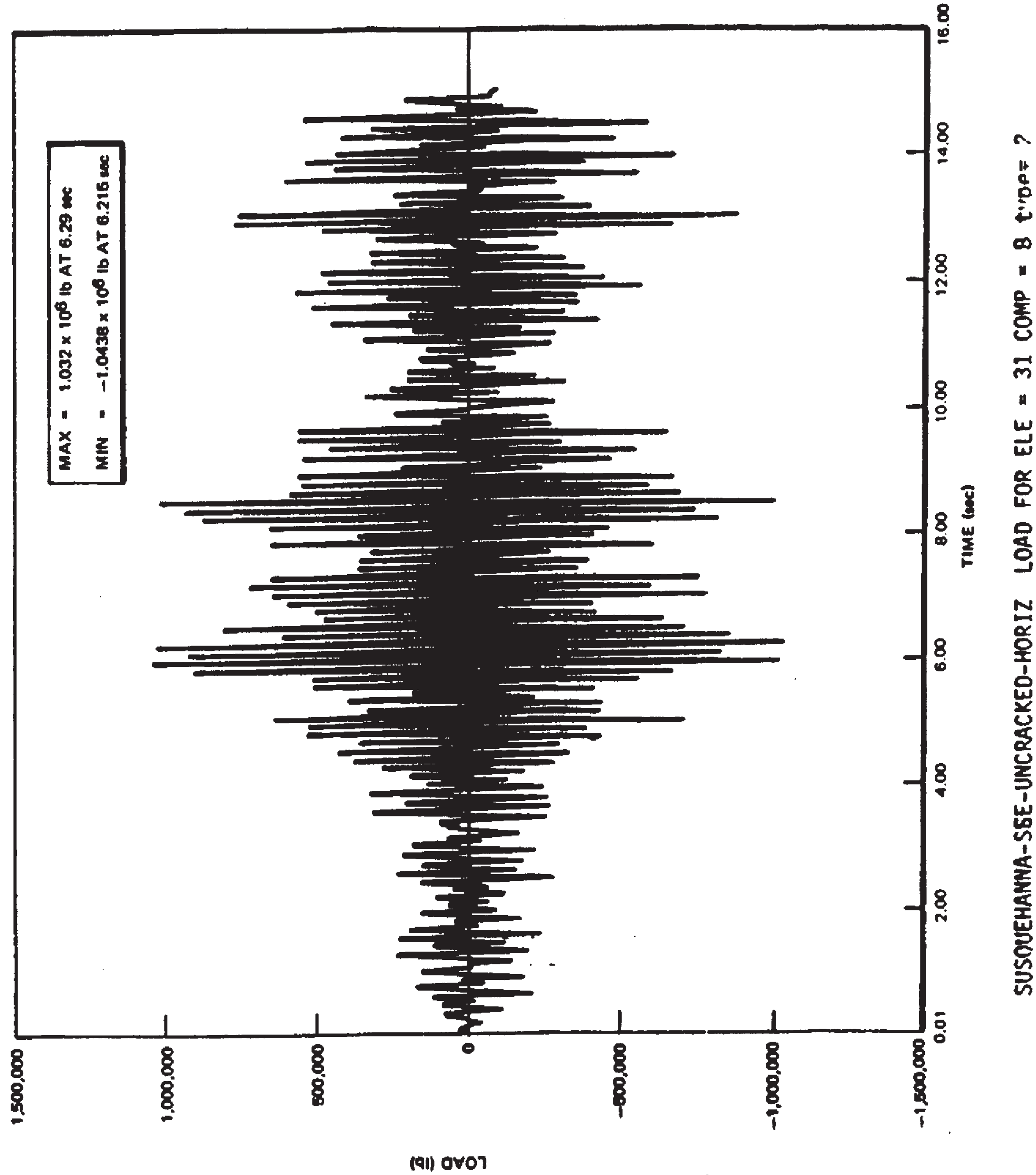
RESPONSE TIME HISTORY PLOT DUE TO HORIZONTAL SSE

FSAR FIGURE 110.41-8

PP&L DRAWING



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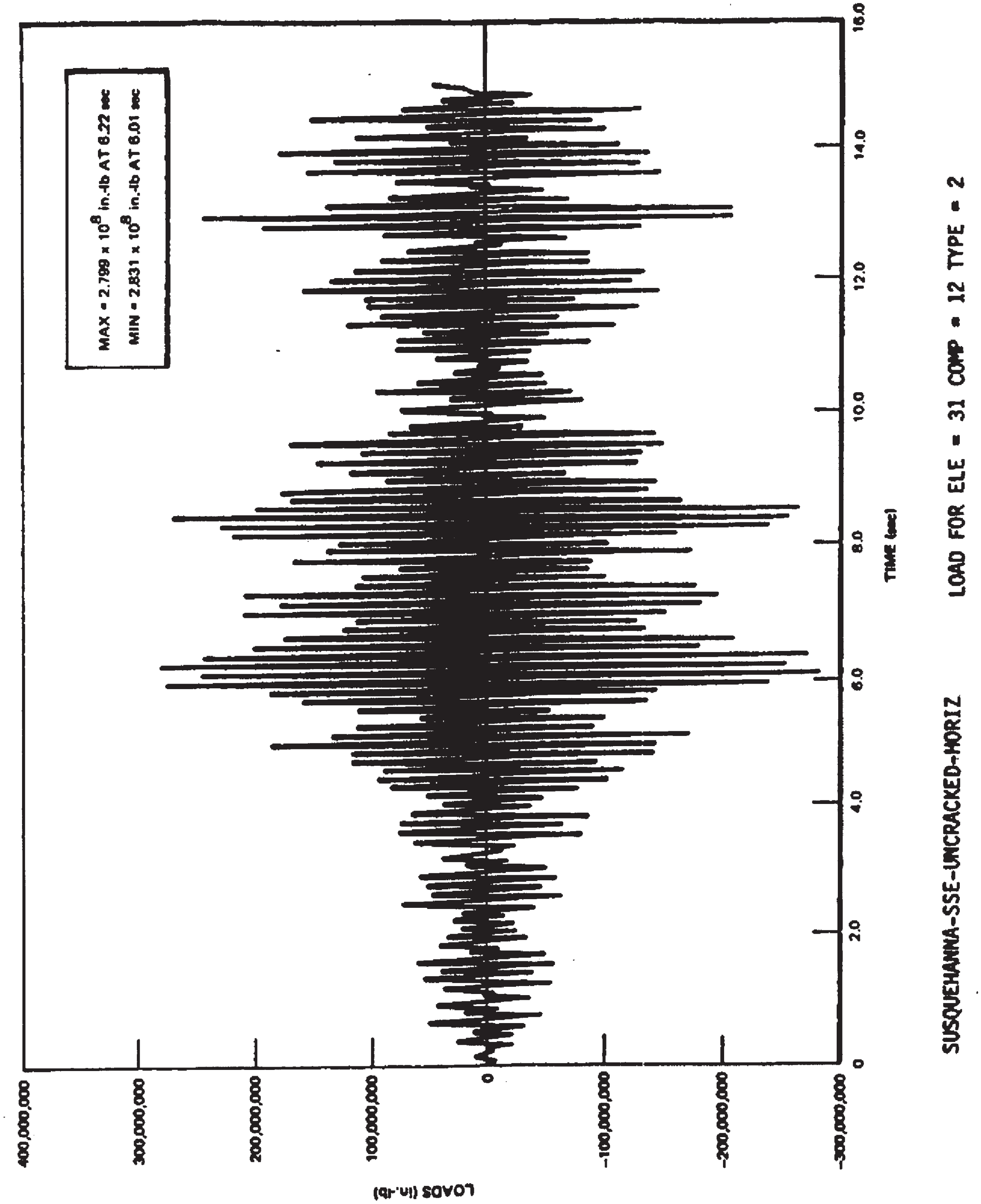
RESPONSE TIME HISTORY PLOT
DUE TO HORIZONTAL SSE

FSAR FIGURE 110.41-9

PP&L DRAWING



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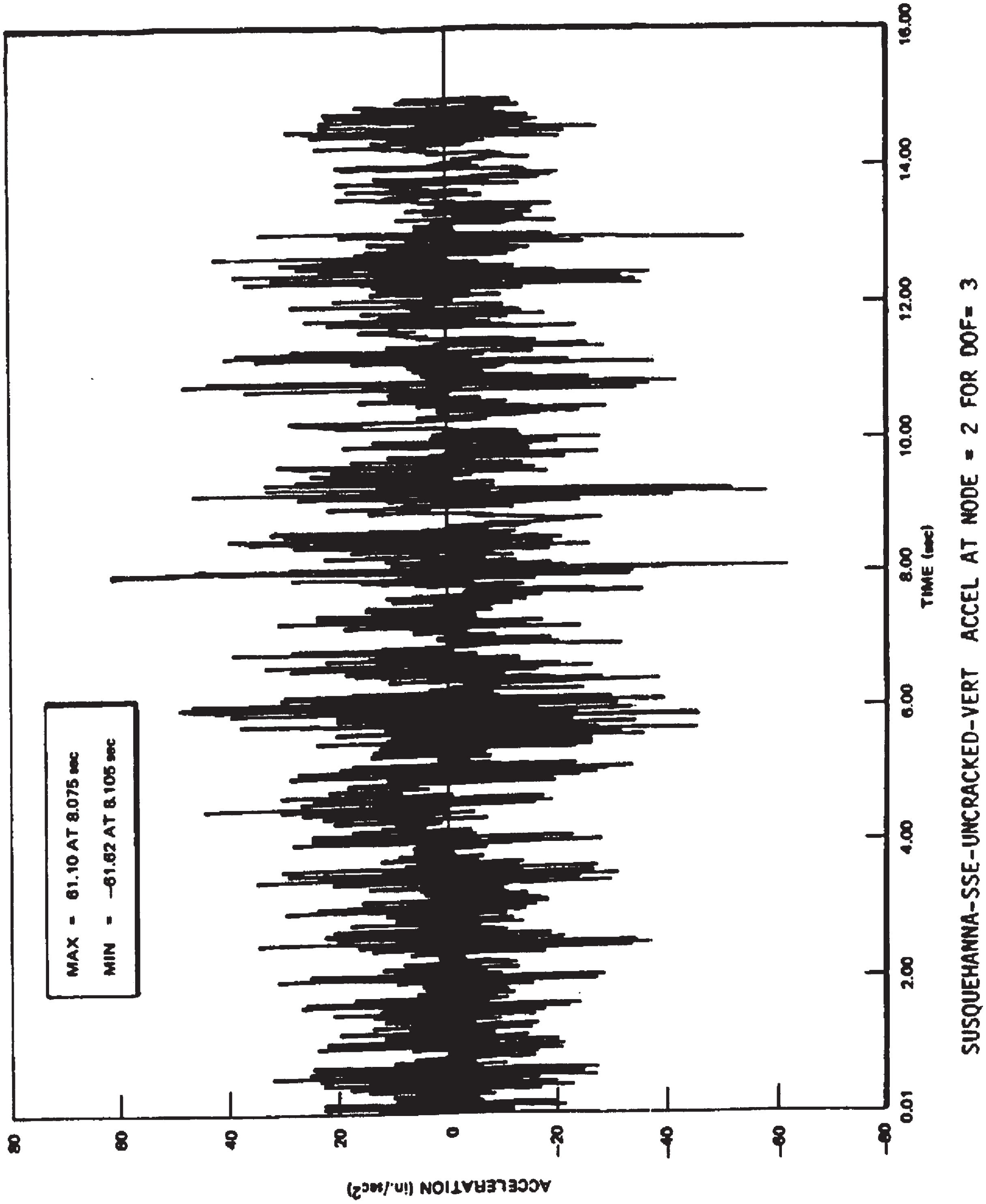
RESPONSE TIME HISTORY PLOT DUE TO HORIZONTAL SSE FSAR FIGURE 110.41-10

PP&L DRAWING



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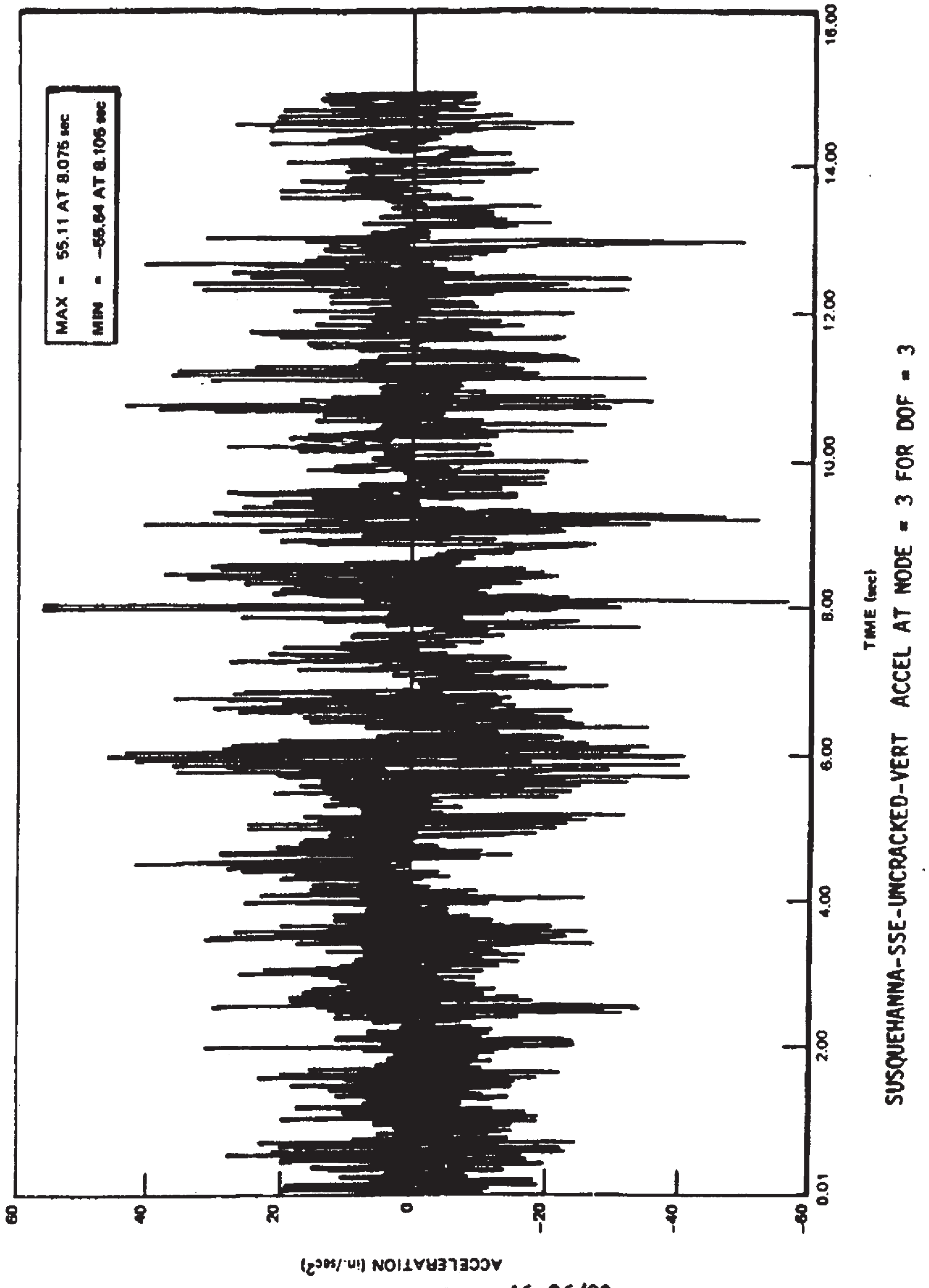
RESPONSE TIME HISTORY PLOT
DUE TO VERTICAL SSE

FSAR FIGURE 110.41-11

PP&L DRAWING



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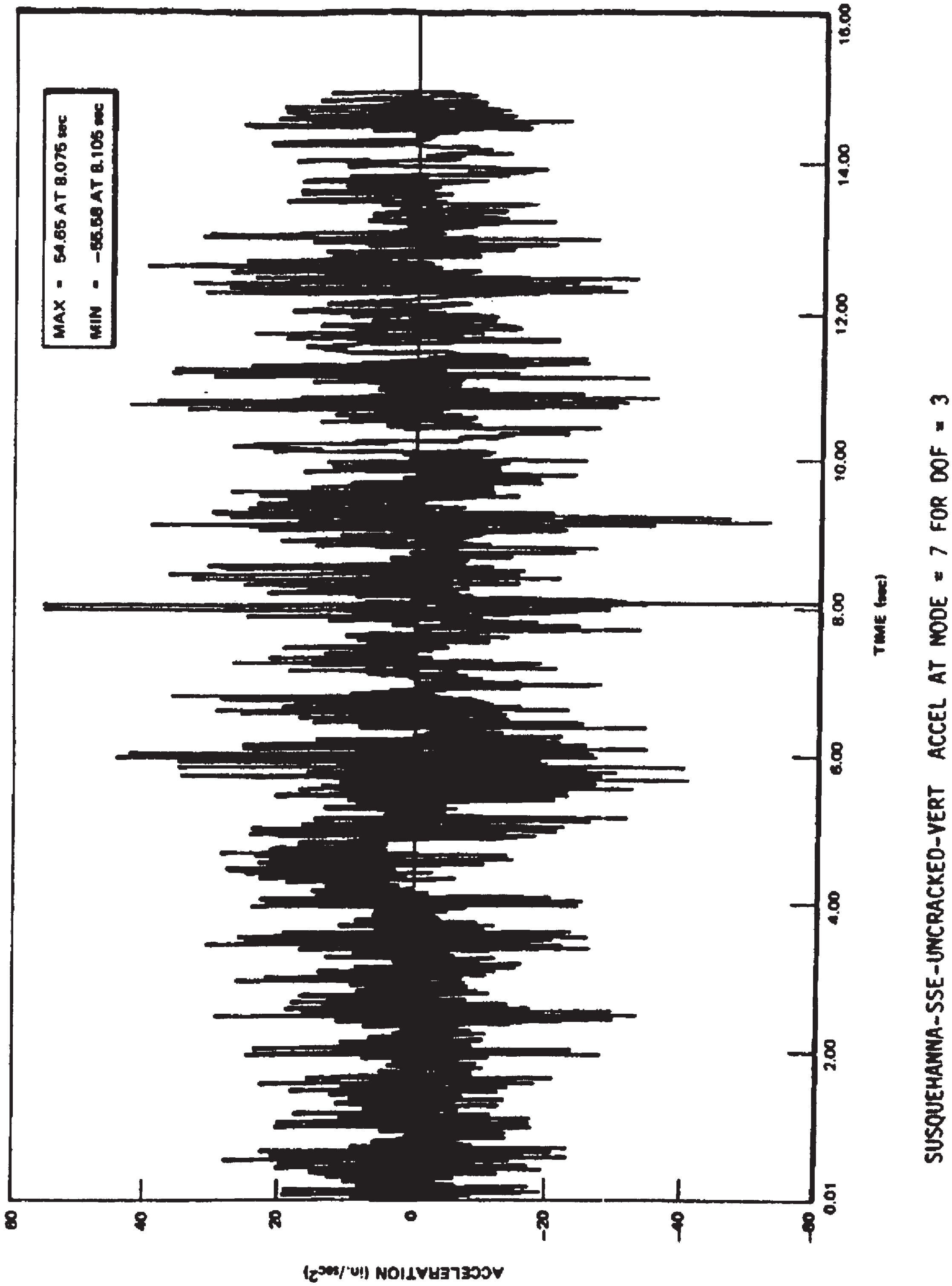
RESPONSE TIME HISTORY PLOT
DUE TO VERTICAL SSE

FSAR FIGURE 110.41-12

PP&L DRAWING



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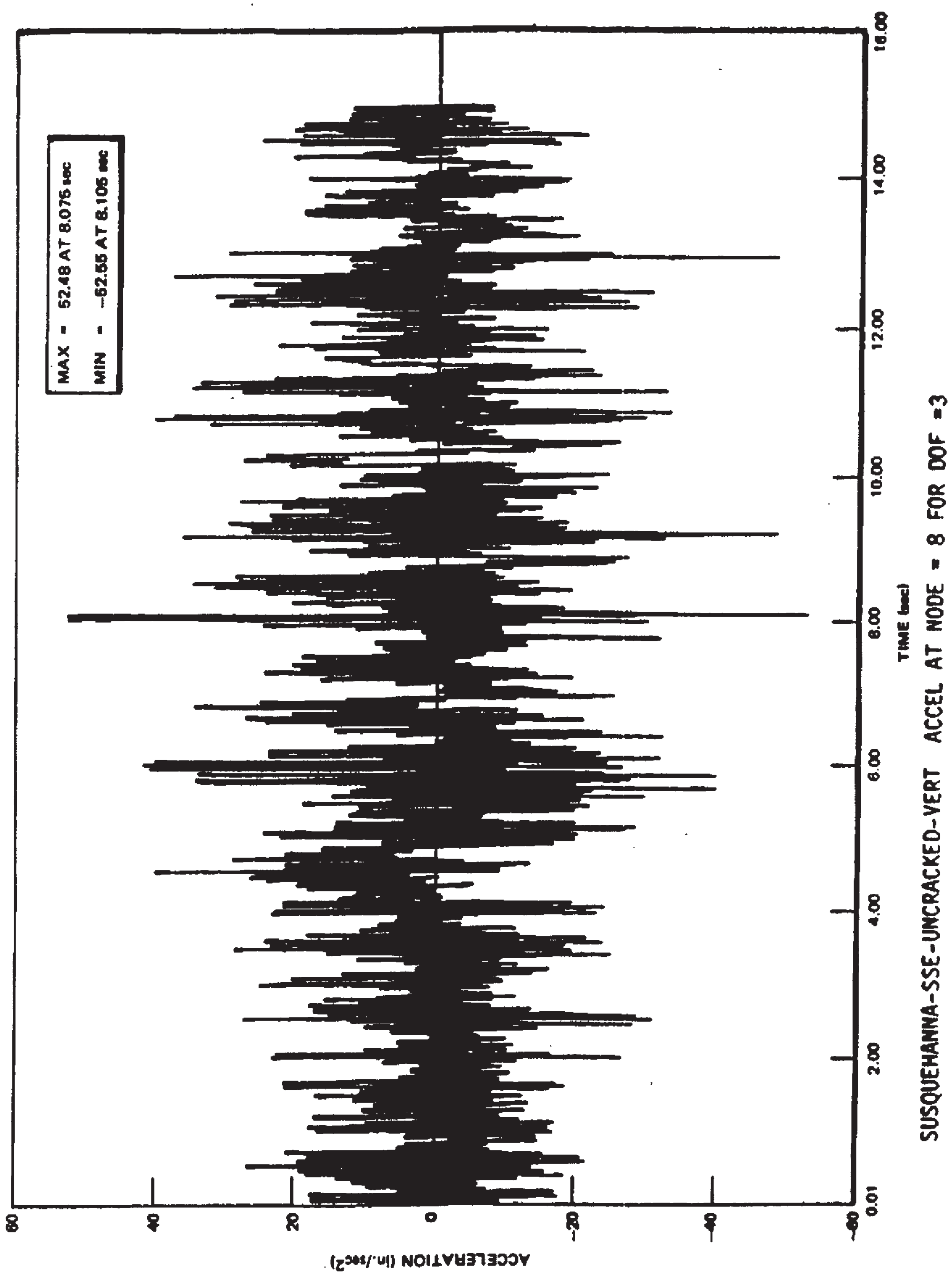
RESPONSE TIME HISTORY PLOT
DUE TO VERTICAL SSE

FSAR FIGURE 110.41-13

PP&L DRAWING



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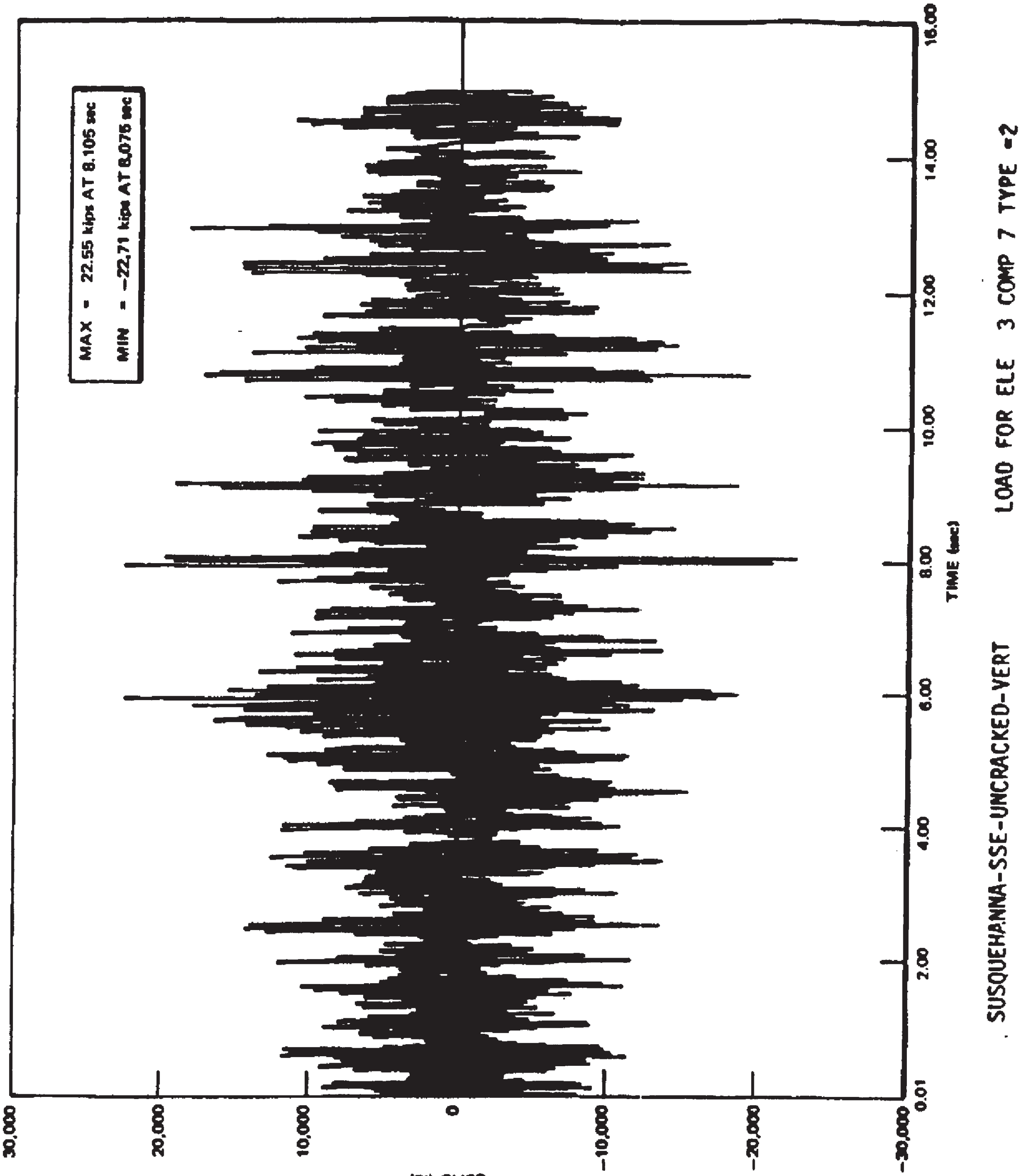
RESPONSE TIME HISTORY PLOT
DUE TO VERTICAL SSE

FSAR FIGURE 110.41-14

PP&L DRAWING



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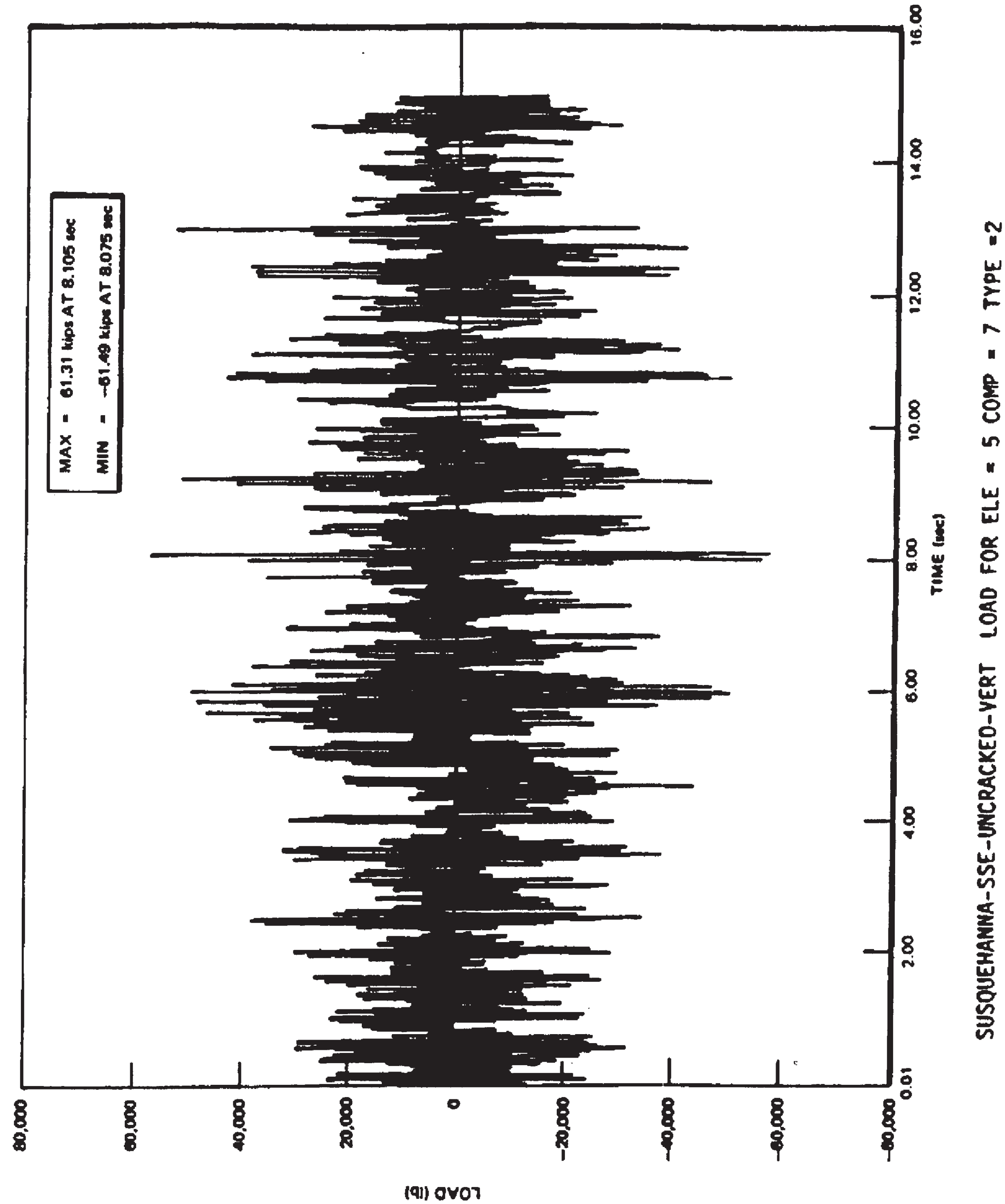
RESPONSE TIME HISTORY PLOT
DUE TO VERTICAL SSE

FSAR FIGURE 110.41-15

PP&L DRAWING



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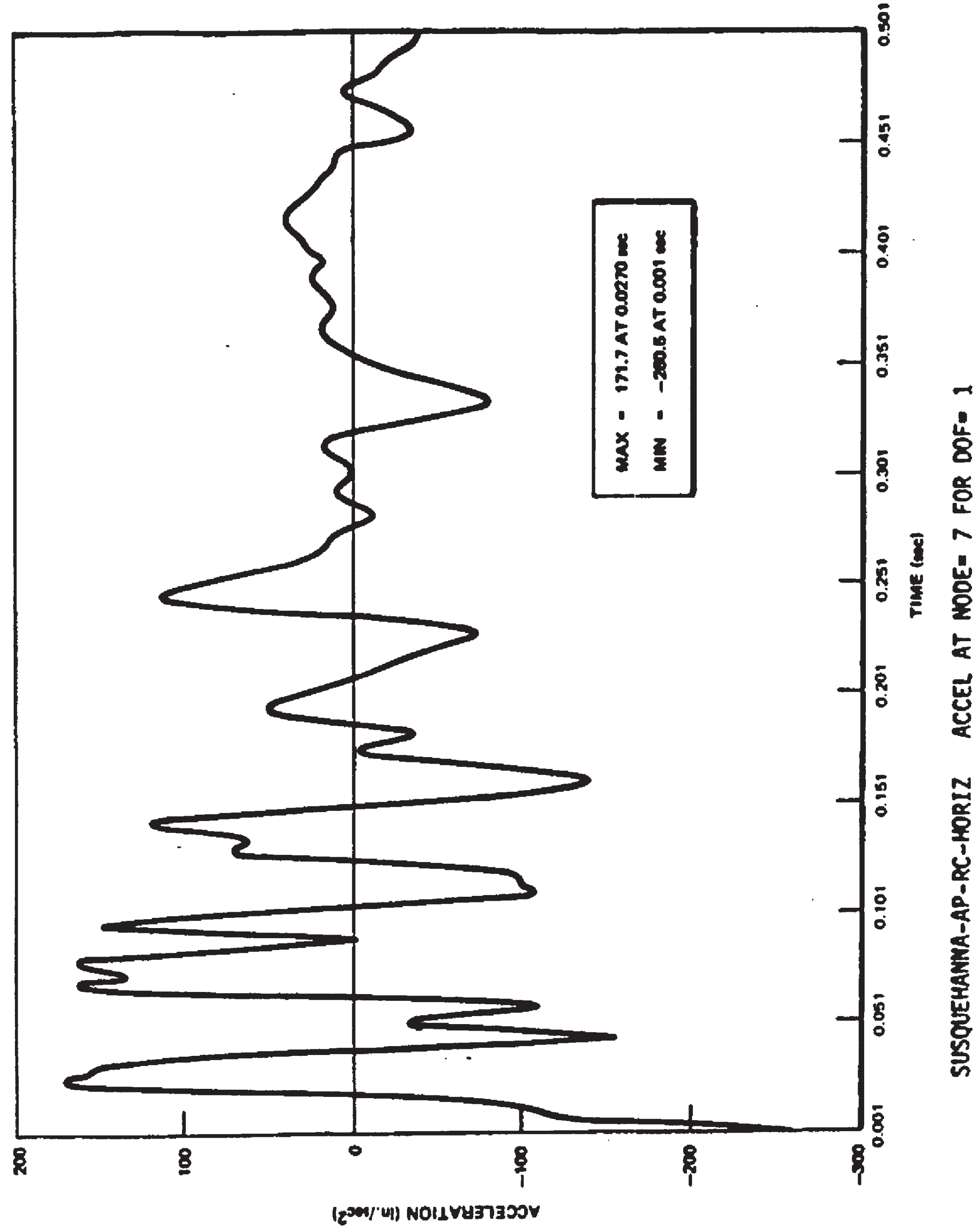
RESPONSE TIME HISTORY PLOT
DUE TO VERTICAL SSE

FSAR FIGURE 110.41-16

PP&L DRAWING



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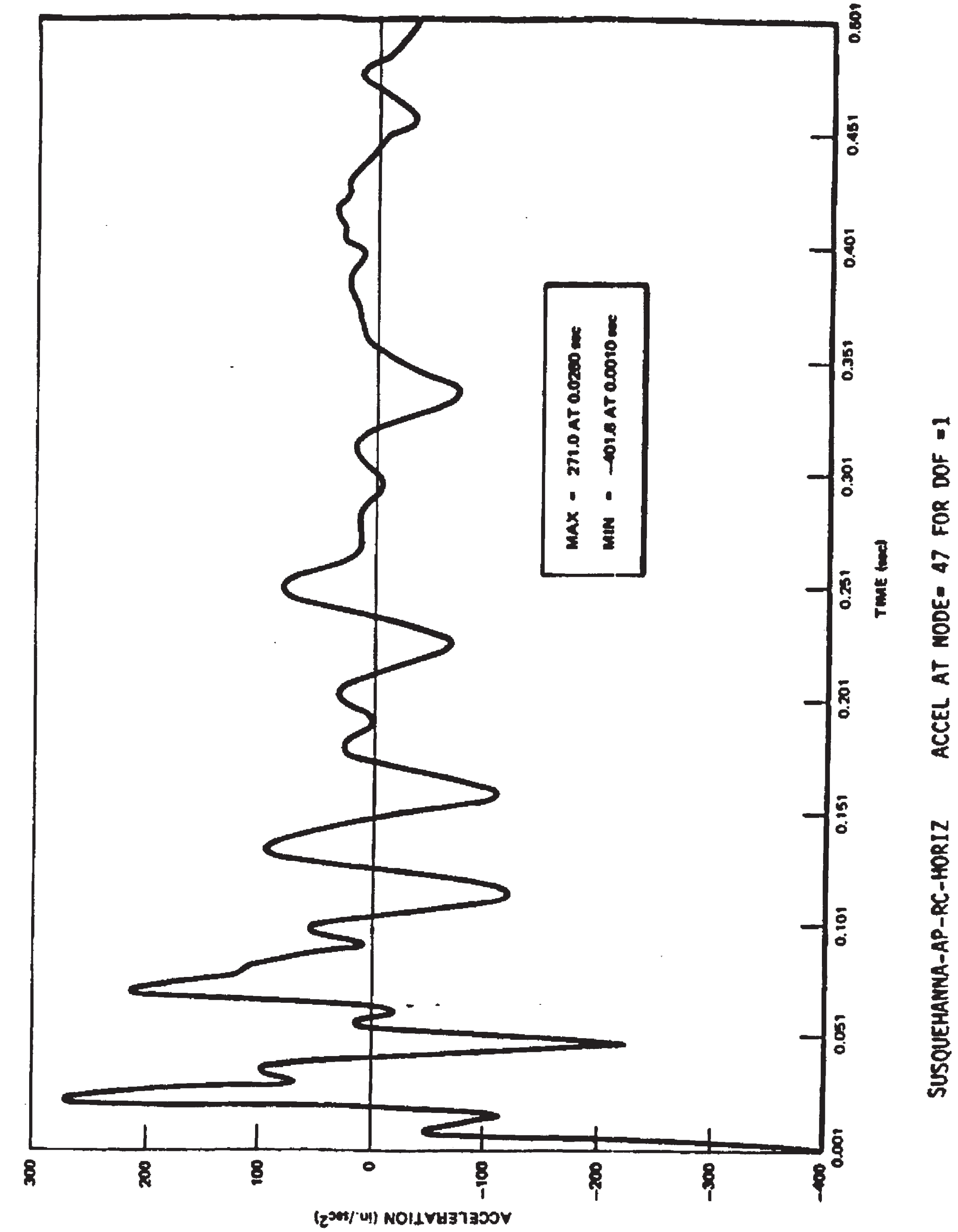
RESPONSE TIME HISTORY PLOT
DUE TO
RECIRCULATION PIPE BREAK

FSAR FIGURE 110.41-17

PP&L DRAWING



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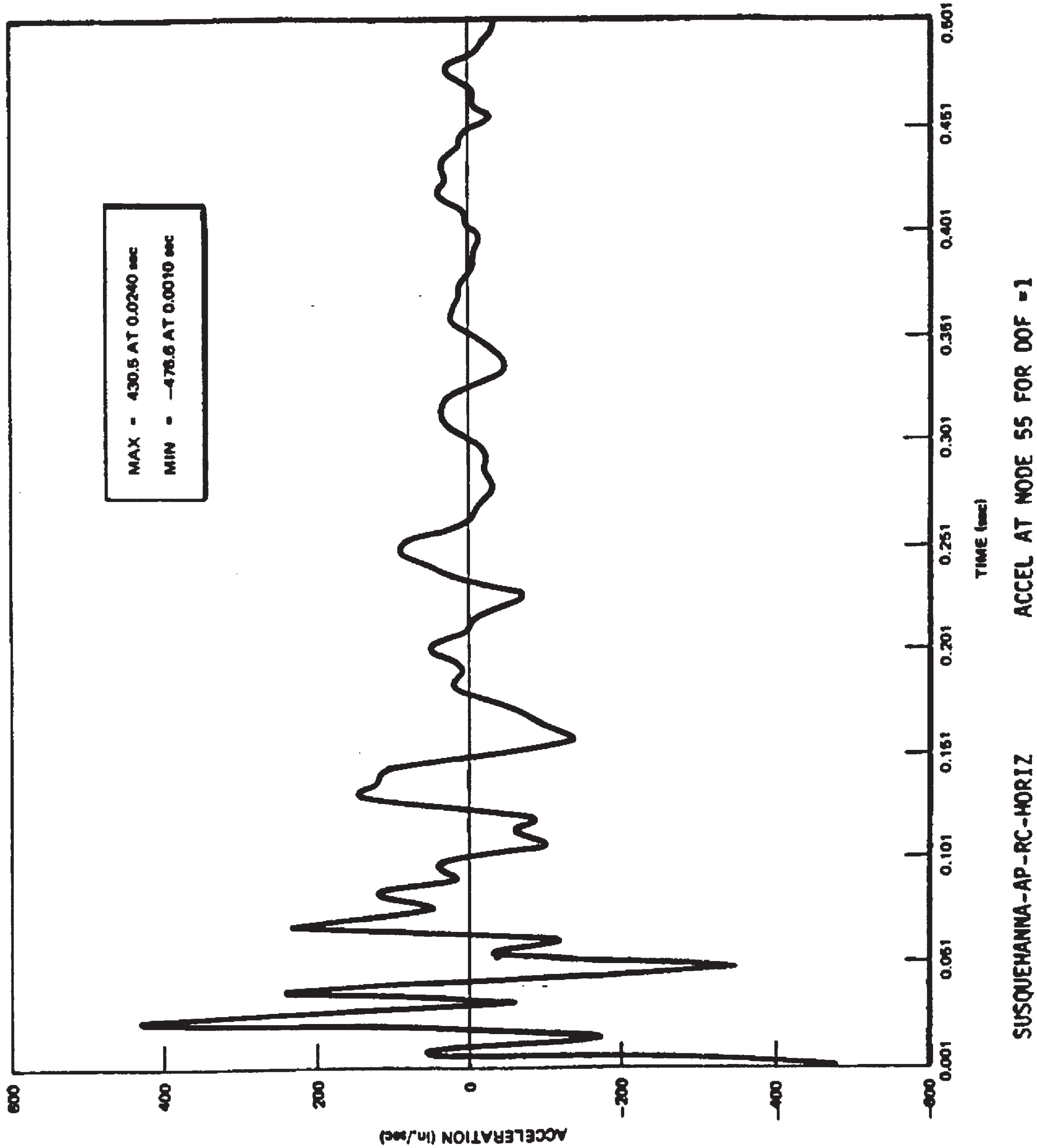
RESPONSE TIME HISTORY PLOT
DUE TO
RECIRCULATION PIPE BREAK

FSAR FIGURE 110.41-18

PP&L DRAWING



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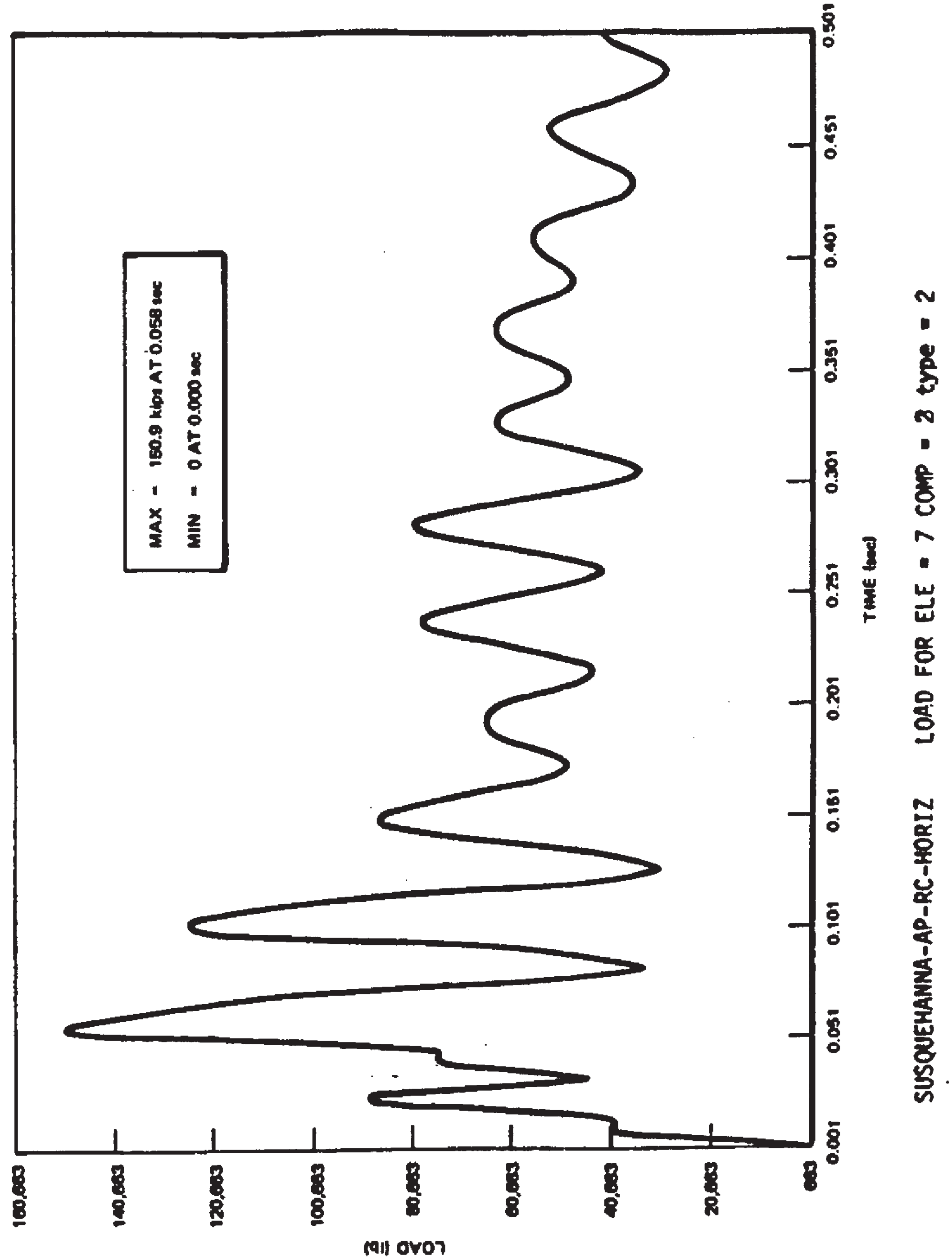
RESPONSE TIME HISTORY PLOT
DUE TO
RECIRCULATION PIPE BREAK

FSAR FIGURE 110.41-19

PP&L DRAWING



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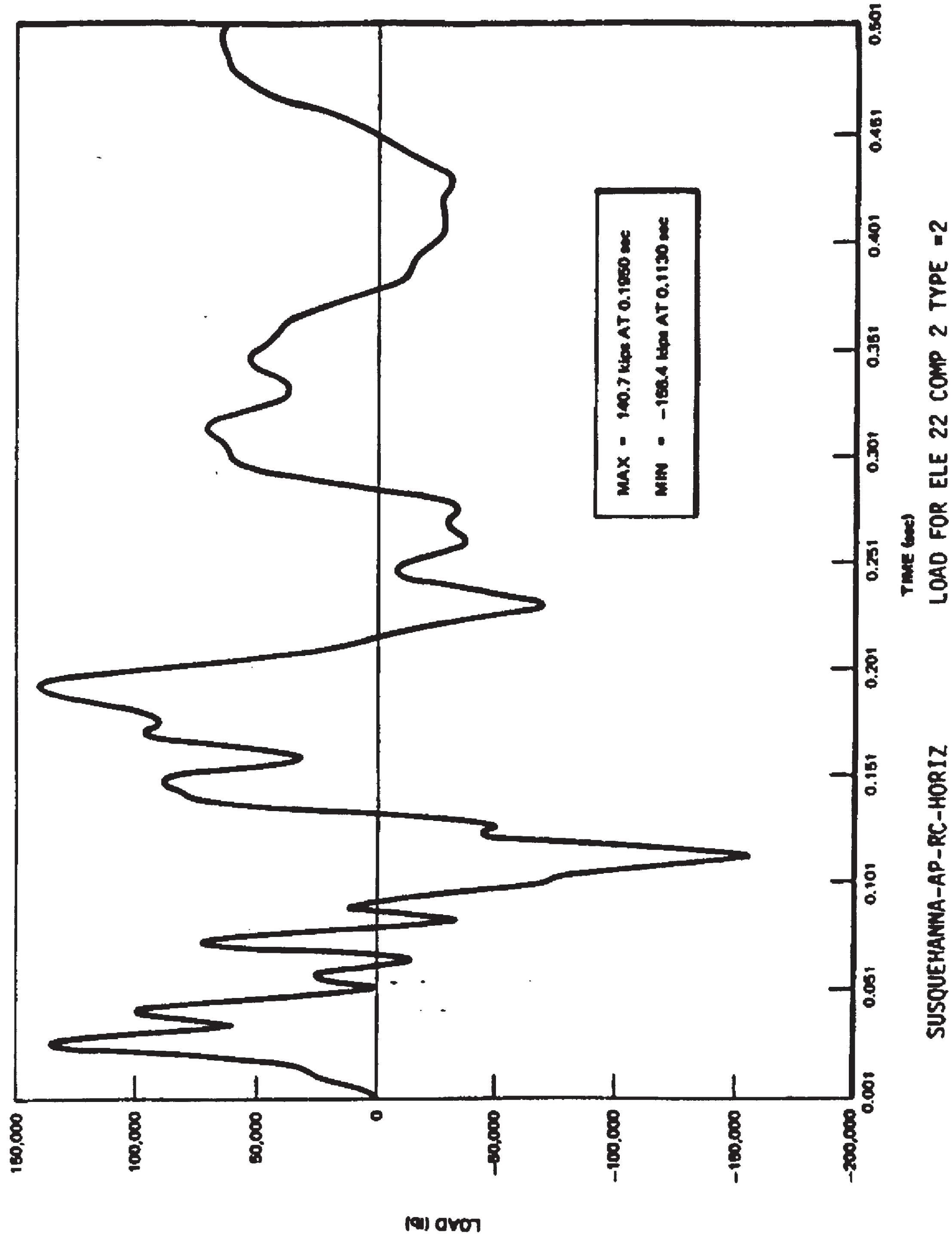
RESPONSE TIME HISTORY PLOT
DUE TO
RECIRCULATION PIPE BREAK

FSAR FIGURE 110.41-20

PP&L DRAWING



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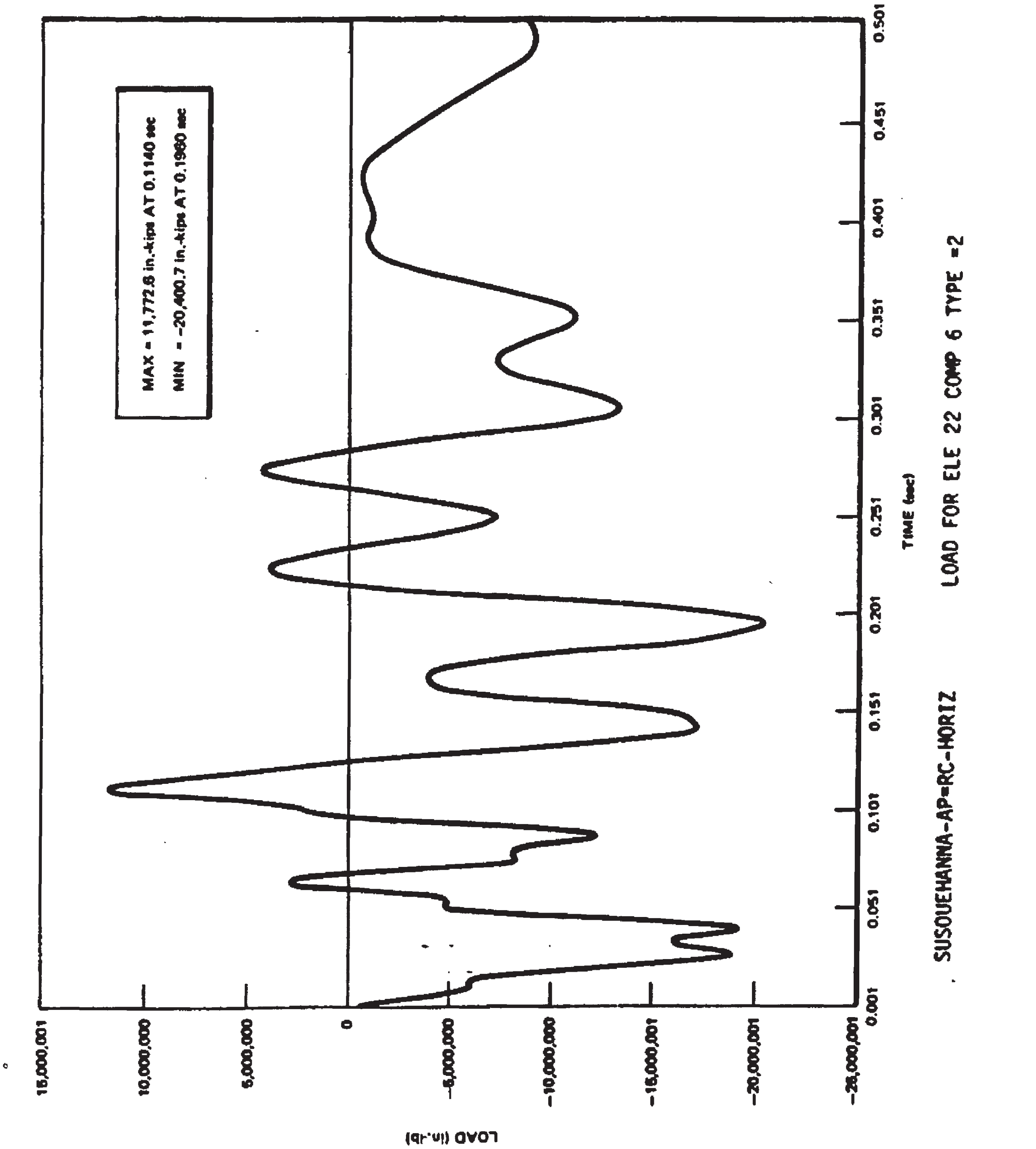
RESPONSE TIME HISTORY PLOT
DUE TO
RECIRCULATION PIPE BREAK

FSAR FIGURE 110.41-21

PP&L DRAWING



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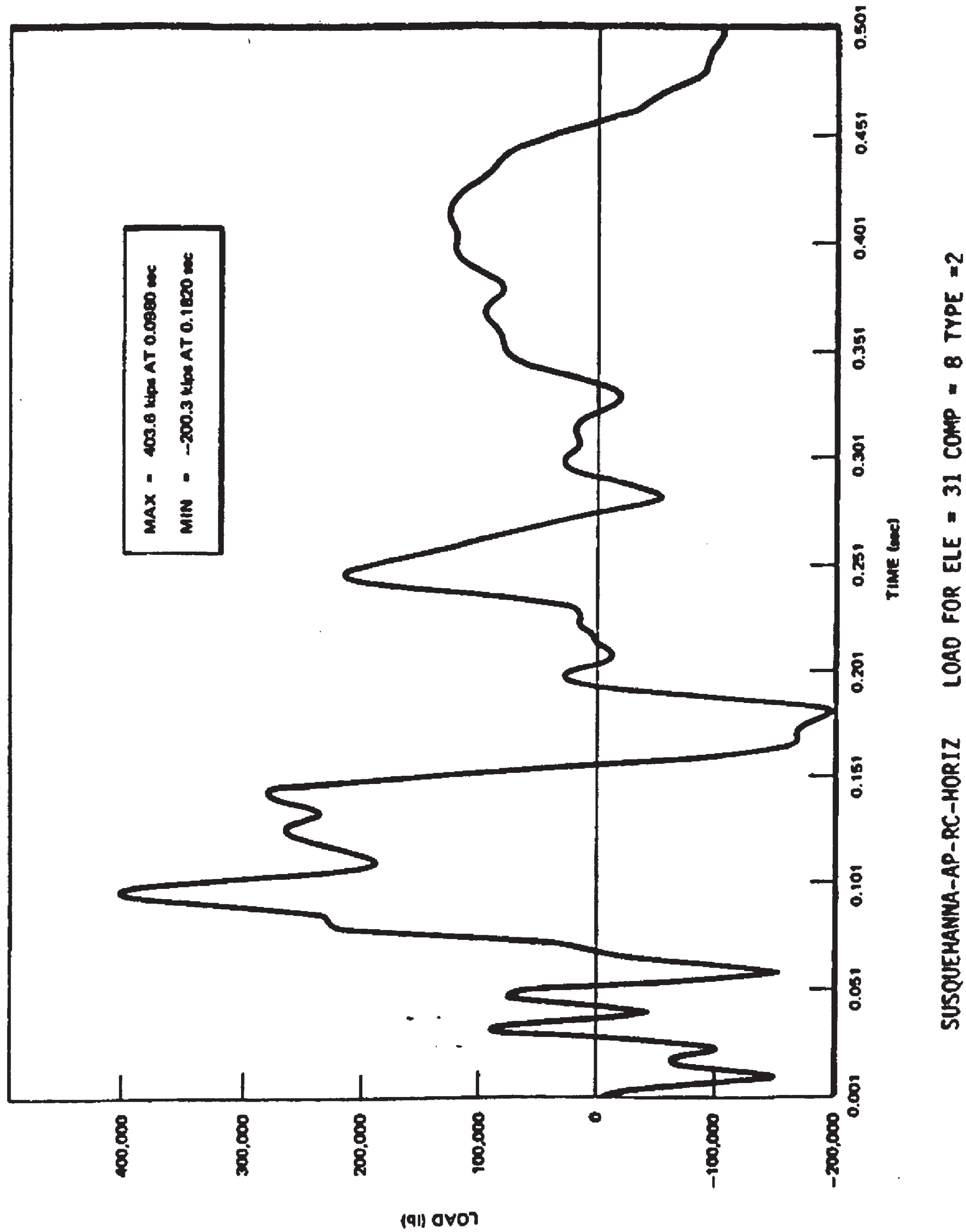
RESPONSE TIME HISTORY PLOT
DUE TO
RECIRCULATION PIPE BREAK

FSAR FIGURE 110.41-22

PP&L DRAWING



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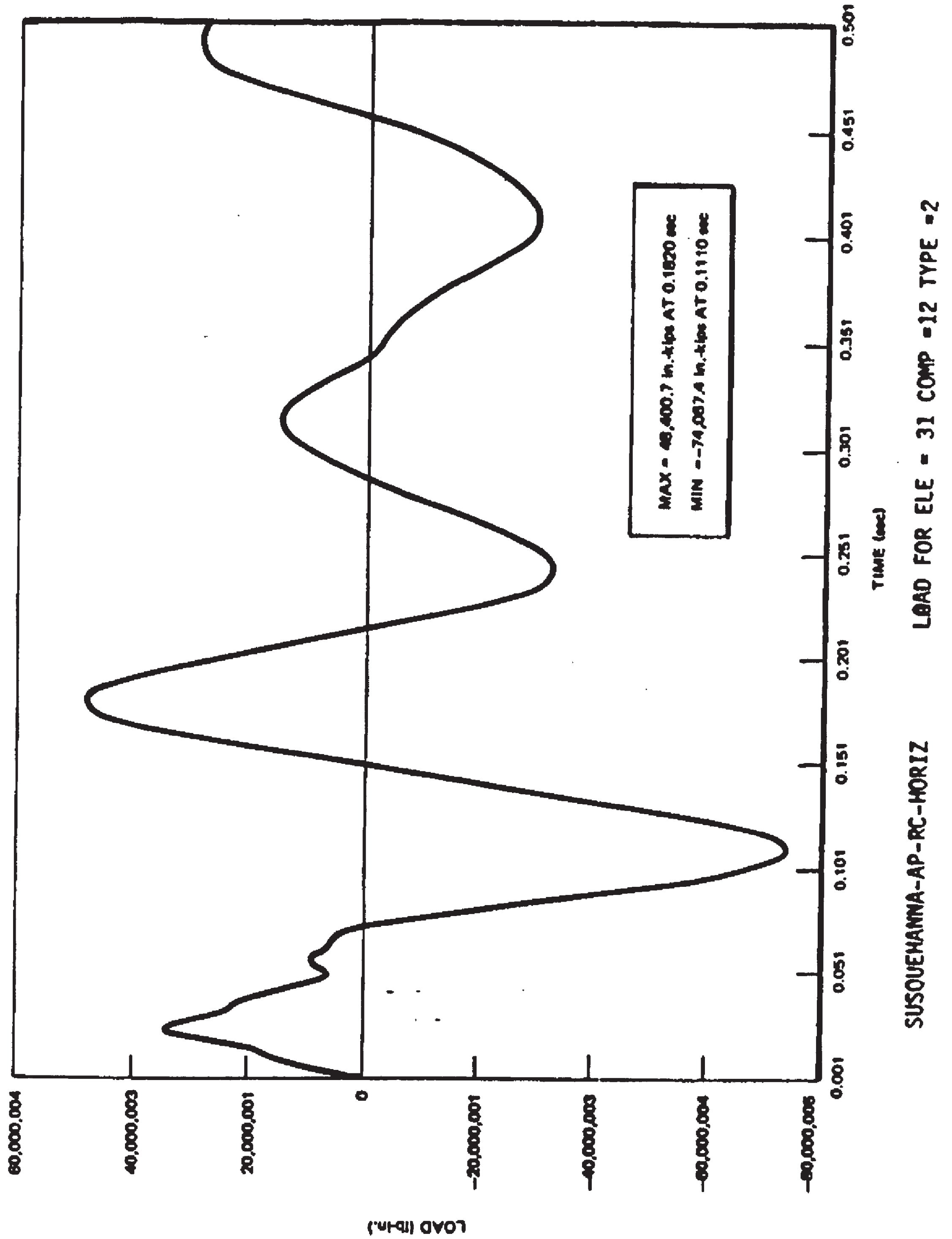
RESPONSE TIME HISTORY PLOT
DUE TO
RECIRCULATION PIPE BREAK

FSAR FIGURE 110.41-23

PP&L DRAWING



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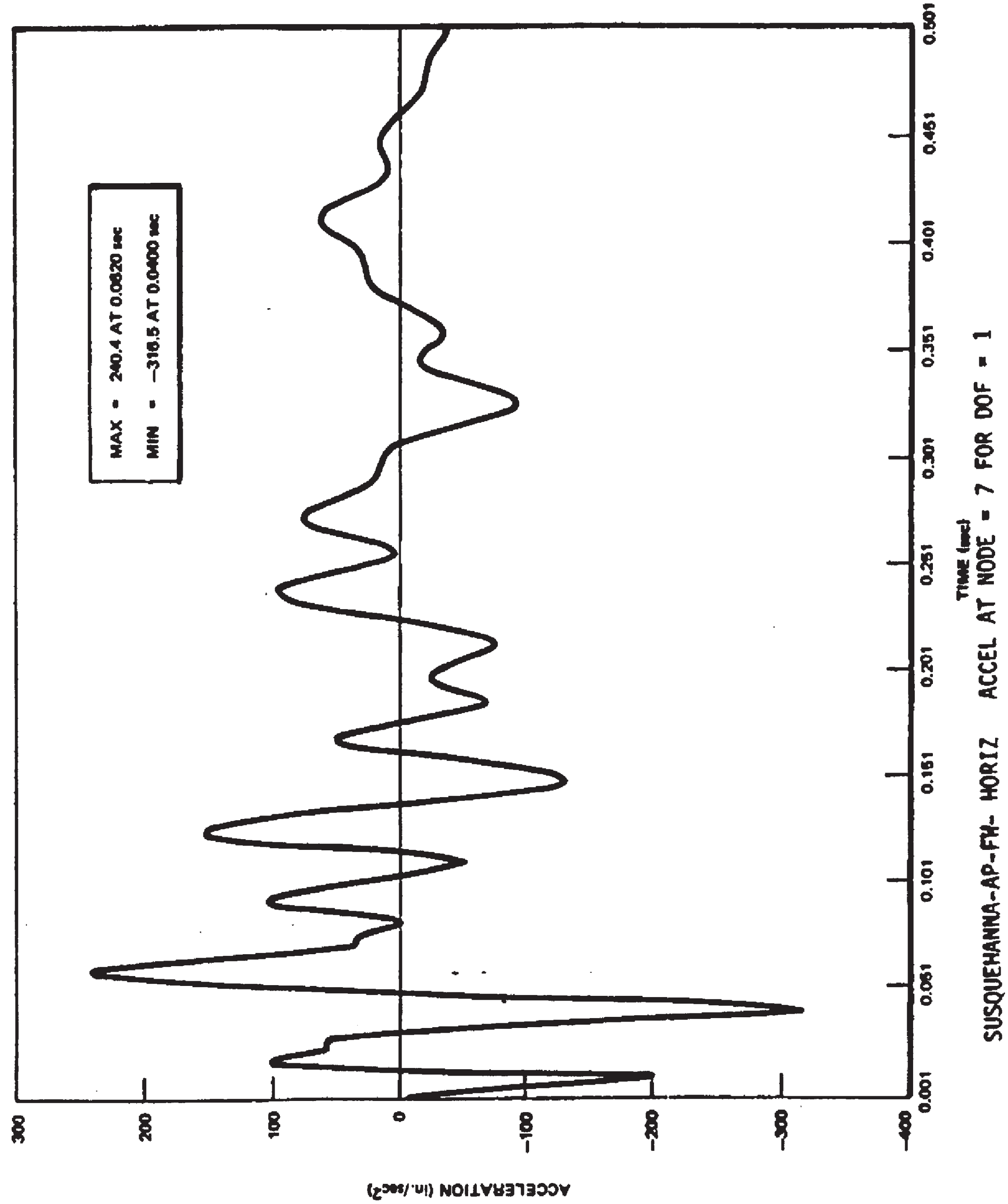
RESPONSE TIME HISTORY PLOT
DUE TO
RECIRCULATION PIPE BREAK

FSAR FIGURE 110.41-24

PP&L DRAWING



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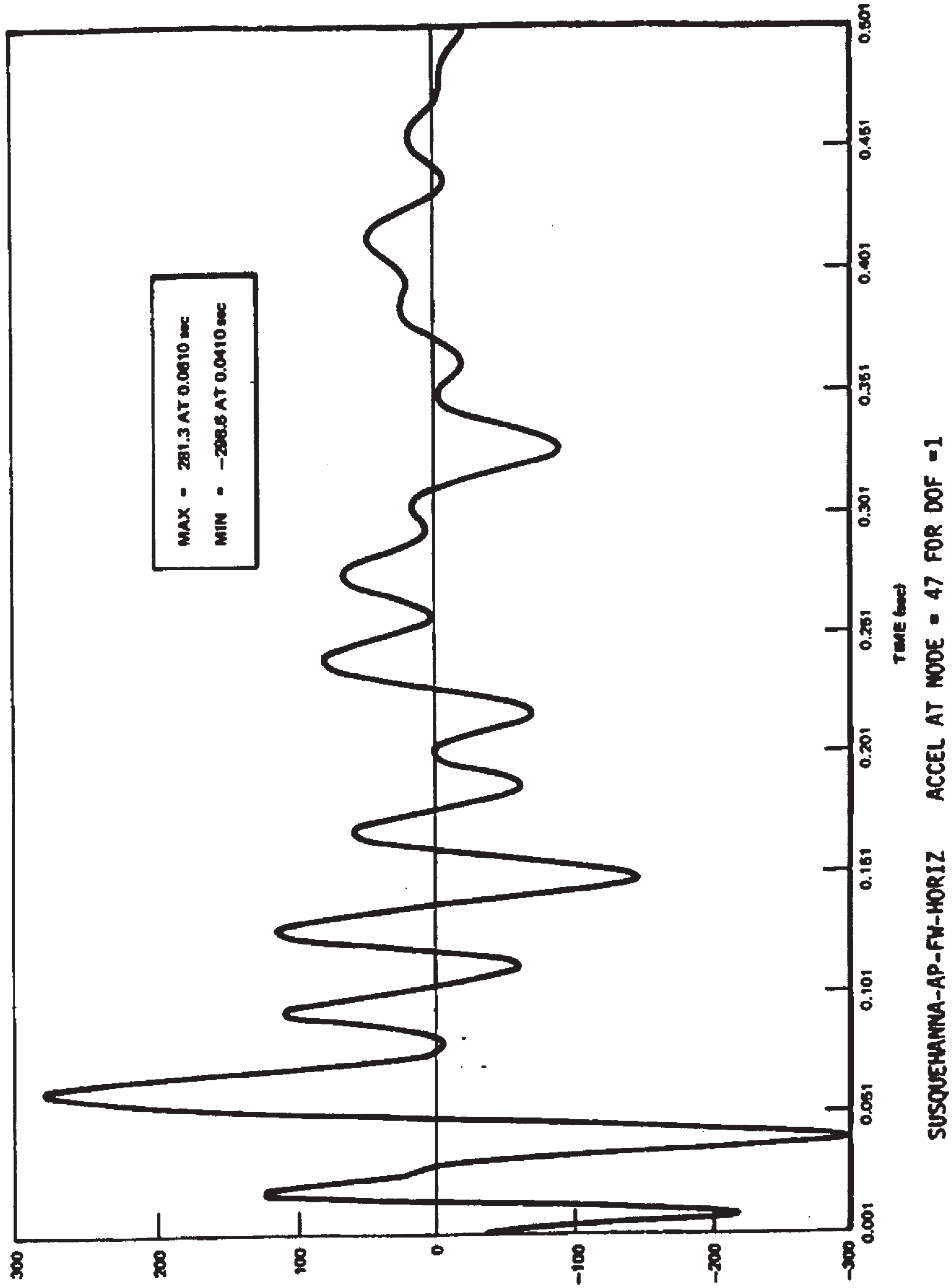
SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 AND 2 FINAL SAFETY ANALYSIS REPORT
ANNULUS PRESSURIZATION (HORIZONTAL) FEEDWATER LINE BREAK
FSAR FIGURE 110.41-25

PP&L DRAWING



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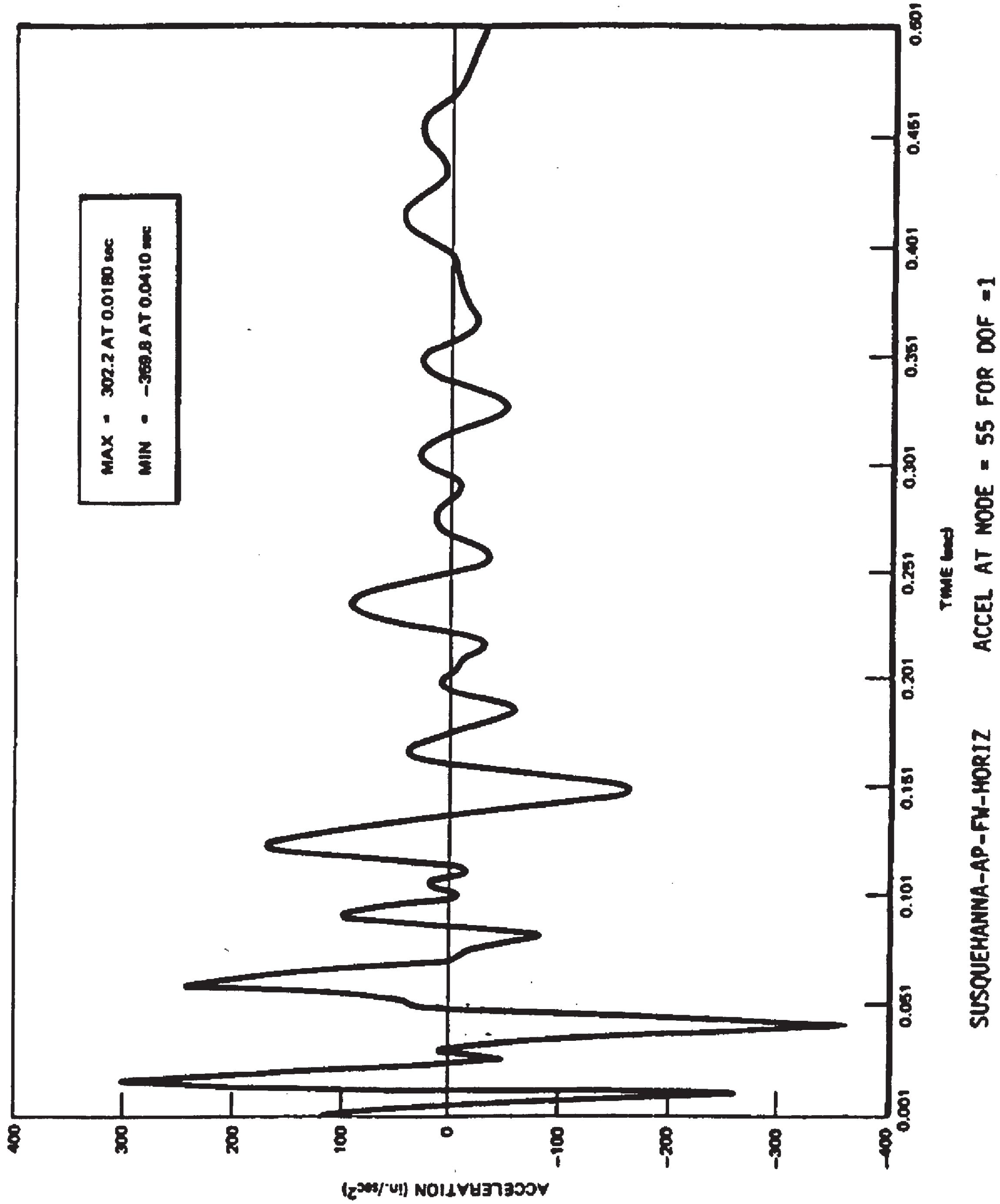
ANNULUS PRESSURIZATION
(HORIZONTAL)
FEEDWATER LINE BREAK

FSAR FIGURE 110.41-26

PP&L DRAWING



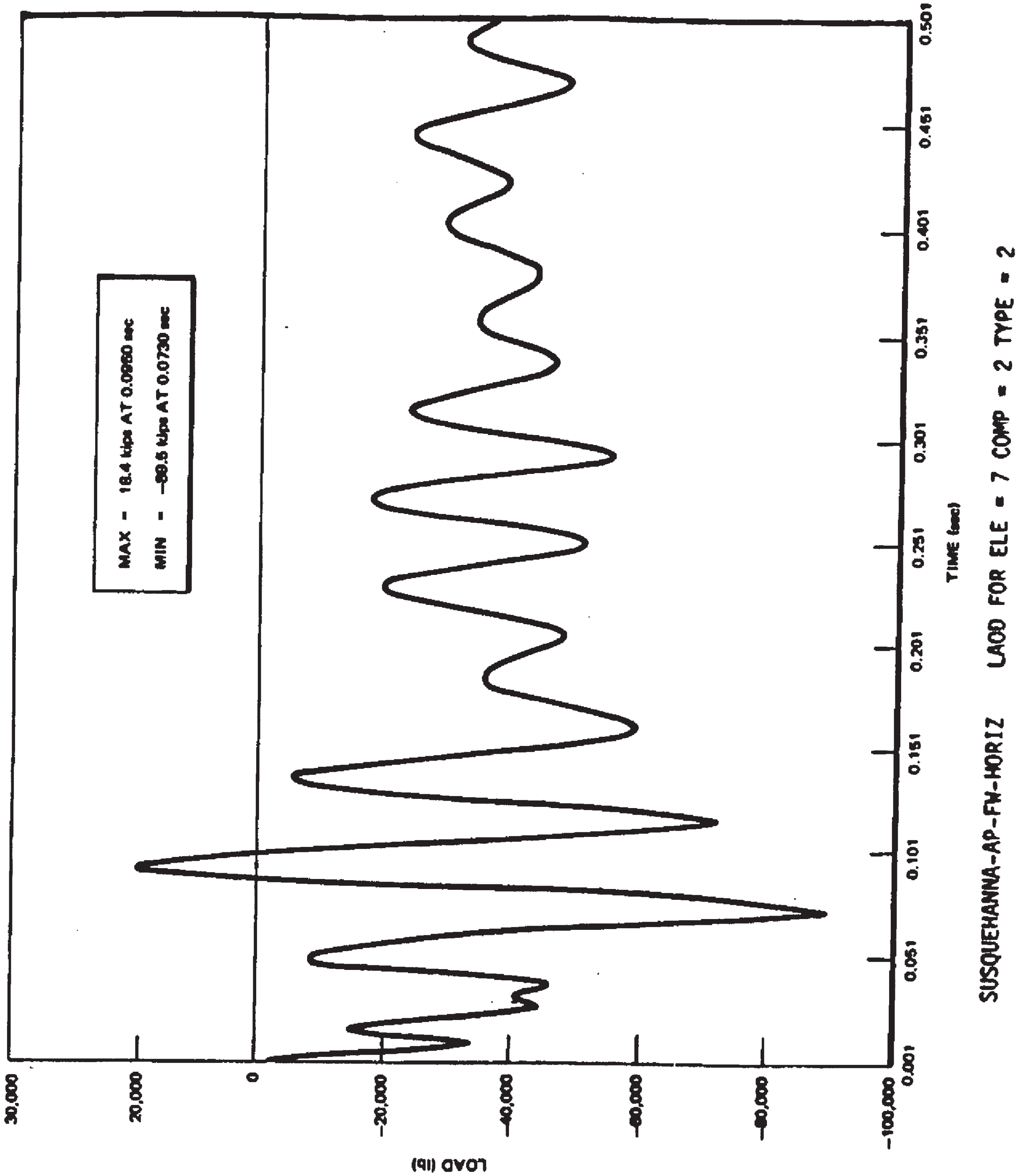
FSAR REV. 46, 06/93

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



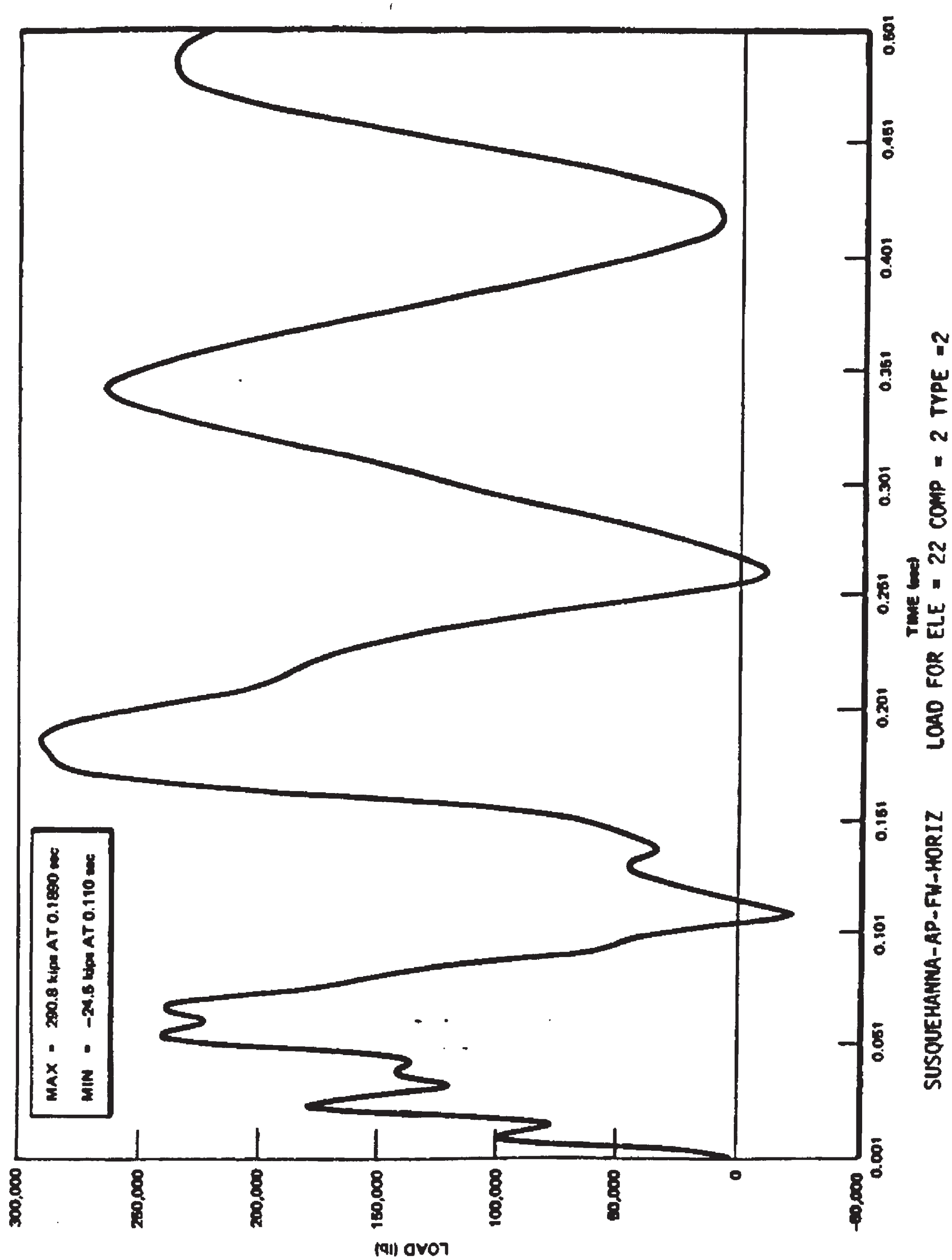
FSAR REV. 46, 06/93

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



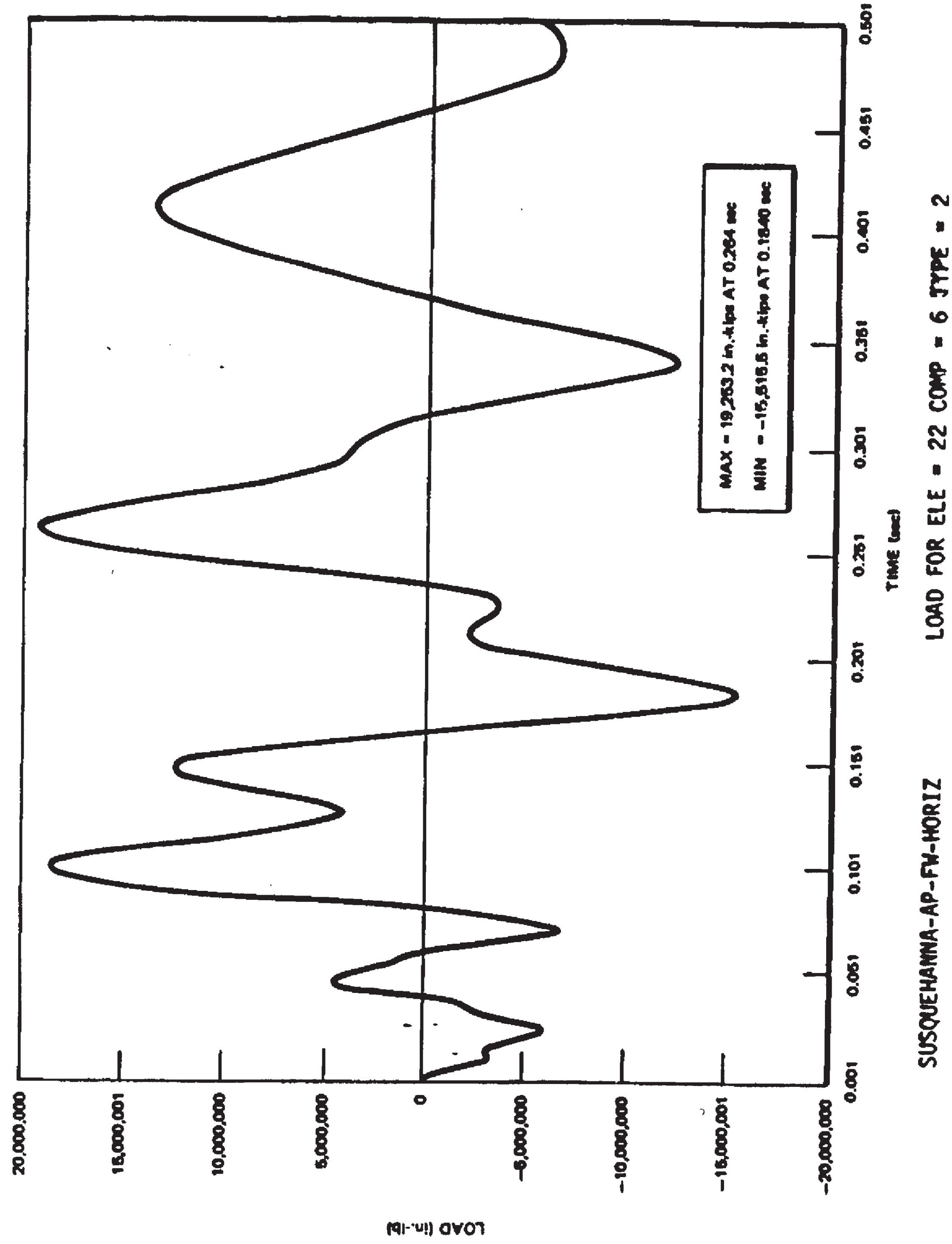
FSAR REV. 46, 06/93

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



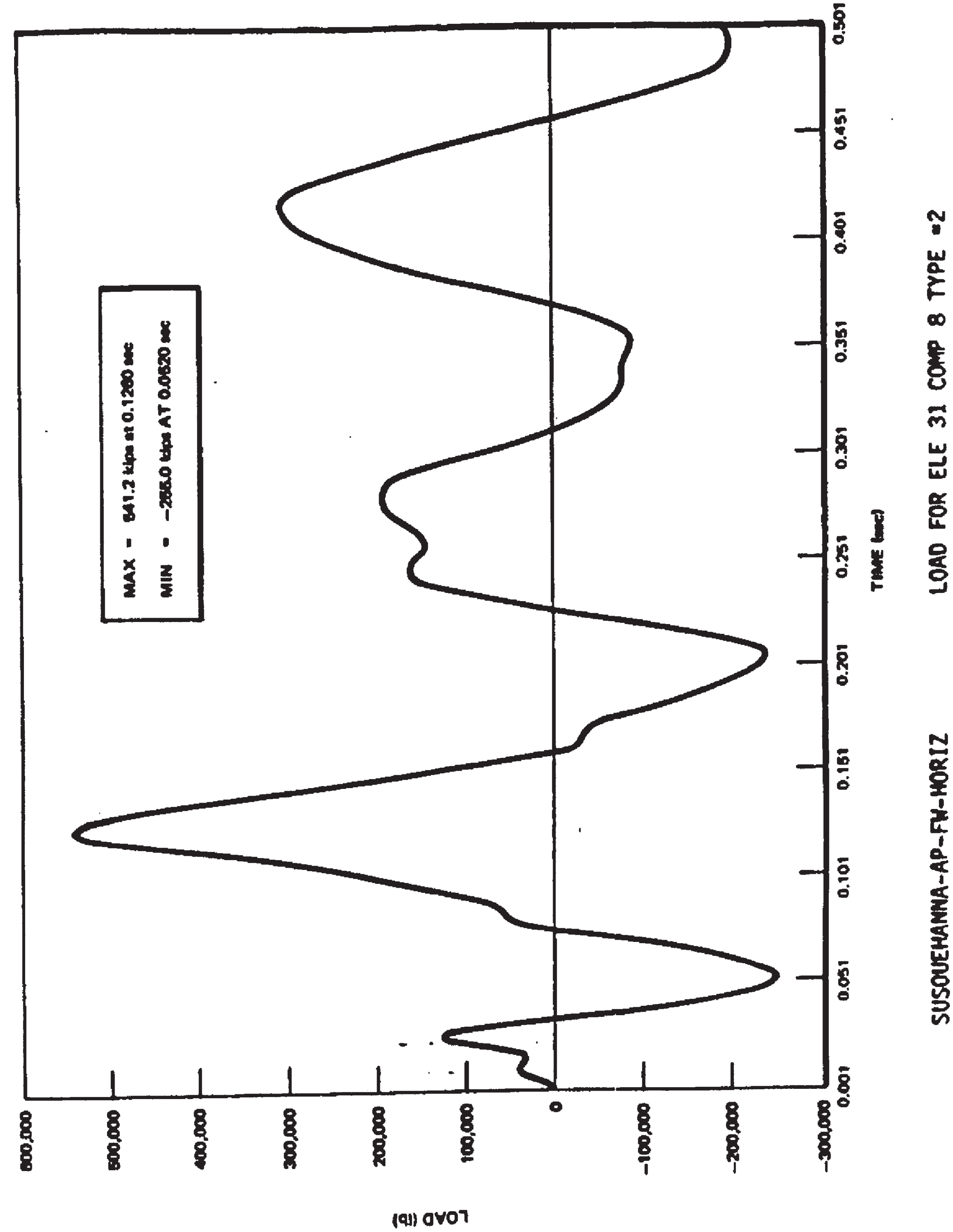
FSAR REV. 46, 06/93

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



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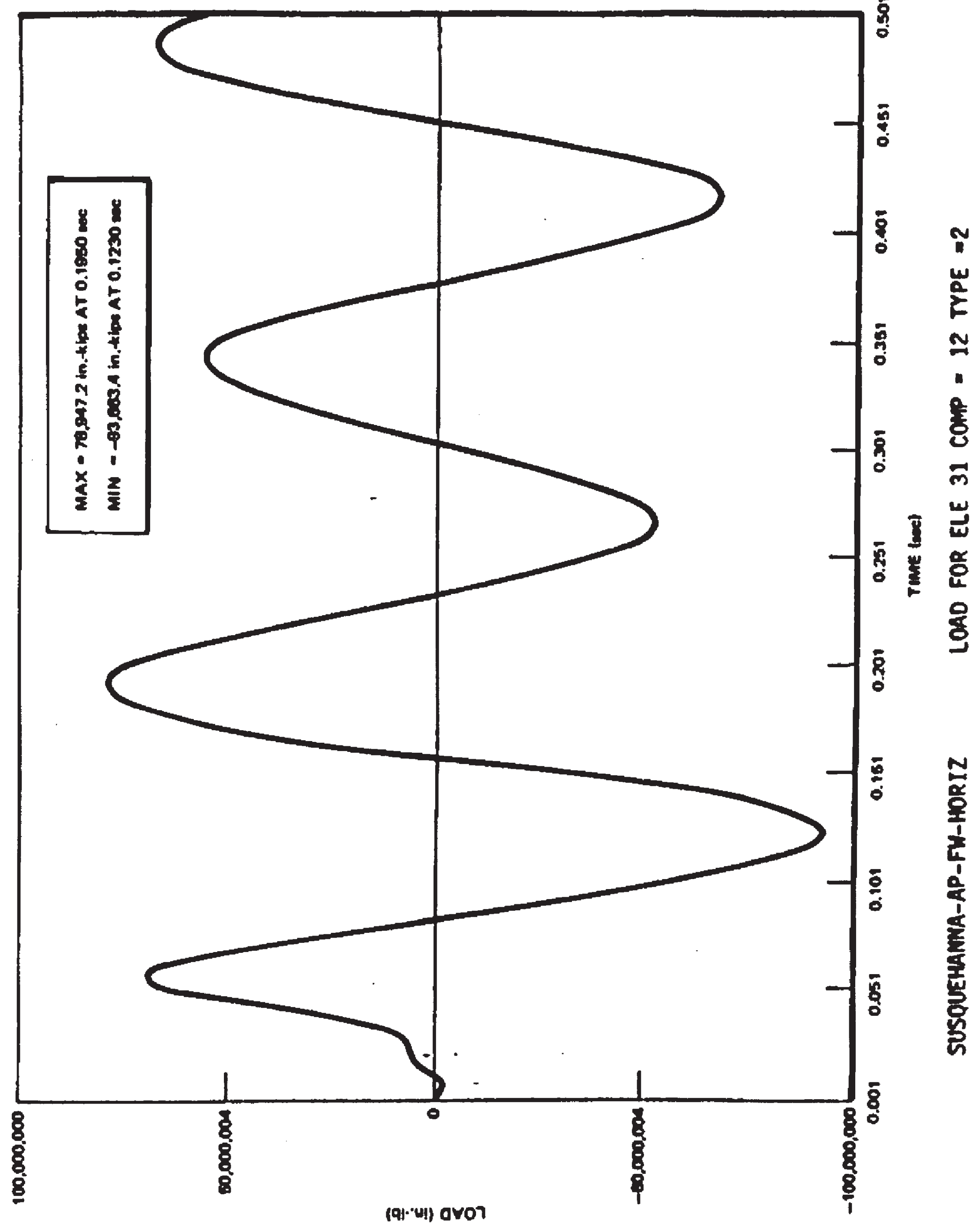
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

ANNULUS PRESSURIZATION
(HORIZONTAL)
FEEDWATER LINE BREAK

FSAR FIGURE 110.41-31

PP&L DRAWING



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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

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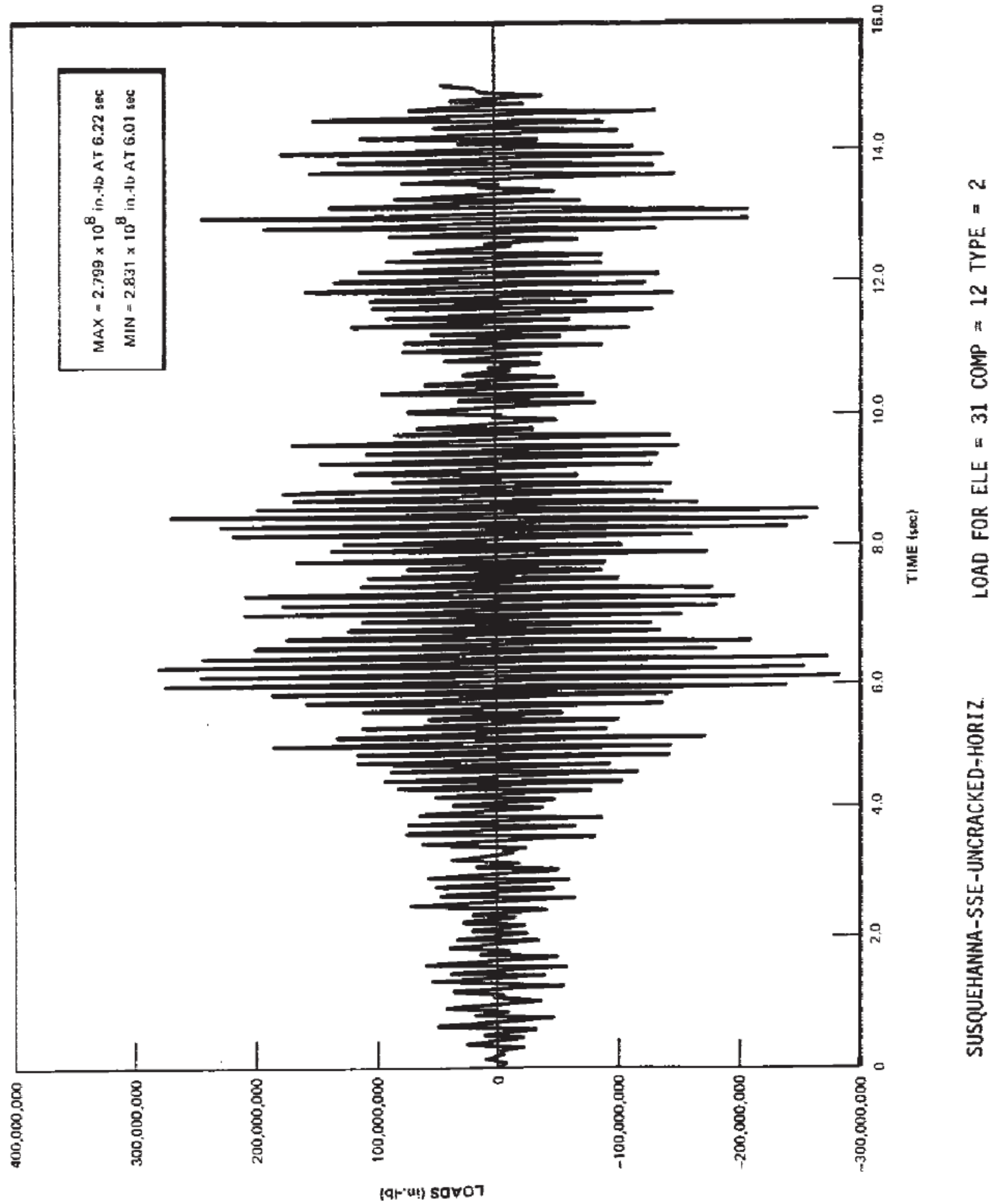
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REACTOR PRESSURE VESSEL AND
INTERNALS HORIZONTAL
MATHEMATICAL MODEL FOR
EARTHQUAKE AND ANNULUS
PRESSURIZATION LOADING

FIGURE 110.41-1, Rev 47

AutoCAD: Figure Fsar 110_41_1.dwg



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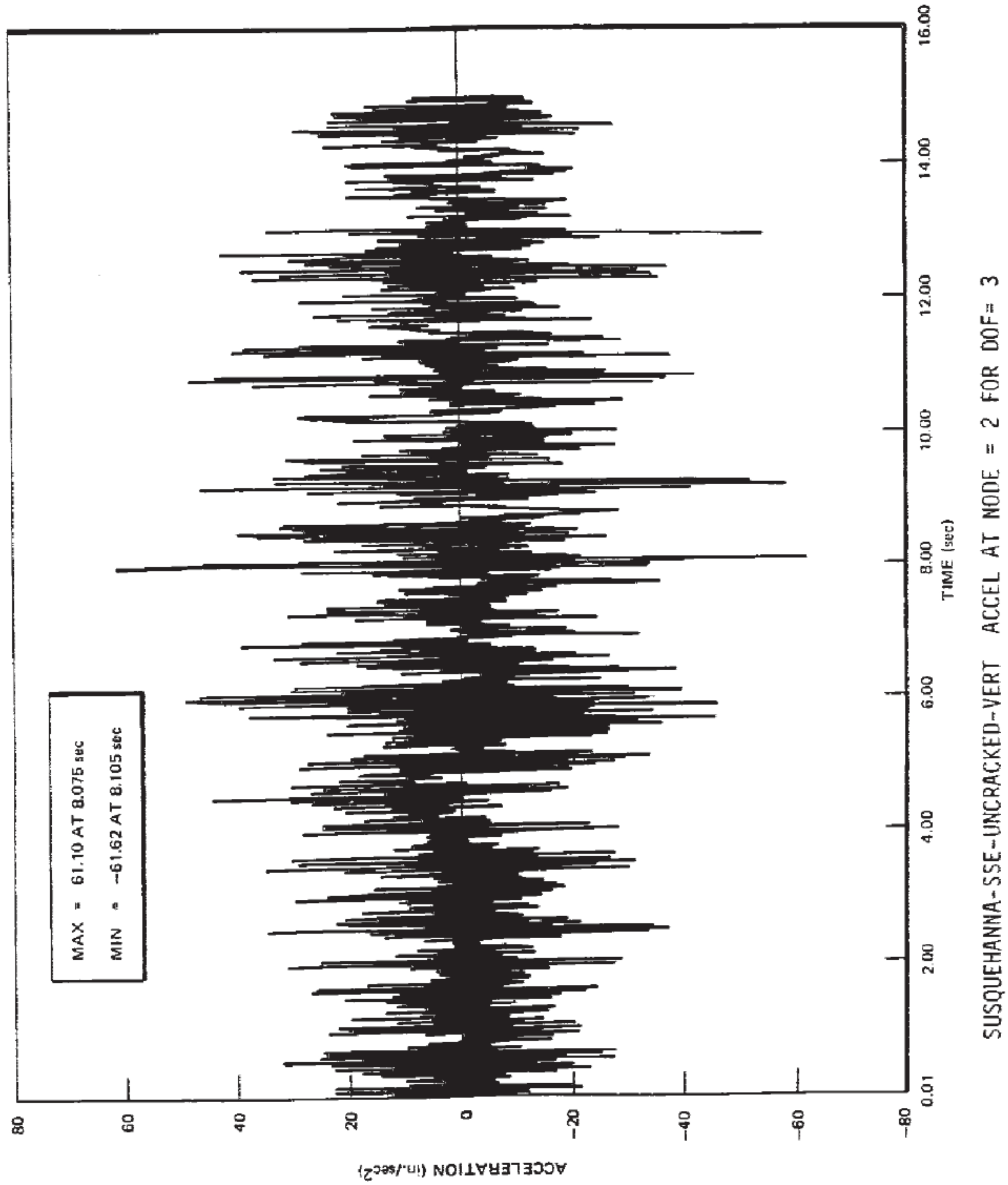
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO HORIZONTAL SSE

FIGURE 110.41-10, Rev 47

AutoCAD: Figure Fsar 110_41_10.dwg



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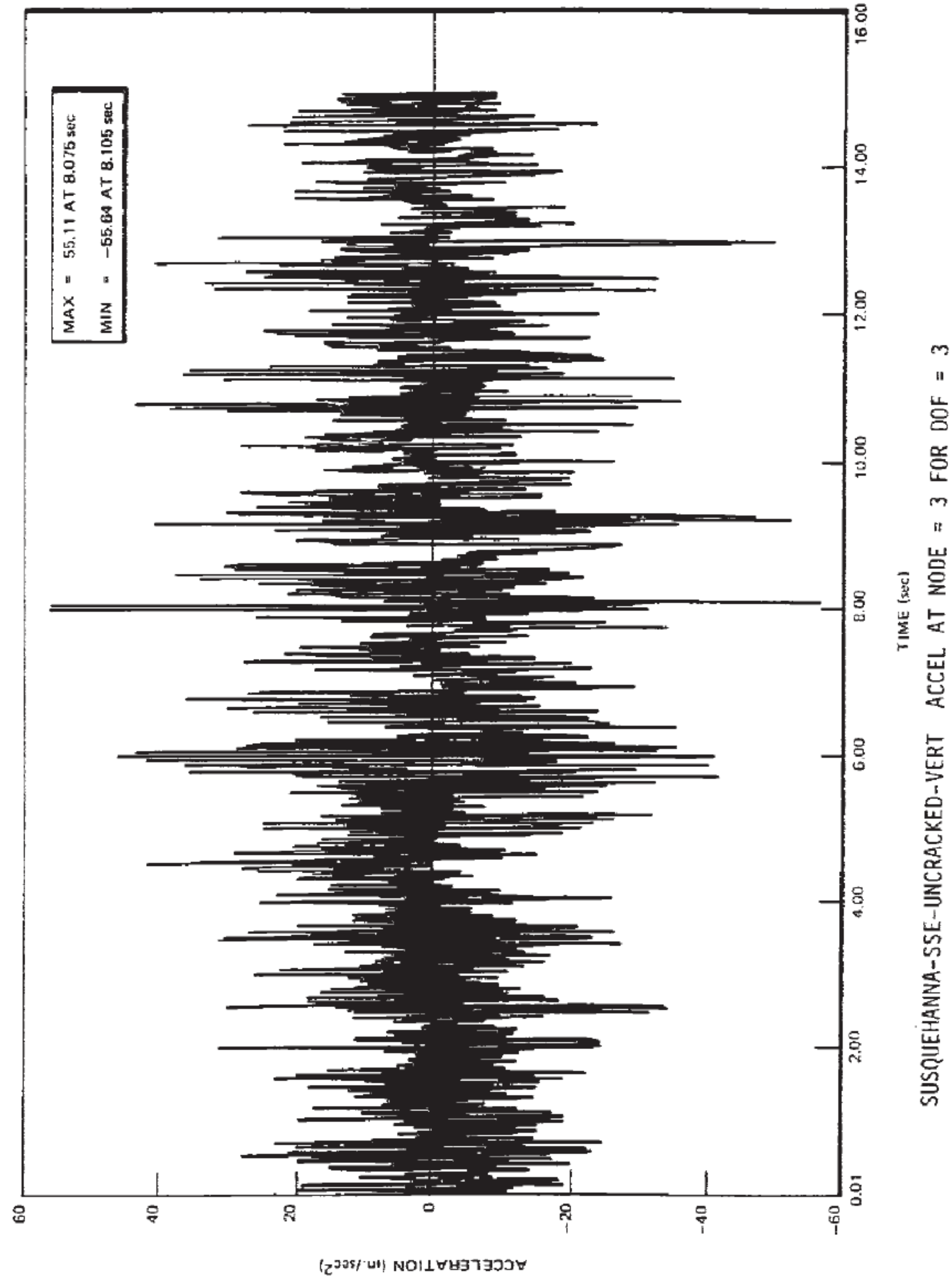
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO VERTICAL SSE

FIGURE 110.41-11, Rev 47

AutoCAD: Figure Fsar 110_41_11.dwg



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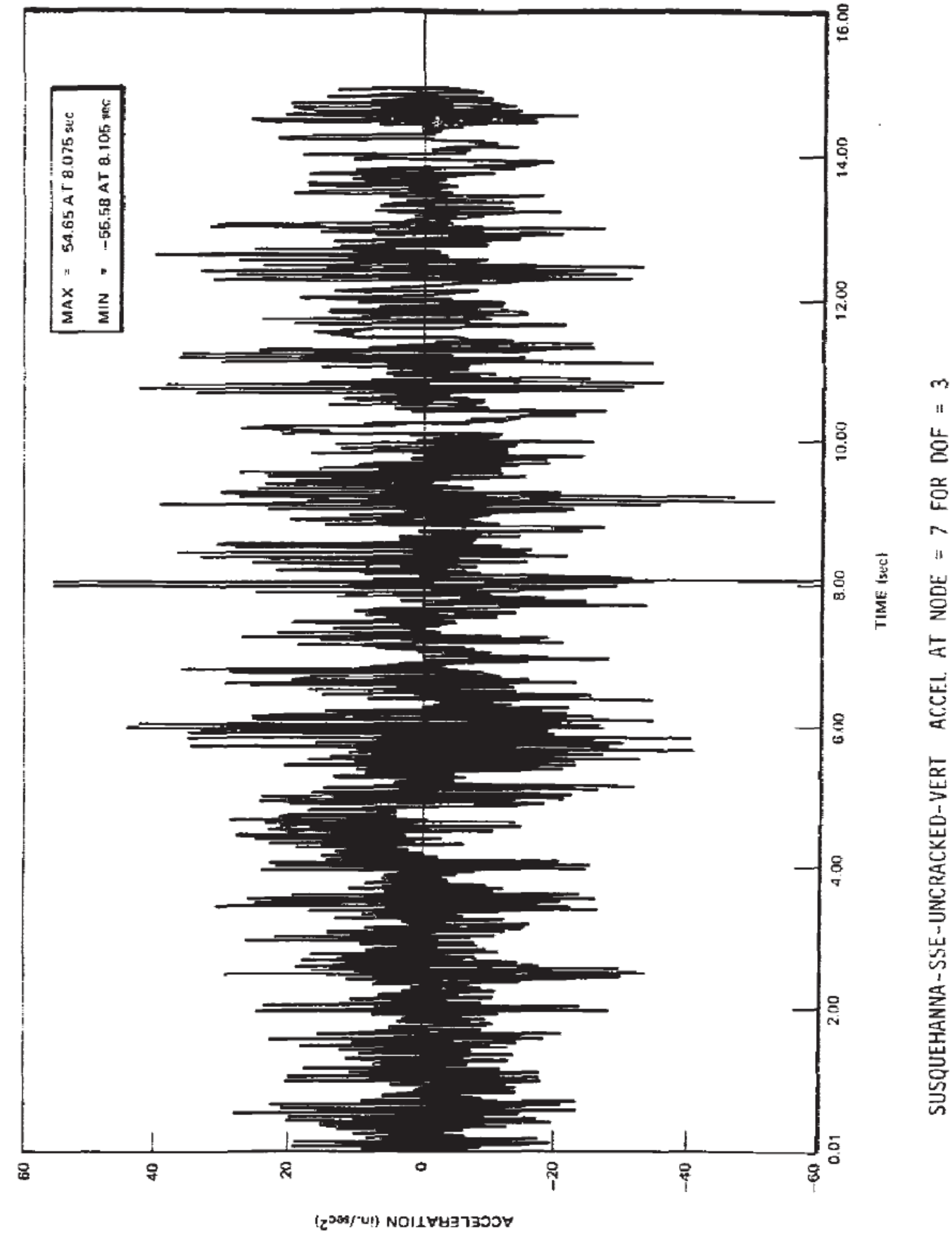
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO VERTICAL SSE

FIGURE 110.41-12, Rev 47

AutoCAD: Figure Fsar 110_41_12.dwg



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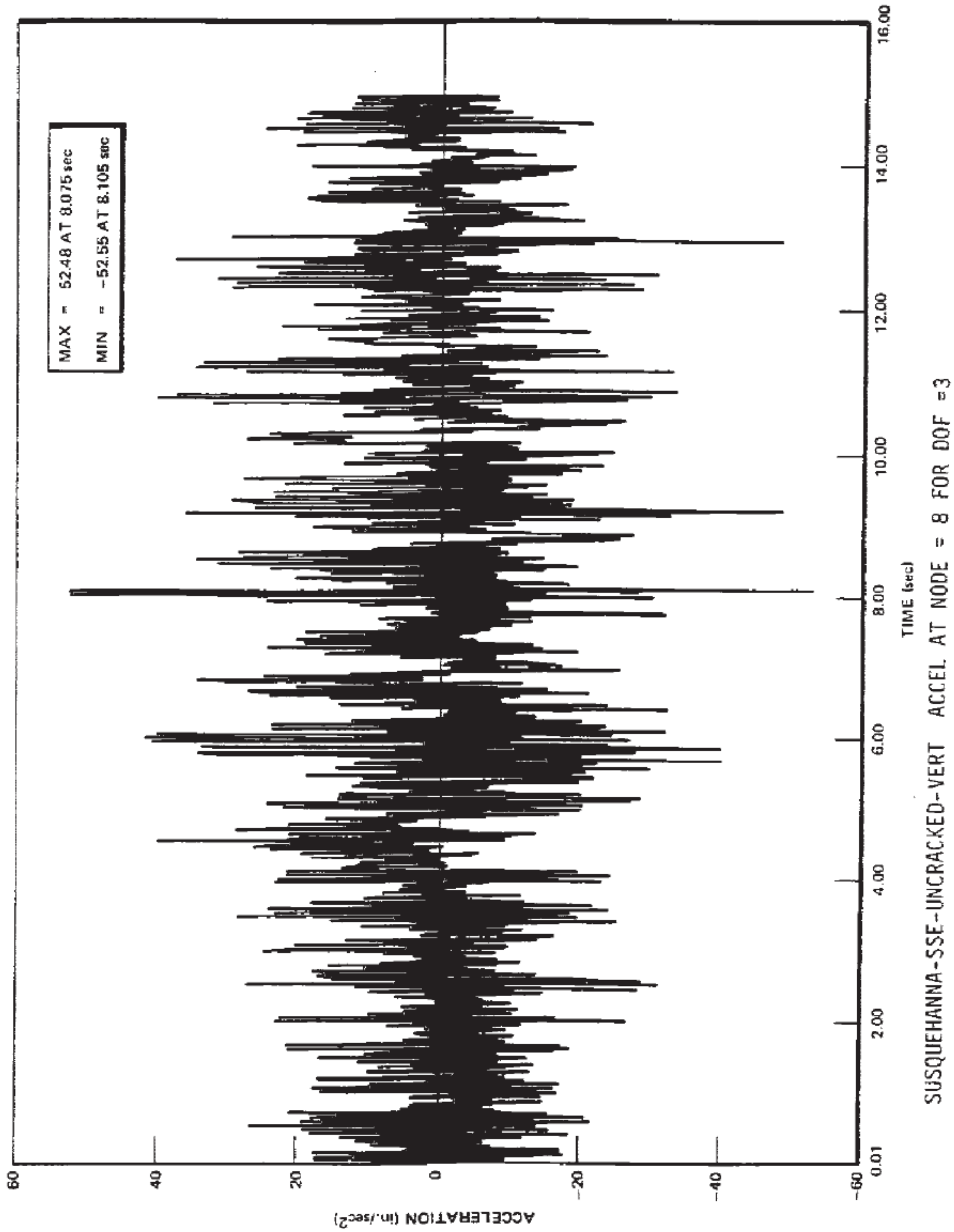
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO VERTICAL SSE

FIGURE 110.41-13, Rev 47

AutoCAD: Figure Fsar 110_41_13.dwg



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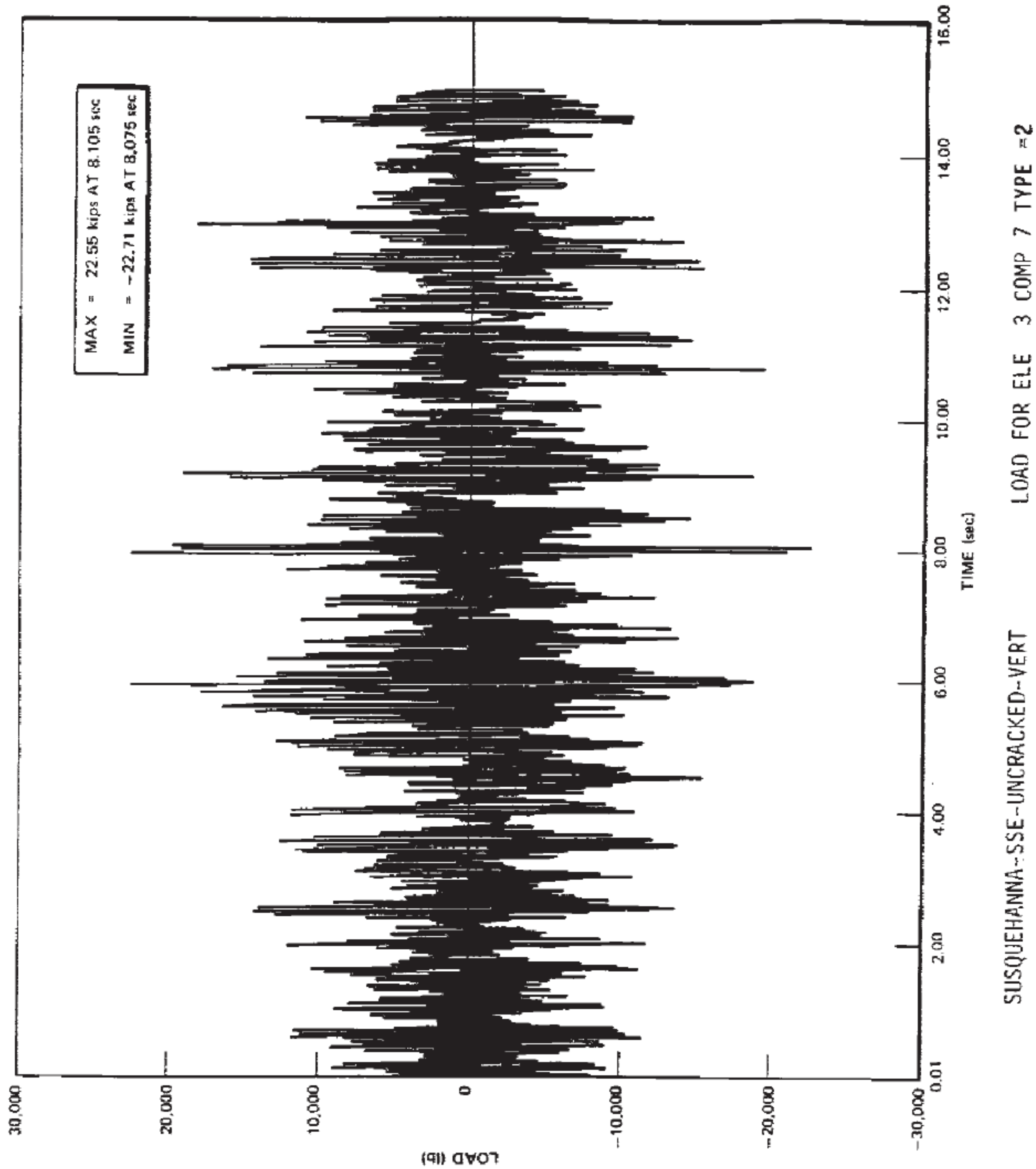
SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
RESPONSE TIME HISTORY PLOT DUE TO VERTICAL SSE FIGURE 110.41-14, Rev 47

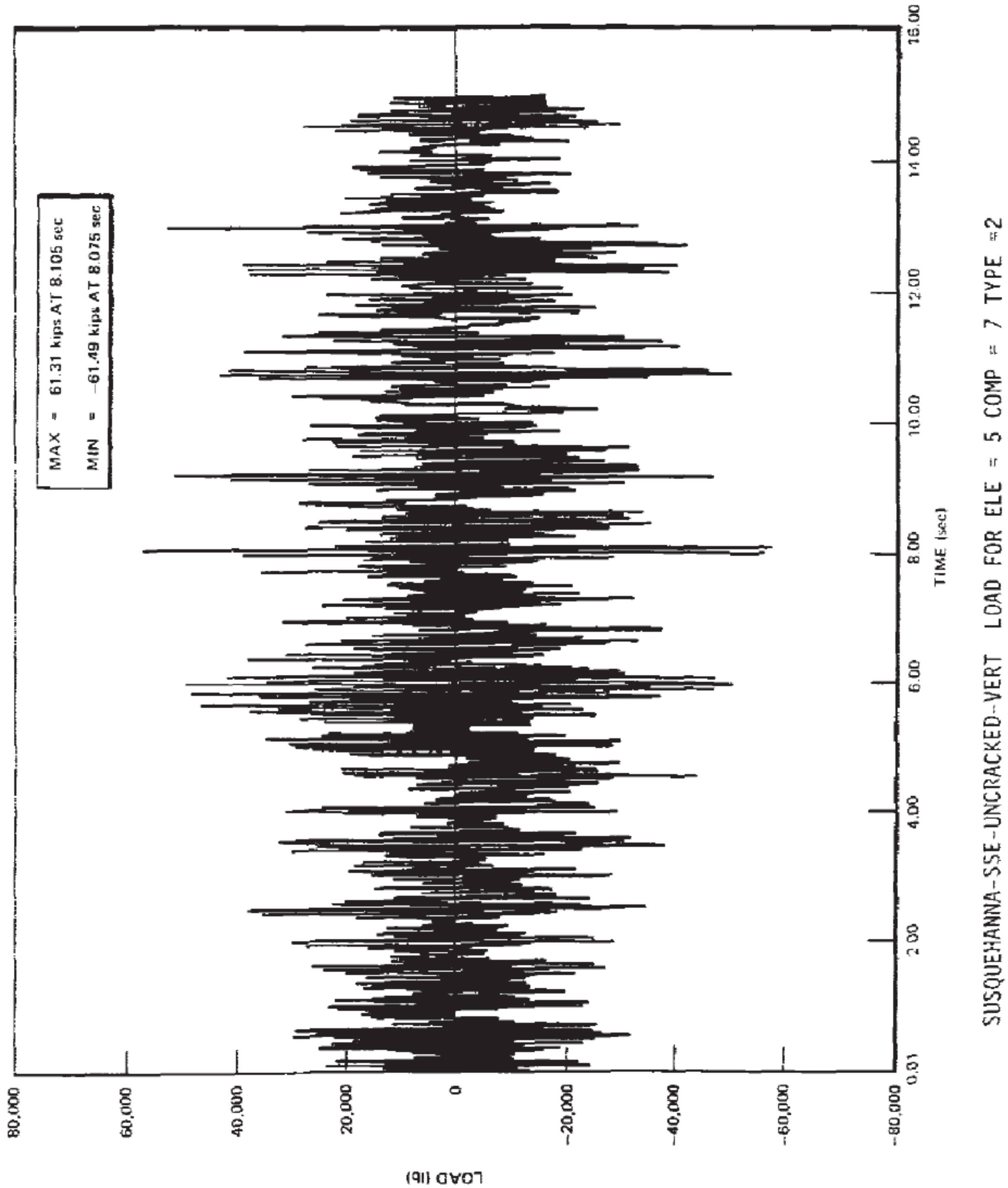
AutoCAD: Figure Fsar 110_41_14.dwg



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

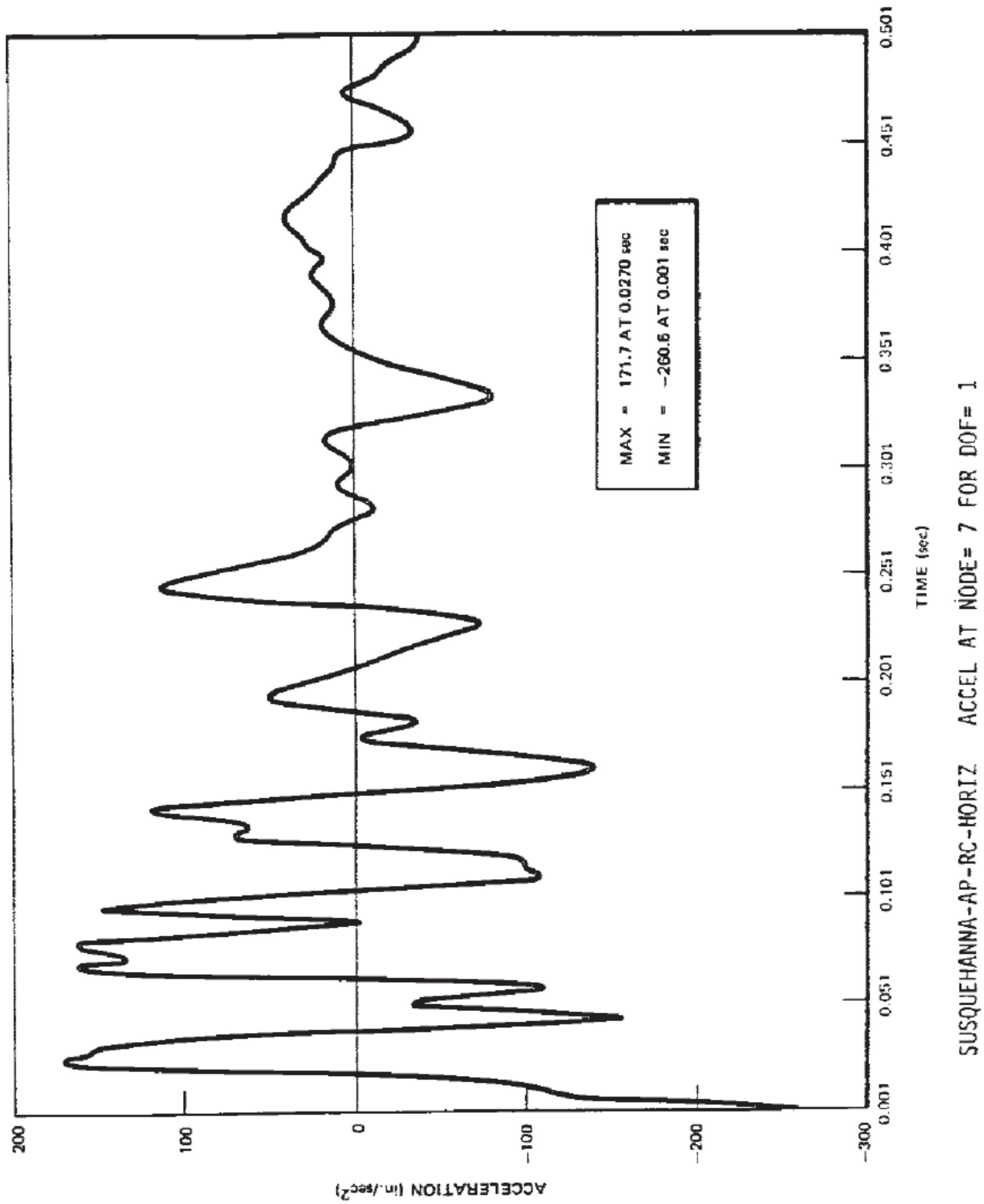
AutoCAD: Figure Fsar 110_41_15.dwg



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



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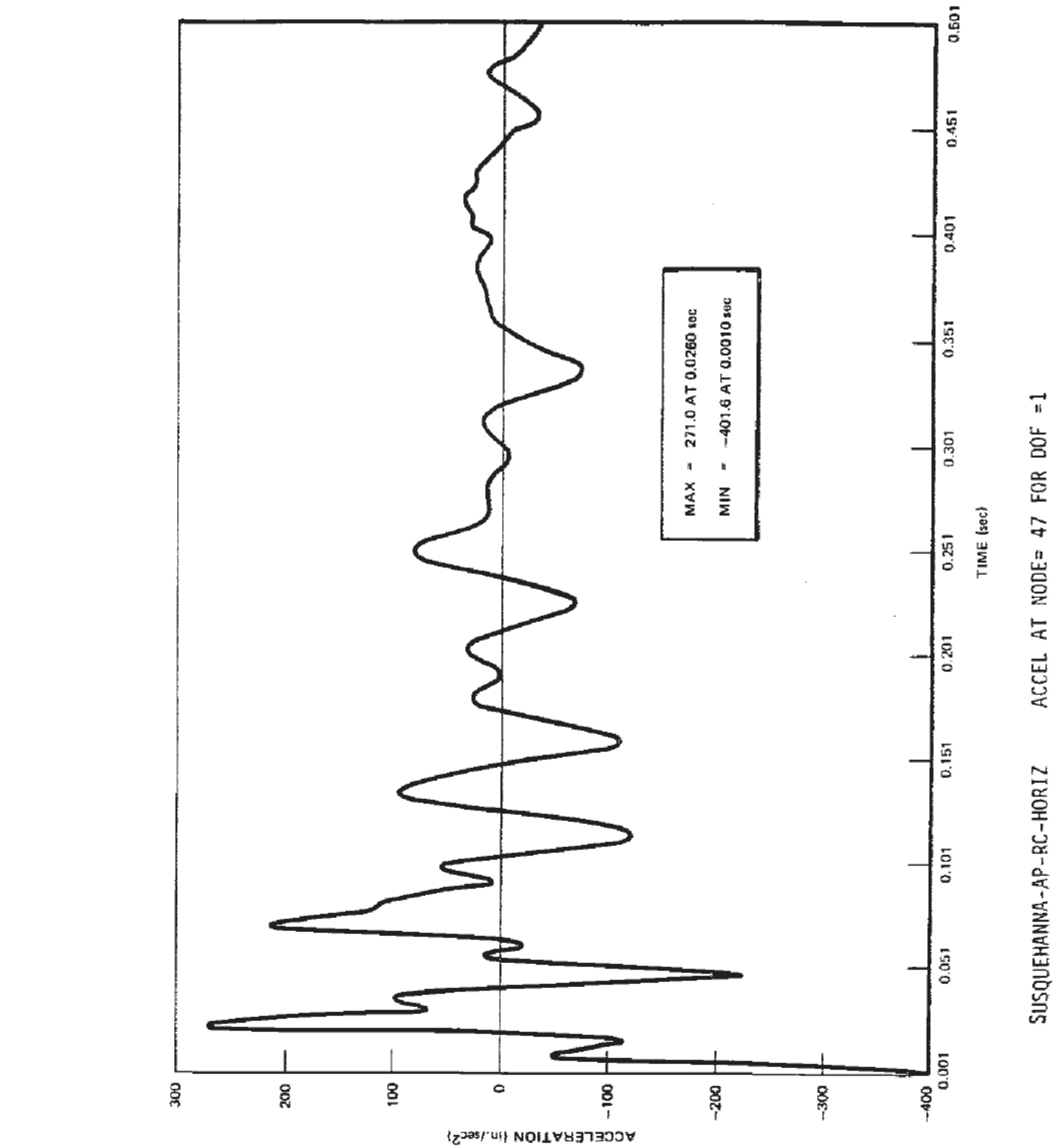
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO
RECIRCULATION PIPE BREAK

FIGURE 110.41-17, Rev 47

AutoCAD: Figure Fsar 110_41_17.dwg



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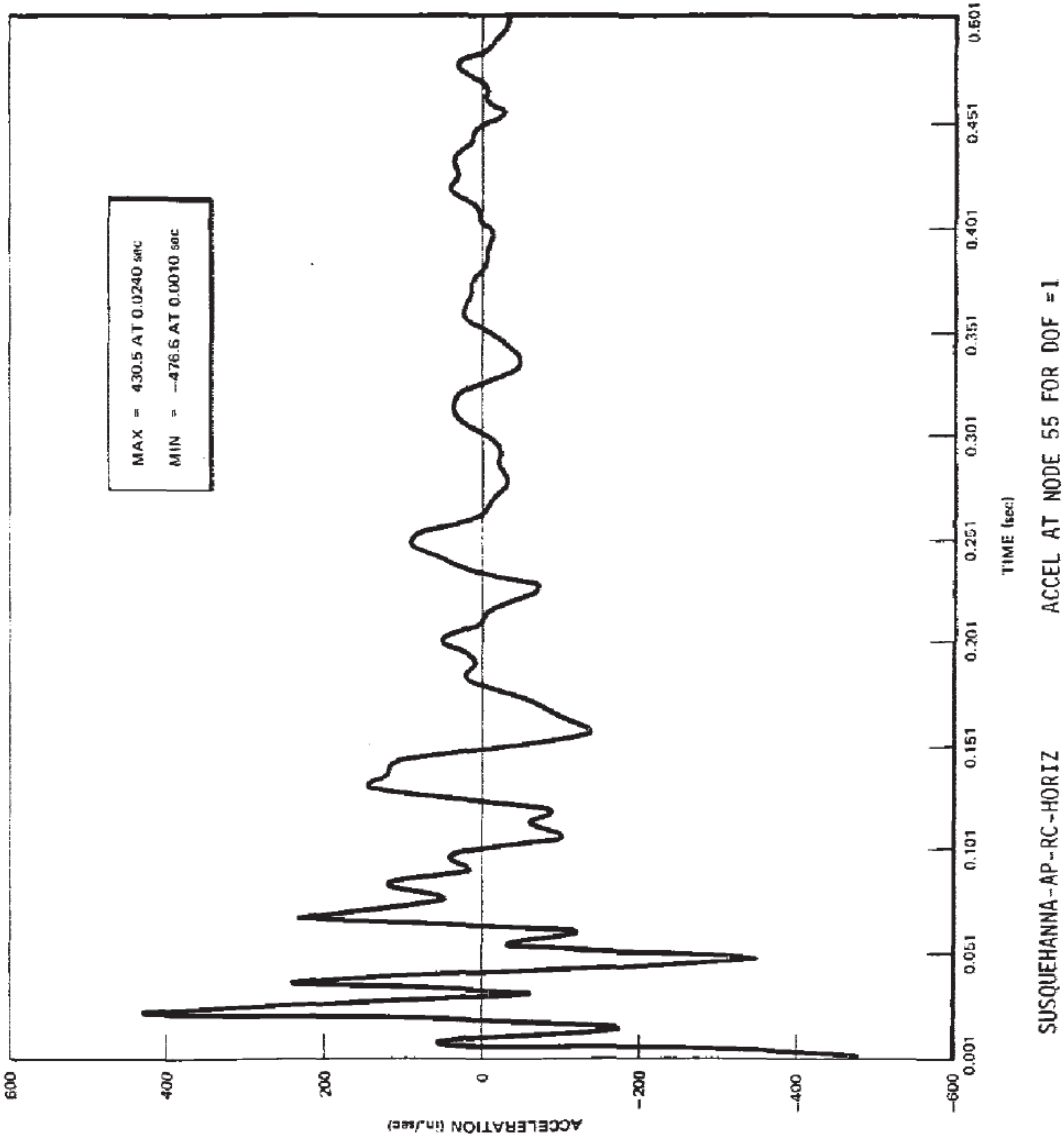
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO
RECIRCULATION PIPE BREAK

FIGURE 110.41-18, Rev 47

AutoCAD: Figure Fsar 110_41_18.dwg



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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 110.41-19, Rev 47

AutoCAD: Figure Fsar 110_41_19.dwg

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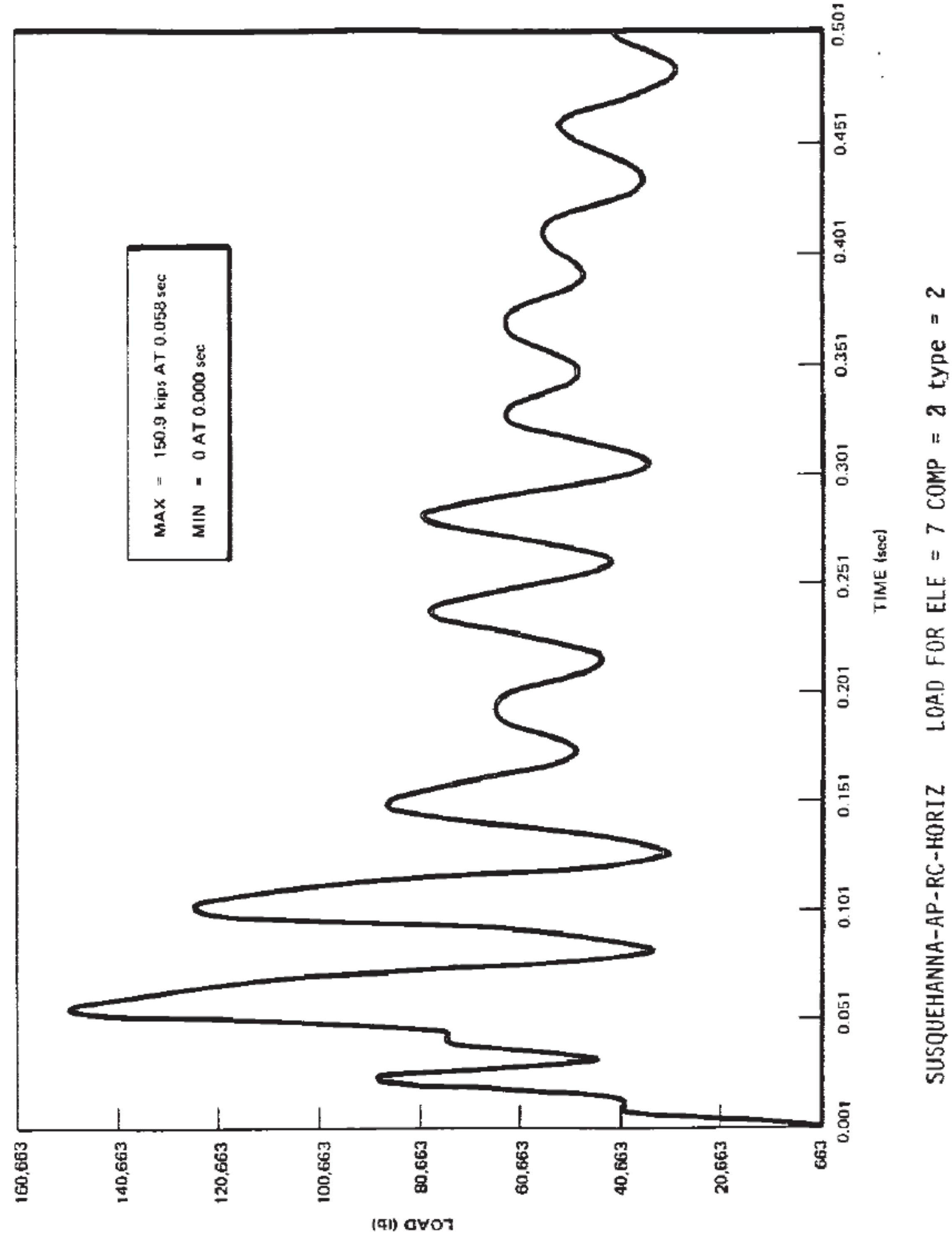
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REACTOR PRESSURE VESSEL
AND INTERNALS
VERTICAL DYNAMIC MODEL
FOR EARTHQUAKE LOADING

FIGURE 110.41-2, Rev 47

AutoCAD: Figure Fsar 110_41_2.dwg



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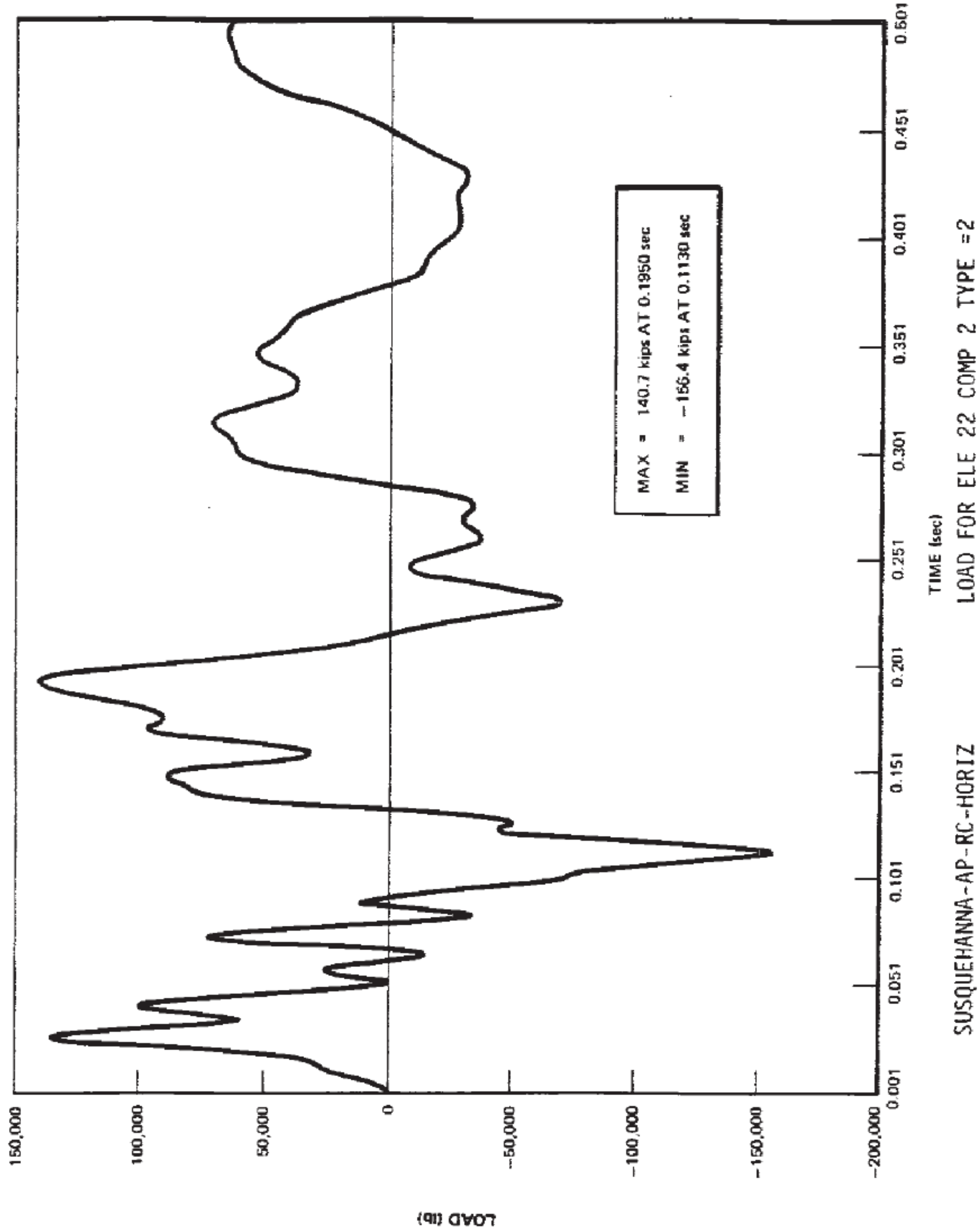
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO
RECIRCULATION PIPE BREAK

FIGURE 110.41-20, Rev 47

AutoCAD: Figure Fsar 110_41_20.dwg



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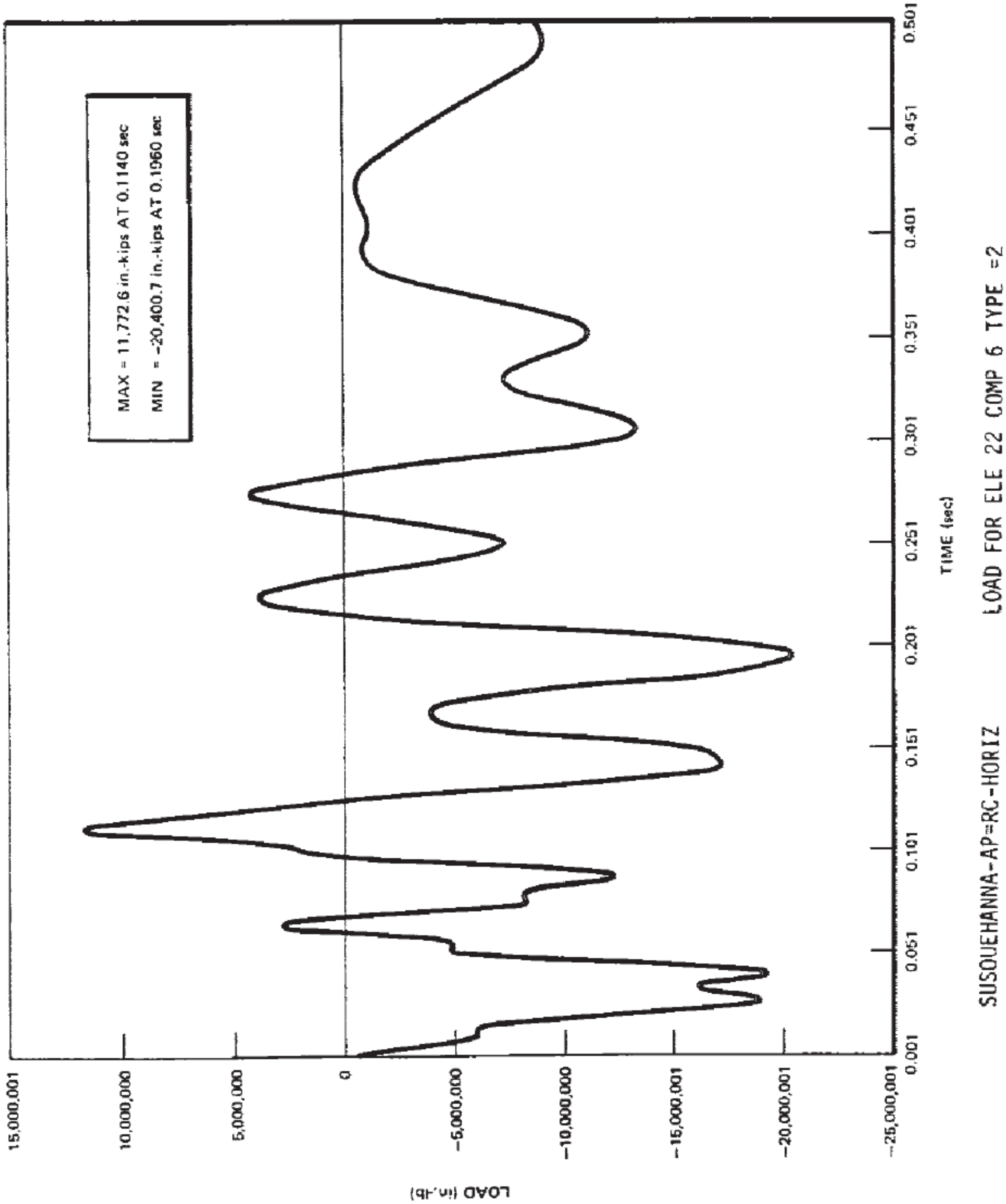
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO
RECIRCULATION PIPE BREAK

FIGURE 110.41-21, Rev 47

AutoCAD: Figure Fsar 110_41_21.dwg



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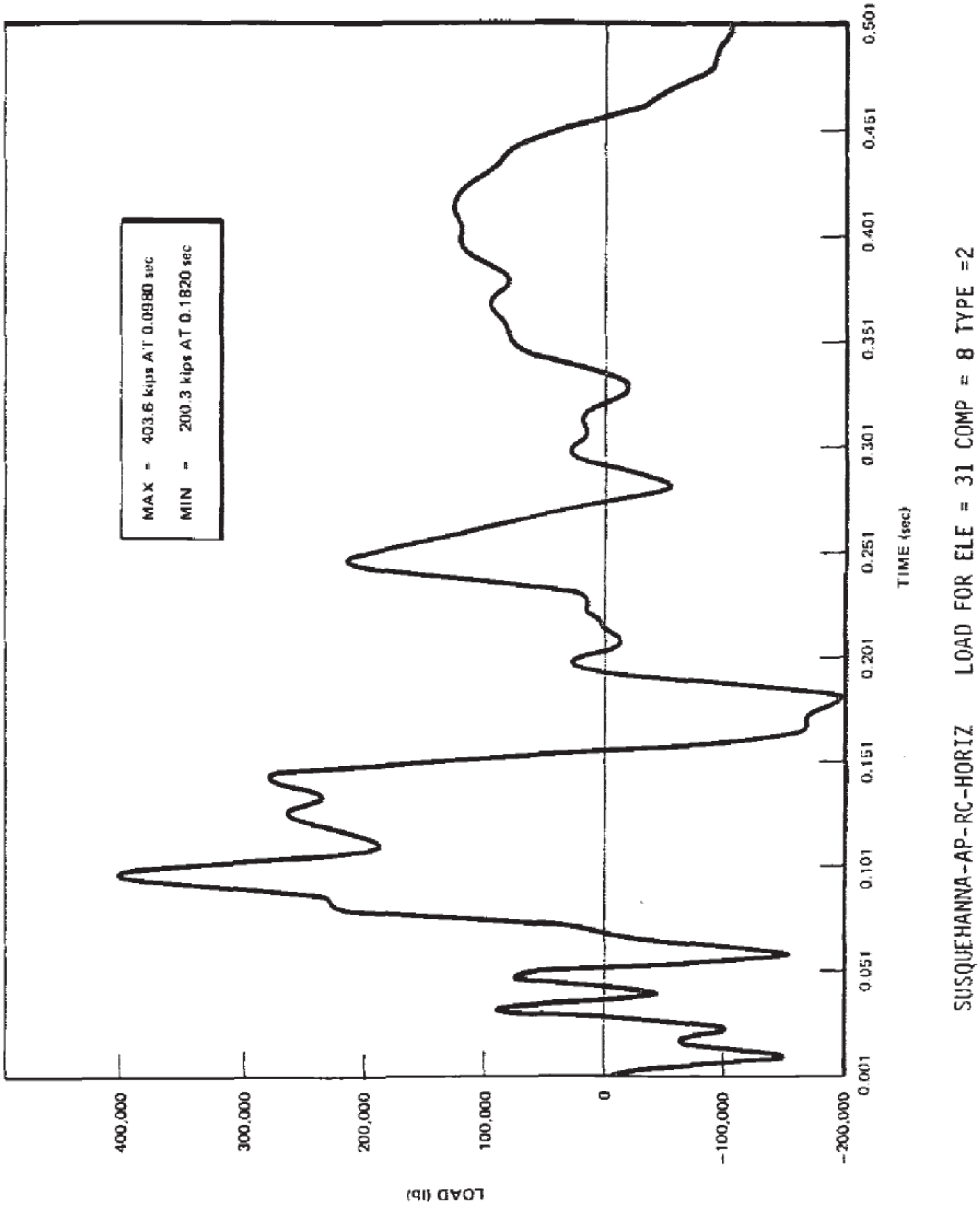
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO
RECIRCULATION PIPE BREAK

FIGURE 110.41-22, Rev 47

AutoCAD: Figure Fsar 110_41_22.dwg



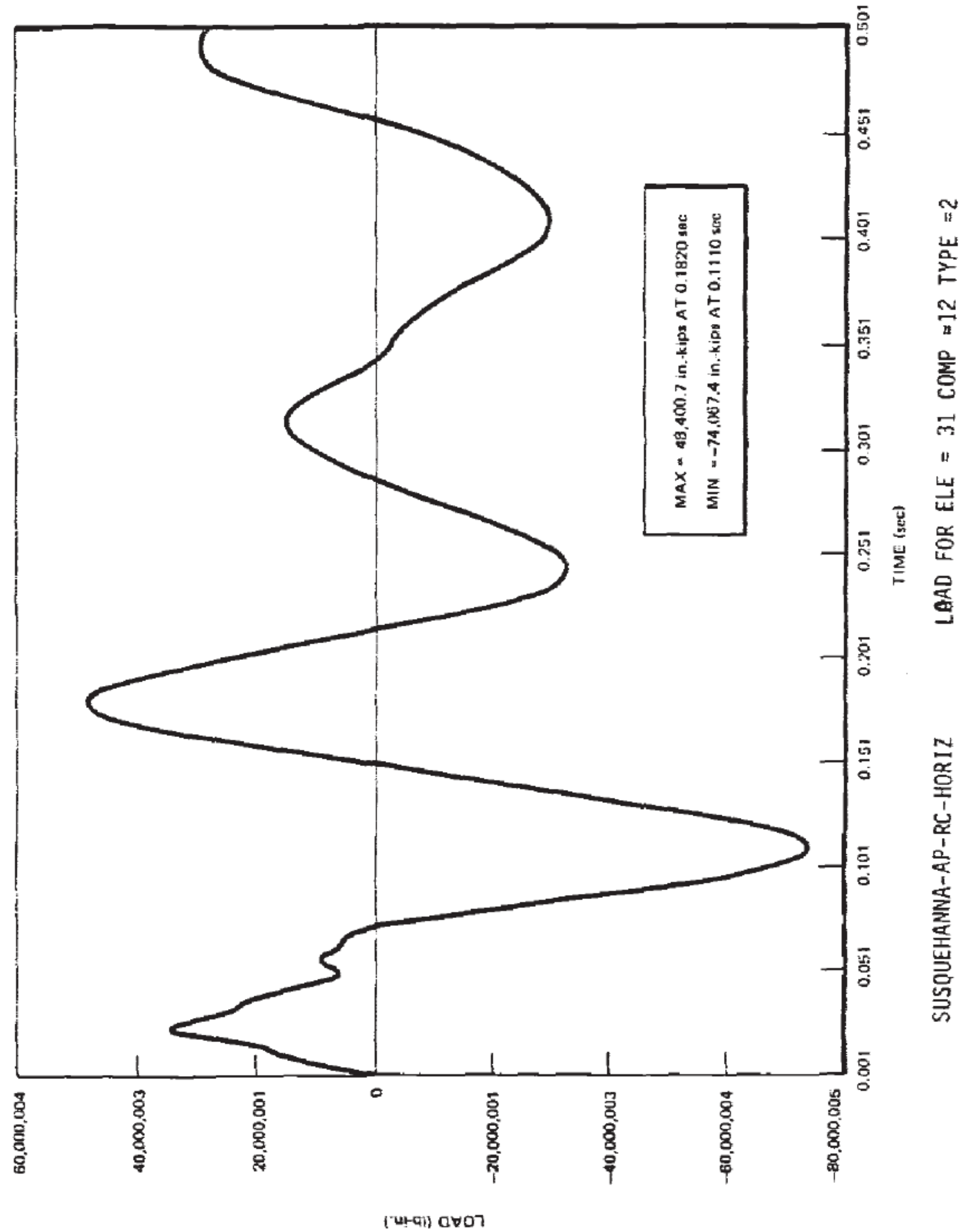
FSAR REV.65

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

RESPONSE TIME HISTORY PLOT
DUE TO
RECIRCULATION PIPE BREAK

FIGURE 110.41-23, Rev 47

AutoCAD: Figure Fsar 110_41_23.dwg



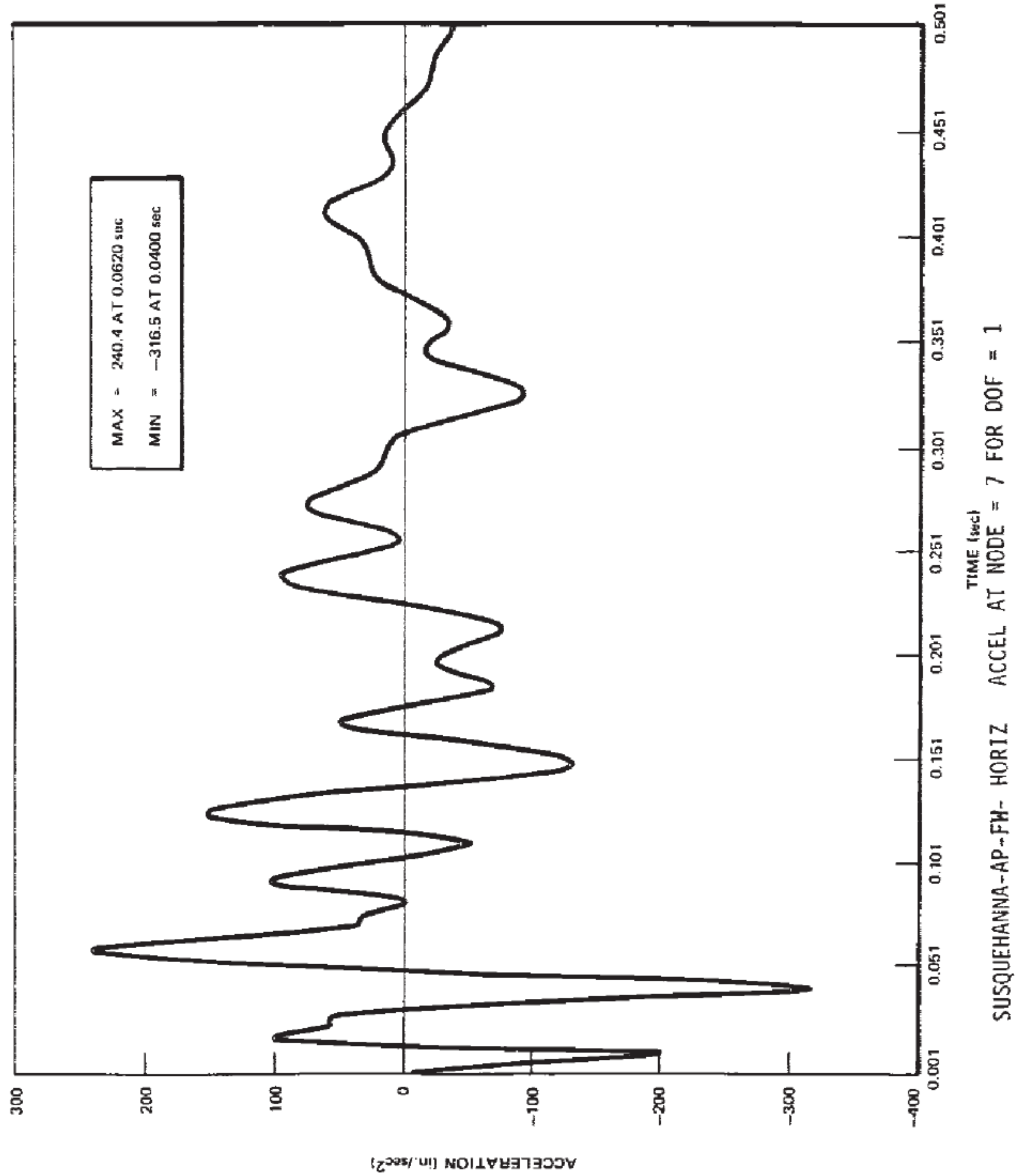
FSAR REV.65

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



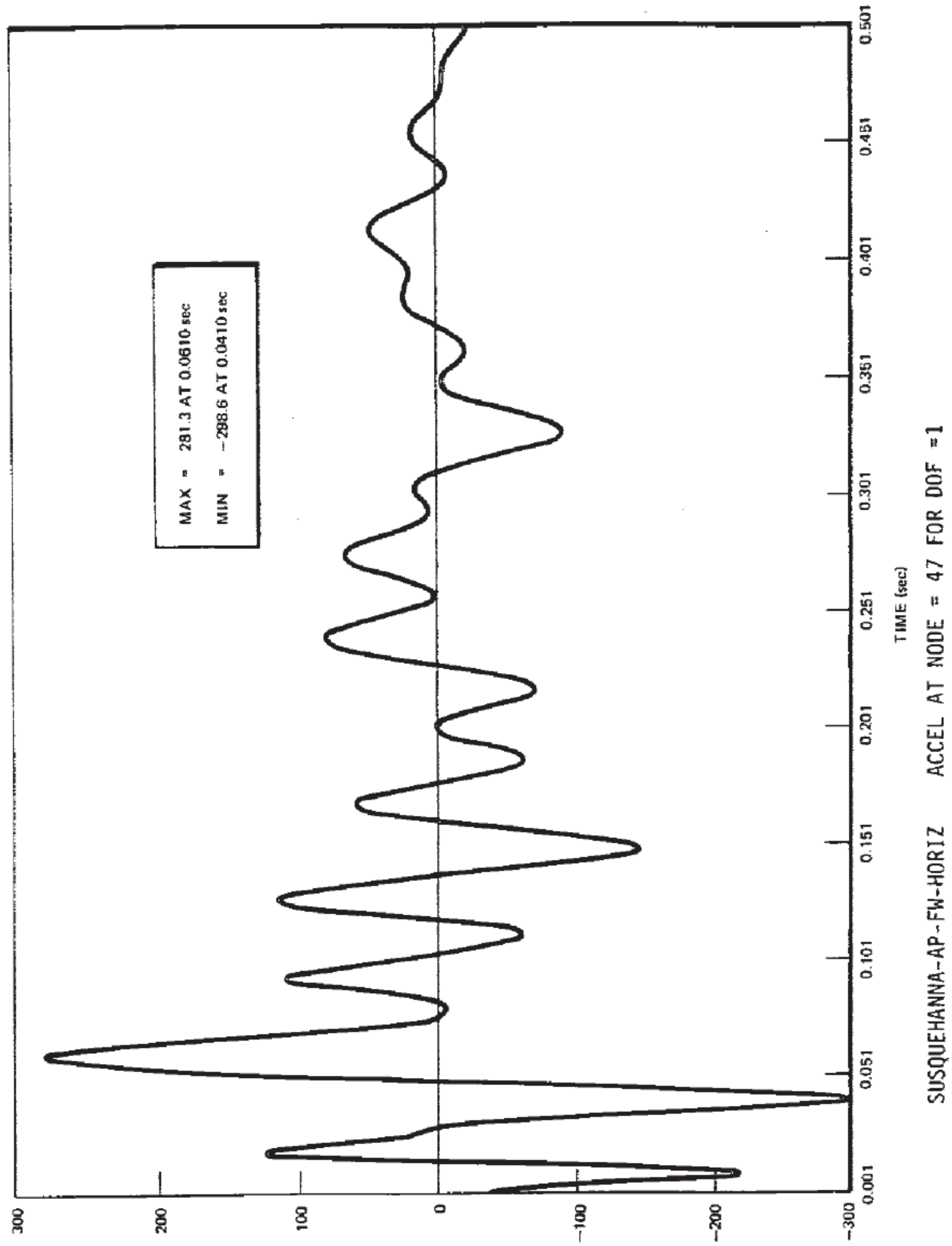
FSAR REV.65

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



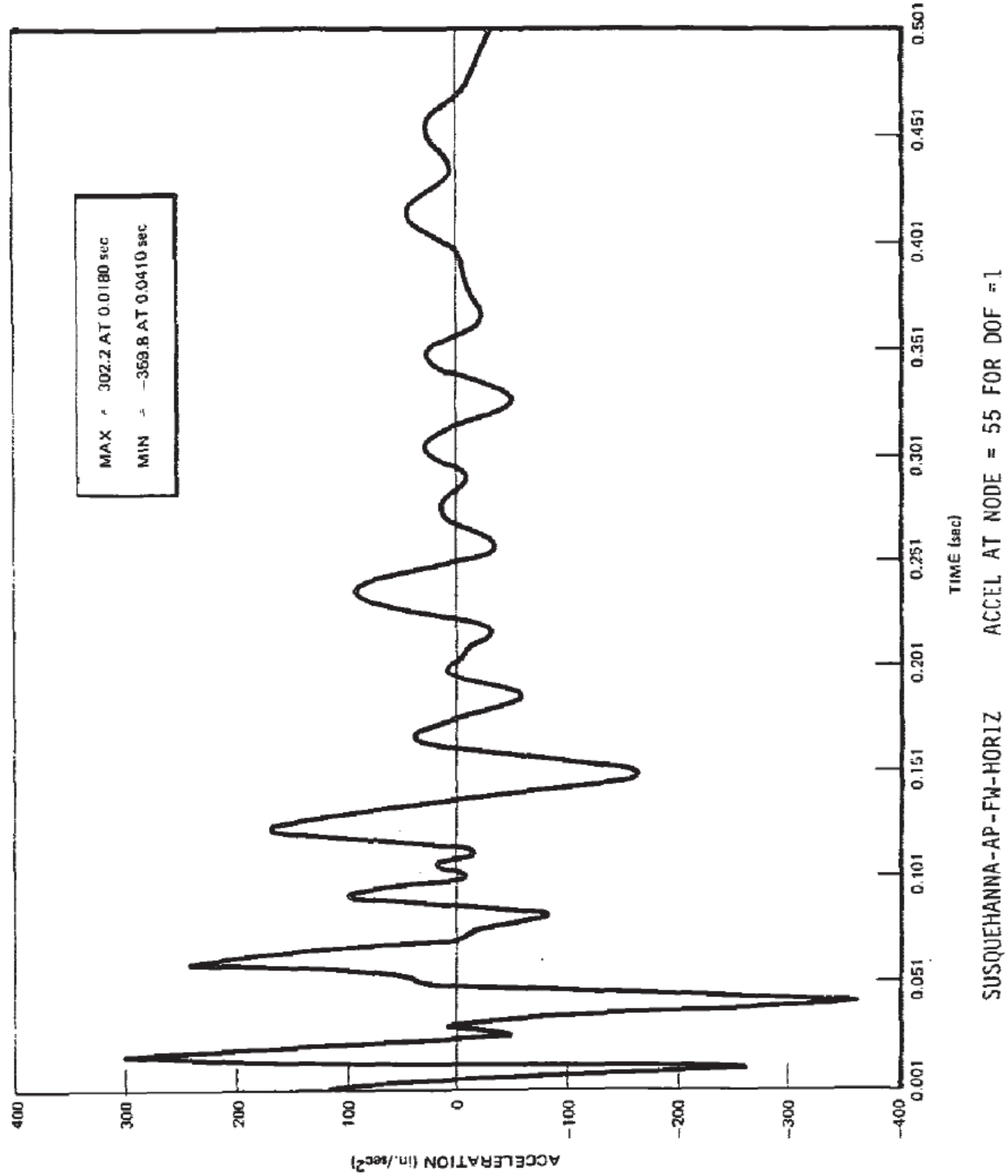
FSAR REV.65

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

ANNULUS PRESSURIZATION
(HORIZONTAL)
FEEDWATER LINE BREAK

FIGURE 110.41-26, Rev 47

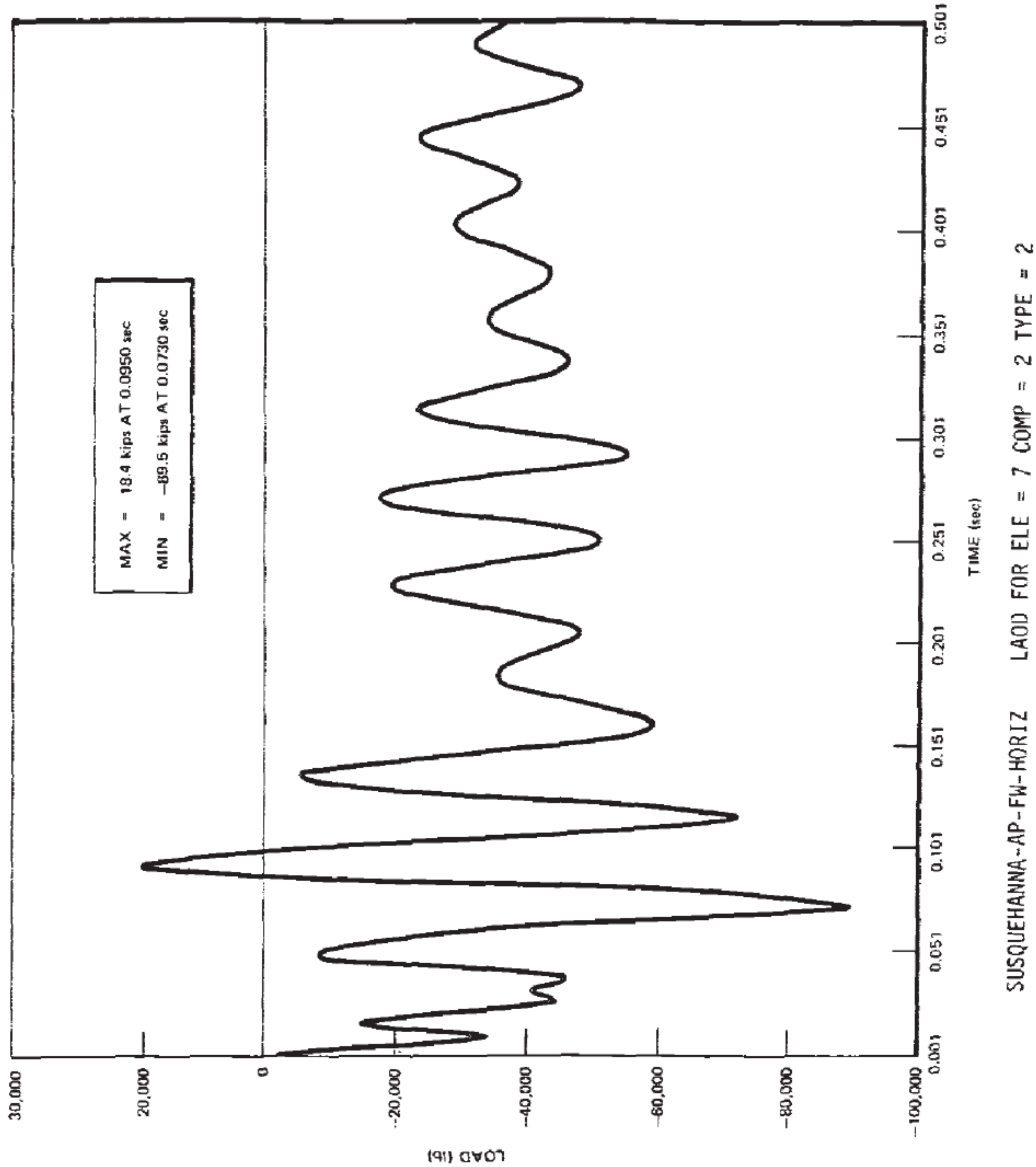
AutoCAD: Figure Fsar 110_41_26.dwg



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



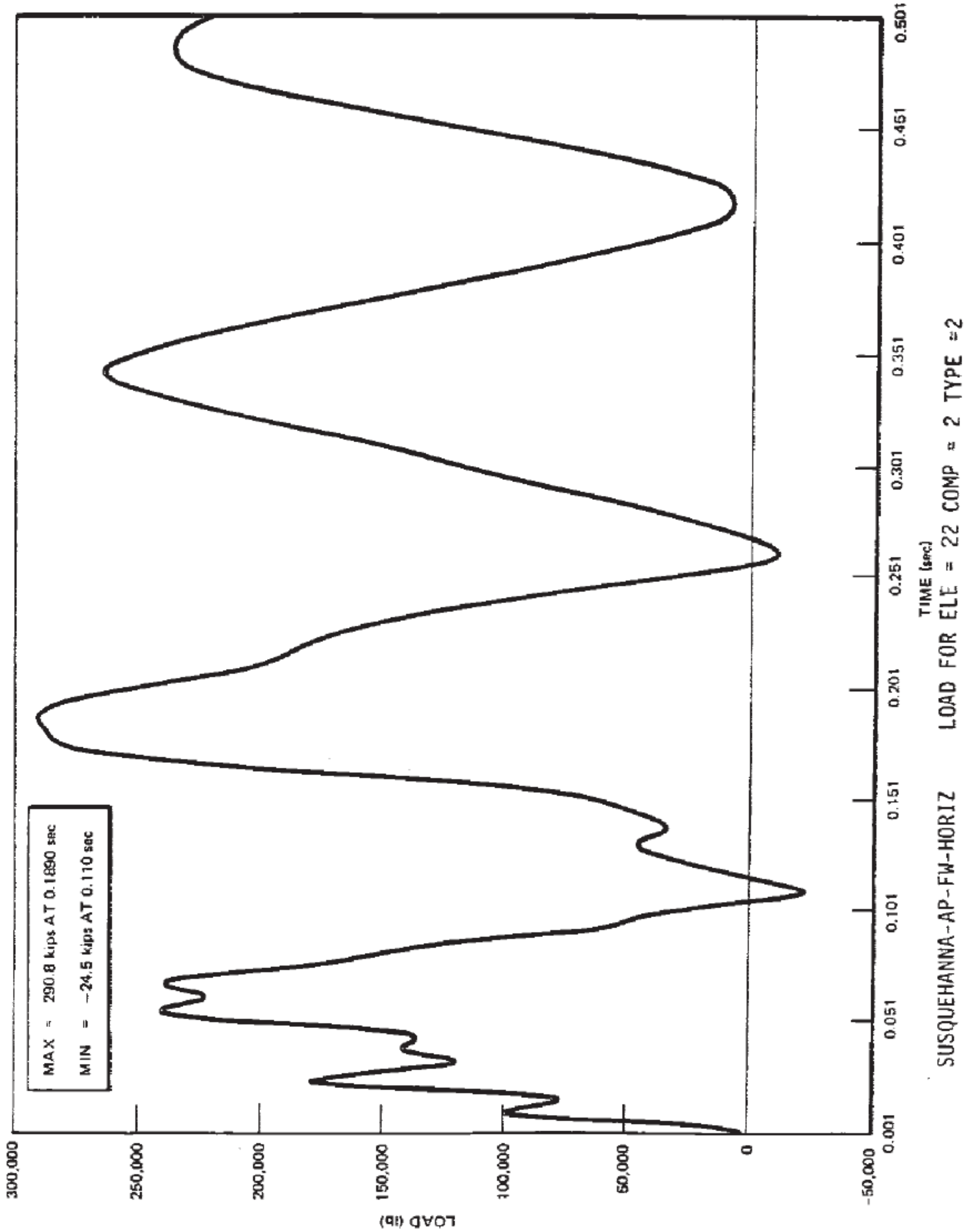
FSAR REV.65

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



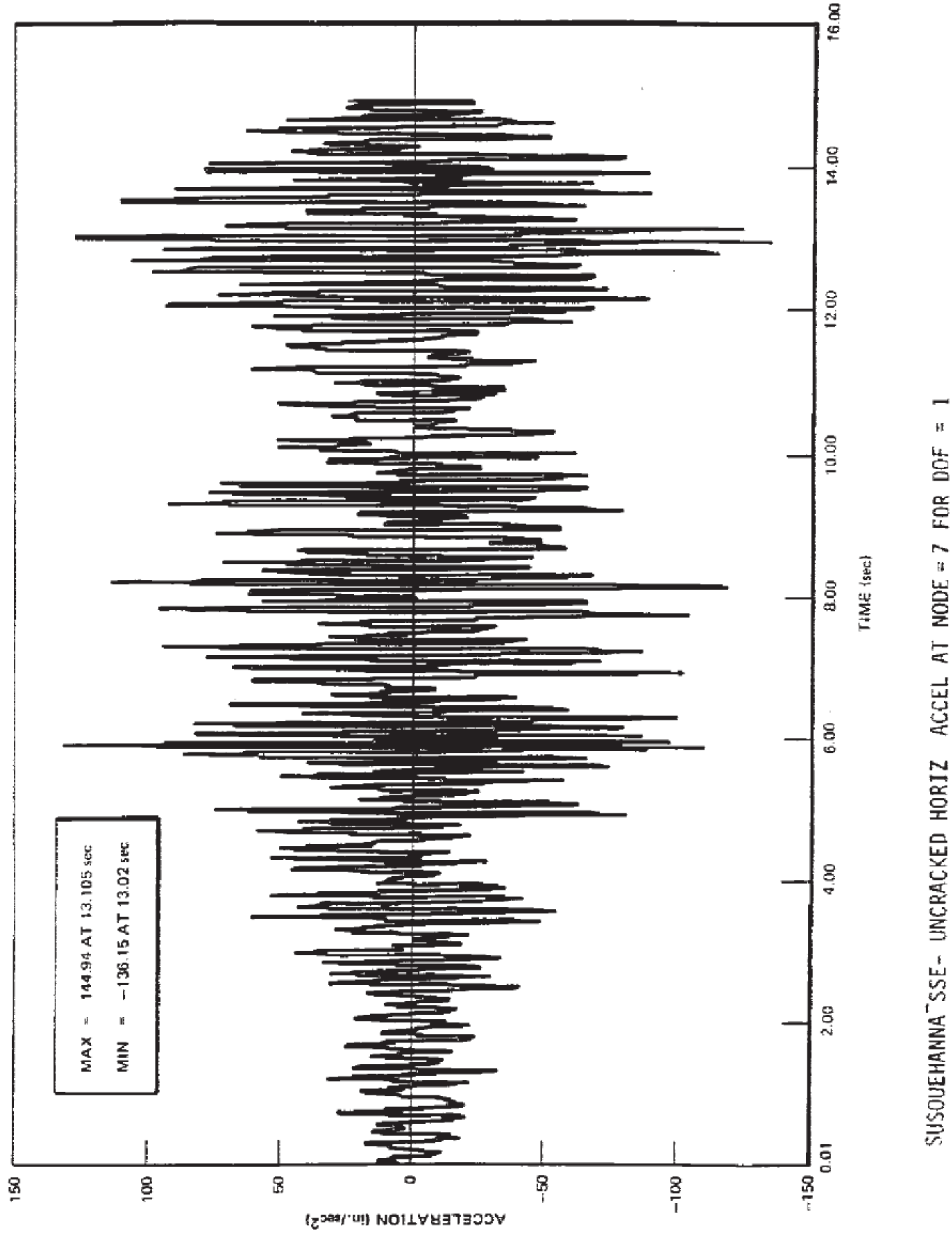
FSAR REV.65

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



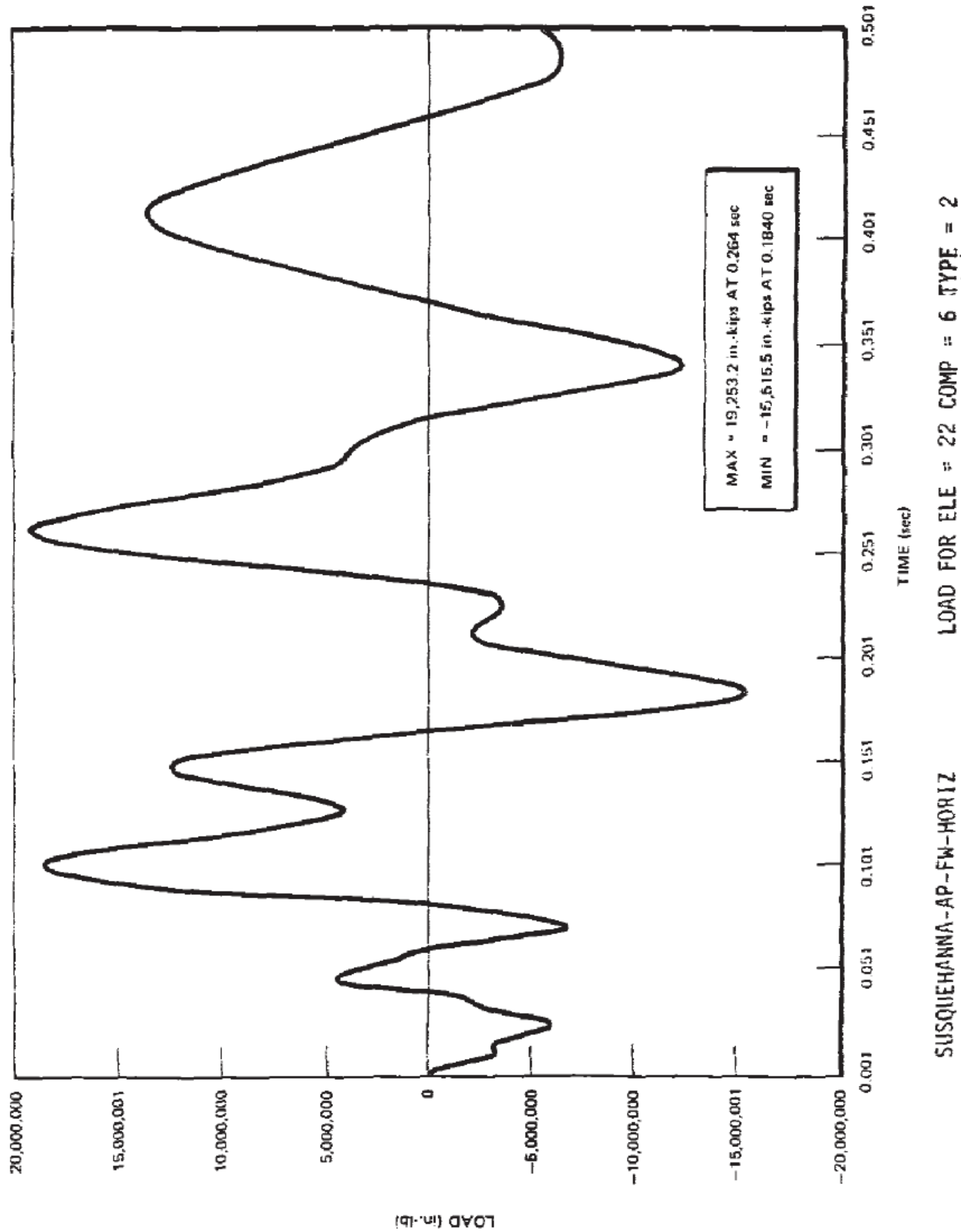
FSAR REV.65

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

RESPONSE TIME HISTORY PLOT
DUE TO HORIZONTAL SSE

FIGURE 110.41-3, Rev 47

AutoCAD: Figure Fsar 110_41_3.dwg



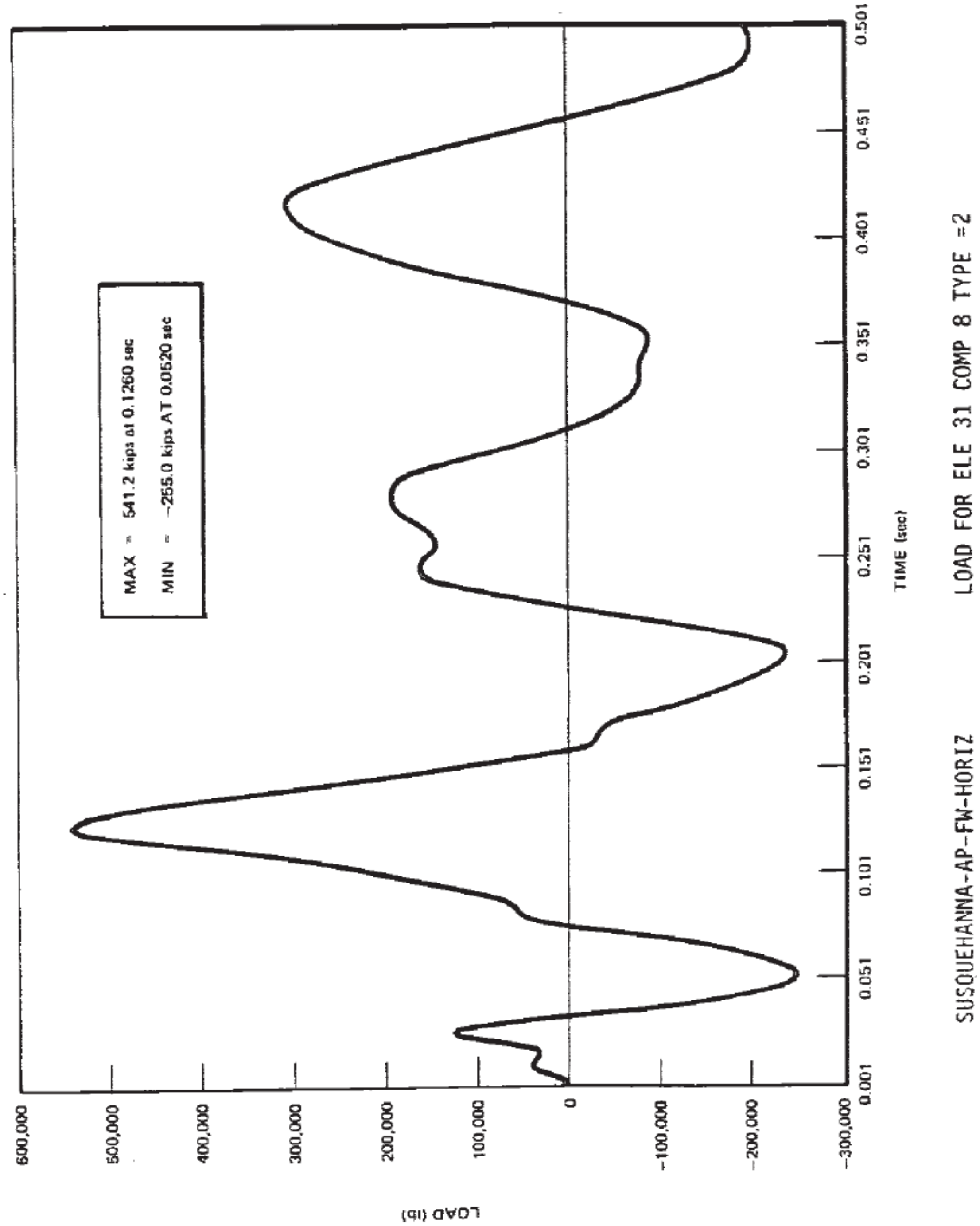
FSAR REV.65

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



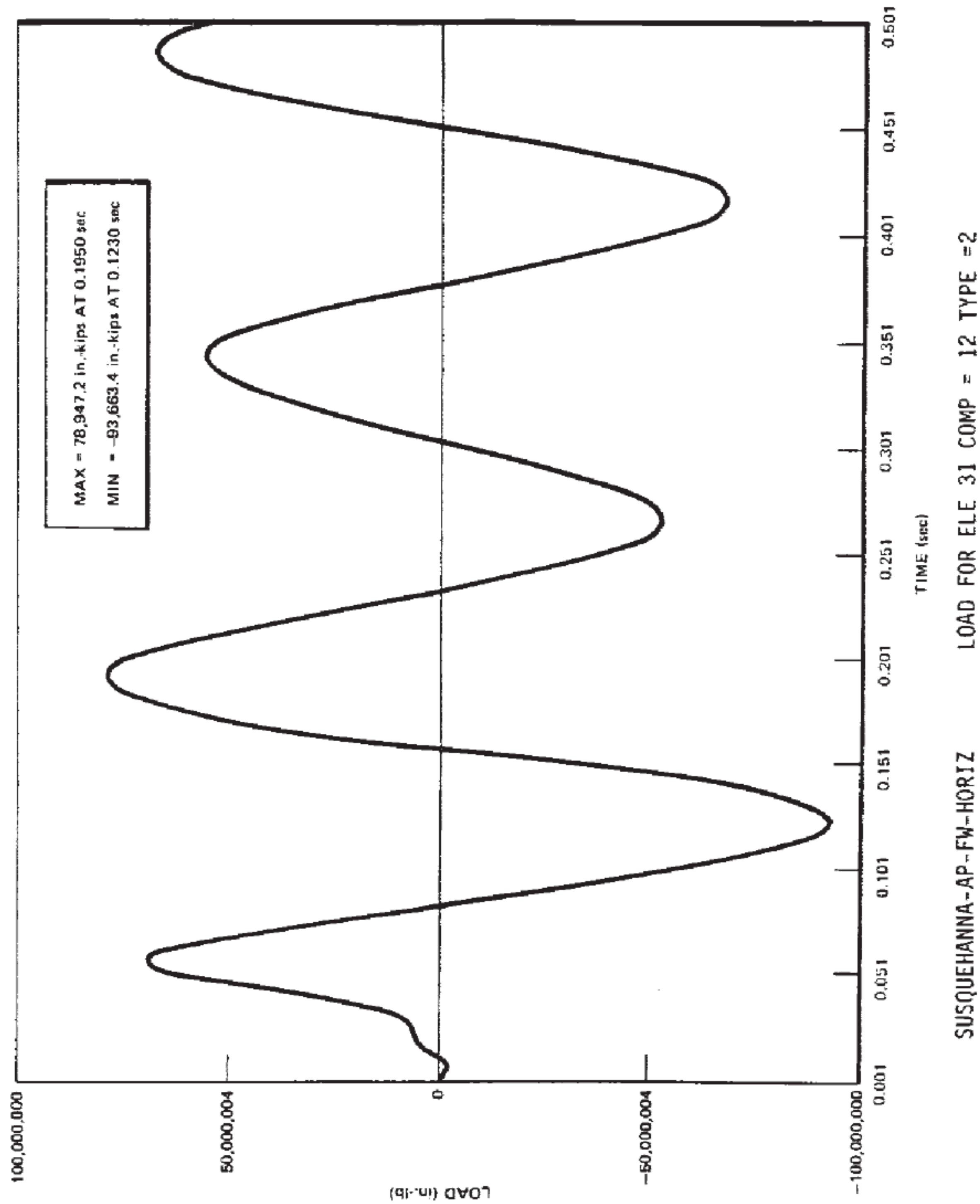
FSAR REV.65

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



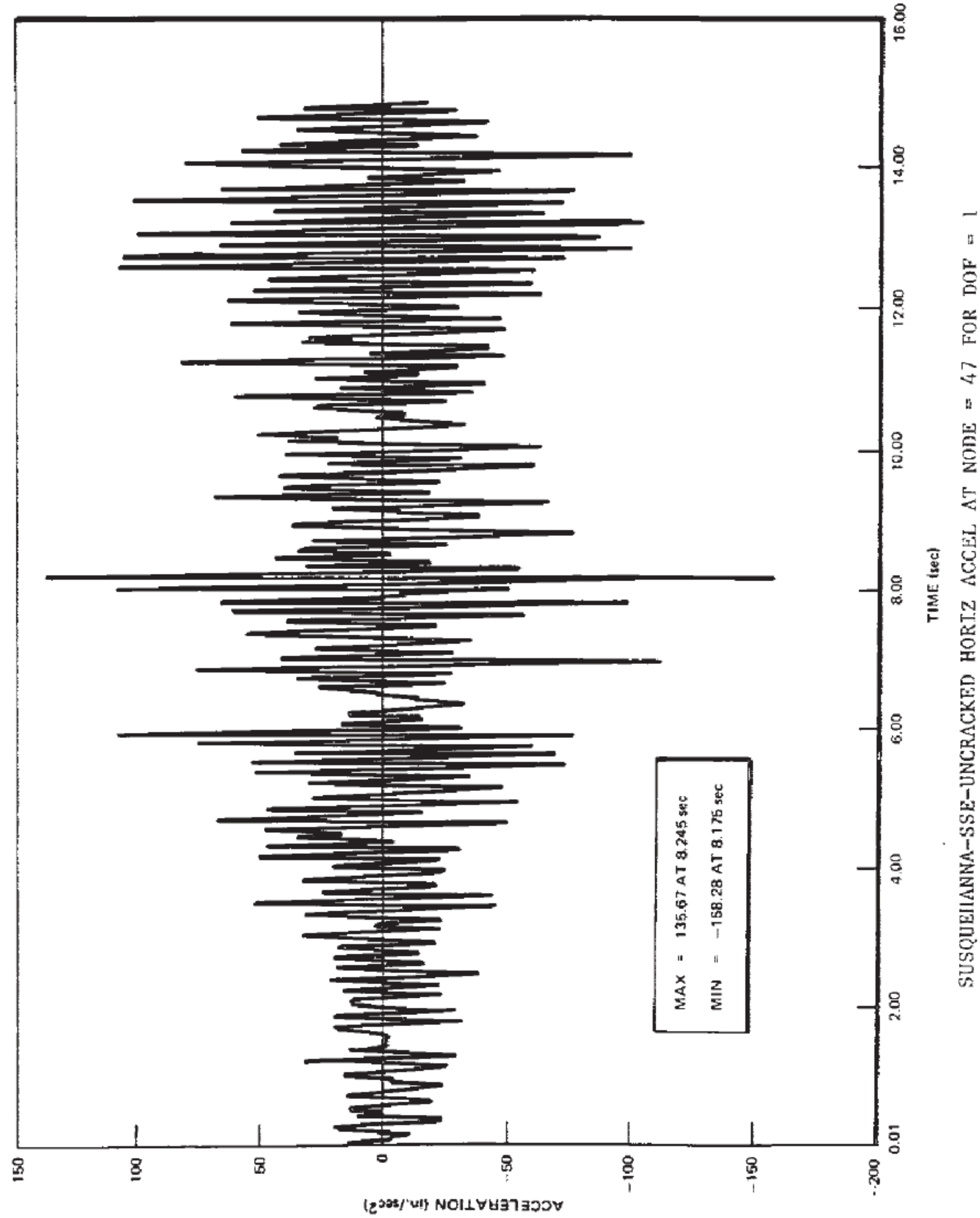
FSAR REV.65

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

ANNULUS PRESSURIZATION
(HORIZONTAL)
FEEDWATER LINE BREAK

FIGURE 110.41-32, Rev 47

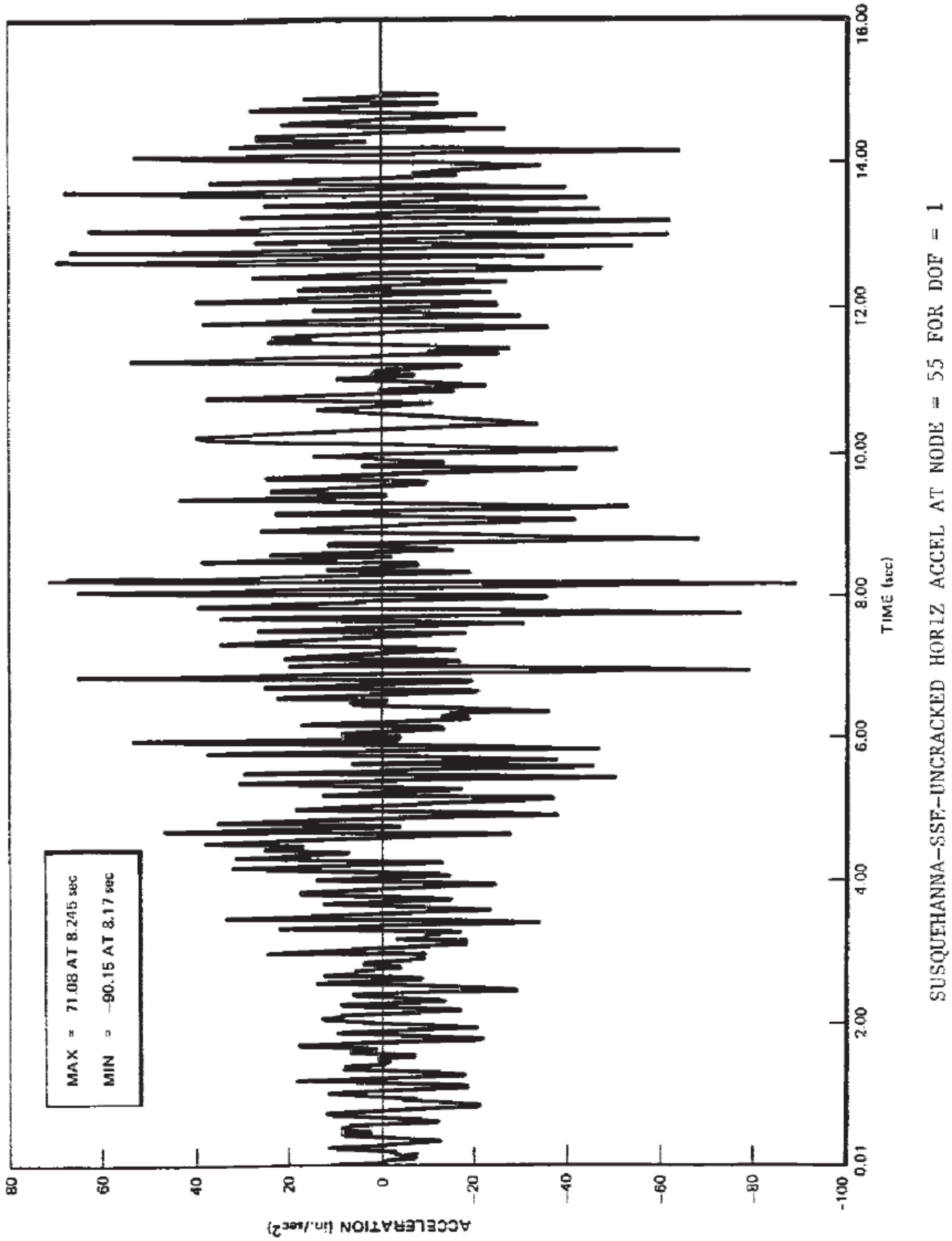
AutoCAD: Figure Fsar 110_41_32.dwg



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



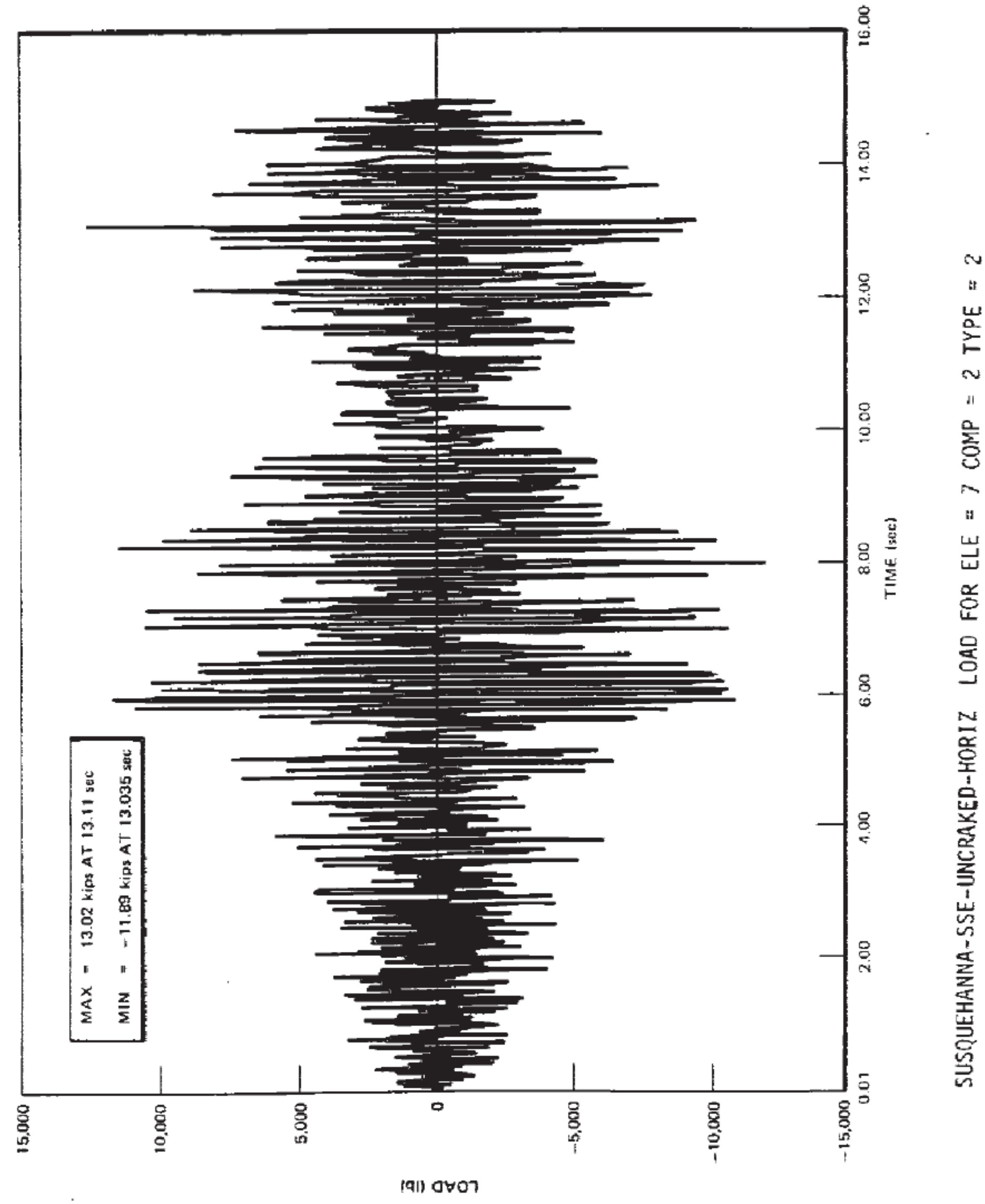
FSAR REV.65

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



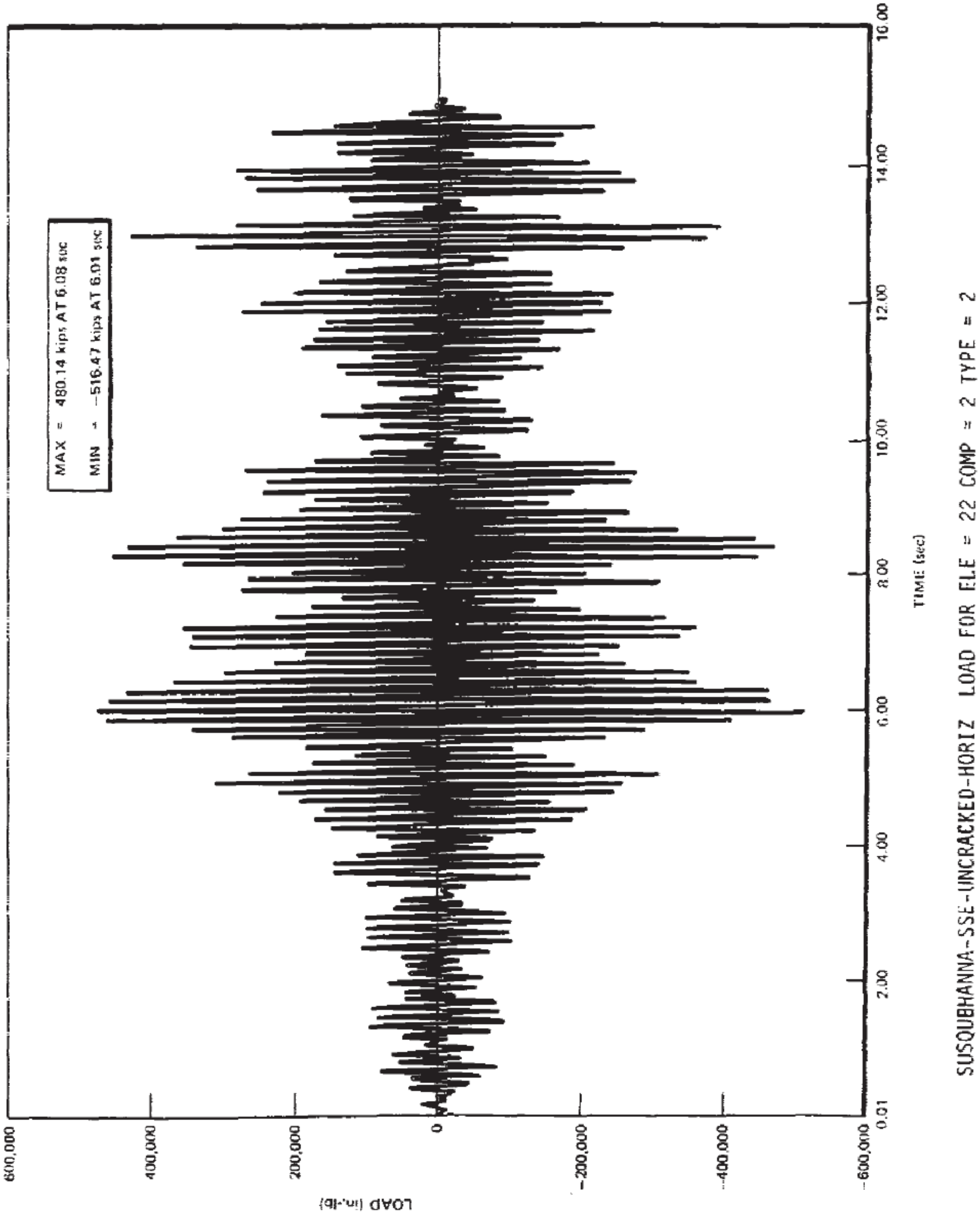
FSAR REV.65

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



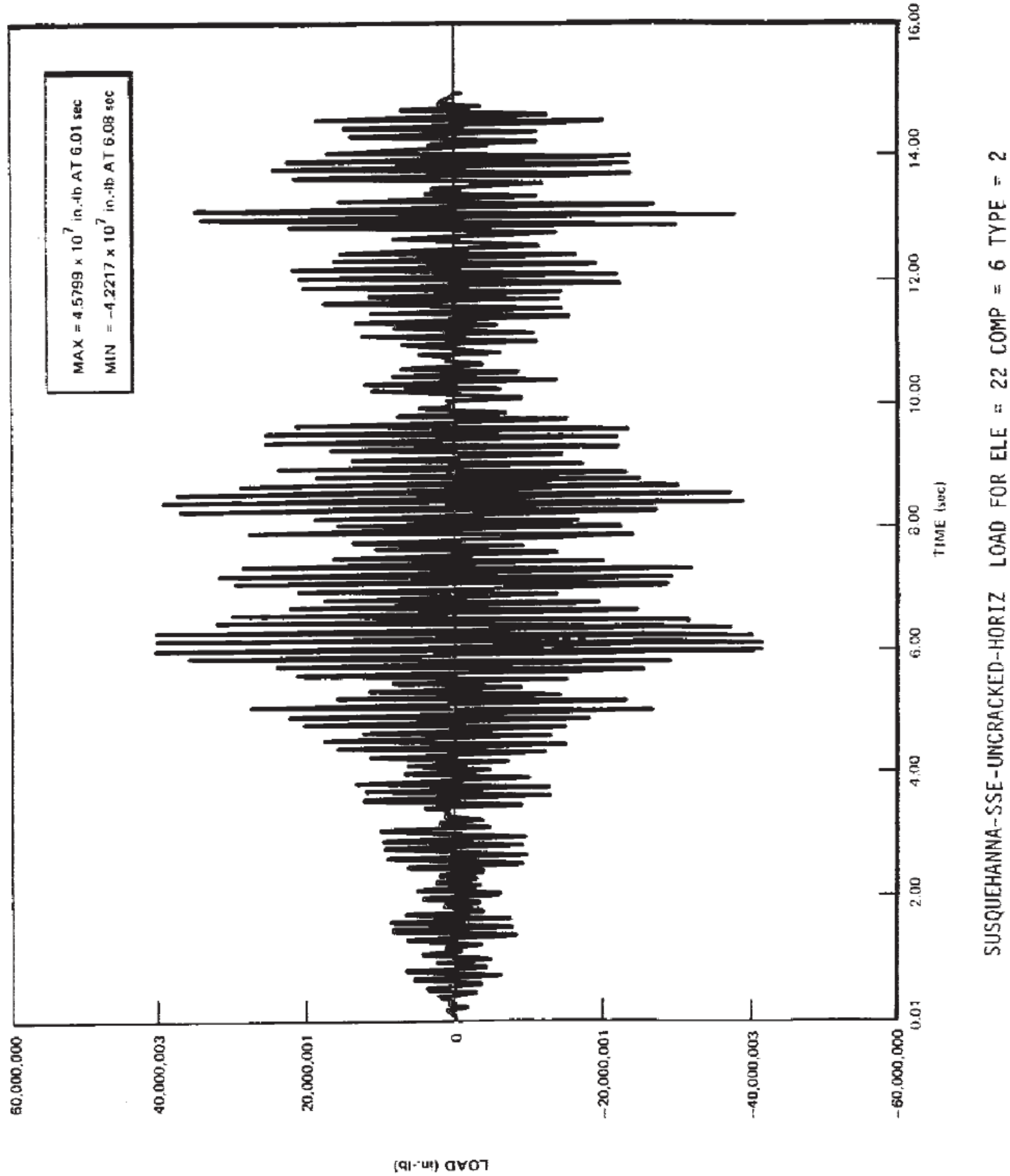
FSAR REV.65

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

RESPONSE TIME HISTORY PLOT
DUE TO HORIZONTAL SSE

FIGURE 110.41-7, Rev 47

AutoCAD: Figure Fsar 110_41_7.dwg



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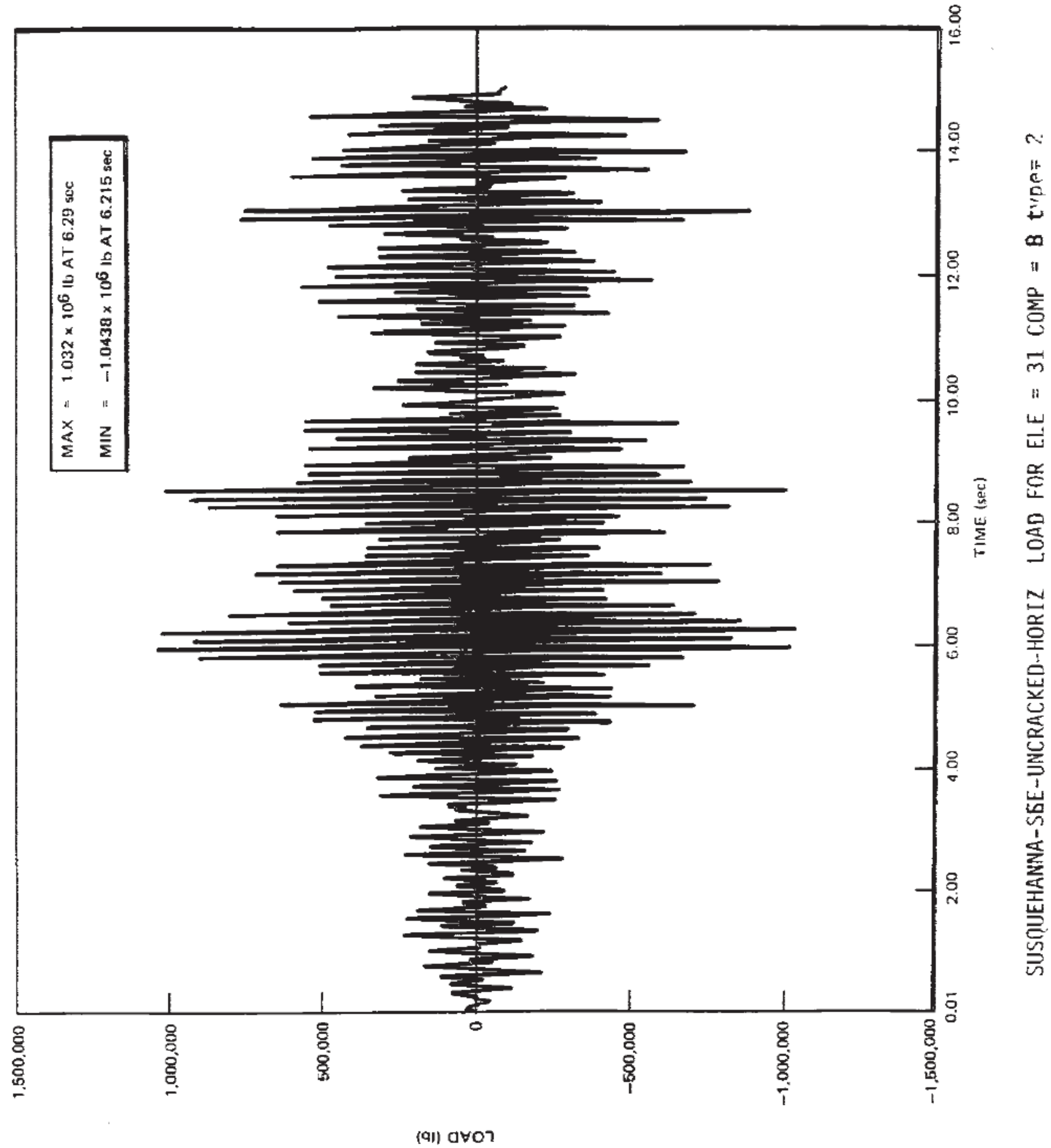
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE TIME HISTORY PLOT
DUE TO HORIZONTAL SSE

FIGURE 110.41-8, Rev 47

AutoCAD: Figure Fsar 110_41_8.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 110.41-9, Rev 47

AutoCAD: Figure Fsar 110_41_9.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_x 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_x (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.

QUESTION 112.1

The information presented in Subsection 3.6.2.1.1 concerning pipe break criteria for piping between containment isolation valves is not completely acceptable. To justify a break exclusion region in piping systems penetrating primary containment, the criteria presented in Branch Technical Position APCSB 3-1, Paragraph B.2d should be specified in addition to the information presented. It is the staff's position that one hundred percent volumetric examination of all process piping welds in this region be performed during each inspection interval.

RESPONSE:

This information is contained in revised Subsections 3.6.2.1.1 and 5.2.4.7.

QUESTION 112.2

Provide sketches showing the locations of the resulting postulated pipe ruptures, including identification of longitudinal, and circumferential breaks, structural barriers, if any, restraint locations, and the constrained direction in each restraint.

Also provide a summary of the data developed to select postulated break locations including, for each point, the calculated stress intensity, the calculated cumulative usage factor, and the calculated primary plus secondary stress range as delineated in SRP Section 3.6.2, Paragraph III.I.b.

RESPONSE:

This information is contained in revised Subsection 3.6.2.1.1 and Figure 3.6-17.

QUESTION 112.3

Provide a summary comparison of the results obtained from the use of each program in Subsection 3.9.1.2 of the FSAR with the results derived from a similar recognized and approved program or results from test problems.

RESPONSE:

Subsection 3.9.1.2 has been revised and Appendix 3.9A has been added to provide the requested information.

QUESTION 112.4

The information presented in Subsections 3.9.2.2b and 3.10 of the FSAR concerning seismic qualification of Category I mechanical and electrical equipment may not be completely acceptable. Criteria which is acceptable to the staff and is currently being implemented on all plants is outlined in the Regulatory Guide 1.100. Provide a comparison of your program with the criteria of the above mentioned Regulatory Guide.

RESPONSE:

The implementation paragraph of this regulatory guide states that the requirements of the position statements will only be applied to plants that received construction permits after November 16, 1976. The Construction Permit for Susquehanna SES was issued in November 1973 and therefore the guidelines of this regulatory guide have not been utilized in the design of this nuclear power station.

Seismic qualification of the NSSS safety related electric equipment has been conducted in accordance with the IEEE Standard 344-1971. Section 3.10a describes the complete qualification methods and procedures that have been utilized.

For an explanation of seismic qualification criteria for non-NSSS equipment, see the following subsections:

a) Mechanical Components - revised Subsection 3.9.2.2b and new Table 3.9-18;

b) Instrumentation - revised Section 3.10b;

c) Electrical Components - revised Subsection 3.10c.2.2 and revised Table 3.10c-1.

QUESTION 112.5

Explain in detail how the loads discussed in Subsections 3.9.2.2a, 3.9.2.2b.2, 3.9.3.1 and Table 3.9-2 of the FSAR are combined for various plant conditions. Table 3.9-2(d) of the FSAR shows that for Emergency and Faulted conditions, the peak loads are combined using the square root of the sum of the squares method. It is the staff's position that these peak loads be combined by absolute sum unless acceptable justification is provided for an alternative method of combination. Provide such justification, or alternatively evaluate the effect of combining responses to dynamic loads by absolute sum on components and supports.

RESPONSE:

For the method of Load Combination for emergency and faulted plant conditions for non-NSSS ASME III Class 1, 2, and 3 components, see Table 3.9-6 and revised Subsections 3.9.3.1.19 and 3.9.2.2b.2.

The combination of loads discussed in Subsections 3.9.2.2a (NSSS components) and 3.9.3 is detailed in Table 3.9-2. Table 3.9-2 is the major part of this section and presents the loading combinations, analytical methods (by reference or example) and also the calculated stress or other design values for the most critical areas in the design of each component. These calculated values are also compared to the applicable code allowables. The combining of two or more peak dynamic loads by the SRSS method in Table 3.9-2(d) is a generic issue and has been recently addressed and resolved by GE and Mark II owners group (see NUREG 0484). The generic resolution applies to Susquehanna SES.

Question 112.6

Recent analyses 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. For the purpose of this request for additional information, the reactor system components that require reassessment shall include:

(1) Reactor Pressure Vessel
(2) Fuel Assemblies, including Grid Structures
(3) Control Rod Drives
(4) ECCS Piping that is attached to the Primary Coolant Piping
(5) Primary Coolant Piping
(6) Reactor Vessel and Pump Supports
(7) Reactor Internals
(8) Biological Shield Wall and Neutron Shield Tank (where applicable)
(9) Pump Compartment Wall

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 reactor cavity structure.

(1) Provide arrangement drawings of the reactor vessel, and pump support systems 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 thermal 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 these systems/components in combinations with all external loadings including safe shutdown earthquake loads, asymmetric cavity pressurization for both the reactor vessel, and recirculation pump which might result from the required postulate. 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 required by 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 of the load transmitted to the backup structures to which these systems are attached.

(5) For all analysis 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) Provide an estimate of the total amount of permanent deformation sustained by the fuel spacer grids. Include a description of the impact testing that was performed in support of your estimate. Address the effects of operating temperatures, secondary impacts, and irradiated material properties (strength and ductility) on the amount of predicted deformation. Demonstrate that the fuel will remain coolable for all predicted geometries.

---

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

(7) Demonstrate that active components will perform their safety function when subjected to the postulated loads resulting from a pipe break in the reactor coolant system.

(8) Demonstrate functionability of any essential piping where service level B limits are exceeded.

RESPONSE:

In accordance with NRC letter from Mr. Olan D. Parr (NRC) to Mr. N. W. Curtis (PP&L) dated February 8, 1979, Question 110.32 replaces Question 112.6, therefore it is no longer necessary to respond to Question 112.6.

QUESTION 112.7

Most of the operating BWR plants have reported finding radial cracks on the reactor vessel feedwater nozzle and the CRD return line. Describe what design modifications will be made to eliminate this problem. In addition, provide a description of the analyses that will be performed to demonstrate the adequacy of the reactor vessel feedwater nozzle and CRD return line to withstand the imposed service condition without the cracking experienced in the operating plants.

RESPONSE:

The mechanisms which have caused cracking in operating BWRs are understood. A summary discussion of the previously observed problems and the solutions incorporated in the Susquehanna design is presented in the following.

A detailed evaluation of the problems of the feedwater nozzle and sparger is presented in NEDE-21821 "BWR Feedwater Nozzle/Sparger Final Report" March 1978. The solution of the feedwater nozzle and sparger cracking problems involves several elements, including material selection and processing, nozzle clad removal, and thermal sleeve and sparger redesign. The following summarizes the problems that have occurred in the nozzle and sparger and shows the solution that eliminates each problem:

Problem	Cause	Fix
Sparger arm cracks	Mechanical fatigue	Eliminate/minimize clearance between thermal sleeve and safe end.
	Thermal fatigue	Eliminate low flow stratification by use of top-mounted elbows.
Flow hole cracks	Thermal fatigue	Eliminates separation by use of converging nozzles.
Nozzle cracks	Thermal	Eliminate clad, control leakage, protect nozzle with multiple sleeves.

The sparger vibration has been attributed to a self-excitation caused by instability of leakage flow through the annular clearance between the thermal sleeve and safe end. Tests have shown that the vibration is eliminated if the clearance is reduced sufficiently or sealed. The solution which has been selected uses a two-stage piston ring seal mounted in the thermal sleeve in conjunction with an interference fit between the sleeve and safe end. This feature is also an essential part of the solution of the nozzle cracking problem, and is described later in more detail. Freedom from vibration over a range of conditions has been demonstrated by the tests reported in NEDE-21821, Section 4.

Sparger arm cracking has also been caused by thermal fatigue, both at the flow holes and adjacent to the tee connection with the thermal sleeve. In both cases, excessive cyclic thermal stresses are caused by the exposure of material in a constrained structure to an unstable boundary between cold feedwater fluid and hot reactor fluid. At low feedwater flow, the presence of exit flow holes at the midplane of the sparger allowed the sparger to be only partially filled with cold fluid. This caused a temperature gradient from the top of the sparger to the bottom, with associated bending stresses which changed directly with changes in the flow gradient. Relocation of the exit flow holes at the top of the sparger allows complete filling of the sparger with the feedwater fluid even at low flow, producing a more stable and homogeneous temperature distribution. As shown by the data reported in NEDE-21821, Section 4.3, stratification has been eliminated over the range of operating flows.

Flow hole cracks occurred partly because the surface of the hole was constrained by the in-plane stiffness of the surrounding sparger material when exposed to the exit flow to reactor coolant gradients, and partly because the gradients themselves were unstable. The instability of the gradients resulted from changing location of the operation point between the cold exit flow and the warmer boundary layer produced by heating of the sparger by reactor fluid.

The result was a high-cycle thermal stress around the edge of the hole. This condition is eliminated by the exit flow elbows which have a long enough exit throat to stabilize the flow separation.

Also, the thermal stress produced by a given gradient is much less with the exit hole in a cylindrical tube, rather than in what previously would behave locally more as a flat plate. Testing, as reported in NEDE-21821, Section 4.3, has shown that the high frequency cycling is eliminated by the new design.

In order to allow for removal of the sparger, it is necessary to provide a sealed joint between the nozzle safe end and the thermal sleeve. This seal is achieved by use of a metal piston ring backed up with a coil spring expander. Even if the piston ring seal was leaktight when initially installed, its long-term sealing ability is unknown. The effects of wear and corrosion on the mating safe end surface would eventually cause leakage to increase to the point where nozzle cracking would initiate. The rate of deterioration is unpredictable but is expected to be short relative to the life of the pressure vessel. To provide protection against seal failure resulting in nozzle cracking, the second piston ring and the added thermal sleeves have been incorporated in the new design. It has been demonstrated by test that the triple thermal sleeve arrangement prevents the leakage flow causing nozzle cracking. This is the result of the concentric sleeve arrangement channeling leakage away from the nozzle end and the fact that the second seal is exposed to very low driving pressures, making leakage past it very small.

As was mentioned earlier, the cracking of the feedwater nozzles is a two-part process. The crack initiation mechanism as discussed above is the result of self-initiated thermal cycling. If this were the only mechanism present, the cracks would initiate, grow to a depth of approximately 0.25 inch, and arrest. This degree of cracking could be tolerated, but unfortunately there is another mechanism which supports crack growth. This mechanism is the system induced transients, primarily the startup/shutdown transients. The triple thermal sleeve arrangement also assists in this area because, even with the piston rings leaking, the heat transfer coefficient between the feedwater and the nozzle are reduced to the point where the thermal stresses in the nozzle are not high enough to cause significant crack growth. Analysis presented in NEDE-21821, Section 4.6, demonstrates this benefit and the benefit of using unclad nozzles.

The cracking of the CRD return nozzles is caused by a mechanism which is very similar to that which caused cracking in the feedwater nozzles - thermally induced fatigue.

The CRD return flow is always at a low temperature. The low flow rate is also low and as the fluid passes through the nozzle it mixes with the hot (540°F) reactor coolant. This mixing is turbulent and results in alternating hot and cold cycling on the nozzle wall. The result is high cycle fatigue which initiates cracking. This mechanism has been demonstrated by test. Tests have also demonstrated that lower-frequency thermal cycles occur in a stagnant CRD return line nozzle.

The fix for this problem is the elimination of the CRD return flow to the vessel nozzle. It has been shown that the CRD system will operate satisfactorily with the return line cut and capped. This has been demonstrated by tests at Peach Bottom, Fitzpatrick, and other operating BWR's.

Stress analyses in keeping with the requirements of the ASME Code, Section III, will be performed to demonstrate the adequacy of the reactor vessel feedwater nozzle (and sparger) and the CRD return line nozzle cap.

QUESTION 112.8

Provide the criteria used in the design of supports for all ASME Class 1, 2 and 3 active pumps and valves to assure that the supports do not deform to the extend that operability of the supported components will not be impaired.

RESPONSE:

The recirculation piping suspension system to be supplied by GE will use three types of component supports. These are hangers, struts and snubbers to support the recirculation pumps and valves.

The design of the hanger supports which carry the load caused by dead weight only has already been completed and is in accordance with the rules and regulations of ANSI Code B31.7. The scope of supply responsibility of the component supports such as struts and snubbers which carry the dynamic loads is presently in the process of being negotiated between the customer and GE. In the event GE is responsible for supplying struts and snubbers, the design will comply with the requirements of ASME Code Section III, Subsection NF. All the component supports are designed, fabricated and assembled so that they cannot become disengaged by the movement of the supported pipe or equipment while performing its function during the various operating conditions of the plant. The design load on each of these component supports is identified as follows:

(a) Hangers

The design load on hangers is the load caused by dead weight. The hangers are calibrated to ensure that they support the design load at both their hot and cold load settings. Hangers provide a specified down travel and up travel in excess of the specified thermal movement.

(b) Struts

The design load on struts includes those loads caused by dead weight. Thermal expansion, primary seismic loads, i.e. operating basis earthquake (OBE) and safe shutdown earthquake (SSE) and system anchor displacements, etc.

(c) Snubbers

The design load on snubbers includes those loads caused by seismic forces (OBE and SSE) and system anchor displacements, etc.

The analyses that are used for the design of these component supports to ensure that all such supports will not deform to the extent that would impair the pressure-retaining integrity of the supported components under normal, upset, emergency and faulted plant conditions, can essentially be divided into three parts as given below:

(a) Piping analysis to determine design loads on component supports.

The piping analysis is performed with GE SAP4 program. SAP4 is general Structural Analysis Program for static and dynamic analysis of linear elastic complex structures. The finite element displacement method is used to solve for the displacements and loads and compute the stresses of each element of the structure. The loads resulting from thermal expansion, dead weight, primary seismic loading (OBE and SSE), and system anchor displacements are first determined individually and then combined under normal, upset, emergency and faulted plant conditions to determine the design load on the respective component support. Piping supports are then designed by the load rating method and, in general, the load combinations for the various plant operating conditions correspond to those used to design the supported pipe.

Design transient cyclic data are not applicable to piping supports as no fatigue evaluation is necessary to meet Code requirements.

(b) Selection from the vendor data.

After determining the design load by piping analysis, component supports are selected from the vendor data that indicates loads are equal to or below the load rating of the components.

(c) Analysis and/or tests to demonstrate acceptability

The vendor performs analyses and/or tests to demonstrate acceptability for his load rating data on component supports. Also, the vendor performs analyses and/or tests to demonstrate that all such component supports will not deform under faulted plant conditions to the extent that would impair the required operability of the supported components to perform a safety function for safe shutdown of the plant.

For non-NSSS supports, please see Subsection 3.9.3.4.6 for a discussion of design criteria.

QUESTION 112.9

Provide the following information in the FSAR:

(1) A tabulation of snubbers utilized in your facility as supports for safety related systems and components including:

   (a) System identification and location
   (b) Type (hydraulic, mechanical)
   (c) Fabricator and rated load capacity
   (d) Function (shock or vibration arrestor, dual purpose)

(2) A summary of the contents of the snubber design specifications.

(3) A description of snubber suppliers performance qualification tests and load tests.

(4) A summary of system and component structural analyses showing:

   (a) Structural analysis model.

   (b) Description of the characterization of snubber mechanical properties used in the structural analysis including considerations such as (i) differences in tension and compression spring rates, (ii) effect of entrapped air and temperature on fluid properties, (iii) other factors affecting snubber characteristics.

   (c) List load conditions and transients analyzed.

   (d) Maximum snubber loads, corresponding piping or component stresses.

   (e) Comparison of computed loads and stresses from (d) above with rated snubber load and component stress intensity limits.

RESPONSE:

See revised Subsection 3.9.3.4.6 for this information.

QUESTION 112.10

To aid us in our licensing review of Susquehanna, you are requested to provide the following information to us:

1. Describe those actions being taken by you to preclude the occurrence of cracking such as described in 1E Bulletin 80-07.

2. Provide a commitment to adopt whatever long-term solution is approved.

3. If you anticipate receiving an Operating License before a long-term solution is agreed upon, describe any short-term actions which you will take to prevent or detect excessive cracking.

   Provide a rationale as to why these actions are sufficient to justify plant operation until a long-term solution is found.

RESPONSE:

1. Those step(s) to be taken to preclude the occurrence of cracking of the jet pump hold-down beams such as described in 1E Bulletin 80-07 are described in PLA-670 N. W. Curtis to B. J. Youngblood, dated March 25, 1981.

2. It is anticipated that there will be alternative long term solutions developed and approved. These approved solutions will be evaluated at the conclusion of current investigative activities and a commitment made at that time.

3. If a long term solution is not agreed upon prior to receipt of an operating license, the following short term actions will be taken to prevent or detect excessive cracking:

   a. A procedure will be instituted to monitor jet pump loop flow/recirculation pump speed ratios daily. A deviation from the normal range may indicate a problem wherein individual jet pump performance data will be used to determine if a problem exists.

   b. A performance monitoring program will be established to obtain and periodically update a "normal" operation data base. Operating data will then be compared to the data base at least weekly to provide an early indication of

potential problems. Calibration checks of the instrumentation used in this program will be performed at least every 18 months or more frequently should there be a tendency for significant drift over the 18 month period.

c. A review of the jet pump operability technical specifications will be performed with the objective of making them more responsive to Operating experience. The recommended Technical Specification includes daily monitoring of Recirculation Pumps Flow/Speed Ratio and Jet Pump Loop Flow/Pump Speed Ratio to detect potential problems.

QUESTION 121.1

Provide a sketch of the Susquehanna reactor vessels (including dimensions) showing all longitudinal and circumferential welds, and all forgings and/or plates. Welds should be identified by a shop control number (such as a procedure qualification number), the heat of filler metal, type and batch of flux, and the welding process. Each forging and/or plate should be identified by a heat number and material specification.

RESPONSE:

Unit 1:

Vessel Beltline Material Identification = Susquehanna SES Unit 1.

A. Lower Shell Course (CBIN Dwg R-1, Rev. 4, Contract No. 683331)

1. Plates

PC #	ID #	MELT #	SLAB #
21	1	CB5083	1
21	2	C0770	2
21	3	C0814	2

2. Welds

The vertical welds and girth weld for this shell course were completed in the "field." Records were not kept on which of the electrodes, identified by heat and lot numbers, were used in the weld-up of the specific field welds of this shell assembly. It is assumed that any of the SMAW electrodes - type 8018 released for field welding could have been used on any or all of the associated seams in the beltline region.

Electrodes released for the field welding of these plates are as follows:

Type	Lot No.	Heat No.
SMAW Electrode Type 8018	B504B27AE	401S0371
SMAW Electrode Type 8018	629616	L320A27AG
SMAW Electrode Type 8018	402K9171	K315A27AE
SMAW Electrode Type 8018	411L3071	L311A27AE
SMAW Electrode Type 8018	494K2351	L307A27AD
SMAW Electrode Type 8018	C115A27A	402C4371
SMAW Electrode Type 8018	J417B27A	412P3611

B. Lower Intermediate Shell Course

1. Plates

PC #	ID #	MELT #	SLAB #
22	1	C0803	1
22	2	C0776	1
22	3	C2433	1

2. Welds

The vertical welds and the girth weld for this shell course were completed in the "field." Records were not kept on which of the electrodes, identified by heat and lot numbers, were used in the weld-up of the specific field welds of the shell assembly. It is assumed that any of the SMAW electrodes (type 8018) released for field welding could have been used on any or all of the associated seams in the beltline region.

Electrodes released for the field welding of these plates are as follows:

Type	Lot No.	Heat No.
SMAW Electrode Type 8018	B504B27AE	401S0371
SMAW Electrode Type 8018	629616	L320A27AG
SMAW Electrode Type 8018	402K9171	K315A27AE
SMAW Electrode Type 8018	411L3071	L311A27AF
SMAW Electrode Type 8018	494K2351	L307A27AD
SMAW Electrode Type 8018	C115A27A	402C4371
SMAW Electrode Type 8018	J417B27AF	412P3611

Unit 2:

Vessel Beltline Material Identification = Susquehanna SES Unit 2

A. Lower Shell Course

1. Plates

PC ID#	MELT #	SLAB #
21-1	6C956	1-1
21-2	6C980	1-1
21-3	6C1053	1-1

2. Welds

The vertical welds and girth weld for this shell course were completed in the "field." Records were not kept on which of the electrodes, identified by heat and lot numbers, were used in the weld-up of the specific field welds of this shell assembly. It is assumed that any of the SMAW electrodes - type 8018 released for field welding could have been used on any or all of the associated seams in the beltline region.

Electrodes released for the field welding of these plates are as follows:

Type	Lot No.	Heat No.
SMAW Electrode Type 8018	B504B27AE	401S0371
SMAW Electrode Type 8018	629616	L320A27AG
SMAW Electrode Type 8018	402K9171	K315A27AE
SMAW Electrode Type 8018	411L3071	L311A27AF
SMAW Electrode Type 8018	494K2351	L307A27AD
SMAW Electrode Type 8018	C115A27A	402C4371
SMAW Electrode Type 8018	J417B27A	412P3611
SMAW Electrode Type 8018	C109A27A	09M057
SMAW Electrode Type 8018	E204A27A	624263
SMAW Electrode Type 8018	F414B27AF	659N315

B. Lower Intermediate Shell Course

1. Plates

PC ID#	MELT #	SLAB #
22-1	C2421	3
22-2	C2929	1
22-3	C2433	2

2. Welds

The vertical welds and girth weld for this shell course were completed in the "field." Records were not kept on which of the electrodes, identified by heat and lot numbers, were used in the weld-up of the specific field welds of this shell assembly. It is assumed that any of the SMAW electrodes (type 8018) released for field welding could have been used on any or all of the associated seams in the beltline region.

Electrodes released for the field welding of these plates are as follows:

Type	Lot No.	Heat No.
SMAW Electrode Type 8018	B504B27AE	401S0371
SMAW Electrode Type 8018	629616	L320A27AG
SMAW Electrode Type 8018	402K9171	K315A27AE
SMAW Electrode Type 8018	411L3071	L311A27AF
SMAW Electrode Type 8018	494K2351	L307A27AD
SMAW Electrode Type 8018	C115A27A	402C4371
SMAW Electrode Type 8018	J417B27AF	412P3611
SMAW Electrode Type 8018	C109A27A	09M057
SMAW Electrode Type 8018	E204A27A	624263
SMAW Electrode Type 8018	F414B27AF	659N315

Security-Related Information
Figure Withheld Under 10 CFR 2.390

FSAR REV. 46, 06/93

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 AND 2 FINAL SAFETY ANALYSIS REPORT
UNIT 1 REACTOR VESSEL FSAR FIGURE 121.1-1

PP&L DRAWING

Security-Related Information
Figure Withheld Under 10 CFR 2.390

FSAR REV. 46, 06/93

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

UNIT 2 REACTOR VESSEL

FSAR FIGURE 121.1-2

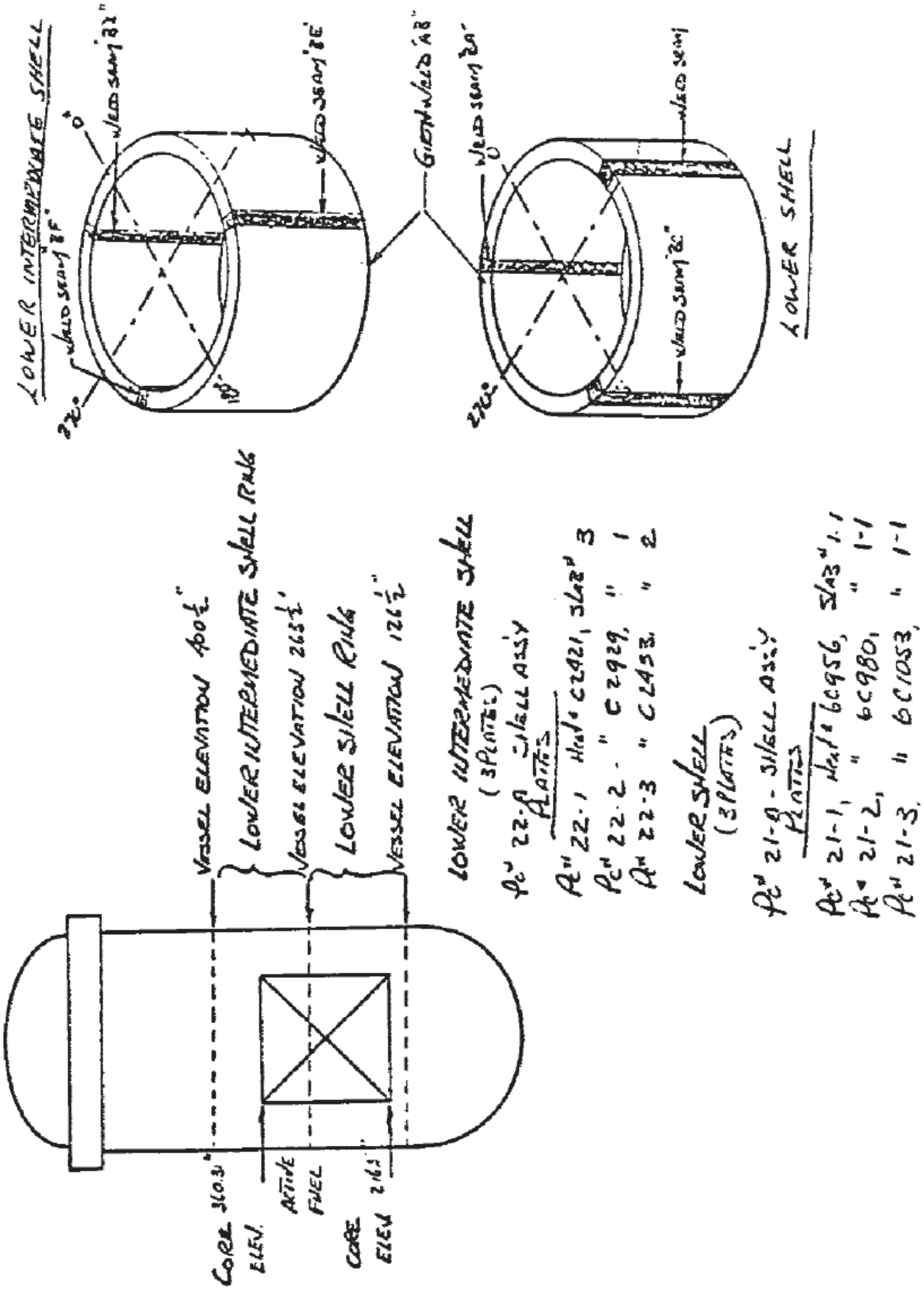
PP&L DRAWING

Security-Related Information
Figure Withheld Under 10 CFR 2.390

FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
UNIT 1 REACTOR VESSEL FIGURE 121.1-1, Rev 47

AutoCAD: Figure Fsar 121_1_1.dwg



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
UNIT 2 REACTOR VESSEL FIGURE 121.1-2, Rev 47

AutoCAD: Figure Fsar 121_2-1.dwg

QUESTION 121.2

Supply the following information for each of the ferritic materials of the pressure retaining components in the reactor coolant pressure boundary of the Susquehanna plant:

(1) The unirradiated mechanical properties as required by the testing programs in Section III of the ASME Code and Appendix G of 10 CFR Part 50 (test results to be presented should include Charpy V-notch, dropweight, lateral expansion, tensile, upper shelf energy, T_{NDT} and RT_{NDT}). If any of these properties have not been determined by a test method required by Appendix G of 10 CFR Part 50, state the actual test procedure used and/or the method used to estimate the test result together with a complete technical justification of the procedure used.

(2) Identify the material(s) in the reactor coolant pressure boundary that will limit the pressure-temperature operating curves at the beginning-of-life.

For each reactor vessel beltline weld, plate or forging provide the following information:

(3) The chemical composition (particularly the Cu, P and S content) and the maximum end-of-life fluence.

(4) The relationship used to predict the shift in RT_{NDT} and percent decrease in upper shelf energy as a function of neutron fluence.

(5) Identify the material(s) in the reactor coolant pressure boundary that will limit the pressure-temperature operating curves and the end-of-life.

RESPONSE:

(1) The Susquehanna SES Unit No. 1 reactor pressure vessel was ordered prior to the issuance of Appendix G 10 CFR Part 50. The ferritic material for the pressure boundary was qualified by dropweight testing for the shell plate material and both dropweight and Charpy V-notch testing for the weld material. The test results, along with the specific requirements prevailing at the time of vessel ordering are summarized in the tables which follow.

Impact Properties of SA533, Grade B, Class 1 Plate Material in Beltline Region		
Location	**Plates**	**Drop Wt. NDTT (°F)**
Lower Shell Course	21-1 21-2 21-3	-10°F -30°F -30°F
Lower Intermediate Shell Course	22-1 22-2 22-3	-10°F -10°F -50°F

Impact Properties of Weld Materials Employed In the Beltline Region				
Weld Material Identification	**Charpy "V" (Ft/lb)**	**Test Temp. °F**	**Required**	**Drop Wt. NDTT °F**
Lot #B504B27AE Ht #401S0371	57, 58, 62	-20	30 ft/lbs at 10°F	-80
Lot #629616 Ht #L320A27AG	51, 52	+10	"	-70
Lot #402K9171 Ht #K315A27AE	58, 58	+10	"	-70
Lot #411L3071 Ht #L311A27AF	51, 67	+10	"	-70
Lot #494K2351 Ht #L307A27AD	87, 96	+10	"	-80
Lot #C115A27A Ht #402C4371	82, 84, 92	+10	"	N/R
Lot #J417B27AF Ht #412P3611	52, 65, 69	-20	"	-80
Note: N/R - Not Reported				

Unirradiated fracture toughness properties (T_{NDT}, RT_{NDT} and upper shelf fracture energy) as required by Appendix G. 10 CFR Part 50, identifying the limiting material in the reactor vessel beltline region.

The Susquehanna SES Unit No. 2 reactor pressure vessel was ordered prior to the issuance of Appendix G, 10 CFR Part 50. The ferritic material for the pressure boundary was qualified by dropweight testing for the shell plate material and both

dropweight and Charpy V-notch testing for the weld material. The test results, along with the specific requirements prevailing at the time of vessel ordering are summarized in the tables which follow.

Impact Properties of SA533, Grade B, Class 1 Plate Material in Beltline Region

Location	Plates	Drop Wt. NDTT (°F)
Lower Shell Ring	21-1 21-2 21-3	-20°F -20°F +10°F
Lower Intermediate Shell Ring	22-1 22-2 22-3	-10°F -20°F -30°F

Impact Properties of Weld Materials Employed In the Beltline Region

Weld Material Identification	Charpy "V" (Ft/lb)	Test Temp. °F	Required	Drop Wt. NDTT °F
Lot #B504B27AE Ht #401S0371	57, 58, 62	-20	30 ft/lbs at 10°F	-80
Lot #629616 Ht #L320A27AG	51, 52	+10	"	-70
Lot #402K9171 Ht #K315A27AE	58, 58	+10	"	-70
Lot #411L3071 Ht #L311A27AF	51, 67	+10	"	-70
Lot #494K2351 Ht #L307A27AD	87, 96	+10	"	-80
Lot #C115A27A Ht #402C4371	82, 84, 92	+10	"	N/R
Lot #J417B27AF Ht #412P3611	52, 65, 69	-20	"	-80
Lot #C109A27A Ht #09MO57	43, 43, 44	+10	"	N/R
Lot #E204A27A Ht #624263	26, 38, 42, 50, 76	-20	"	-70
Lot #F414B27AF Ht #659N315	74, 76, 77	-10	"	-80

Note : N/R - Not Reported

(2) Intermediate shell plate 23-3, Heat Slab No. C1232-2 is initially limiting for Susquehanna Unit 1. Core beltline plates 22-1, 22-2, & 22-3, Heat & Slab Nos. 6C956-1-1, 6C980-1-1, & 6C1053-1-1 respectively are initially limiting for Susquehanna Unit 2 with respect to pressure-temperature operating curves at the beginning-of-life. The feedwater nozzles will also be limiting at lower pressures as indicated on Figures 5.3-4 and 5.3-5 for Units 1 & 2.

(3) Vessel Plate Material Susquehanna SES Unit 1

	C	Mn	P	S	Cu	Si	Ni	Mo
	(Values are shown in percent)							
Lower Shell								
PC 21-1, Melt #B5083, Slab #1	.21	1.27	.010	.019	.14	.25	.48	.51
PC 21-2, Melt #CO770, Slab #2	.22	1.23	.008	.016	.14	.19	.50	.49
PC 21-3, Melt #CO814, Slab #2	.20	1.36	.011	.016	.13	.26	.51	.51
Lower Intermediate Shell								
PC 22-1, Melt #CO803, Slab #1	.21	1.30	.009	.019	.09	.24	.53	.52
PC 22-2, Melt #CO776, Slab #1	.22	1.34	.010	.010	.12	.27	.48	.48
PC 22-3, Melt #C2433, Slab #1	.18	1.30	.009	.015	.10	.23	.63	.57

	C	Mn	P	S	Cu	Si	Ni	Mo	Cr	Vn
	(Values are shown in percent)									
Weld Material - Unit 1										
Type SMAW **Electrode 8018** **Lot # and Heat #**										
Lot #B504B27AE Ht #401S0371	.05	1.18	.013	.012	.03	.37	1.04	.56	.03	.02
Lot #629616 Ht #L320A27AG	.05	1.17	.015	.018	.04	.44	.99	.55	.05	.02

	C	Mn	P	S	Cu	Si	Ni	Mo	Cr	Vn
Lot #402K9171 Ht #K315A27AE	06	1.15	.015	.016	.03	.36	.98	.53	.05	.02
Lot #411L3071 Ht #L311A27AF	.05	1.20	.016	.019	.03	.46	.93	.50	.04	.02
Lot #494K2351 Ht #L307A27AD	.05	1.18	.015	.017	.04	.37	1.1 0	.57	.04	.02
Lot #C115A27A Ht #402C4371	.033	1.22	.009	.014	.02	.49	.92	.57	N/R	N/R
Lot #J417B27AF Ht #412P3611	.07	1.10	.016	.019	.03	.36	.93	.47	.03	.02
Note: N/R - Not Reported										

Vessel Plate Material - Unit 2

	C	Mn	P	S	Cu	Si	Ni	Mo
	(Values are shown in percent)							
Lower Shell								
PC 21-1, Melt #6C956, Slab #1-1	.18	1.43	.012	.006	.11	.22	.55	.52
PC 21-2, Melt #6C980, Slab #1-1	.19	1.35	.011	.006	.10	.22	.56	.51
PC 21-3, Melt #6C1053, Slab #1-1	.18	1.37	.012	.010	.10	.30	.58	.50
Lower Intermediate Shell								
PC 22-1, Melt #C2421, Slab #3	.19	1.22	.007	.011	.13	.25	.68	.55
PC 22-2, Melt #C2929, Slab #1	.20	1.27	.006	.015	.13	.22	.64	.56
PC 22-3, Melt #C2433, Slab #2	.18	1.30	.009	.015	.10	.23	.63	.57

	C	Mn	P	S	Cu	Si	Ni	Mo	Cr	Vn
	(Values are shown in percent)									
Weld Material - Unit 2										
Type SMAW **Electrode 8018** ***Lot # and Heat #***										
Lot #B504B27AE Ht #401S0371	.05	1.18	.013	.012	.03	.37	1.04	.56	.03	.02
Lot #629616 Ht #L32OA27AG	.05	1.17	.015	.018	.04	.44	.99	.55	.05	.02
Lot #402K9171 Ht #K315A27AE	.06	1.15	.015	.016	.03	.36	.98	.53	.05	.02
Lot #411L3071 Ht #L311A27AF	.05	1.20	.016	.019	.03	.46	.93	.50	.04	.02
Lot #494K2351 Ht #L307A27AD	.05	1.18	.015	.017	.04	.37	1.10	.57	.04	.02
Lot #C115A27A Ht #402C4371	.033	1.22	.009	.014	.02	.49	.92	.57	N/R	N/R
Lot #J417B27AF Ht #412P3611	.07	1.10	.016	.019	.03	.36	.93	.47	.03	.02
Lot #C109A27A Ht #09M057	.063	1.18	.009	.021	.03	.47	.89	.53	N/R	N/R
Lot #E204A427A Ht #624263	.051	1.08	.010	.023	.06	.38	.89	.60	N/R	N/R
Lot #F414B27AF Ht #659N315	.05	1.14	.015	.013	.04	.35	1.00	.49	.05	N/R
Note: N/R - Not Reported										

The maximum end-of-life fluence is 1.4×10^{18} n/cm^2 at 1/4T depth of the vessel beltline material for Units 1 and 2.

(4) This information is contained in Subsection 5.3.1.4.1.7.

(5) Core beltline plates 22-2 & 22-3, Heat.& Slab Nos. C0776-1 and C2433-1 are limiting at end-of-life for Susquehanna Unit 1. For Susquehanna SES Unit 2, the core beltline plates which were limiting at the beginning-of-life (see response to 121.2 (2) are also limiting at the end-of-life. The feedwater nozzles are also limiting at lower pressures for Units 1 & 2 as shown on Figures 5.3-4 and 5.3-5.

QUESTION 121.3

The FSAR states that compliance with Appendix G of 10 CFR Part 50 and Appendix G of Section III of the ASME Code was not possible for components purchased prior to the issuance of the Summer 1972 Addenda of the ASME Code without replacement of large amounts of materials, reworking of fabricated components and the revision of most all of the design analyses for the components.

The details of the method of compliance as stated in the FSAR are insufficient to identify the areas of noncompliance with Appendix G of 10 CFR Part 50. The applicant should state specifically those sections in which strict compliance with the regulations was not achieved.

The technical bases for the proposed alternate methods used to satisfy the requirements of those sections of Appendix G of 10 CFR Part 50 where strict compliance was not achieved should be presented. These bases should include technical justification to demonstrate that the proposed alternatives provide acceptable safety margins relative to the Appendix G requirements.

RESPONSE:

See revised Subsection 5.3.1.5.

QUESTION 121.4

Paragraph II.C.2 of Appendix H. 10 CFR Part 50 states: "Surveillance capsules containing the surveillance specimens shall be located near but not attached to the inside vessel wall in the beltline region..." FSAR Section 5.3 indicates that the capsule holder brackets were welded to the reactor pressure vessel inner wall. Present sufficient design and fabrication detail to demonstrate that the capsule attachments were designed and constructed in accordance with accepted standards, such as the ASME Code Section III rules for attachments to vessels.

RESPONSE:

The surveillance brackets are welded to the clad material which surfaces the pressure vessel and are, therefore, not attached to the pressure boundary directly. As attached, the brackets do not have to comply with specifications of the ASME Pressure Vessel Code. See Figure 121.4-1.

Security-Related Information
Figure Withheld Under 10 CFR 2.390

FSAR REV. 46, 06/93

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

SURVEILLANCE
SPECIMEN BRACKETS

FSAR FIGURE 121.4-1

PP&L DRAWING

Security-Related Information
Figure Withheld Under 10 CFR 2.390

FSAR REV.65

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

SURVEILLANCE
SPECIMEN BRACKETS

FIGURE 121.4-1, Rev 47

AutoCAD: Figure Fsar 121_4_1.dwg

QUESTION 121.5

In FSAR Sections 1.6 and 5.3 General Electric Report NEDO-20631, "Mechanical Property Surveillance of Reactor Pressure Vessels for General Electric BWR-6 Plants," dated March 1975, is referenced.

This report has not been submitted for review. Therefore the information referenced by the GE report must be provided in the FSAR so that an evaluation of compliance with Appendix H of 10 CFR Part 50 can be made for this plant.

RESPONSE:

The report NEDO-20631 has been withdrawn as a reference for Susquehanna SES, and was replaced by report NEDO-21708, "Radiation Effects in Boiling Water Reactor Pressure Vessel Steels," dated October 1977. The NRC staff has been provided NEDO-21708 for review. NEDO-21708 addresses the requirements of Appendix H to 10CFR Part 50 and supports the current application of Regulatory Guide 1.99.

QUESTION 121.6

To provide assurance that high energy turbine missiles will not be produced at operating speed or design overspeed, provide documentation (including the results of material property testing) to show the degree of conformance of the turbine-generator with the guidelines in SRP 10.2.3, "Turbine Disk Integrity," Paragraph II, "Acceptance Criteria."

RESPONSE:

Turbine disk integrity is discussed in Section 10.2.3. Results from tests on the disks are given below.

High Pressure Rotor

The high pressure turbine rotors were forged from vacuum degassed NiCrMoV steel. Final rotor properties were verified by tests after a suitable quench and temper. The measured rotor properties, together with a 100% volumetric (ultrasonic) evaluation, form the basis for rotor material acceptance.

Material properties of high pressure rotors on turbines 170 X 592 and 170 X 593 have been examined and the rotors were found acceptable for their intended application. In particular, the high pressure rotors on above units have bore measured room temperature Charpy energies in excess of 50 foot pounds and bore measured 50% fracture appearance transition temperatures (FATT) below 50°F.

The ratio of fracture toughness K_{IC} to maximum tangential stress for above rotors meets or exceeds $2\sqrt{in.}$ K_{IC} for the above ratio was calculated from acceptance data by methods which are more conservative than methods described by J. A. Begley and W. A. Longsdon in Westinghouse Scientific Paper 71-1E7-MSLRF-P1. Rotor bore tangential stresses for the above ratio consist of the sum of centrifugal stress at 115%* of rated speed, thermal stress and shrink stress.

The tangential stresses, as calculated above, were compared with measured rotor yield strength, as adjusted to rotor operating temperature, and it was found that stresses are lower than .75 of yield strength, as required by design criteria.

LOW PRESSURE ROTOR

The low pressure turbine wheels were forged from vacuum degassed NiCrMoV steel. Final wheel properties were verified by tests after a suitable quench and temper. The measured wheel properties, together with a 100% volumetric (ultrasonic) evaluation, form the basis for wheel material acceptance.

Material properties of low pressure wheels on turbines 170 X 592 and 170 X 593 have been examined and the wheels were found acceptable for their intended application. In particular, all wheels on above units have surface measured room temperature Charpy energies in excess of 60 foot pounds and surface measured 50% fracture appearance transition temperatures (FATT) below 0°F.

The ratio of fracture toughness K_{IC} to maximum tangential stress for all above wheels meets or exceeds $2\sqrt{in}$. K_{IC} for the above ratio was calculated from acceptance data by methods which are more conservative than methods described by J. A. Begley and W. A. Longsdon in Westinghouse Scientific Paper 71-1E7-MSLRF-P1. Wheel bore tangential stresses for the above ratio consist of the sum of centrifugal stress at 115%* of rated speed, thermal stress and shrink stress.

The tangential stresses, as calculated above, were compared with measured wheel yield strength, as adjusted to wheel operating temperature, and it was found that stresses are lower than .75 of yield strength, as required by design criteria.

*Note: The highest speed anticipated from loss of load with normal operation of the control system is 110% of rated speed.

QUESTION 121.7

(Reference Hatch Nuclear Plant Unit No. 2 response to items 121.15 and 121.18). The following information is necessary to demonstrate that the Susquehanna Unit Nos. 1 and 2 feedwater inlet nozzle thermal sleeve/sparger design has been evaluated with due consideration to nozzle cracking due to thermal cycling and that a program of schedule augmented inservice inspections, with a sensitive method that will assure detection, has been developed:

(1) The technical basis to assure the structural integrity of both the feedwater inlet nozzle and the sparger.

(2) An evaluation of the feasibility of automated ultrasonic testing (UT) fixtures installed on all feedwater inlet nozzles with particular attention on examination of the nozzle bore region.

(3) An evaluation of the feasibility of performing the internal surface examination by magnetic particle methods.

Your response should contain:

a) a description of the nozzle and sparger design including dimensions, materials of construction and weld locations.

b) description of analyses and test data, referencing if necessary data previously submitted to the staff where directly appropriate for this plant.

c) projected crack growth rates, stress levels and usage factors for both the nozzle and the sparger should be described in detail.

d) any plant modifications that are planned to reduce the feedwater to reactor water temperature differential during low power operation.

e) any instrumentation that will be installed in the reactor to verify the conclusions of the design analysis should be identified.

Several ultrasonic testing concepts and procedures have been used to examine the feedwater inlet nozzle regions in operating plants. Define the specific ultrasonic testing procedure that will be used for Susquehanna Unit Nos. 1 and 2. Discuss the influence of local grindouts on crack detection on your ultrasonic testing method.

In addition, provide a description of the augmented inservice inspection (ISI) program to be implemented including scheduled surface examination, ultrasonic testing and verification of the leak tight integrity of the thermal sleeve to safe end joint on all nozzles. The essential elements of an acceptable program are given below:

Augmented Inservice Inspection Plan

(1) Preservice Examination - Preservice UT examination should include all nozzle inner radius, bore, and safe end regions. In addition, a preservice surface examination should be performed on the accessible regions of all nozzle inner radii.

(2) Inservice Examination - To confirm the continuing structural integrity, the following examinations should be performed:

(a) At each scheduled refueling outage, an external UT examination of all feedwater nozzle inner radii, bore and safe end regions.

(b) After 50 startup/shutdown cycles but prior to 70 cycles, a surface examination of the accessible regions of all nozzle inner radii. The definition of startup/shutdown cycles and the procedure for liquid penetrant examination is contained in report NUREG-0312, "Interim Technical Report on BWR Feedwater and Control Rod Drive Return Line Nozzle Cracking."

(c) Subsequent surface examinations of the accessible region of all nozzle inner radii should be performed at the earlier of (i) every other scheduled refueling outage, or (ii) at the scheduled refueling outage after 20 but prior to 40 startup/shutdown cycles after the last surface examination.

(3) Thermal Sleeve to Safe End Joint - An examination method, such as a leak test should be developed to confirm the continuing structural and leak-tight integrity of the thermal sleeve to safe end joint.

Acceptance Standards

(1) All UT indications elevated to be cracks should be verified by appropriate surface examination and removed by local grinding.

(2) All surface indications evaluated to be service induced cracks should be removed by local grinding.

(3) The UT inspection personnel should be required to demonstrate supplemental qualifications by either (i) past successful experience in locating and identifying cracks in BWR feedwater inlet nozzles or (ii) performing a qualification test on a full size unclad nozzle mockup.

Recording and Reporting Standards

Requirements for recording of indications and reporting of inspection results are contained in report NUREG-0312.

RESPONSE:

Note: All responses are presented in the order they are found in the Question.

(1) Discuss the technical basis to assure the structural integrity of both the feedwater inlet nozzle and the sparger, including:

(a) A description of the nozzle and sparger design including dimensions, materials of construction and weld locations.

Description of feed water inlet nozzle.

Part	Material
Nozzle Forging	SA508CL II
Safe End	SA508CL I

Dimensions, location of weld and other details are provided in the following drawings:

79E902, Sheet 1	Susquehanna SES Units 1 and 2 - Figure(s) 121.1-1a, b
137C5543 PT No. 4	Susquehanna SES Units 1 and 2 - Figure(s) 121.7-2

Description of sparger material, basic dimensions and weld locations are presented in the drawings:

Susquehanna Unit 1 - Figure(s) 121.7-3a,b,c
Susquehanna Unit 2 - Figure(s) 121.7-4a,b,c

(b) A description of analysis and test data referencing, if necessary, data previously submitted to the staff where directly appropriate for this.

The information for part b has been provided in the response and reference document cited for Question 112.7.

(c) Projected crack growth rates, stress levels and usage factors for both the nozzle and the sparger should be described in detail.

The information for part c has been provided in the response and reference document cited for Question 112.7

(d) Any plant modifications that are planned to reduce the feedwater to reactor water temperature differential during low power operation.

Susquehanna SES is currently evaluating specific modifications to fluid systems and operating procedures as discussed in General Electric Document NEDE-21821A.

(e) Any instrumentation that will be installed in the reactor to verify the conclusions of the design analysis should be identified.

Due to demonstrated benefits from nozzle cracking fixes, no instrumentation has been installed for design verification.

(2) Evaluate the feasibility of automated ultrasonic testing (UT) fixtures installed on all feedwater inlet nozzles with particular attention on examination of the nozzle bore region.

Currently, automated ultrasonic examination of the feedwater nozzle inner radii, safe-end, and bore region is not feasible. Preservice examinations on the nozzles will be performed utilizing a General Electric developed ultrasonic testing (UT) procedure. This procedure divides the nozzle inner surface into three regions, each

of which is examined separately by a single angle beam shear wave technique. Current state of the art technology does not allow for automation of this technique due to the complexity of the technique and various scanning patterns involved; also, computer assisted signal discrimination equipment is not yet available for field usage.

Scanning solely of the nozzle bore region may be accomplished from the cylindrical section of the nozzle forging utilizing a temporary, removable track scanner. In terms of radiation exposure, (examination/set up time) and examination coverage, automation of only this portion of the examination is not beneficial and is not being considered for preservice activities.

(3) Evaluate the feasibility of performing the internal surface examination by magnetic particle methods.

Magnetic prod inspection methods are not acceptable in this area. Due to limited access in which to perform the examination, maintenance of proper prod contact with the nozzle surface is difficult, possibly resulting in arc-strikes below the electrodes. These surface defects are localized heat affected zones of higher hardness than the surrounding metal. Should the arc strike be accompanied by localized cracking, then surface grinding would be necessary to restore the nozzle to its original surface condition.

Handheld magnetic yokes will not readily fit in the area between the sparger body and the nozzle radius while maintaining proper contact with the nozzle surface and still allow adequate access to perform the examination. Based on the above, magnetic particle examination methods are not considered feasible inside the reactor vessel with the present sparger configuration.

(4) Define the specific ultrasonic testing procedure that will be used for Susquehanna Unit Nos. 1 and 2. Discuss the influence of local grindouts on crack detection on your ultrasonic testing method.

Susquehanna SES will perform preservice examinations of all feedwater nozzle inner radii, safe end, and bore regions to provide a baseline for routine augmented inservice inspections outlined later in this response. Feedwater nozzle safe end examinations will be performed in accordance with ASME Section XI requirements to General Electric Company Procedure #ISE-QAI-322 "Ultrasonic Examinations of Similar and Dissimilar Metal Welds." The inner radii and bore regions will be performed in accordance with General Electric Company Generic Procedures listed below:

TP-508-0173	Rev. D	"Procedure for Nozzle Inner Radii Zone I Ultrasonic Examination"
TP-508-0174	Rev. D	"Procedure for Nozzle Inner Radius Zone 2 Ultrasonic Examination"
E50YP14	Rev. 0	"Procedure for Nozzle Inner Radius Zone 3 Ultrasonic Examination"

Susquehanna SES site specific procedures technically in accordance with these generic procedures are being generated.

For examination purposes, the inner surface has been divided into three regions' each of which is examined separately by a single angle beam shear wave technique. Examination of the nozzle inner radius will be performed by pulse-echo ultrasonic techniques from the exterior of the reactor pressure vessel by contacting the vessel plate surface. The nozzle inner bore region shall be examined from the outer blend radius and the cylindrical portion of the nozzle - the former requiring a special transducer wedge that complements the curvature of the contact area radius.

Should local grindouts be made in the examination surface creating a depression with definable sides, depth, and length, the ultrasonic techniques being used would obtain reflections from these cavities. Such reflections will be minimized by blending the grind cavity into the surrounding base metal in accordance with ASME

requirements. This should result in improved detection sensitivity to actual cracking in the grindout area by eliminating spurious geometric indications from the grindout.

(5) Provide a description of the augmented inservice inspection (ISI) program to be implemented including scheduled surface examination, ultrasonic testing, and verification of the leak tight integrity of the thermal sleeve to safe end joint on all nozzles.

AUGMENTED INSERVICE INSPECTION PROGRAM

Susquehanna SES will implement the reactor feedwater (RFW) nozzle inspection program described below. Justification for any deviations from recommended inspections in NUREG 0619 are presented following the response.

PRESERVICE EXAMINATION

Susquehanna SES will perform a preservice ultrasonic examination of reactor feedwater nozzle inner radii, bore, and safe end regions. All U.T. personnel and procedures will be fully qualified as required. In addition, a preservice liquid penetrant examination will be performed on accessible areas of all Unit #1 feedwater nozzle inner radius surfaces. Also, all nozzle forgings have previously been fully shop magnetic particle inspected and have met ASME Section III requirements. Full liquid penetrant examination will be performed on all Unit #2 feedwater nozzle forgings prior to installation of the spargers.

Inservice Examination

Susquehanna SES-1 will perform the following routine inservice inspections as follows:

1. Ultrasonic examination of the reactor feedwater nozzle inner radii and bore region will be performed every two (2) refueling cycles on one (1) RFW nozzle. The inspection interval begins with the first refueling cycle since the unclad nozzle and triple sleeve sparger was installed prior to plant start up. Safe end examinations will continue to be performed in accordance with ASME Section XI requirements.

2. Penetrant testing of the nozzle inner radii and bore region will be performed only as required to verify and characterize U.T. indications.

3. Visual inspection of the sparger will continue to be performed in accordance with ASME Section XI requirements.

4. Verification of the thermal sleeve to safe end joint shall be made by performance of in-vessel physical leak testing or some alternate method such as on-line leaking monitoring. Susquehanna SES is presently pursuing the feasibility of the later.

In the event an indication is discovered by UT and found to result from service induced cracks propagating from the nozzle inner surfaces, the following action will be taken:

All accessible areas of remaining feedwater nozzles will be examined using penetrant techniques during the refueling outage in which the cracking is verified.

All surface indications determined to be service induced cracks will be removed by local grinding.

A RFW nozzle examination program for subsequent refueling outages will include the external ultrasonic examination of all feedwater nozzle inner radii, bore and safe end regions for each scheduled refueling outage for 3 consecutive outages. If no new indications are discovered, or if new indications are determined to not result from service induced cracks at the nozzle inner surfaces, the aforementioned program will be resumed. If after 3 additional outages no new indications resulting from surface induced cracks are detected, subsequent examinations will be scheduled in accordance with normal ASME Section XI requirements.

The conduct of surface examinations of accessible nozzle inner radius surfaces will continue to be used throughout plant life only to confirm or characterize new ultrasonic indications which are suspected to result from service induced cracks at the nozzle inner surfaces.

Thermal Sleeve to Safe End Joint

Susquehanna SES shall verify the integrity of the thermal sleeve-to-safe end joint by performance of in-vessel physical leakage testing or alternate methods such as on-line periodic leakage monitoring.

Recording and Reporting Standards

Susquehanna SES-1 will record crack indications and report inspection results in compliance with the requirements stated in NUREG 0619.

JUSTIFICATION OF DEVIATION FROM RECOMMENDATIONS

Ultrasonic Examinations Frequency

Susquehanna SES will ultrasonically examine one RFW nozzle every second refueling outage. This is justified for the following reasons, which reflect a significant advance in the Susquehanna SES design and operating procedures towards the long term solution of the BWR nozzle cracking problems per NUREG 0619.

a. Improved Design: The Susquehanna SES RFW thermal-sleeve-to-safe-end joint provides a near zero leakage design. This design essentially eliminates the primary historical initiating source of nozzle cracking in BWRs.

b. No Nozzle Cladding: The Susquehanna SES-1 RFW nozzle surfaces are not clad. The likelihood of crack initiation in unclad nozzles is considerably reduced such that elimination of the nozzle cladding and installation of the triple sleeve sparger design may be all that is necessary to suppress cracking within the design lifetime.

c. Proven Examination Technique: The ultrasonic examination equipment and personnel to be used in performing both baseline and inservice ultrasonic examinations will be qualified on a full scale mock-up of the nozzle, simulating the nozzle geometry and anticipated fatigue crack defects. Since the Susquehanna SES-1 reactor feedwater nozzles are unclad as stated in b) above, a more sensitive examination is possible due to lack of clad/basemetal interface.

d. Augmented Examination Frequency: The above stated program provides RFW nozzle examination coverage at approximately one and one half times the frequency of the ASME Section XI requirements, i.e., all RFW nozzles will be examined within approximately seven years rather than within ten years.

The above factors, when combined, provide adequate assurance that the factors which have led historically to BWR RFW nozzle cracking have been virtually eliminated. Furthermore, any cracking which might occur from unanticipated sources will be discovered before propagating to a significant depth utilizing an augmented examination schedule with state-of-the-art qualified ultrasonic examination techniques.

Surface Examinations

Susquehanna SES-1 will perform liquid penetrant examinations of the accessible internal surfaces of all RFW nozzles during the preservice inspection activities. In-service surface examinations necessitating removal of the spargers, will be performed only when indications of service induced cracking are discovered ultrasonically. This is justified as follows:

a. Reduced probability of crack initiation and growth as stated in the justification above (Ultrasonic Examinations Frequency a thru f).

b. Access: In order to obtain access to perform a penetrant surface examination of the RFW nozzle surfaces during a refueling outage, the vessel water level would have to be lowered below the level of the spargers and hydrolaser decontamination performed. Special shielded platforms would be required to minimize exposures.

c. Removal of the current design sparger for routine penetrant examination may result in damage to the thermal sleeve sealing surface, resulting in increased likelihood of leakage.

Acceptance Standards

(1) All UT indications evaluated to be cracks should be verified by appropriate surface examination and removed by local grinding.

(2) All surface indications evaluated to be service induced cracks should be removed by local grinding.

(3) The UT inspection personnel should be required to demonstrate supplemental qualifications by either (i) past successful experience in locating and identifying cracks in BWR feedwater inlet nozzles or (ii) performing a qualification test on a full size unclad nozzle mock-up.

Response:

(1) Susquehanna SES will comply with this criteria as stated in 2(a) above.

(2) Susquehanna SES will comply with this criteria as stated in 2(a) above.

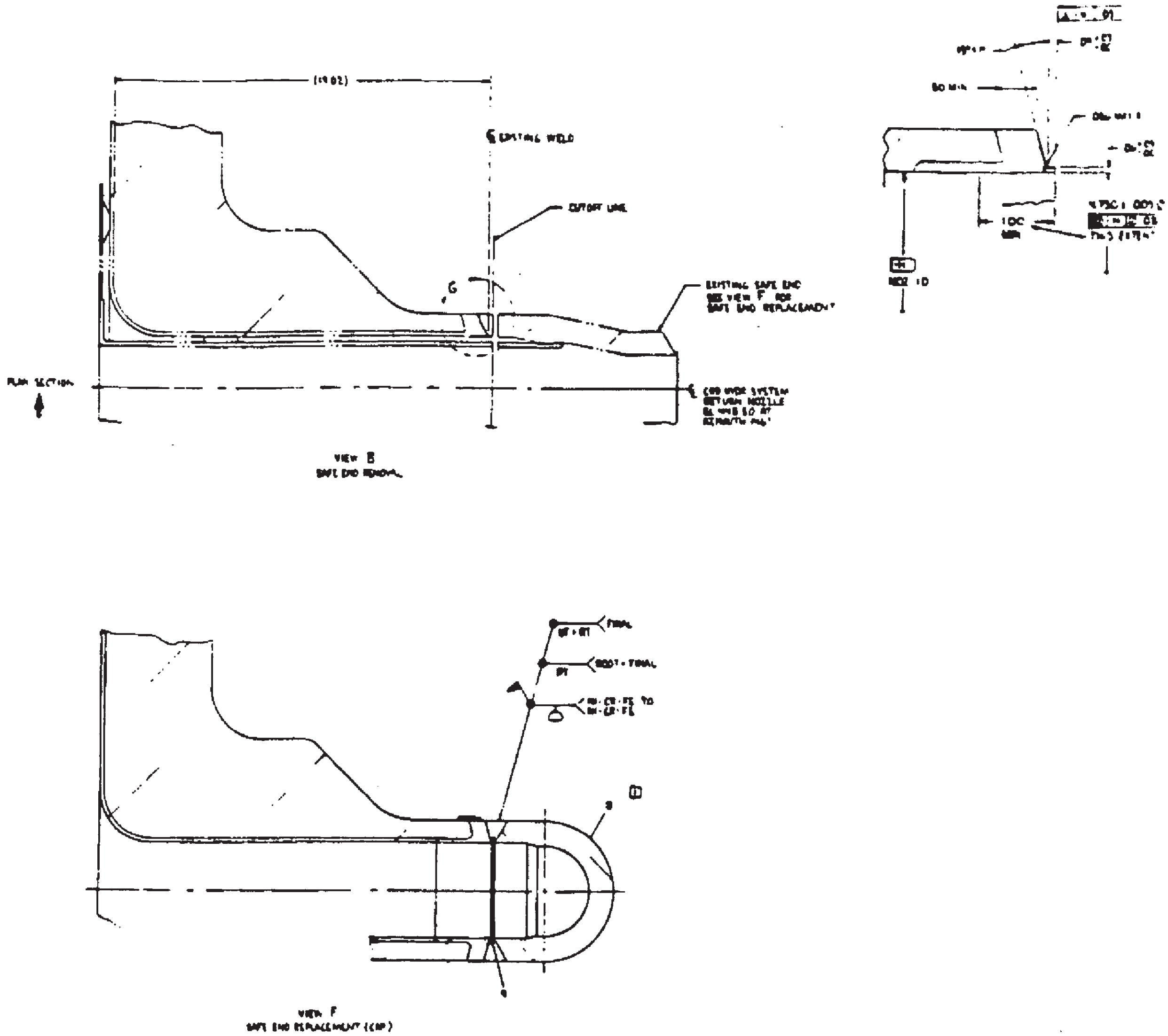
(3) Susquehanna SES will utilize General Electric Company qualified procedures previously referenced. All personnel performing examinations at Susquehanna SES will be qualified in accordance with these procedures on a full nozzle mock-up.

Security-Related Information
Figure Withheld Under 10 CFR 2.390

FSAR REV. 46, 06/93

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 AND 2 FINAL SAFETY ANALYSIS REPORT
FEEDWATER INLET NOZZLE SAFE END
FSAR FIGURE 121.7-1a

PP&L DRAWING



FSAR REV. 46, 06/93

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

FEEDWATER
INLET NOZZLE
SAFE END

FSAR FIGURE 121.7-1b

PP&L DRAWING

Security-Related Information
Figure Withheld Under 10 CFR 2.390

FSAR REV. 46, 06/93

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

FEEDWATER
INLET NOZZLE
SAFE END

FSAR FIGURE 121.7-2

PP&L DRAWING

Security-Related Information
Figure Withheld Under 10 CFR 2.390

FSAR REV. 46, 06/93

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

UNIT 1
IMPROVED INTERFERENCE FIT
FEEDWATER SPARGER ASSEMBLY
SOLUTION ANNEALED AFTER ALL
MACHINING & WELDING (EXCLUDING
THERMAL SLEEVE & BRACKETS)
FSAR FIGURE 121.7-3a

PP&L DRAWING

Security-Related Information
Figure Withheld Under 10 CFR 2.390

FSAR REV. 46, 06/93

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 AND 2 FINAL SAFETY ANALYSIS REPORT
UNIT 1 IMPROVED INTERFERENCE FIT FEEDWATER SPARGER ASSEMBLY NOZZLE AND THERMAL SLEEVE ARRANGEMENT FSAR FIGURE 121.7-3b

PP&L DRAWING

Security-Related Information
Figure Withheld Under 10 CFR 2.390

FSAR REV. 40, 06/95

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 AND 2 FINAL SAFETY ANALYSIS REPORT
UNIT 1 IMPROVED INTERFERENCE FIT FEEDWATER SPARGER THERMAL SLEEVE INTERFERENCE FIT DETAILS
FSAR FIGURE 121.7-3c

PP&L DRAWING

Security-Related Information
Figure Withheld Under 10 CFR 2.390

FSAR REV. 46, 06/93

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 AND 2 FINAL SAFETY ANALYSIS REPORT
UNIT 2 IMPROVED INTERFERENCE FIT FEEDWATER SPARGER ASSEMBLY FSAR FIGURE 121.7-4a

PP&L DRAWING

Security-Related Information
Figure Withheld Under 10 CFR 2.390

FSAR REV. 46, 06/93

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 AND 2 FINAL SAFETY ANALYSIS REPORT
UNIT 2 IMPROVED INTERFERENCE FIT FEEDWATER SPARGER ASSEMBLY NOZZLE AND THERMAL SLEEVE ARRANGEMENT FSAR FIGURE 121.7-4b

PP&L DRAWING

Security-Related Information
Figure Withheld Under 10 CFR 2.390

FSAR REV. 46, 06/93

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

UNIT 2
IMPROVED INTERFERENCE FIT
FEEDWATER SPARGER
THERMAL SLEEVE INTERFERENCE
FIT DETAILS

FSAR FIGURE 121.7-4c

PP&L DRAWING

Security-Related Information
Figure Withheld Under 10 CFR 2.390

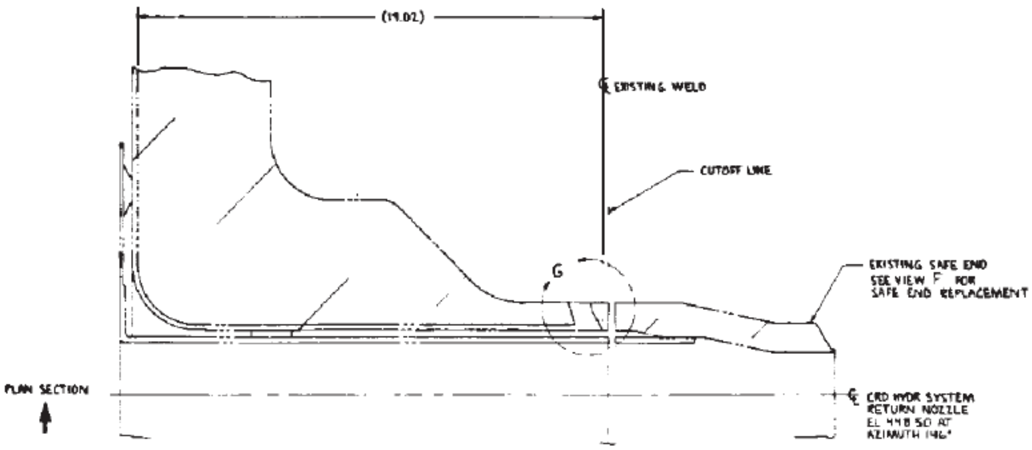
FSAR REV.65

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

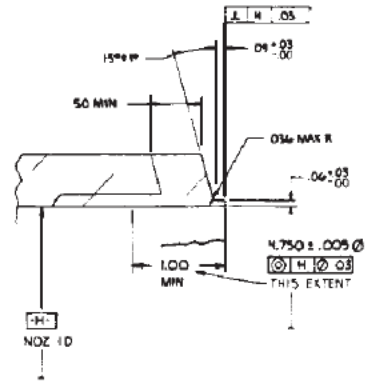
FEEDWATER
INLET NOZZLE
SAFE END

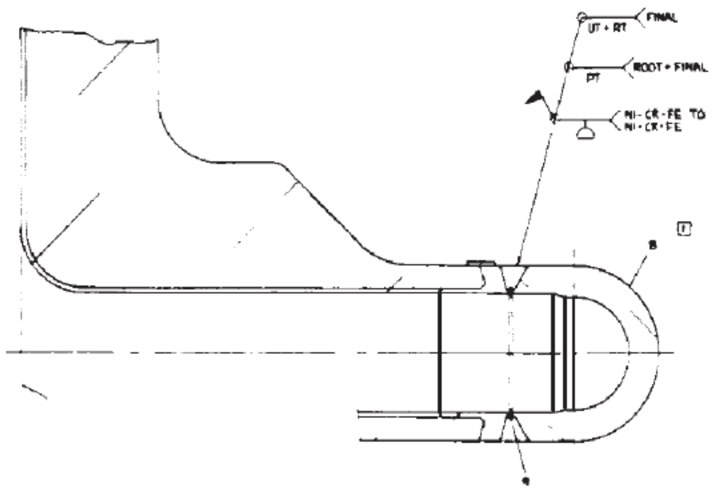
FIGURE 121.7-1A, Rev 47

AutoCAD: Figure Fsar 121_7_1A.dwg



VIEW B
SAFE END REMOVAL





VIEW F
SAFE END REPLACEMENT (CAP)

FSAR REV.65

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

FEEDWATER
INLET NOZZLE
SAFE END

FIGURE 121.7-1B, Rev 47

AutoCAD: Figure Fsar 121_7_1B.dwg

Security-Related Information
Figure Withheld Under 10 CFR 2.390

FSAR REV.65

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

FEEDWATER
INLET NOZZLE
SAFE END

FIGURE 121.7-2, Rev 47

AutoCAD: Figure Fsar 121_7_2.dwg

Security-Related Information
Figure Withheld Under 10 CFR 2.390

FSAR REV.65

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

UNIT 1
IMPROVED INTERFERENCE FIT FEEDWATER
SPARGER ASSEMBLY SOLUTION ANNEALED
AFTER ALL MACHINING & WELDING (EXCLUDING
THERMAL SLEEVE & BRACKETS)

FIGURE 121.7-3A, Rev 47

AutoCAD: Figure Fsar 121.7-3A.dwg

Security-Related Information
Figure Withheld Under 10 CFR 2.390

FSAR REV.65

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

UNIT 1
IMPROVED INTERFERENCE FIT
FEEDWATER SPARGER ASSEMBLY
NOZZLE AND THERMAL SLEEVE ARRANGEMENT

FIGURE 121.7-3B, Rev 47

AutoCAD: Figure Fsar 121_7_3B.dwg

Security-Related Information
Figure Withheld Under 10 CFR 2.390

FSAR REV.65

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

UNIT 1
IMPROVED INTERFERENCE FIT
FEEDWATER SPARGER
THERMAL SLEEVE INTERFERENCE
FIT DETAILS

FIGURE 121.7-3C, Rev 47

AutoCAD: Figure Fsar 121_7_3C.dwg

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
UNIT 2 IMPROVED INTERFERENCE FIT FEEDWATER SPARGER ASSEMBLY FIGURE 121.7-4A, Rev 47

AutoCAD: Figure Fsar 121_7_4A.dwg

Security-Related Information
Figure Withheld Under 10 CFR 2.390

FSAR REV.65

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

UNIT 2
IMPROVED INTERFERENCE FIT
FEEDWATER SPARGER ASSEMBLY
NOZZLE AND THERMAL SLEEVE ARRANGEMENT

FIGURE 121.7-4B, Rev 47

AutoCAD: Figure Fsar 121_7_4B.dwg

Security-Related Information
Figure Withheld Under 10 CFR 2.390

FSAR REV.65

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

UNIT 2
IMPROVED INTERFERENCE FIT
FEEDWATER SPARGER
THERMAL SLEEVE INTERFERENCE
FIT DETAILS

FIGURE 121.7-4C, Rev 47

AutoCAD: Figure Fsar 121_7_4C.dwg

QUESTION 121.8

We will require that your inspection program for Class 1, 2 and 3 components be in accordance with the revised rules in 10 CFR Part 50, Section 50.55a, paragraph (g) published in the February 12, 1976 issue of the FEDERAL REGISTER.

To evaluate your inspection program, the following minimum information is necessary for our review:

(1) A preservice inspection plan to consist of the applicable ASME Code Edition and the exceptions to the Code requirements.

(2) An inservice inspection plan submitted within six months of anticipated commercial operation.

The preservice inspection plan will be reviewed to support the safety evaluation report finding on compliance with preservice and inservice inspection requirements. The basis for the determination will be compliance with:

(1) The Edition of Section XI of the ASME Code stated in your PSAR or later Editions of Section XI referenced in the FEDERAL REGISTER that you may elect to apply.

(2) All augmented examinations established by the Commission when added assurance of structural reliability was deemed necessary. Examples of augmented examination requirements can be found in NRC positions on (a) high energy fluid systems in SRP Section 3.2, (b) turbine disk integrity in SRP Section 10.2.3, and (c) feedwater inlet nozzle inner radii.

Your response should define the applicable Section XI Edition(s) and subsections. If any examination requirements of the Edition of Section XI in your PSAR can not be met, a relief request including complete technical justification to support your conclusion must be provided.

The inservice inspection plan should be submitted for review within six months of anticipated commercial operation to demonstrate compliance with 10 CFR Part 50, Section 50.55a, paragraph (g). This plan will be evaluated in a safety evaluation report supplement. The objective is to incorporate into the inservice inspection program Section XI requirements in effect six months prior to commercial operation and any augmented examination requirements established by the

Commission. Your response should define all examination requirements that you determine are not practical within the limitations of design, geometry, and materials of construction of the components.

Attached are detailed guidelines for the preparation and content of the inspection programs and relief requests to be submitted for staff review.

RESPONSE:

The inspection program for Class 1, 2 and 3 components has been provided (PLA-619, N. W. Curtis to B. J. Youngblood dated 1/27/81).

QUESTION 121.9

Paragraph IV.A.2.a, Appendix G, 10 CFR Part 50, requires that a reference temperature, RT_{NDT}, be determined for each ferritic material of the reactor vessel and that this reference temperature be used as a basis for providing adequate margins of safety for reactor operation. Previously-submitted data are inadequate to define an RT_{NDT} for the reactor vessel ferritic materials; therefore, supply the following additional information:

(a) If both CVN and dropweight tests were conducted for vessel beltline shell plates as stated in FSAR Subsection 5.3.1.5.1.2, supply the CVN test results in addition to the previously submitted dropweight test results (per response to Question 121.2). Calculate an RT_{NDT} for every shell plate, and explain in detail the method used to establish each RT_{NDT} value.

(b) If only dropweight tests were conducted for vessel beltline shell plates as stated in the response to Question 121.2, explain in detail the method(s) used to establish an RT_{NDT} value for each vessel plate.

(c) Supply both CVN and dropweight test results for every other ferritic vessel plate not addressed by items (a) and (b). This should include the upper shell and both lower and upper vessel heads. Calculate an RT_{NDT} value for each plate and explain in detail the method used to establish the RT_{NDT} values.

(d) Identify every ferritic weld seam in the reactor vessel by weld wire, heat number, flux type, lot of flux and welding process. This should include any ferritic weld in the beltline region, upper shell, and lower and upper vessel heads. Submit CVN and dropweight test results in addition to the previously submitted beltline weld data. Calculate an RT_{NDT} for every ferritic weld seam, and explain in detail the method(s) used to establish each RT_{NDT} value.

(e) Submit the correlation data used to establish an RT_{NDT} value of no less than -50°F when dropweight results are not available for weld material. This data should include weld wire and flux types, welding process, and heat treatment for each correlation weldment specimen. Explain in detail the analysis used to establish the -50°F value.

RESPONSE:

This information was provided by letter dated 5/20/81 (PLA-796, Curtis to Youngblood).

QUESTION 121.10

Paragraph IV.A.3, Appendix G, 10 CFR Part 50, requires that materials for piping, pumps and valves meet the impact energy requirements of Paragraph NB-2332 of the ASME Code. Materials for bolting must meet the requirements of Paragraph NB-2333 of the ASME Code. To demonstrate compliance with Paragraph IV.A.3, supply all impact test data for the ferritic materials of these components. Identify each material by its ASME specification, heat or lot number, and dimensions when applicable. If any of the above data are not available, submit data from the literature and/or further tests, and analyses to demonstrate compliance with Appendix G.

RESPONSE:

This information was provided by letter dated 5/20/81 (PLA-796, Curtis to Youngblood).

QUESTION 121.11

Paragraph IV.B, Appendix G, 10 CFR Part 50, requires the reactor vessel beltline materials have a minimum upper shelf energy of 75 ft-lbs in the transverse direction. Insufficient data have been supplied to demonstrate that all the beltline plates and welds meet this upper shelf requirement. Submit the following information to demonstrate compliance with Paragraph IV.B:

(a) Impact energy data for all beltline plates (21-1, -2, and -3 of Unit No. 1 and 21-1, -2, and -3 of Unit No. 2) that will demonstrate that the plates in the vessel beltline will have 75 ft-lbs (in the transverse direction) for unirradiated material or that the upper shelf energy will not fall below 50 ft-lbs at the design fluency level. If these data are not available, submit data from the literature and/or further tests on similar base metal, and analyses used to define the upper shelf energy level.

(b) Impact energy data for the following beltline weld materials that will demonstrate that the weld seams in the vessel beltline will have 75 ft-lbs for unirradiated material or that the upper shelf energy will not fall below 50 ft-lbs at the design fluency level. These welds, identified by lot number/heat number are: 629616/L320A27AG, 411L3071/L311A27AF, J417B27AF/412P3611, C109A27A/09M057 and E204A27A/624263. If these data are not available, submit data from the literature and/or further tests of weld material of the same weld wire and flux type, and analyses used to define the upper shelf energy level.

RESPONSE:

This information was provided by letter dated 5/20/81 (PLA-796, Curtis to Youngblood).

QUESTION 121.12

The materials surveillance program uses three specimen capsules, that should contain reactor vessel steel specimens of the limiting base material, weld metal and heat-affected zone material. To help demonstrate compliance with Appendix H, 10 CFR Part 50, provide a table that includes the following information for each specimen:

(1) Actual surveillance material;

(2) Origin of each surveillance specimen (base metal: heat number, plate identification number; weld metal: weld wire, heat of filler material, production welding conditions, and plate material used to make weld specimens);

(3) Test specimen and type;

(4) Fabrication history of each test specimen;

(5) Chemical composition of each test specimen.

Provide the location, lead factor and withdrawal time for each specimen capsule calculated with respect to the vessel inner wall. The above information should be submitted in tabular form as illustrated in Enclosure 1.

RESPONSE:

This information was provided by letter dated 5/20/81 (PLA-796, Curtis to Youngblood).

QUESTION 121.13

Paragraph III.A of Appendix G, requires that ferritic materials of the reactor coolant pressure boundary be impact tested by means of Charpy V-notch and dropweight (when required by the ASME Code) tests. Supply the impact test data for the vessel nozzles, flanges and shell regions near geometric discontinuities to demonstrate compliance with Paragraph III.A. Each component material must be identified by heat number and location within the reactor coolant pressure boundary. Impact test data should include test temperatures, CVN energy, and/or mils lateral expansion.

RESPONSE:

This information was provided by letter dated 5/20/81 (PLA-796, Curtis to Youngblood).

QUESTION 121.14

The applicant has not submitted a Pre-service Inspection (PSI) Program for review. To evaluate compliance with 10 CFR Part 50.55a(g)(2), we will require a complete response to Question 121.8 concerning the PSI. All pre-service examination requirements defined in Section XI of the ASME Code that have been determined to be impractical must be identified and a supporting technical justification must be provided. The PSI program should include at least the following information:

(a) For ASME Code Class 1 and 2 components, provide a table similar to IWB-2600 and IWC-2600 confirming that either the entire Section XI pre-service examination was performed on the component or relief is requested with a technical justification supporting your conclusion.

(b) Where relief is requested for pressure retaining welds in the reactor vessel, identify the specific welds that did not receive a 100% pre-service ultrasonic examination and estimate the extent of the examination that was performed.

(c) Certain ASME Code Class 1 and 2 vessel and piping system welds, that are 1/2 inch or less in nominal wall thickness, are subject to a Section XI pre-service volumetric examination. Confirm that a 100% volumetric examination was performed on these thin-wall weldments.

(d) Where relief is requested for piping system welds (Examination Category B-J, C-F and C-G), provide a list of the specific welds that did not receive a complete Section XI pre-service examination including a drawing or isometric identification number, system, weld number, and physical configuration, e.g., pipe to nozzle weld, etc. Estimate the extent of the pre-service examination that was performed, the primary reason a complete examination is impractical, alternative and/or supplemental examination performed and that method of fabrication examination.

(e) Describe the extent and method of pre-service volumetric examination of Class 1 integrally-welded supports in Examination Categories B-H and B-K-1.

(f) Describe the extent and method of pre-service examination of Class 2 pressure-retaining bolting 2 inches in diameter or less.

RESPONSE:

The response to this question was submitted by letters dated 5/19/81 (PLA-813, Curtis to Youngblood), 6/16/81 (PLA-846, Curtis to Schwencer), and 4/23/82 (PLA-1053, Curtis to Schwencer).

QUESTION 121.15

Paragraph 50.55a(b)(2)(iv) requires that ASME Code Class 2 piping welds in the Residual Heat Removal Systems, Emergency Core Cooling Systems and Containment Heat Removal Systems shall be examined. List the lines in these systems that were exempted from preservice volumetric and/or surface examination based on Paragraphs IWB-1220 and IWC-1220 of Section XI and provide a technical justification. The control of water chemistry to minimize stress corrosion described in Paragraph IWC-1220(c) of Section XI is not an acceptable basis for exempting ECCS components from examination because practical evaluation, review and acceptance standards cannot be defined. To satisfy the inspection requirements of General Design Criteria 36, 39, 42, and 45, the in-service inspection program must include periodic volumetric and/or surface examination of a representative sample of welds in the RHR, ECCS and Containment Heat Removal Systems.

RESPONSE:

The response to this question was submitted by letters dated 5/19/81 (PLA-813, Curtis to Youngblood), 6/16/81 (PLA-846, Curtis to Schwencer), and 4/23/82 (PLA-1053, Curtis to Schwencer).

QUESTION 121.16

List the systems and line sizes that were exempted from pre-service examination based on Paragraph IWB-1220(b)(1) of the 1974 Edition of Section XI based on "normal makeup systems using on-site power."

RESPONSE:

The response to this question was submitted by letters dated 5/19/81 (PLA-813 Curtis to Youngblood), 6/16/81 (PLA-846, Curtis to Schwencer), and 4/23/82 (PLA-1053, Curtis to Schwencer).

QUESTION 121.17

Your inservice inspection program must be revised to include periodic volumetric and/or surface examination of a representative sample of welds in the RHR, ECCS, and Containment Heat Removal Systems. In addition, we will require the following information concerning the examination of piping system welds:

A) Your inservice inspection program is based upon the 1977 Edition through Summer 1978 Addenda of Section XI. However, in lieu of performing stress analysis on piping runs relative to weld sample selection for Category B-J and C-F welds as required by the Summer 1978 Addenda, you have elected to substitute the sample requirements of the 1974 Edition of Section XI. Describe the selection criteria used to determine your weld sample.

B) Your program exempts under IWB-1220(a) components ≤ 4 inches nominal pipe size that are above the normal reactor pressure vessel water line. Provide the technical basis for applying this exclusion for examination.

C) Your program exempts lines 18-GBB-109 and 6-GBB-109 on the basis of fluid chemistry control (IWC-1220(c) and IWC-1220(b)). Our position is that IWC-1220(c) is not a technically valid basis for exemption of inservice inspection requirements. Describe the exclusion from examination of these lines based only on IWC-1220(b). Discuss the operating conditions under which these lines are stagnant.

RESPONSE:

A. Section 3.0 of the Susquehanna SES Unit #1 Inservice Inspection Plan addresses the applicable Code Edition and Addenda to be utilized in preparation of the specific weld sampling program for Class 1 Category BJ and Class 2 Category CF, piping welds. The extent of examination is further illustrated by composite Code tables found on Pages 3-3 through 3-7 of the ISI Plan. The weld selection criteria used will be that found in the specific paragraphs referenced in this section.

For all Code Class 1 pipe welds, examination selection criteria will be determined by Table IWB-2500 and Table IWB-2600 of the 1974 Edition, Summer 1975 Addenda.

For all Code Class 2 pipe welds, including the RHR, ECCS, and CHR systems, the weld sampling program to be

utilized is specified by paragraph IWC-2411 of the 1974 Edition, Summer 1975 Addenda (Pages 3-6, 3-7 of Section 3.0).

These requirements are in compliance with 10 CFR 50.55 a (b) (2)(iv).

B. Components 4-inch nominal pipe size and under that are above normal reactor pressure vessel water line are exempt per paragraph IWB-1220(a) based on reactor coolant make-up. The calculation for normal make-up capacity was performed by General Electric as described in G.E. Specification 22A2750, thereby, allowing 2" and 4" exemption below and above normal reactor water level.

C. IWC-1220(c) was erroneously referenced here; the correct reference should be IWC-1220(b). The proper exemption, systems or portions of systems that do not operate during normal plant conditions, is being taken for RHR system piping containing a test return line and the connection to the containment spray header. These lines are isolated from the RHR system during normal plant operation and do not perform a safety function during normal operating modes.

QUESTION 121.18

Identify the specific model(s) of General Electric low pressure turbine(s) installed in Susquehanna SES Units 1 & 2. Provide tables showing, for each of the seven wheels in a Susquehanna low pressure turbine, the weight and location of the wheels relative to the turbine center, and the (a) dimensions, (b) shapes, (c) weights and d) initial energy (or velocity) ranges of missiles postulated to be representative of missile-producing turbine wheel rupture at

i. design overspeed, and

ii. destructive overspeed.

RESPONSE:

The turbine numbers for SSES are 170 x 592 and 170 x 593 for units 1 & 2 respectively. Specifics of the turbine wheels are insignificant compared to the characteristics of the missile fragments, and were considered in their generation.

As stated in Section 3.5.1.3.2, the characteristics of the missiles postulated by the General Electric Company's Large Steam Turbine Division (GE)for destructive overspeed failures of the wheels in their 38" last stage buckets machines are given in Table 3.5-1 and Figure 3.5-1.

As stated in Section 3.5.1.3.4, GE has determined that the probability of a design overspeed failure in their 38" last stage bucket machines is insignificant compared with the probability of a destructive overspeed failure. Consequently, GE does not supply characteristics of missiles which could result from a wheel failure at design overspeed.

The details of the methodology used by GE in determining failure probabilities for the wheels in their 38" last stage buckets machines and in determining the characteristics of the missiles resulting from destructive overspeed failures are given in References 3.5-1, 3.5-2, 3.5-3 and 3.5-4.

QUESTION 121.19

Provide tables listing all barriers and safety-related targets considered in the calculation of P2 (or P2 x P3). Provide schematic drawings (to scale) which show the location and orientation of barriers and targets relative to the turbine train.

RESPONSE:

No barriers where originally considered in the calculation of P2 x P3 except the targets structural walls and floor slabs. No credit was taken for the moisture separator or the radiation shield wall.

The targets considered in the original calculation were the control structure, the reactor buildings, the ESSW pumphouse, the steam tunnels, and the spent fuel pools. See response to Question 121.21 for revised evaluation of essential targets.

Scaled layout drawings are provided in Section 1.2 of the FSAR which show location and orientation of barriers, turbines, and targets.

QUESTION 121.20

The derivations presented in 3.5.1.3.6 of the FSAR need clarification; numerous variables are poorly defined or not defined at all. The probability P3 must either be re-derived or an available document referenced which contains an adequate derivation.

RESPONSE:

Refer to revised Section 3.5.1.3 for the derivation.

QUESTION 121.21

Recalculate the P2 x P3 probabilities associated with each safety-related target for design overspeed and destructive overspeed, and show that the total P2 x P3 for each of these failure conditions is less than 10^{-3} per failure, as stipulated in Regulatory Guide 1.115.

RESPONSE:

I. Introduction

As noted in Section 3.13 of the FSAR, SSES was issued its construction permit over three years before Regulatory Guide 1.115 was formalized.

Because of this, we have proposed an alternate approach in the FSAR. However, in order to be responsive to this question we have evaluated our alternate method against Regulatory Guide 1.115.

A re-evaluation of our previous calculation changes P2 x P3 from 52×10^{-3} to $.60 \times 10^{-3}$. This reduction was made based on redundancy, credit for missile shielding, and more precise knowledge of essential shutdown equipment locations.

The data presented in this response is based on destructive overspeed missile spectrum data only. To date no missiles have been assumed by the turbines vendor to arise from a design overspeed event. However, for low trajectory missiles, the range of missile velocities which are higher than the minimum velocity necessary to cause damage is greater for destructive overspeed failures than for design overspeed failures. Therefore, destructive overspeed events result in higher total probabilities for damage. Furthermore, the destructive overspeed event encompasses the design overspeed event.

II. Reactor Building

An inspection of Table 3.5-3 shows that the major risk is the reactor building at 45×10^{-3} for P2x P3 . This has two components, the risk for the roof slab at 818' and the risk for the west face of reactor building along column line "P".

A. Roof Slab

The original calculation assumed that unacceptable damage would occur if a turbine missile caused any spallation of the roof slab at elevation 818; the roof slab risk was about 3×10^{-3} from our original calculation. However, there is no essential safe shutdown equipment at elevation 818' or on the two floors below at elevation 799' and 779'.

The computer code employed in the original calculation conservatively assumed that the missile velocity normal to the roof slab would be the same as if the roof slab were at the same elevation as the turbine. Actually, the difference in elevations between the 818' roof slab and the turbine axis is nearly 85 ft.; thus, the actual missile velocity normal to the roof slab should be reduced by approximately 75 ft./sec.

Because the floor slabs have difference thicknesses at different locations, the sum of the floor slab thickness at 818', 799' and 779' ranges from a minimum of 5'3" to a maximum of 8'3" of reinforced concrete. This mass of concrete, plus the reduction in strike velocity normal to the roof of 75 ft./sec. means that the roof slab risk to essential safe shutdown equipment below elevation 779' is negligible.

The reactor vessel is protected by massive shield plugs and the steel primary containment head assembly in the vertical direction, and is considered safe.

The only target at elevation 818' is the spent fuel pool. The P2 x P3 datum from our original calculation is unchanged at $.118 \times 10^{-3}$.

B. Face Wall

The reactor building face wall is primarily subject to low trajectory missiles with a risk of 42×10^{-3} from our original calculation. A review of FSAR Dwgs. M-223, Sh. 1 M-226, Sh. 1 shows that all low trajectory missiles would be subject to interference and slowing from the moisture separator and the radiation shield walls between the turbine low pressure hoods and the reactor building wall.

1. Credit for Shielding

The radiation shield wall is made of concrete block 3'3" thick which is equivalent to 2 ft. of poured concrete per modified NDRC formulae. The moisture separator has a carbon steel shell thickness of 1.25 inch. As each inch of carbon steel is equivalent to one foot of poured concrete, the total equivalent thickness for the moisture separator and the shield wall is greater than four feet of poured concrete. No credit was taken for the extensive internal steel baffling of the moisture separator.

The net effects of the equivalent concrete can be estimated using the methods in the SRP and our previously calculated data.

For low trajectory missiles P2 x P3 will in effect be proportional to:

$V_2 - V_s$
$V_2 - V_1$

Where V_2 = Maximum missile ejection strike velocity

Where V_1 = Minimum missile ejection strike velocity

Where V_s = Minimum missile ejection strike velocity required to spall the 3'0" reactor building wall.

The velocities are dependent on missile type, so it is necessary to consider the three stage groups (defined in FSAR Subsection 3.5.1.3.2) separately.

a) State Group I Missiles:

The analysis is based on the 'worst-case' missile which has V_2 = 470 ft./sec., V_1= 0 ft./sec., a weight of 2000 lbs. and minimum projected rim area equivalent to a diameter of 16.47 in.

Assuming no barriers between the turbine and the reactor building, these missiles contribute 1.66×10^{-2} to P2 x P3. The minimum velocity required to spall a 3'0" wall is 150 ft./sec. This is V_s for the case with no barriers.

The minimum missile velocity required to perforate a 4'0" thick wall is 434 ft./sec. The residual velocity for a missile striking the barrier at 470 ft./sec. is 50 ft./sec. Since this is below the spalling threshold of 150 ft./sec. for the 3 ft. thick reactor building wall, the Stage Group I contribution to P2 x P3 will be zero.

b) Stage Group II Missiles:

The 'worst-case' missile has V_2 = 550 ft./sec., V_1 = 0 ft./sec., a weight of 3000 lb and a minimum projected rim area equivalent to a diameter of 19.48 in.

Assuming no barriers these missiles contribute 2.01×10^{-2} to P2 x P3.

The minimum velocity required to spall a 3'0" wall is 117.0 ft./sec. This is V_s for the no barrier case. The minimum velocity required to perforate a 4'0" wall with a residual velocity of 117.0 ft./sec. is 434.3 ft./sec. This is V_s for the barrier case.

The value of P2 x P3 would thus become

$$P2 \times P3 = \frac{550 - 434.3}{550-117} \times 2.01 \times 10^{-2}$$

$$= 5.37 \times 10^{-3}$$

c) Stage Group III Missiles

The 'worst-case' missile has V_2 = 610 ft./sec. V_1 = 400 ft./sec., a weight of 6500 lb and a minimum projected rim area equivalent to a diameter of 26.63 in. The maximum projected rim area is

equivalent to a diameter of 39.99 in. Assuming no barriers these missiles contribute 5.01 x 10^{-3} to P2 x P3.

The minimum velocity required to spall a 3'0" wall is 71.5 ft./sec. For the maximum projected area the threshold for spalling is 84 ft./sec. The residual velocity for a missile striking a 4'0" wall at 400 ft./sec. (i.e., at V_1) is 154 ft./sec. for the minimum area. For the maximum projected rim area the residual velocity for a strike at 400 ft./sec. is 112 ft./sec. Hence the minimum residual velocity exceeds the maximum spalling threshold velocity for the reactor building wall. A similar analysis for penetration shows that stage III missiles will penetrate a 3'0" wall in 80% of the strikes after first penetrating a 4'0" wall.

2. Target Area Reduction

In the original calculation the target area for the reactor building face was a rectangle 90 feet high and 59 feet wide for a total area of 530 ft.

A review of essential shutdown cable and equipment has been performed for updating our Fire Protection Review Report for the new 10 CFR 50 Appendix R. This review is summarized in Q40.95. From this data base the spaces behind the target wall were examined for effects from spallation and penetration.

a. No target area above el 749' contained essential shutdown equipment.

b. Elevation 719':

(1) On elevation 719' of the reactor building, damage could occur for the hydraulic control units (HCU) on only one side of the core. Moreover, the Standby Liquid Control can be considered a back-up. Also, three essential raceways, originally considered in the target area, E1K883, E1KJ23, and E1KJ24, are located about one hundred feet east of the target area. Only one essential raceway, E1K833, is in the target area. The actual target area of the wall is six feet wide between the stairwell and the elevator shaft on the northwest corner of the reactor building as shown on Dwg. M-243, Sh. 1, of the FSAR. Since there are no internal barriers which could restrict or contain the spallation products, for this target area the damage criterion is spallation. That is, any spallation is assumed to cause unacceptable damage, and, therefore, wall perforation would not add to the risk. The height of the target area is from 729' at the turbine deck to 749' at the floor slab or twenty feet.

Therefore, the actual essential spallation target area is 6' x 20' or 120 $ft.^{2}$

The area correction factor for spallation is:

$$\frac{120\text{ ft}}{5310\text{ ft.}} = .022$$

Applying this to each state group,

	P2 X P3	Area Correction Factor	Subtotal
Stage I0.0		.022	= 0.0
Stage II	5.37×10^{3} x	.022	$= .118 \times 10^{-3}$
Stage III	5.01×10^{-3}	.022	$= .110 \times 10^{-3}$
Risk Total for the 120 ft. target area			$= .228 \times 10^{-3}$

(2) No credit is taken for the 1'0" walls of the elevator shaft and the stairwell stopping the Stage III missile. The 2'0" concrete wall of the containment instrument gas room will resist missiles that enter the reactor building. Spallation products are assumed to be contained by either 1'0" or 2'0" walls. The damage criterion for this area is wall perforation only.

The area of risk for wall penetration is 23' wide. However, 6 ft. of this width has been accounted for above so that the net area is (23'-6') or 17' wide. Height is 20' (from elevation 729' to elevation 749'), for an area of 17' x 20' or 340 ft.

The area correction factor is then

$$\frac{340 \text{ ft.}^2}{5310 \text{ ft.}^2} = .064$$

Stage Group III missiles can penetrate the 3'0" reactor building wall 80% of the time after penetrating the 4'0" of shielding, depending on exit velocity and missile orientation. Thus the effective correction factor is 80% x .064 = .051.

Missiles	P2 x P3	Area Correction Factor	Penetration Probability
Stage I	0.0		
Stage II	0.0		
Stage III	5.01×10^{-3} X	.051	= $.257 \times 10^{-3}$

Total Reactor Building Risk

120 ft. area risk	$.228 \times 10^{-3}$
	$.257 \times 10^{-3}$
+340 ft. area risk	

Reactor Building total $.485 \times 10^{-3}$

III. Other Buildings

The other buildings as listed in table 3.5-3 contributed 7×10^{-3} for P2 x P3 in the original calculation. However, no credit was taken for redundancy of essential equipment. A "lob" missile is the only credible type for the diesel generator buildings. With this trajectory (low trajectories being prevented by the turbine pedestal) only one diesel generator could be damaged per turbine destruction event due to the substantial walls that separate the units and the near vertical trajectory. Losing one diesel would not prevent cold shutdown.

A similar argument can be applied to the steam tunnel. The outboard MSIVs could be lost but the primary containment's six feet of steel-lined concrete, the steam tunnel walls, the moisture separator and the radiation shield walls would protect the inboard MSIVs. The total risk of the diesel generator building and the steam tunnel was 5.5×10^{-3}. Applying the concept of redundant equipment reduces this to zero.

The control structure risk was originally calculated as $.73 \times 10^{-3}$ per turbine destructive event. This was based on the whole building volume above el 729' as a target. Again from our Appendix R review no essential equipment exists above elevation 783'. The roof and floor slab thicknesses equals a minimum of 5' 10.5" to a maximum of 8' 10.5". Also note that gravitational deceleration slows missiles by 75 feet per second at the roof slab elevation.

Including a factor for reduction of low trajectory missiles because of the moisture separator and radiation shield walls reduces the total control structure P2 x P3 to essentially zero for both high and low trajectory missiles.

IV. Quantified Revised Estimates

Applying the correction factors described above revises the P2 x P3 values given in FSAR Table 3.5-2 to the estimates below.

Target	P2 x P3	
	Was	Revised Estimate
Reactor Bldg	$45. \times 10^{-3}$	$.485 \times 10^{-3}$
Spent Fuel Pool	$.118 \times 10^{-3}$	$.118 \times 10^{-3}$
Steam Tunnel	$3. \times 10^{-3}$	0
Control Structure	$.73 \times 10^{-3}$	0
Diesel Generator Bldgs	2.5×10^{-3}	0
TOTAL	52×10^{-3}	$.603 \times 10^{-3}$

V. Non-Quantified Conservatism

The damage probability, P3, has numerous conservatisms as originally calculated. The method used was based on a modification of the NDRC method. We assumed that the rotor fragment always hits on edge and never on the flat face. A flat face hit would considerably reduce the delivered energy per unit area of the target which, intuitively, lowers the probability of spallation and/or penetration.

Energy transfer at impact was conservatively assumed to be 100% of the energy normal to the target face at ejection. No corrections were made for air resistance of the large, tumbling fragments, or less than complete mechanical coupling from glancing blows, ricochets, or the like.

We had further assumed that any spallation or penetration prevented safe shutdown. It is obvious that not all wall damage results in complete equipment failure and prevention of cold shutdown.

VI. Summary

In summary, examining both quantified and non-quantified conservatism in our calculations convinces us that the total P2 x P3 figure for SSES is well within the 1×10^{-3} number stipulated in Regulatory Guide 1.115.

QUESTION 123.1

Pursuant to General Design Criterion 2, safety-related structures, systems and components are to be designed for appropriate load combinations arising from accidents and severe natural phenomena. With regard to the vibratory loads attributed to the feedback of hydrodynamic loads from the pressure suppression pool of the containment, the staff requires that safety-related mechanical, electrical, instrumentation and control equipment be designed and qualified to withstand effects of hydrodynamic vibratory loads associated with either safety relief valve (SRV) discharge of LOCA blowdown into the pressure suppression containment combined with the effects of dynamic loads arising from earthquakes.

The criteria to be used by the staff to determine the acceptability of your equipment qualification program for seismic and dynamic loads are IEEE Std. 344-1975 as supplemented by Regulatory Guides 1.100 and 1.92, and Standard Review Plan Sections 3.9.2 and 3.10. State the extent to which the equipment in your plant meets these requirements and the above requirements to combine seismic and hydrodynamic vibratory loads. For equipment that does not meet these requirements provide justification for the use of other criteria.

RESPONSE:

I. BOP

For Susquehanna Project, all BOP Safety related mechanical, electrical, instrumentation and control equipment located inside Primary Containment, Reactor and Control buildings, is being qualified for Seismic loads in combination with hydrodynamic vibratory loads associated with SRV discharge and LOCA blowdown. Although the SRSS method of combination of seismic and hydrodynamic loads is acceptable, for the project to be conservative, the loads are combined by absolute sum method. The cases which have deviations from the absolute sum method of combination will be identified in the qualification reports.

The criteria for the qualification of BOP equipment for seismic loads is described in Section 3.7b.3 of the FSAR. The criteria for load combinations and methodology for the design assessment and qualification of Safety related BOP equipment for seismic and hydrodynamic loads have been described in Sections 5.7 and 7.1.7 of the Design Assessment Report (DAR) Rev. 2. Basically the requirements of IEEE Std. 344-1975 as

Supplemented by Regulatory guides 1.100 and 1.92 and SRP Sections 3.9.2 and 3.10 are covered in the criteria with the following exception for spatial combination of three components of dynamic motion as stated in Section 7.1.7.1.3 of the DAR. The criteria states "the response at any point is the maximum value obtained by adding the response due to vertical dynamic load with the larger value of the responses due to one of the horizontal dynamic loads by the absolute sum method."

All Susquehanna BOP equipment is being qualified for the criteria discussed above.

II. NSSS

LOAD COMBINATIONS:

These were transmitted to the NRC on 8/28/80 as Page 3 of Attachment N to PLA-536. This was in response to NRC Question 110.42.

IMPLEMENTATION OF LOAD COMBINATIONS:

The GE SQRT Program uses outputs from the GE Equipment Adequacy Evaluation Program which combines dynamic loads by SSES as accepted by the NRC in NUREG-0484.

The individual items associated with the load combinations are added as described below:

Steady State Events (e.g., Dead Load, Pressure) - Absolute Sum

Time Varying Components (e.g., Maximum Seismic, Maximum Hydrodynamic) - SRSS

Components of Events (e.g., Maximum X-Load Due to Y-Earthquake) - SRSS

Modal Response-SRSS, except for closely spaced modes where effects are combined by Absolute Sum, Double Sum, or Grouping.

Details for each item of equipment are contained in that equipment's Design Record File which is available for audit.

QUESTION 123.2

Provide the following information:

(i) Two summary equipment lists (one for NSSS supplied equipment and one for BOP supplied equipment). These lists should include all safety related mechanical components, electrical, instrumentation, and control equipment, including valve actuators and other appurtenances of active pumps and valves. In the lists, the following information should be specified for each item of equipment.

(1) Method of qualification used:

(a) Analysis of test (indicate the company that prepared the report, the reference report number and date of the publication).

(b) If by test, describe whether it was a single or multi-frequency test and whether input was single axis or multi-axis.

(c) If by analysis, describe whether static or dynamic, single or multiple-axis analysis was used. Provide natural frequency (or frequencies) of equipment.

(2) Indicate whether the equipment has met the qualification requirements.

(3) Indicate the system in which the equipment is located and whether the equipment is required for:

a) hot stand-by

b) cold shutdown

c) both

d) neither

(4) Location of equipment, i.e., building, elevation.

(5) Availability for inspection (Is the equipment already installed at the plant site?)

(ii) An acceptable scenario of how to maintain hot standby and cold shutdown based on the following assumptions:

(1) SSE or OBE

(2) Loss of offsite power

(3) Any single failure

(iii) A compilation of the required response spectra (RRS) for all applicable vibratory loads (individual and combined if required) for each floor of the nuclear station under consideration.

RESPONSE:

The response to this question was submitted via PLA 627 (Curtis to Youngblood) dated February 5, 1981.

QUESTION 123.3

Identify those items of nuclear steam supply system and balance-of-plant equipment requiring reevaluation and specify why reevaluation is necessary (i.e. because the original qualification used the single frequency, single axis methodology, because equipment is affected by hydrodynamic loads, or because both of the above conditions were present) for each item of equipment.

RESPONSE:

Originally almost all Safety related BOP equipments for Susquehanna had been qualified for only Seismic loads. This equipment has been re-evaluated due to the inclusion of new hydrodynamic (SRV & LOCA) loads, and are being re-qualified with respect to the criteria described in DAR Section 7.17. The qualification program for the BOP Safety related equipment is being executed in the following four phases.

Phase-I: Qualification of Equipment for Only Seismic Loads:

The only known dynamic load at the time of execution of this phase of the program was Seismic loads. During this phase, the vendors supplying the equipment were required to qualify the equipment in accordance with the requirements specified in FSAR Subsection 3.7b.3.

Phase-II: Evaluation for Combined Seismic and Hydrodynamic (SRV & LOCA) Loads:

This phase was undertaken to evaluate if the existing Seismic qualification of all Safety related BOP equipment could be extended to the combined Seismic and hydrodynamic loads. The criteria used for the re-evaluation is described in DAR Section 7.1.7. The general problem areas identified during this evaluation and the proposed action to mitigate these problems are shown below.

Problem	Action
Additional hydrodynamic loads	. Retest and/or reanalysis.
	. Modifications to equipment of their supports if required.
Flexibility of equipment Support not considered	. Provide response spectre considering support flexibility

	. Include Support Conditions during analysis or testing.
Inadequate Modelling	. Correct during reanalysis.
Inadequate Testing	. Retest.
	. Qualification by analysis.

Phase III: Requalification Efforts:

Specifically, the Problem areas identified in the previous phase are resolved during this phase by taking appropriate actions. The requalification reports demonstrate that the criteria of DAR Section 7.1.7 have been complied with.

Phase IV: Modifications to Equipment or Equipment Supports:

Equipment or their Supports needing modifications identified during the regulations efforts of Phase III are executed during this phase.

The following are NSSS equipment:

SYSTEM	MPL #
Safety Relief Valve	B21F013
MSIV	B21F022/ F028
Flow Element	B21N051/ 52/53/54
Recirc. Pump Motor	B31C001
Gate Valve	B31F023/ 31/32
HCU	C12D001
CRD Valves	C112F009/ 10/11/12
SLC Storage Tank	C41A001
SLC Accumulator	C41A003
SLC Pump	C41C001
SLC Explosive Valve	C41F004
RHR Heat Exchanger	E11B001
RHR Pump	E11C002
Flow Orifice Assembly	E11N012/ N014
LPCS Pump & Motor	E21C001

Flow Orifice Assembly	E21N002
MSIV Heater	E32B001
MSIV Blower	E32C001/ C002
HPCI Pump	E41C001
HPCI Turbine	E41C002
Flow Orifice Assembly	E41N007
RCIC Pump	E51C001
RCIC Turbine	E51C002
Flow Orifice Assembly	E51N001
Fuel Prep Machine	E18E001
Gen. Purpose Grapple	F18E011
Dryer & Separator Sling	F19E008
Head Strong Back	F19E009
Control Rod Grapple	F20E002
Refueling Platform	E21E003
In Vessel Rack	F22E006
Def. Fuel Storage Cont.	F22E009
Fuel Storage Vault	F22E012

CONTROL ROOM PANELS

Reactor Core Cooling BB	H12-P601
Power Range Monitoring Cabinet	H12-P608
RPS Div. 1 and 2 Log VB	H12-P609
RPS Div. 2 and 3 Logical VB	H12-P611
NSSS Temperature Recorder VB	H12-P614
Feedwater & Recirculation Instrument Panel	H12-P612
NSSS Process Instrument Panel	H12-P613
Div 1 RHR/HPCI Relay VB	H12-P617
Div 2 RHR/HPCI Relay VB	H12-P6118
ADS Ch A Relay VB	H12-P628
MSIV Leakage Control Div 2 VB	H12-P654
HPCI Relay VB	H12-P620
RCIC Relay VB	H12-P621
Inboard Valve Relay Board	H12-P622
Outboard Valve Relay VB	H12-P623
Div 1 CS Relay VB	H12-P626
Div 2 CS Relay VB	H12-P627
ADS Ch B Relay VB	H12-P631
MSIV Leakage Control Div 1 VB	H12-P655
Radiation Monitoring Instrument Panel A	H12-P606
Radiation Monitoring Instrument Panel B	H12-P633
Operating BB	H12-P680
Termination Cabinets	H12-P700 Series
Plant Operation Benchboard	H12-P853

NUCLEAR BOILER

Condensing Chamber	B21-D002
Condensing Chamber	B21-D004AB
Condensing Chamber	B21-D006AD
Condensing Chamber	B21-D007AD
Condensing Chamber	B21-D008AD
Condensing Chamber	B21-D009AD

LOCAL PANELS

Reactor Water Clean-Up	H23-P002
Reactor Vessel Level and Pressure (A)	H23-P004
Reactor Vessel Level and Pressure (B)	H23-P005
Recirculation Pump A	H23-P009
Jet Pump B	H23-P010
High Pressure Coolant Injection B	H23-P014
Reactor Core Isolation Cooling A	H23-P017
Residual Heat Removal Channel A	H23-P018
Residual Heat Removal Div. 2 Channel B	H23-P021
Recirculation Pumps	H23-P022
Drywell Pressure Local Panel A	H23-P057
Drywell Pressure Local Panel B	H23-P058
Main Steam Isolation Valve Leakage Control	H23-P074 Div. 2
Core Spray Local Panel A	H23-P001
Standby Liquid Control	H23-P011
Main Steam Flow A/B	H23-P015
High Pressure Coolant Injection Leak Det.	H23-P016
Core Spray Channel B	H23-P019
Main Steam Flow C/D	H23-P025
High Pressure Coolant Injection	H23-P036
Reactor Core Isolation Cooling Leak Det.	H23-P038 Div. 2(B)
Main Steam Flow A/B	H23-P041
Main Steam Flow C/D	H23-P042
Main Steam Isolation Valve Leakage Con.	H23-P073 Div. 1
High Pressure Coolant Injection Div. 1 A	H23-P034
Reactor Core Isolation Cooling Div. 2 B	H23-P037
SRM/IRM	H23-P030/ 31/3/2/33

NUCLEAR BOILER

Temperature Element	B21-N004
Temperature Element	B21-N010AD
Temperature Element	B21-N014AD
Pressure Switch	B21-N015AD
Temperature Element	B21-N016AD
Temperature Element	B21-N017

Vacuum Switch	B21-N056AD
Temperature Element	B21-N064
Differential Pressure Transmitter	B31-N014CD
Temperature Element	B31-N023AB
Differential Pressure Transmitter	B31-N024AB
Level Switch	C12-N013AD
Level Switch	C12-N013EF
Temperature Switch	C41-N003
Pressure Transmitter	C41-N004
Pressure Indicator	C41-R003
Valve, Guide Tube	C51-J004AE
Miscellaneous Parts	C51-5110001
Pressure Switch	C72-N003AD
Pressure Switch	C72-N005AD
Limit Switch	C72-N006AD
Limit Switch	C72-N008AD
Level Transmitter	E11-N008AB
Temperature Element	E11-N009AD
Differential Pressure Transmitter	E11-N013
Differential Pressure Transmitter	E11-N015A
Differential Pressure Transmitter	E11-N015B
Pressure Switch	E11-N018
Switch	E11-N021AB
Pressure Switch	E11-N022AB
Level Switch	E11-N023AB
Level Switch	E11-N024
Temperature Element	E11-N029AD
Temperature Element	E11-N030AD
Flow Indicating Switch	E11-N033AB
Differential Pressure Transmitter	E21-N003AB
Switch	E21-N006AB
Pressure Switch	E21-N007AB
Flow Meter	E32-N006
Level Switch	E41-N002
Level Switch	E41-N003
Level Switch	E14-N014
Level Switch	E41-N015AB
Level Switch	E41-N018
Temperature Element	E41-N024AB
Temperature Element	E41-N025AH
Temperature Element	E41-N028AB through E41-N030AB
Temperature Indicator	E41-R002
Level Switch	E51-N010
Temperature Element	E51-N011AB
Temperature Element	E51-N021AB
Temperature Element	E51-N022AB
Temperature Element	E51-N023AB

Temperature Element	E51-N025AD through E51-N027AD
Temperature Indicator	E51-R005
Temperature Element	G33-N016AF
Temperature Element	G33-N022AF
Temperature Element	G33-N023AF
Switch	G33-N044A

QUESTION 123.4

Describe the methods and criteria used to determine the acceptability of the original equipment qualification to meet the required response spectra of item 2. (iii). - 123.2 (iii).

RESPONSE:

I. BOP

For cases where the original spectra for which an equipment was qualified enveloped the combined Seismic and hydrodynamic load spectra of Item 123.2 (iii), the equipment is considered qualified. Otherwise (which is true for most cases) the equipment is requalified for the combined spectra to meet the criteria discussed in response to Questions 123.1. These criteria are described in Section 7.1.7 of the Design Assessment Report.

II. NSSS

The methods and criteria used to determine the acceptability of the original equipment qualification may be found in General Electric Company's Proprietary reports: NEDE-24788, "Seismic Qualification Review Team (SQRT) Technical Approach for Re-Evaluation of BWR 4/5 Equipment"; and NEDE-25250 "Generic Criteria For High-Frequency Cutoff of BWR Equipment."

QUESTION 123.5

Describe the methods and criteria used to address the vibration fatigue cycle effects on the affected equipment due to required loading conditions.

RESPONSE:

I. BOP

As described in Subsection 3.7b.3.2 of FSAR, in general, the design of equipment is not fatigue controlled since the number of cycles in an earthquake is low.

For combined Seismic and hydrodynamic loads for equipment qualified by analysis, the fatigue effects are implicitly considered since the stresses due to SRV (which are generally controlling for fatigue) are a small contribution to the overall equipment stresses.

Fatigue effects in BOP equipment qualified by testing are accounted for by repetition of the tests. Typically tests are done for 5 OBE (or 5 upset conditions, i.e., OBE + SRV + LOCA) followed by 1 SSE (or 1 faulted condition, i.e., SSE + SRV + LOCA) in each of front-to-back/vertical and side-to-side/vertical biaxial configurations. In addition, on some selected pieces of equipment, vibratory table testing is carried out for an extended duration of time (such as 30 to 60 minutes) beyond the combined loading test. The input motions for the extended duration tests will be such that the generated test response spectra for any segment of the extended duration tests will envelope the SRV spectra. Furthermore, it will be ascertained that the equipment performs its intended function before, during and after the vibratory table tests. The results of the extended duration tests will be documented in the respective qualification reports.

II. NSSS

Vibration fatigue cycle effects for NSSS equipment designed to ASME code requirements was reviewed at GE by NRC consultants from Battelle Pacific Northwest Laboratories on October 7, 1980. The consultants stated satisfaction with the GE approach which encompasses OBE, SRV, thermal and pressure cycles.

Non-ASME Code components qualified by test address the "strong motion" phase of seismic and SRV dynamic motion sufficient to generate maximum equipment response. These loads are controlling. GE testing generally consists of 5 upset and 1 faulted test of 30 seconds each which is about 50% greater than required to address strong motion vibration.

Non-ASME Code components qualified by analysis generally have not, in the past, had to address vibration fatigue cycle effects. In most cases, such effects are not now part of the qualification record.

QUESTION 123.6

Based on the methods and criteria described in items 4 and 5, provide the results of the review of the original equipment qualification with identification of (1) equipment which has failed to meet the required response spectra and required requalification, and (2) equipment which was found acceptable, together with the necessary information to justify the adequacy of the original qualification.

RESPONSE

I. BOP

For cases where the original seismic reports can be extended to qualify an equipment for combined seismic and hydrodynamic loads by inspection and subsequent concurrence by vendor, such documents form a part of the qualification package. The following pieces of equipment bought under the indicated purchase order (P.O.) fall into this category:

(1) Cooling and chilled water pumps (P.O. #M-327)

(2) Expansion Tanks and Air Separator Tanks (P.O. #M-302)

(3) Nitrogen Gas Accumulators (P.O. #M-156)

The rest of the BOP equipment is being qualified for the criteria described in Section 7.1.7 of the Design Assessment Report. The qualification reports for this equipment will provide the appropriate documentation.

II. NSSS

Refer to the Response to Question 123.3 for the list of equipment reevaluated by GE on the Susquehanna SQRT Program. All of the equipment listed in qualified to SQRT Criteria with the exception of the following:

B211-F022/ F028	MSIV	Data required from vendor
B31-F031/ F032	Gate Valve	Operability deflection analysis required
C12-F009/F012 F011/F012	CRD Valve	Operability deflection analysis required

C41-A003	SLC Accumulator	A/E pipe accelerations required
C41-F004	SLC Explosive Valve	A/E pipe accelerations required
E32-B001	MSIV Heater	Test required
E41-C002	HPCI Turbine	Test required
E51-C002	RCIC Turbine	Analysis of lube oil piping required
F22-E006	Invessel Rack	Analysis required
F22-E009	Def. Fuel Storage Cont.	Analysis required
H12-P608	Power Range Monitoring Cabinet	Test required
H23-P030 -P031 -P032 -P033	SRM/IRM Panels	Test required
163C1158	Flow Transmitter on H23-P074	Test required
272A8005	Switch on H12-P853	Test required
272A8006	Switch on H12-P853	Test required

Information to justify qualification of the equipment selected by the NRC for the Site Audit will be available at the site for NRC inspection. Information to justify qualification of the remainder of the equipment is available for NRC audit at GE-San Jose.

QUESTION 123.7

Describe procedures and schedule for completion of each item identified in item 6.(1) 123.6 (1) that requires requalification.

RESPONSE:

I. BOP

Typically, the qualification program is executed in the following steps.

- o Determine Qualification Awards
  - Request Vendor (or Consultant) Quote
  - Receive and Evaluate Quote
  - Place Purchase Order
- o Perform Qualification
  - Review Test Procedure
  - Review Analysis Methodology
  - Begin Analysis or Testing
- o Final Completion
  - Receive and review Requalification Reports
  - Final Approval of the Report

The schedule for the completion of the qualification program is shown in the attached Table 123.7-1.

II. NSSS

The response to Question 123.6 lists the equipment found by GE to require requalification along with a statement defining the work to be performed. All requalification will be completed on a schedule sufficient to permit NRC review prior to fuel load.

TABLE 123.7-1

SCHEDULE FOR COMPLETION OF EQUIPMENT REQUALIFICATION

SQRT Form No.	Equipment	No. of Items/ 2 Units	Completion Date
E-109-1	4 kV Switchgear	12	3-13-81
E-109-2	4 kV Switchgear Sub-Components	12	5-15-81
E-112	ESW and RHR Pump Motors	8	Complete
E-117-1	480 V Safe-Guard Load Center Unit Substations	8	3-27-81
E-118	480 V Motor Control Centers	24	4-17-81
E-119A-1	Battery Monitors	20	3-27-81
E-119A-2	Battery Fuse Boxes	16	3-27-81
E-119A-3	Battery Chargers	22	3-27-81
E-119BC	24 Vdc, 125 Vdc & 250 Vdc Battery Cells & Racks	8	5-29-81
E-120-1	125 Vdc Distribution Panels	16	3-20-81
E-120-2	24 Vdc Distribution Panels	4	4-10-81
E-121-1	125 V & 250 Vdc Load Centers	12	3-27-81
E-121-2	250 Vdc Control Centers	6	4-10-81
E-135-1	Electrical Penetration (Medium Voltage)	12	5-15-81
E-135-2	Electrical Penetration (Low Voltage)	32	5-15-81
E-136	AC Instrument Transformers	14	3-27-81
E-151	Motor Generator Sets and Control Cabinet	4 Sets	Complete
E-152	Automatic Transfer Switches	8	Complete
E-155	Control Switches	44	6-15-81
J-038A	Field Mounted Electronic Pressure Transmitters	32	Complete
J-03B-1 thru J-03B-14	Panel - Mounted Instruments	242	4th quarter 1981

TABLE 123.7-1

SCHEDULE FOR COMPLETION OF EQUIPMENT REQUALIFICATION (Cont'd.)

SQRT Form No.	Equipment	No. of Items/ 2 Units	Completion Date
J-05A-14, 31,33,37, 10A & B, 43,47,49,92, 93,95 & 97	Control Panels & Devices	31	5-30-81 (panels) 6-15-81 (devices)
J-05B-1	Remote Shutdown Control Panel	1	5-30-81 (panels) 6-15-81 (devices)
J-27	Reactor Coolant Pressure Boundary Leak Detection System	2	Complete (panels) 6-15-81 (devices)
J-31	Annubar Flow Elements	2	Complete
J-59-1 thru J-59-10	RTD's	54	5-22-81
J-65-1 thru J-65-4	Control Valves in Nuclear Service	28	3-27-81
J-65B-1 thru J-65B-11	Control Valves in Nuclear Service	86	3-27-81
J-69-1 & 2	Pilot Solenoid Valves	8	5-15-81
J-69B-1 thru 6	Pilot Solenoid Valves	74	5-15-81
J-70-1	Pressure Regulating Valves	8	5-15-81
J-70-2	Process Solenoid Valves	76	5-15-81
J-92-1 thru J-92-5	Excess Flow Check Valves	238	5-01-81
J-98	Carrier Modulator (Isolator)	-	6-15-81
M-11	ESW Pumps	4	Complete
M-12	RHR Suction Water Pumps	4	Complete
M-22-1 & 2	Reactor Building Cranes	2	4-03-81
M-30 (78 forms)	Diesel Generator	4 Sets	Complete
M-30 (6 forms)	Diesel Generator	4 Sets	2-27-81
M-55	Reactor Vessel Top Head Insulation Support Steel	2	Complete
M-58	Diesel Oil Transfer Pumps	4	Complete

TABLE 123.7-1

SCHEDULE FOR COMPLETION OF EQUIPMENT REQUALIFICATION (Cont'd.)

SQRT Form No.	Equipment	No. of Items/ 2 Units	Completion Date
M-60	Buried Diesel Generator Fuel Oil Storage Tanks	4	3-27-81
M-87-1	Containment Hydrogen Recombiners	8	5-15-81
M-87-2	Hydrogen Recombiner Power Supply	4	Complete
M-90	Fuel Pool Skimmer Surge Tanks	2	4-27-81
M-149	Containment Vacuum Relief Valves	20	5-22-81
M-151	Suppression Pool Suction Strainers	32	Complete
M-156	Containment Nitrogen Gas Accumulators	60	Complete
M-159-1 thru M-159-21	Nuclear Safety & Relief Valves	58	5-01-81
M-160AC	SRV Discharge Line & RHR Relief Valve F055 Discharge Line Vacuum Breakers	68	5-15-81
M-164	CRD Vent Valve Platform	2	Complete
M-192	High Density Spent Fuel Pool Racks	48 Modules	Complete
M-302	Expansion Tanks & Air Separators	4	Complete
M-307-1 thru M-307-3	Centrifugal Fans	6	3-13-81
M-308-1	Vane Axial Fans, Reactor Building	2	5-01-81
M-308-2	Vane Axial Fans, Diesel Generator Building	4	Complete
M-308-3 & 4	Vane Axial Fans, ESSW Pumphouse	8	Complete
M309-1 thru M-309-4	Air Handling Units	12	4-17-81
M-310	Centrifugal Water Chillers	2	5-22-81
M-315	Reactor Building Unit Coolers	24	5-29-81
M-317	Drywell Unit Coolers	12	3-27-81

TABLE 123.7-1

SCHEDULE FOR COMPLETION OF EQUIPMENT REQUALIFICATION (Cont'd.)

SQRT Form No.	Equipment	No. of Items/ 2 Units	Completion Date
M-320-1	Chlorine Detectors	2	6-15-81
M-320-2-1A & 1B	Flow Switches	28	6-15-81
M-320-2-2A	Flow Switches	2	6-15-81
M-320-3	Level Gauge	2	6-15-81
M-320-4	Pressure Differential Switches	4	6-15-81
M-370-5A & 5B	Temperature Switches	24	6-15-81
M-320-6-1A & 1B	Temperature Switches	4	6-15-81
M-320-6-2A	Temperature Switches	4	6-15-81
M-320-6-3A & 7	Temperature Switches	10	6-15-81
M-320-8	Pressure Differential Transmitter	18	6-15-81
M-320-9	Temperature Detector Unit	2	6-15-81
M-320-10	Level Switches	4	6-15-81
M-321-1	Standby Gas Treatment System - Housing	2	2-20-81
M-321-2	Standby Gas Treatment System - Deluge Drain Valves	8	5-01-81
M-321-3	Standby Gas Treatment System - Control Panels	4	3-06-81
M-232C-1	Air Flow Monitoring Unit	1	3-13-81
M-323C-2	SGTS Exhaust Vent Flow Conditioning & Sampling Probe System	1	3-13-81
M-325	High Efficiency Ventilation Filters	2	Complete
M-327-1	Chilled Water Pump	2	Complete
M-327-2	Cooling Water Pump	2	Complete
M-334-1 thru M-334-5	HVAC Control Panels & Devices	12	5-30-81 (panels) 6-15-81 (devices)
M-336A	HVAC Dampers	195 Units	5-08-81
M-362	SGTS Centrifugal Fans	2	Complete

TABLE 123.7-1

SCHEDULE FOR COMPLETION OF EQUIPMENT REQUALIFICATION (Cont'd.)

SQRT Form No.	Equipment	No. of Items/ 2 Units	Completion Date
M-365	Chilled Water Relief Valves	2	5-01-81
P-10A-1	Motor Operated Gate Valves, 600#	5	6-15-81
P-10A-2	Motor Operated Gate Valves, 900#	15	6-15-81
P-10A-3	Motor Operated Globe Valves, 900# & 600#	9	6-15-81
P-10B	Motor Operated Stop Check Valves, 900#	2	6-15-81
P-11A-1	Motor Operated Gate Valves, 900#	2	6-15-81
P-11A-2	Air Operated Testable Check Valves, 900#	2	6-01-81
P-12A-1	Motor Operated Gate Valves, 150#	24	6-15-81
P12A-2	Motor Operated Globe Valves, 300#	11	6-15-81
P12A-3	Motor Operated Gate Valves, 300#	20	6-15-81
P-12A-4	Gear Operated Gate & Globe Valves, 300#	7	6-01-81
P-12B-1	Motor Operated Gate Valves, 150# & 300#	4	6-15-81
P-12B-2	Air Operated Gate Valves, 150#	14	6-01-81
P-12B-3	Gear Operated Gate & Globe Valves, 150#	13	6-01-81
P-14A	Motor Operated Globe Valves, 1500#	1	6-15-81
P-14B	Motor Operated Globe Valves, 1500#	1	6-15-81
P-15A	Motor Operated Globe Valves, 1500#	11	6-15-81
P15B-1	Motor Operated Gate Valves, 1500#	18	6-15-81
P-15B-2	Air Operated Gate Valves, 1500#	2	6-01-81
P-16A-1	Motor Operated Butterfly Valves, 150#	28	6-15-81

TABLE 123.7-1

SCHEDULE FOR COMPLETION OF EQUIPMENT REQUALIFICATION (Cont'd.)

SQRT Form No.	Equipment	No. of Items/ 2 Units	Completion Date
P16A-2	Air Operated Butterfly Valves, 150#	8	6-01-81
P-16A-3	Gear Operated Butterfly Valves, 150#	12	6-01-81
P-17A-1	Motor Operated Gate Valves, 900#	7	6-15-81
P-17A-2	Motor Operated Globe Valves, 900#	1	6-15-81
P-17A-3	Air Operated Testable Check Valves, 900#	2	6-01-81
P-17A-4	Gear Operated Gate Valves, 900#	5	6-01-81
P-17B	Air Operated Testable Check Valves, 900#	2	6-01-81
P-18A	Gear Operated Gate Valves, 150#	1	6-01-81
P-31A	Air Operated Butterfly Valves, 150#	9	6-01-81

TABLE 123.7-1

SCHEDULE FOR COMPLETION OF EQUIPMENT REQUALIFICATION

SQRT Form No.	Eauilsment	No. of Items/ 2 Units	Completion Date
Ξ-109-1	4 kV Switchgear	12	3-13-81
Ξ-109-2	4 kV Switchgear Sub-Components	12	5-15-81
Ξ-112	ESW and RHR Pump Motors	8	Complete
Ξ-117-1	480 V Safe-Guard Load Center Unit Substations	8	3-27-81
Ξ-118	480 V Motor Control Centers	24	4-17-81
Ξ-119A-1	Batterv Monitors	20	3-27-81
Ξ119A-2	Battery Fuse Boxes	16	3-27-81
Ξ-119A-3	Battery Chargers	22	3-27-81
Ξ-119BC	24 Vdc, 125 Vdc & 250 Vdc Battery Cells & Racks	8	5-29-81
Ξ-120-1	125 Vdc Distribution Panels	16	3-20-81
Ξ-120-2	24 Vdc Distribution Panels	4	4-10-81
Ξ-121-1	125 V & 250 Vdc Load Centers	12	3-27-81
Ξ-121-2	250 Vdc Control Centers	6	4-10-81
Ξ-135-1	Electrical Penetration (Medium Voltage)	12	5-15-81
Ξ-135-2	Electrical Penetration (Low Voltage)	32	5-15-81
Ξ-136	AC Instrument Transformers	14	3-27-81
Ξ-151	Motor Generator Sets and Control Cabinet	4 Sets	Complete
Ξ-152	Automatic Transfer Switches	8	Comlslete
Ξ-155	Control Switches	44	6-15-81
I-038A	Field Mounted Electronic Pressure Transmitters	32	Complete
I-03B-1 thru I-03B-14	Panel – Mounted Instruments	242	4th quarter 1981
I-05A-14 ɜ1,33,37, IOA & B ɫ3,47,49,92, ɔ3.95 & 97	Control Panels & Devices	31	5-30-81 (panels) 6-15-81 (devices)
I-05B-1	Remote Shutdown Control Panel	1	5-30-81 (panels) 6-15-81 (devices)
1-27	Reactor Coolant Pressure Boundary Leak Detection System	2	Complete (panels) 6-15-81 (devices)

TABLE 123.7-1

SCHEDULE FOR COMPLETION OF EQUIPMENT REQUALIFICATION

SQRT Form No.	Equipment	No. of Items/ 2 Units	Completion Date
J-31	Annubar Flow Elements	2	Complete
J-59-1 thru J59-10	RTD's	54	5-22-81
J-65-1 thru J-65-4	Control Valves in Nuclear Service	28	3-27-81
J65B-1 thru J65B-11	Control Valves in Nuclear Service	86	3-27-81
J-69-1& 2	Pilot Solenoid Valves	8	5-15-81
J-69B-1 thru 6	Pilot Solenoid Valves	74	5-15-81
J-70-1	Pressure Regulating Valves	8	5-15-81
J-70-2	Process Solenoid Valves	76	5-15-81
J-92-1 thru J92-5	Excess Flow Check Valves	238	5-01-81
J-98	Carrier Modulator (Isolator)	-	6-15-81
M-11	ESW Pumps	4	Complete
M-12	RHR Suction Water Pumps	4	Complete
M-22-1 & 2	Reactor Building Cranes	2	4-03-81
M-30 (78 forms)	Diesel Generator	4 Sets	Complete
M-30 (6 forms)	Diesel Generator	4 Sets	2-27-81
M-55	Reactor Vessel Top Head Insulation Support Seal	2	Complete
M-58	Diesel Oil Transfer Pumps	4	Complete
M-60	Buried Diesel Generator Fuel Oil Storage Tanks	4	3-27-81
M-87-1	Containment Hydrogen Recombiners	8	5-15-81
M-87-2	Hydrogen Recombiner Power Supply	4	Complete
M-90	Fuel Pool Skimmer Surge Tanks	2	4-27-81
M-149	Containment Vacuum Relief Valves	20	5-22-81
M-151	Suppression Pool Suction Strainers	32	Complete
M-156	Containment Nitrogen Gas Accumulators	60	Complete
M-159-1 thru M-159-21	Nuclear Safety & Relief Valves	58	5-01-81

TABLE 123.7-1

SCHEDULE FOR COMPLETION OF EQUIPMENT REQUALIFICATION

SQRT Form No.	Equipment	No. of Items/ 2 Units	Completion Date
M-160AC	SRV Discharge Line & RHR Relief Valve F055 Discharge Line Vacuum Breakers	68	5-15-81
M-164	CRD Vent Valve Platform	2	Complete
M-192	High Density Spent Fuel Pool Racks	48 Modules	Complete
M-302	Expansion Tanks & Air Separators	4	Complete
M-307-1 thru M-307-3	Centrifugal Fans	6	3-13-81
M-308-1	Vane Axial Fans, Reactor Building	2	5-01-81
M-308-2	Vane Axial Fans, Diesel Generator Building	4	Complete
M-308-3 & 4	Vane Axial Fans, ESSW Pumphouse	8	Complete
M-309-1 thru M-309-4	Air Handling Units	12	4-17-81
M-310	Centrifugal Water Chillers	2	5-22-81
M-315	Reactor Building Unit Coolers	24	5-29-81
M-317	Drywell Unit Coolers	12	3-27-81
M-320-1	Chlorine Detectors (Note)	2	6-15-81
M-320-2-1A & 1B	Flow Switches	28	6-15-81
M-320-2-2A	Flow Switches	2	6-15-81
M-320-3	Level Gauge	2	6-15-81
M-320-4	Pressure Differential Switches	4	6-15-81
M-370-5A & 5B	Temperature Switches	24	6-15-81
M-320-6-1A & 1B	Temperature Switches	4	6-15-81
M-320-6-2A	Temperature Switches	4	6-15-81
M-320-6-3A & 7	Temperature Switches	10	6-15-81
M-320-8	Pressure Differential Transmitter	18	6-15-81
M-320-9	Temperature Detector Unit	2	6-15-81
M-320-10	Level Switches	4	6-15-81
M-321-1	Standby Gas Treatment System- Housing	2	2-20-81
M-321-2	Standby Gas Treatment System- Deluge Drain Valves	8	5-01-81

TABLE 123.7-1

SCHEDULE FOR COMPLETION OF EQUIPMENT REQUALIFICATION

SQRT Form No.	Equipment	No. of Items/ 2 Units	Completion Date
M-321-3	Standby Gas Treatment System- Control Panels	4	3-06-81
M-232C-1	Air Flow Monitoring Unit	1	3-13-81
M-323C-2	SGTS Exhaust Vent Flow Conditioning & Sampling Probe System	1	3-13-81
M-325	High Efficiency Ventilation Filters	2	Complete
M-327-1	Chilled Water Pump	2	Complete
M-327-2	Cooling Water Pump	2	Complete
M-334-1 thru M-334-5	HVAC Control Panels & Devices	12	5-30-81 (panels) 6-15-81 (devices)
M-336A	HVAC Dampers	195 Units	5-08-81
M-362	SGTS Centrifugal Fans	2	Complete
M-365	Chilled Water Relief Valves	2	5-01-81
P-10A-1	Motor Operated Gate Valves, 600#	5	6-15-81
P-10A-2	Motor Operated Gate Valves, 900#	15	6-15-81
P-10A-3	Motor Operated Globe Valves, 900# & 600#	9	6-15-81
P-10B	Motor Operated Stop Check Valves, 900#	2	6-15-81
P-11A-1	Motor Operated Gate Valves, 900#	2	6-15-81
P-11A-2	Air Operated Testable Check Valves, 900#	2	6-01-81
P-12A-1	Motor Operated Gate Valves, 150#	224	6-15-81
P-12A-2	Motor Operated Globe Valves, 300#	11	6-15-81
P-12A-3	Motor Operated Gate Valves, 300#	20	6-15-81
P-12A-4	Gear Operated Gate & Globe Valves, 300#	7	6-01-81
P-12B-1	Motor Operated Gate Valves, 150# & 300#	4	6-15-81
P-12B-2	Air Operated Gate Valves, 150#	14	6-01-81
P-12B-3	Gear Operated Gate & Globe Valves, 150#	13	6-01-81
P-14A	Motor Operated Globe Valves, 1500#	1	6-15-81
P-14B	Motor Operated Globe Valves, 1500#	1	6-15-81

TABLE 123.7-1

SCHEDULE FOR COMPLETION OF EQUIPMENT REQUALIFICATION

SQRT Form No.	Equipment	No. of Items/ 2 Units	Completion Date
P-15A	Motor Operated Globe Valves, 1500#	11	6-15-81
P15B-1	Motor Operated Gate Valves, 1500#	18	6-15-81
P-15B-2	Air Operated Gate Valves, 1500#	2	6-01-81
P-16A-1	Motor Operated Butterfly Valves, 150#	28	6-15-81
P16A-2	Air Operated Butterfly Valves, 150#	8	6-01-81
P-16A-3	Gear Operated Butterfly Valves, 150#	12	6-01-81
P-17A-1	Motor Operated Gate Valves, 900#	7	6-15-81
P-17A-2	Motor Operated Globe Valves, 900#	1	6-15-81
P-17A-3	Air Operated Testable Check Valves, 900#	2	6-01-81
P-17A-4	Gear Operated Gate Valves, 900#	5	6-01-81
P-17B	Air Operated Testable Check Valves, 900#	2	6-01-81
P-18A	Gear Operated Gate Valves150#	1	6-01-81
P-31A	Air Operated Butterfly Valves, 150#	9	6-01-81

Note: Chlorine detection system has been removed and detectors have been deleted.

QUESTION 123.8

Describe plans for a confirmatory in-situ impedance test and an in-plant SRV test program or other alternatives to characterize the ability of equipment to accommodate hydrodynamic loading.

RESPONSE:

In situ tests are being performed for the determination of structural dynamic characteristics of the equipment for in-service condition. This in-situ information is being used as supporting evidence for (a) validating a mathematical model for qualification by analysis, or (b) simulating the in-service condition on the vibratory table tests for qualification by testing. The results and the usage of in-situ testing will be described in the respective qualification reports, whenever such tests are performed.

All safety related BOP equipment for Susquehanna project is being qualified for combined seismic and hydrodynamic loads for the criteria described in Section 7.1.7 of DAR.

Susquehanna has no plans to perform an in-plant SRV test for equipment qualifications. An air bubble test was conducted in the suppression pool in an attempt to simulate the effects of an SRV air clearing transient load. The data from this test are being studied in an effort to determine the extent of conservations in the analytical prediction of applied hydrodynamics loads.

QUESTION 123.9

To confirm the extent to which the safety related equipment meets the requirements of General Design Criterion 2, the Seismic Qualification Review Team (SQRT) will conduct a plant site review. For selected equipment, SQRT will review the combined required response spectra (RRS) or the combined dynamic response, examine the equipment configuration and mounting, and then determine whether the test or analysis which has been conducted demonstrates compliance with the RRS if the equipment was qualified by test, or the acceptable analytical criteria if qualified by analysis.

The staff requires that a "Qualification Summary of Equipment" as shown on the attached pages be prepared for each selected piece of equipment and submitted to the staff two weeks prior to the plant site visit. The applicant should make available at the plant site for SQRT review all the pertinent documents and reports of the qualification for the selected equipment. After the visit, the applicant should be prepared to submit certain selected documents and reports for further staff review.

RESPONSE:

Susquehanna SQRT pre-visit information required for the SQRT site review has been submitted for all BOP and NSSS equipment. "Qualification Summary of Equipment" and the pertinent documents, reports, vendor prints and all necessary information as required are available for SQRT review.

QUESTION 130.1

The use of a SRSS methodology to combine primary and secondary stresses is questioned. Provide background and engineering justification for SRSS usage for this application.

RESPONSE:

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

QUESTION 130.2

The use of a low ($\pm$5%) peak broadening factor is contrary to related current practice. Provide detailed engineering justification for this deviation.

RESPONSE:

Please refer to the generic response to NRC Question SEB 2 provided by GE in a May 1, 1978 letter from L. J. Sobon (GE) to J. T. Knight (NRC).

QUESTION 130.3

The usage of SRSS for the combining of closely spaced mode response is not generally considered as good engineering practice.

Provide engineering justification for the adoption of this criteria.

RESPONSE:

Please refer to the generic response to NRC Question SEB 3 provided by GE in a May 1, 1978 letter from L. J. Sobon (GE) to J. T. Knight (NRC).

QUESTION 130.4

Explain the manner by which random bubble frequency samples will be obtained and enveloped.

RESPONSE:

Please refer to Subsection 4.1.3 of the Susquehanna Steam Electric Station, Design Assessment Report.

QUESTION 130.5

The horizontal and vertical design response spectra in the FSAR do not conform to the requirements of Regulatory Guide 1.60. Compare, via graphs, the horizontal and vertical design response spectra proposed in the Susquehanna FSAR and that required by R.G. 1.60. Note that in regards to the vertical design response spectra, the SEB Branch position is that the vertical design response spectra values may be taken as two-thirds the horizontal design response spectra only for sites located in the Western United States. The Susquehanna site does not meet this criteria. Justify the deviations that exist between the horizontal and vertical response spectra provided and those obtained by using R.G. 1.60.

RESPONSE:

Comparisons between the Susquehanna Project design response spectra (as last amended in February, 1972) and those in Regulatory Guide 1.60 (December, 1973) may be found in Figures 3.7b-102, 3.7b-103, 3.7b-104 and 3.7b-105, and described in FSAR Subsection 3.7b.1.1. These figures show that the Regulatory Guide Spectra are generally higher than the project design spectra.

In the report WASH-1255 (Ref. 1) N. M. Newmark has shown based upon 14 strong motion records that the ratio of vertical to the horizontal ground acceleration is 2/3 on an average. Although only 3 of 14 earthquake records considered were on rock, the ratio of the vertical to the horizontal accelerations is less in rocks than alluvium. From their additional research based on the analysis of 30 vertical recordings made on "hard" or rock sites, the authors P. C. Rizzo, D. E. Shaw, and M. D. Snyder report (Ref. 2) that the ratio of 2/3 between the vertical and horizontal accelerations is conservative. Their study which also includes the sites located in eastern United States, such as Blue Mountain Lake, New York, show that the provisions of the Regulatory Guide 1.60 are overly conservative for "hard" or rock sites. It is stated in their report that the requirement of the Regulatory Guide 1.60 envelopes both rock and soil sites.

All principal Category I structures of the Project are founded on competent rock and have been designed for a maximum SSE ground acceleration of 0.1g and 0.067g in the horizontal and vertical directions respectively. Engineered Safeguards and Service Water (ESSW) Pumphouse and Spray Pond which are founded on soil have the ground acceleration levels in both directions increased by 50%.

For discussions of additional conservatism in maximum earthquake potential and safe shutdown earthquake, please refer to Sections 2.5.2.4 and 2.5.2.6 of the FSAR.

References:

(1) "A Study of Vertical and Horizontal Earthquake Spectra." USAEC Contract No. AT(49-5)-2667, WASH-1255. N. M. Newmark Consulting Engineering Services, Urbana, Illinois, April 1973.

(2) "Vertical Seismic Response Spectra," P. C. Rizzo, D. E. Shaw, and M. D. Snyder, Journal of Power Division, ASCE, January 1976, pp. 121-141.

QUESTION 130.6

The damping value for welded steel structures is given as 5 percent of critical for the SSE condition in FSAR Table 3.7b-2. Regulatory Guide 1.61 states that this value should not exceed 4 percent of critical. Justify using the less conservative value.

RESPONSE:

Table C.2.1 of the PSAR provides for a damping value of 5.0% of critical for the welded steel structures at a stress level at or just below the Yield Point. This was based on a technical paper by Dr. N. M. Newmark, cited as Reference 3.7b-2 in the FSAR. Since the stress levels for SSE conditions are allowed close to the yield point, the damping value of 5.0% for SSE has been considered appropriate and hence, has been provided in Table 3.7b-2. Therefore, the above criteria for the damping value is used for the design of the structures with appropriate design margins.

QUESTION 130.7

SRP Section 3.7.2.1 requires that a sufficient number of modes in the dynamic model of the structures must be considered, so that the inclusion of additional modes does not result in more than a 10 percent increase in the response. Show that this requirement has been taken into consideration in the design. If it has not been considered, justify your method in light of the 10 percent requirement.

RESPONSE:

As described in Subsection 3.7b.2.1 the Seismic analysis of the Seismic Category I Structures considers all modes whose frequencies are up to 33 cps. However, if a structure has only one or two modes with a natural frequency below 33 cps, then the three lowest modes are used. For example, see Table 3.7b-5 which lists the natural frequencies used for the Vertical Seismic analysis of the containment. If a structure has three or less degrees of freedom, then all modes are considered in the analysis. The design spectra for the project (Figures 3.7b-1 and 3.7b-2) show that the maximum spectral accelerations occur for frequencies between 2 cps and 6.67 cps, and reach an asymptotic value of ground acceleration of 0.05 g for OBE and 0.10g for SSE.

Analysis shows that the total effective mass of all the modes included is more than 90% of the total actual mass. From this discussion, it can be concluded that sufficient number of modes have been considered to satisfy the SRP requirements.

QUESTION 130.8

Describe the method employed for system/subsystem decoupling.

RESPONSE:

The procedure used for modeling considers Seismic Category 1 structures to be "Seismic Systems," whereas Seismic Category 1 piping systems, equipment and the components are considered "Seismic Sub-systems." As described in Subsection 3.7b.2.3, all equipment, components and piping systems are lumped into the supporting structure mass except for the reactor vessel, which is analyzed using a coupled model of the containment structure and the reactor vessel (See Figures 3.7b-7 and 3.7b-8). It is described in Section 3.2 of reference 3.7b-3 that any equipment, a component or a piping system is usually lumped into the Supporting Structure mass if its estimated mass is less than one-tenth (1/10) that of the supporting mass or, for supporting structures having continuous mass distributions, 0.03 of the fundamental mode effective mass. This equipment, component or piping system is later analyzed using response spectra generated at the supporting level.

Subsection 3.7b.2.3 has been revised accordingly.

QUESTION 130.9

The SRP defines closely spaced modes as modes with frequencies within 10 percent of each other, while the FSAR defines them as being within 0.5 cps of each other. Taking, for example, 15 cps the SRP would consider closely spaced modes to lie in the range of 13.5 cps to 16.5 cps. The FSAR, on the other hand, only admits those in the 14.5 cps to 15.5 cps range.

Justify, using the less conservative approach.

RESPONSE:

See revised Subsection 3.7b.2.7.

QUESTION 130.10

The loading combinations and allowable stress limits for the concrete containment must meet the provisions of SRP section 3.8.1 which reference the ASME Boiler and Pressure Vessel Code, Section III, Division 2, April 1973 issue entitled, "Proposed Standard Code for Concrete Reactor Vessels and Containment." There appear to be deviations between the ASME Code Table CC-3200-1, as modified on SRP page 3.8.1-6, and the load given in FSAR Table 3.8-2. Looking first at the loads themselves, Table 3.8-2 lacks "P_v," subatmospheric minimum pressure load. Also, it is not clear whether the FSAR expression "$T_o + T_a$" is equivalent to "$T_a + R_a$" as defined in the SRP. Also, is "T_o" from the FSAR equal to "$T_o + R_o$" as defined in the SRP? In comparing the load combinations the "construction" and "extreme environmental" loadings are absent in Table 3.8-2. Clarify these discrepancies.

Table CC-3200-1 of the ASME Code is based on working stress criteria as indicated in SRP Section 3.8.1.11.5, while FSAR Table 3.8-2 is based on ultimate strength design. Assess the extent to which the containment design satisfies the working stress criteria.

RESPONSE:

The containment was analyzed for the external pressure load, "P_v," acting alone. The magnitude of this load is defined in Subsection 3.8.1.3.2.7. Since this load is small, it may be combined with other loads without affecting the design. Table 3.8-2 has been revised accordingly to include "P_v."

"T_o" and "T_a" as used include piping loads during operating and accident conditions respectively. Therefore, "$T_o + T_a$" as defined in the FSAR is equivalent to "$T_a + R_a$" as defined in the SRP. Also, "T_o" as defined is equivalent to "$T_o + R_o$" as defined in the SRP.

See Subsection 3.8.1.5.1.1 for the discussion regarding the extent to which the containment design satisfies the working stress criteria.

QUESTION 130.11

Explain the apparent contradiction in the statements concerning tangential shears on FSAR pages 3.8-11 and 3.8-13.

Page 3.8-11 states that "Tangential shears caused by seismic loads are totally resisted by helical reinforcing bars and concrete in compression." Page 3.8-13 indicates, however, that tangential shear is not permitted. Note that tangential shear stress is governed by subsection CC-3400 of the ASME-ACI 359 Code with exceptions listed in SRP Section 3.8.1. II.5. In those exceptions, Vc is increased from 40 psi to 60 psi for the 7th combination of Table CC-3200-1.

RESPONSE:

Seismic tangential shears are totally resisted by helical reinforcing bars and concrete acting together as a truss. The reinforcing bars resist diagonal tension and the concrete resists diagonal compression. Although SRP Section 3.8.1. II.5 allows concrete to resist tangential shear, the helical reinforcing bars were designed assuming no tangential shear was taken by the concrete (as stated in Subsection 3.8.1.5.1.2 a) 4)).

Subsection 3.8.1.4.2 has been revised to clarify the design procedure.

QUESTION 130.12

Provide, in tabular form, the concrete compressive strengths used for the containment including its interior structures, "other Category I structures" of Section 3.8.4, and their associated foundation mats.

RESPONSE:

Table 3.8-11 is added, and Subsections 3.8.1.6.1, 3.8.3.6.1, 3.8.4.6.1 and 3.8.5.6 are revised accordingly.

QUESTION 130.13

The loading combinations of FSAR Table 3.8-3 for the drywell head assembly, equipment hatches, personnel lock, suppression chamber access hatches, and CRD removal hatch are not in agreement with those of SRP Section 3.8.2.II.3.b.

For example, Table 3.8-3 does not contain the contributions from "T_e," "R_e," or "P_e" as defined in the SRP. Also, the allowable stresses are as given in SRP Table 3.8.2-1 based on subsection NE of the ASME Code, Section III, Division 2. FSAR Table 3.8-3, however, bases allowable stresses on subsection NB of the Code. Address the load combinations and stress limits of the SRP and discuss the degree of conservatism that exists between the SRP guidelines and FSAR Table 3.8-3.

RESPONSE:

Table 3.8-3a provides a comparison of FSAR and SRP load combinations and allowable stresses for ASME Class MC components. Subsection 3.8.2.3.1 has been revised accordingly.

QUESTION 130.14

The drywell floor, like the concrete containment, must meet the provisions of the ASME "Proposed Standard Code for Concrete Reactor Vessels and Containment," April 1973 issue. This includes the load combinations of Table CC-3200-1, with exceptions given in SRP Section 3.8.3.3, pg. 3.8.3-16, and the allowable stress limits of Subsection CC-3400 with exceptions on SRP page 3.8.3-22. In a similar manner to Question 130.10 assess the degree to which the drywell floor satisfies the loading conditions and stress allowables of the SRP.

RESPONSE:

For a complete response to this question, see the response to Question 130.10.

The allowable stress criteria for the drywell floor, as given in Subsection 3.8.1.5.1.2, meets the provisions of ASME Subsection CC-3400 with exceptions on SRP page 3.8.3-22.

QUESTION 130.15

There appear to be three deviations in the load combinations for the reactor pedestal between those in SRP Section 3.8.3.3, page 3.8.3-14 and those provided in FSAR Table 3.8-2: The "Normal" and "Normal/Severe" loadings provided in the FSAR does not meet our requirements. The following two loadings should be addressed instead:

a) $1.4D + 1.7L + 1.9E = U$
b) $(0.75)\ (1.4D + 1.7L + 1.9E + 1.7T_o + 1.7R_o) = U$

Address the two load cases by showing that they will not cause overstress in the existing design.

Note that, as stated in Question 130.10, it is assumed that "T_o" from the FSAR equals "$T_o + R_o$" from the SRP, and "$T_o + T_a$" from the FSAR equals "$T_a + R_a$" from the SRP.

RESPONSE:

The reactor pedestal has been analyzed for the two (2) additional load combinations listed above without any overstress occurring. Refer to revised Subsections 3.8.3.3.1 and 3.8.3.3.2, revised Table 3.8-2, and new Table 3.8-2a.

Refer to the response to Question 130.10 for a clarification of loads "T_o", "T_a", and "R_a".

QUESTION 130.16

Indicate whether the reactor shield wall is designed using composite steel-concrete design. If so, describe the provisions made to allow for adequate shear transfer.

RESPONSE:

The concrete in the reactor shield wall is used for radiation shielding only and is not relied upon as a structural element. The inner and outer steel plates resist all structural loadings without any assistance from the concrete; i.e., composite action between steel and concrete is not assumed.

Subsection 3.8.3.1.3 has been revised to include this information.

QUESTION 130.17

The qualitative statement that the abnormal loadings govern the design of the suppression chamber columns because they include the design basis accident pressure load does not demonstrate that our requirements for load combinations are met by FSAR Table 3.8-5. State whether the loading combinations (along with allowable stresses) given below (SRP page 3.8.3-15) are satisfied.

Allowable Stress	Load Combination
Y	$1.7D + 1.7L + 1.7E$
Y	$1.3\ (D + L + E + T_o + R_o)$
.9Y	$D + L + T_o + R_o + E'$

RESPONSE:

The suppression chamber columns have been analyzed for the three (3) additional loading combinations listed above without any overstress occurring.

Table 3.8-5 has been revised to include these additional load combinations.

QUESTION 130.18

The seismic loads due to the dead weight of the drywell platforms has been ignored, although seismic reaction loads of piping and equipment supported by the box girders of the platforms is included.

Justify the omission of seismic loads due to dead weight by comparing the masses (or weights) of the piping and equipment with that of the drywell platforms themselves. Also, clarify whether the seismic input for the piping and equipment is the SSE.

RESPONSE:

The drywell platforms are not designed for seismic loads due to their own dead weight for the following reasons:

1. For the box beams, seismic loads due to the dead weight of the beams are insignificant relative to the pipe rupture loads. Seismic loads on the box beams due to their own dead weight are less than 10 kips. Equivalent static pipe rupture loads on the box beams range from approximately 350 to 850 kips depending on the pipe diameter.

2. For the framing beams, seismic loads due to dead weight of the beams are small since these beams are laterally braced by other framing beams and by the grating.

The omission of seismic loads on the drywell platforms due to their own weight is based upon the above reasons and not on the relative weights of the platforms and the piping and equipment supported by the platforms.

The seismic input for the piping and equipment is the SSE.

Subsection 3.8.3.3.5 has been revised accordingly.

QUESTION 130.19

Explain the basis of the load combination for reinforced concrete in Table 3.8-8 for the reactor building, which applies to "Structural Elements Carrying Mainly Seismic Forces." It implies that if seismic loads exceed a certain proportion of the total then that load combination is used in lieu of the others.

RESPONSE:

Table 3.8-8 has been revised to clarify the design procedure. For response see revised Table 3.8-8.

QUESTION 130.20

The Susquehanna FSAR Section 3.7b.2.1 indicates that both a flexible base model and fixed base model were utilized for the seismic analysis of the containment building. Discuss and explain the rationale for using two different models for the seismic analysis. Demonstrate the equivalency of the two models by comparing their dynamic characteristics on the results from the two analyses.

RESPONSE:

For response see Subsection 3.7b.2.1.1.

QUESTION 130.21

In Torsional Analysis of Diesel Generator Building and ESSW pumphouse: Justify the use of static analysis for a dynamic phenomenon.

RESPONSE:

See Subsections 3.7b.2.11.1 and 3.7b.2.11.2 for response.

TABLE 130.21-2
TABLE 130.21-3

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TABLE 130.21-2
TABLE 130.21-3

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QUESTION 130.22

Explain why the analysis for the torsional effect was not done for the Reactor Building.

RESPONSE:

See Subsection 3.7b.2.11.2 for response.

QUESTION 130.23

In Figure 7-6, which shows downcomer bracing system details, it appears that the bracing is welded to the liner plate through the use of an embedded plate without any anchorage to the containment concrete wall. Since the steel liner plate is not a structural component, indicate how the pulling forces from the bracing can be resisted and how the leaktight integrity of the liner can be maintained.

RESPONSE:

Downcomer bracing forces are resisted by embedded anchorages in the containment concrete wall. This design assures the leaktight integrity of the liner plate is maintained.

Question 130.24

In your response to Question 2, you indicated that Sm is the allowable stress as specified in UBC. For extreme and/or abnormal loading combinations, you increase the allowable stress by a factor of 1.67, which is in conformance with the practice of SRP Sections 3.8.3 and 3.8.4, for reinforced concrete structures. However, concrete masonry walls are quite different from reinforced concrete walls, particularly the unreinforced ones, the use of such a practice may not result in an adequate design. Depending on the types of stress, that is, tensile, shearing or axial compressive, the factor may vary from 0 to 2.5 (see enclosure 2). Specify the masonry design strength f'_m used in Susquehanna masonry walls and the allowable values for all types of stresses.

Response:

Masonry walls are described in Section 3.8.

In accordance with the meeting with NRC's Structural Branch of May 29, 1981, and subsequent phone call of June 2, 1981, PLA-831 (Curtis to Schwencer dated June 3, 1981) transmitted the requested information.

TABLE 130.24-1
Page 1 and 2

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Question 130.25

In the note to you response to Question 2, you stated that the allowable shear or tension between masonry block and concrete or grout infill is considered to be equal to three percent of the compressive strength of the block. The allowable shear or tension as specified by you is in the staff's opinion too high. To specify the allowable shear or tension of the vertical joint between why these in terms of the compression strength of the block is in the first place unconservative and the use of seemingly low percentage of 3% may actually result in an allowable shearing stress greater than its corresponding strength. Therefore, revision of the stress criterion is required.

Response:

Masonry walls are described in Section 3.8.4.

In accordance with the meeting with NRC's Structural Branch of May 29, 1981, and subsequent phone call of June 2, 1981, PLA-831 (Curtis to Schwencer dated June 3, 1981) transmitted the requested information.

Question 130.26

In your response to Question 4: (1) It is indicated that response spectrum method is used for the dynamic analysis of the concrete masonry walls. However, there is no mention as to which of the response spectra is used, upper floor or lower floor response spectrum or the average of the two. It is required that an upper bound envelope of the individual floor is used. (2) Through the use of ACI-318 formula the cracking of concrete masonry wall is considered. The use of such a formula is questionable in view of the fact that in a concrete masonry wall the weakest section is the bed joint and the modulus of rupture is equal to that of neither the concrete block nor the mortar.

Indicate how the modulus of rupture is established in your computation.

Response:

Masonry walls are described in Section 3.8.

In accordance with the meeting with NRC's Structural Branch of May 29, 1981, and subsequent phone call of June 2, 1981, PLA-831 (Curtis to Schwencer dated June 3, 1981) transmitted the requested information.

Question 130.27

In response to Question 5, it is stated that when the design stresses of masonry walls exceed the allowable stresses, fixes are designed such that the criteria is satisfied.

Indicate the number of walls where such fixes are needed and provide examples.

Response:

Masonry walls are described in Section 3.8.

In accordance with the meeting with NRC's Structural Branch of May 29, 1981, and subsequent phone call of June 2, 1981, PLA-831 (Curtis to Schwencer dated June 3, 1981) transmitted the requested information.

TABLE 130.27-1

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Question 130.28

Provide justification for any deviation from the attached staff's interim criteria in you design and analysis of the masonry walls.

RESPONSE:

Masonry walls are described in Section 3.8.

In accordance with the meeting with NRC's Structural Branch of May 29, 1981, and subsequent phone call of June 2, 1981, PLA-831 (Curtis to Schwencer dated June 3, 1981) transmits the requested information.

TABLE 130.28-1
TABLE 130.28-2

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TABLE 130.28-1
TABLE 130.28-2

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