NORTHEAST UTILITIES

THE CONNECTICUT LIGHT AND POWER COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY
HOLYOKE WATER POWER COMPANY
NORTHEAST UTILITIES SERVICE COMPANY
NORTHEAST NUCLEAR ENERGY COMPANY

General Offices • Selden Street, Berlin, Connecticut

P.O. BOX 270
HARTFORD, CONNECTICUT 06141-0270
(203) 666-6911

February 6, 1984

Docket No. 50-423
B11026

Director of Nuclear Reactor Regulations
Mr. B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

References:

1. J. H. Bickel letter to Abel Garcia, dated November 9, 1983.
2. W. G. Counsil letter to B. J. Youngblood, dated January 10, 1984.
3. B. J. Youngblood letter to W. G. Counsil, dated January 6, 1984.

Gentlemen:

Millstone Nuclear Power Station, Unit No. 3
Probabilistic Safety Study (PSS)

Reference 1 and 2 represent Northeast Nuclear Energy Company (NNECO) response to questions previously raised by the Commission and their Consultants. In reference 3, the Commission requested NNECO submit additional information which resulted from the staffs preliminary review of the Millstone Probabilistic Safety Study. Enclosed please find a documentation of all questions posed to NNECO in the above Reference 3, along with our formal responses herein. We trust you will find this information fully responsive.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

W. G. Counsil

W. G. Counsil
Senior Vice President

Boo1
1/1

STATE OF CONNECTICUT)
)ss. Berlin
COUNTY OF HARTFORD)

Then personally appeared before me W. G. Counsil, who being duly sworn, did state that he is Senior Vice President of Northeast Nuclear Energy Company, Licensee herein, that he is authorized to execute and file the foregoing information in the name and on behalf of the Licensee herein and that the statements contained in said information are true and correct to the best of his knowledge and belief.

Lorraine J. D'Amico
Notary Public

My Commission Expires March 31, 1988

Question 720.1: The analysis assumes that if HPR or LPR is available, containment spray recirculation is not required. Previous PRAs have assumed that cooling the core during recirculation does not in and of itself provide cooling of the containment. Thus, containment failure can result followed by a loss of core recirculation due to sump flashing and pump cavitation. What is the justification for this study not considering the need for CSR in sequences where HPR/LPR succeeds? The specific sequences of concern are those which dump most of the latent/decay heat to the containment (i.e., large and medium LOCAs, small LOCAs without AFWS, transients with bleed and feed).

Response: A formal technical response to this question has already been provided on Novemb[illegible], 1983 in Reference 1.

Question 720.2: The analysis assumes that HPI is capable of providing sufficient injection during large LOCAs. Previous PRAs have considered HPI to be incapable of this because of considerations of flow capacity and/or concerns that the pumps are susceptible to pump runout when pumping against very low pressures. What is the justification for eliminating these concerns at Millstone 3?

Response: A formal technical response to this question has already been provided on November 9, 1983 in Reference 1.

Question 720.3: Provide a summary of Westinghouse Calc Note CN-PRA-83-022 which identifies computer codes used, key assumptions and transient plots used in the analysis.

Response: Westinghouse Calc Note CN-PRA-83-022 provides the necessary calculations to assess the reactor coolant system response to postulated small and medium break loss of coolant accidents for which only minimum safety injection capability is available. The results of these analyses were utilized along with other similar analyses to determine system success criteria for the Millstone Unit 3 PRA.(1) This calc note analyzed breaks of 2- and 6-inch diameters. For both breaks, the location was assumed to occur at the RCS cold leg which is the limiting break location for Westinghouse PWRs.(1)

The NOTRUMP(2) computer code was used to perform the necessary calculations. NOTRUMP is a generalized one-dimensional network code which is used to model operational and severe transients for Westinghouse PWRs. An input model was developed for Millstone Unit 3 and a steady state full power simulation was established and referenced. The two LOCA events analyzed were: Case A models a 2-inch diameter cold leg break from full power conditions and Case B models a 6-inch diameter cold leg break from full power conditions. In both of these cases, it was assumed that no feedwater (normal or auxiliary) was available. This assumption has minimal impact on system response. The minimum safety injection flow to the RCS for both cases was approximately 75% of the capacity of one Millstone Unit 3 high head safety injection pump. This flow rate was chosen to bound the success criteria for Millstone which uses a combination of high head safety injection and charging pumps to accomplish high pressure injection. One charging pump provides approximately 75% of the flow of one SI pump in the pressure range of 0-1200 psig.

Case A calculations were carried out to 2000 seconds of transient time. While this particular calculation was not carried out to a time when the system is in a long term stable condition, the calculation was taken out far enough to show that the trend of the transient would not result in core damage/melt.

---

1. WCAP 9600, Report on Small Break Accidents for Westinghouse NSSS Systems, June, 1979.
2. WCAP 8913, Rev. 3 "NOTRUMP - a Nodel Transient Model Boiler, Steam Generator, and General Network Code, P. E. Mezer, G. H. Frick, Jan. 29, 1982. Proprietary Class, II.

Case B calculations were carried out to 1000 seconds which was adequate to show the establishment of a long-term stable condition. This calculation shows that the minimum safety injection flow and accumulator flow is sufficient to maintain vessel mixture level above the core and prevent core melt.

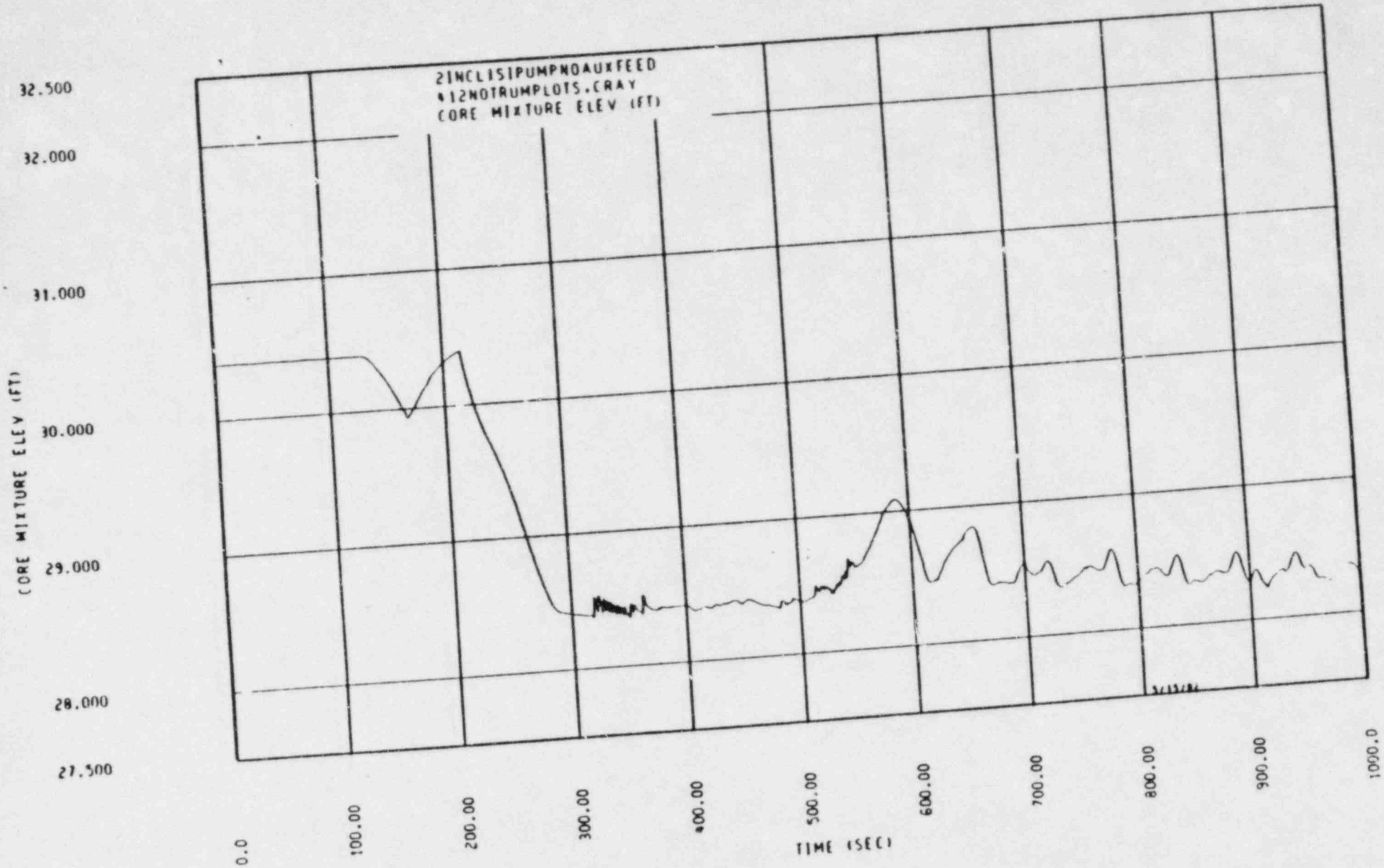
These two analyses show that while the flow minimum safety injection flow cannot maintain the RCS inventory above the break location, the flow is sufficient to maintain the vessel mixture level core the core. These conclusions were consistent with conclusions reached for similar LOCA analyses.(1) A detailed description of small break LOCA phenomena can be found in WCAP 9600. The transient time plots for key plant parameters for both cases are provided herein.

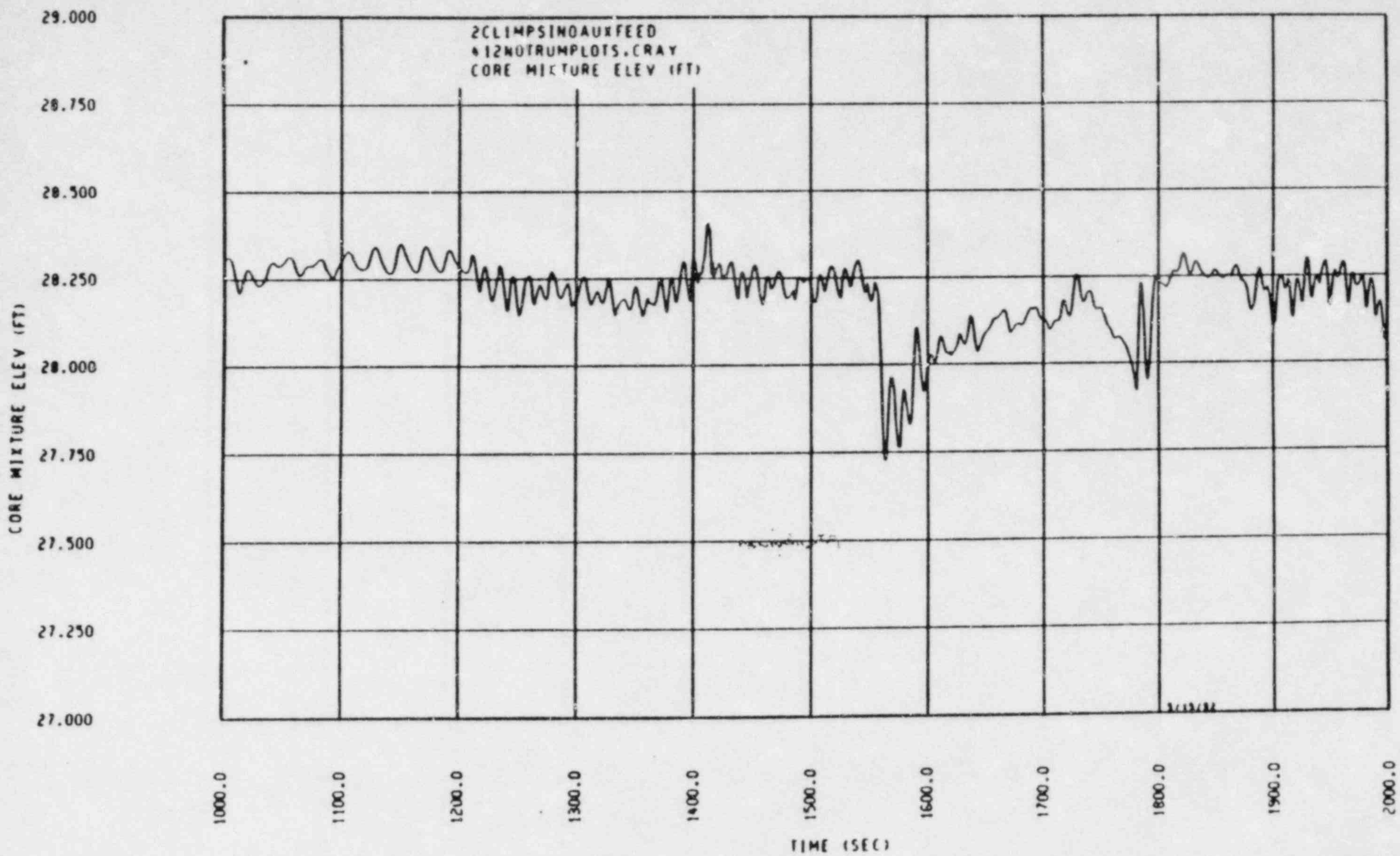
CASE A

2-INCH COLD LEG BREAK PLOTS

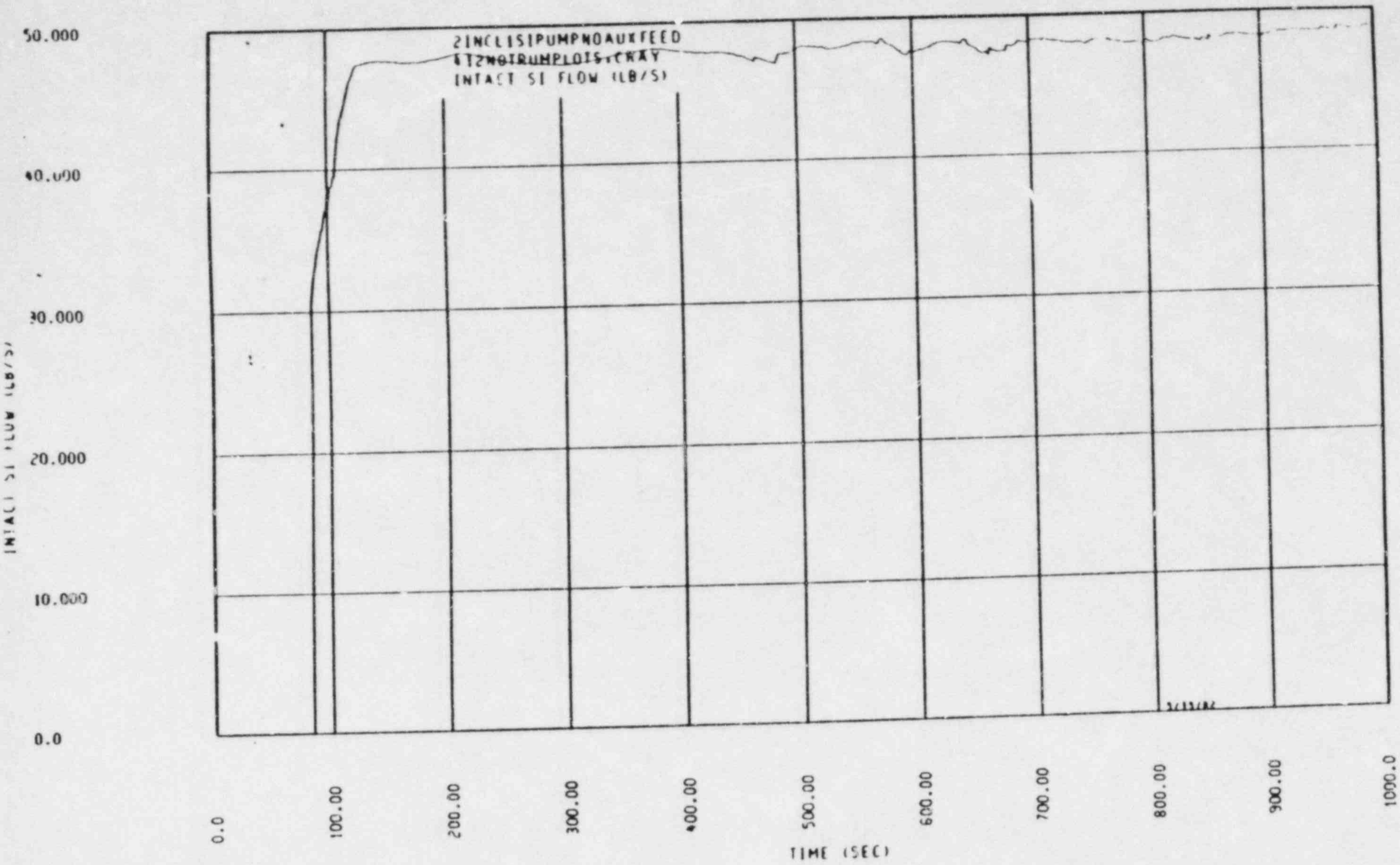
2INCLISIPUMPNOAUXFEED
412NOTRUMPLOTS.CRAY
CORE PRESSURE (PSIA)
2500.0
2000.0
1500.0
1000.0
500.00
0.0
CORE PRESSURE (PSIA)
0.0
100.00
200.00
300.00
400.00
500.00
600.00
700.00
800.00
900.00
1000.0
TIME (SEC)


2CLIMPSINOAUXFEED
412NOTRUMPLOTS.CRAY
CORE PRESSURE (PSIA)
CORE PRESSURE (PSIA)
1200.0
1175.0
1150.0
1125.0
1100.0
1075.0
1050.0
1025.0
1000.0
1000.0
1100.0
1200.0
1300.0
1400.0
1500.0
1600.0
1700.0
1800.0
1900.0
2000.0
TIME (SEC)

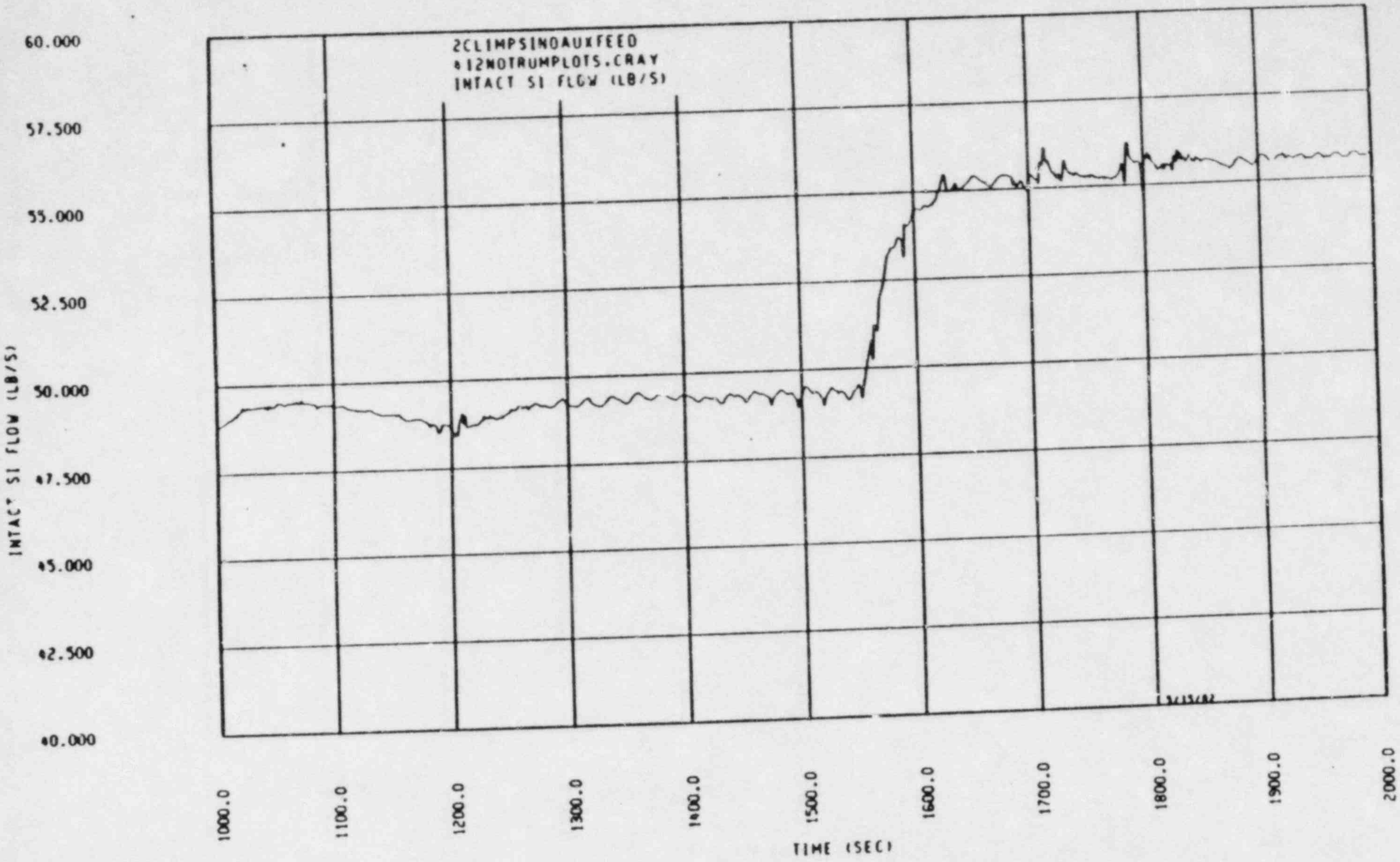

2INCLISIPUMPNOAUXFEED
412NOTRUMPLOTS.CRAY
CORE MIXTURE ELEV (FT)
32.500
32.000
31.000
30.000
29.000
28.000
27.500
CORE MIXTURE ELEV (FT)
0.0
100.00
200.00
300.00
400.00
500.00
600.00
700.00
800.00
900.00
1000.0
TIME (SEC)

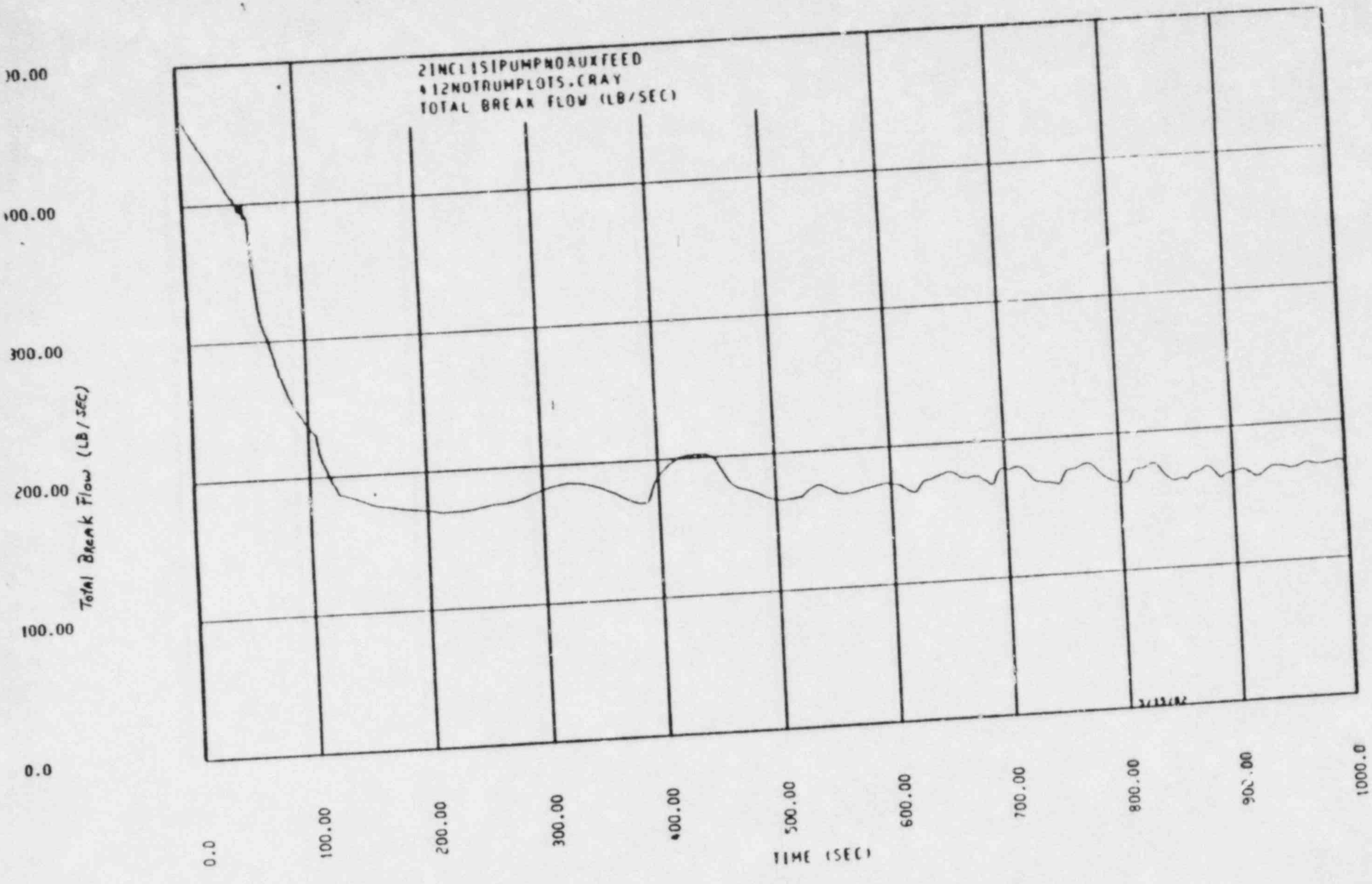


2CLIMPSINOAUXFEED
412NOTRUMPLOTS.CRAY
CORE MIXTURE ELEV (FT)
29.000
28.750
28.500
28.250
28.000
27.750
27.500
27.250
27.000
CORE MIXTURE ELEV (FT)
1000.0
1100.0
1200.0
1300.0
1400.0
1500.0
1600.0
1700.0
1800.0
1900.0
2000.0
TIME (SEC)



2INCLISIPUMPNOAUXFEED
412NOTRUMPLOTS.CRAY
INTACT SI FLOW (LB/S)
50.000
40.000
30.000
20.000
10.000
0.0
INTACT SI FLOW (LB/S)
0.0
100.00
200.00
300.00
400.00
500.00
600.00
700.00
800.00
900.00
1000.0
TIME (SEC)

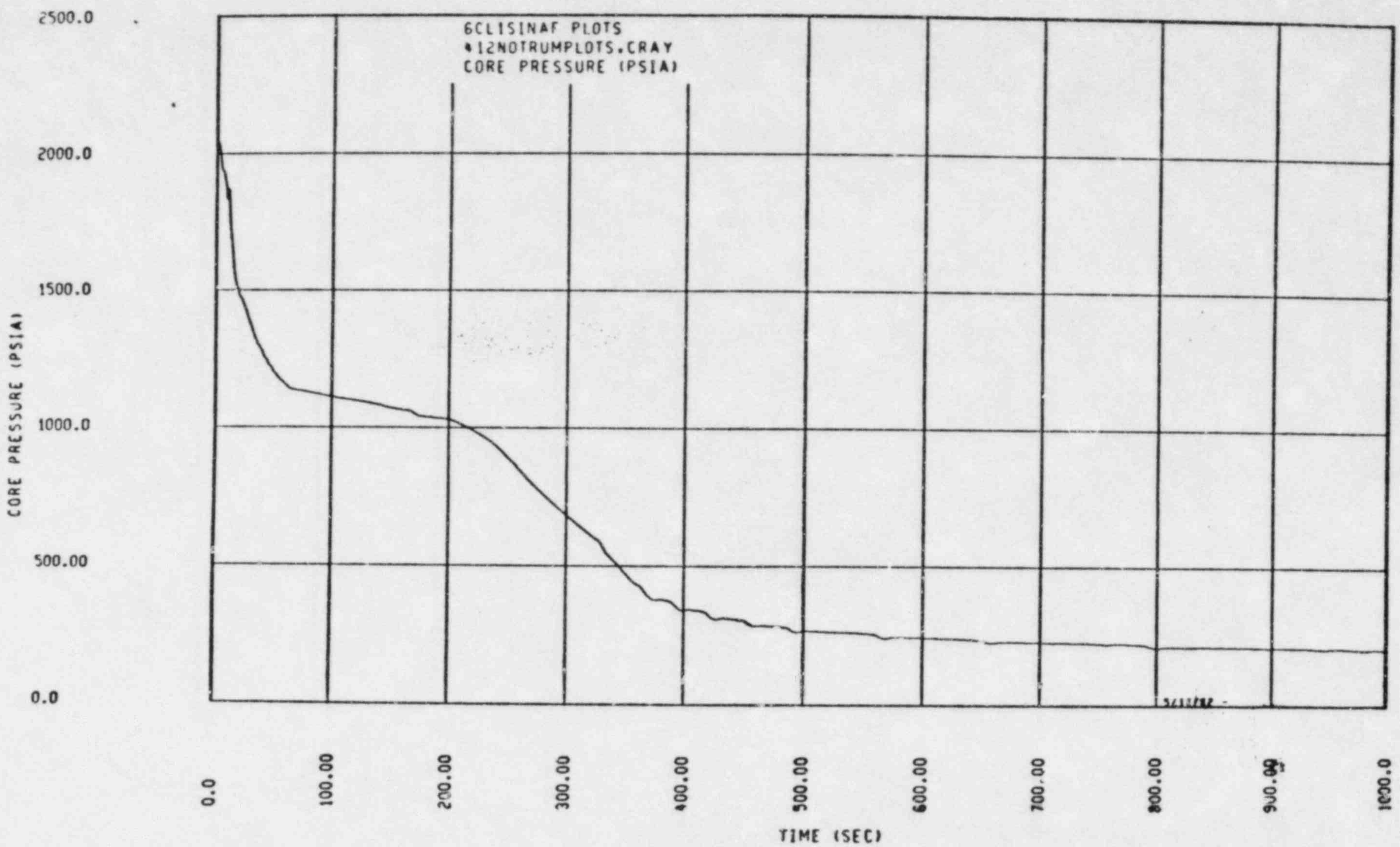

2CLIMPSINOAUXFEED
&12NOTRUMPLOTS.CRAY
INTACT SI FLOW (LB/S)
60.000
57.500
55.000
52.500
50.000
47.500
45.000
42.500
40.000
INTACT SI FLOW (LB/S)
1000.0
1100.0
1200.0
1300.0
1400.0
1500.0
1600.0
1700.0
1800.0
1900.0
2000.0
TIME (SEC)


21NCL1SIPUMPNOAUXFEED
412NOTRUMPLOTS.CRAY
TOTAL BREAK FLOW (LB/SEC)
)0.00
400.00
300.00
200.00
100.00
0.0
Total Break Flow (LB/SEC)
0.0
100.00
200.00
300.00
400.00
500.00
600.00
700.00
800.00
900.00
1000.0
TIME (SEC)

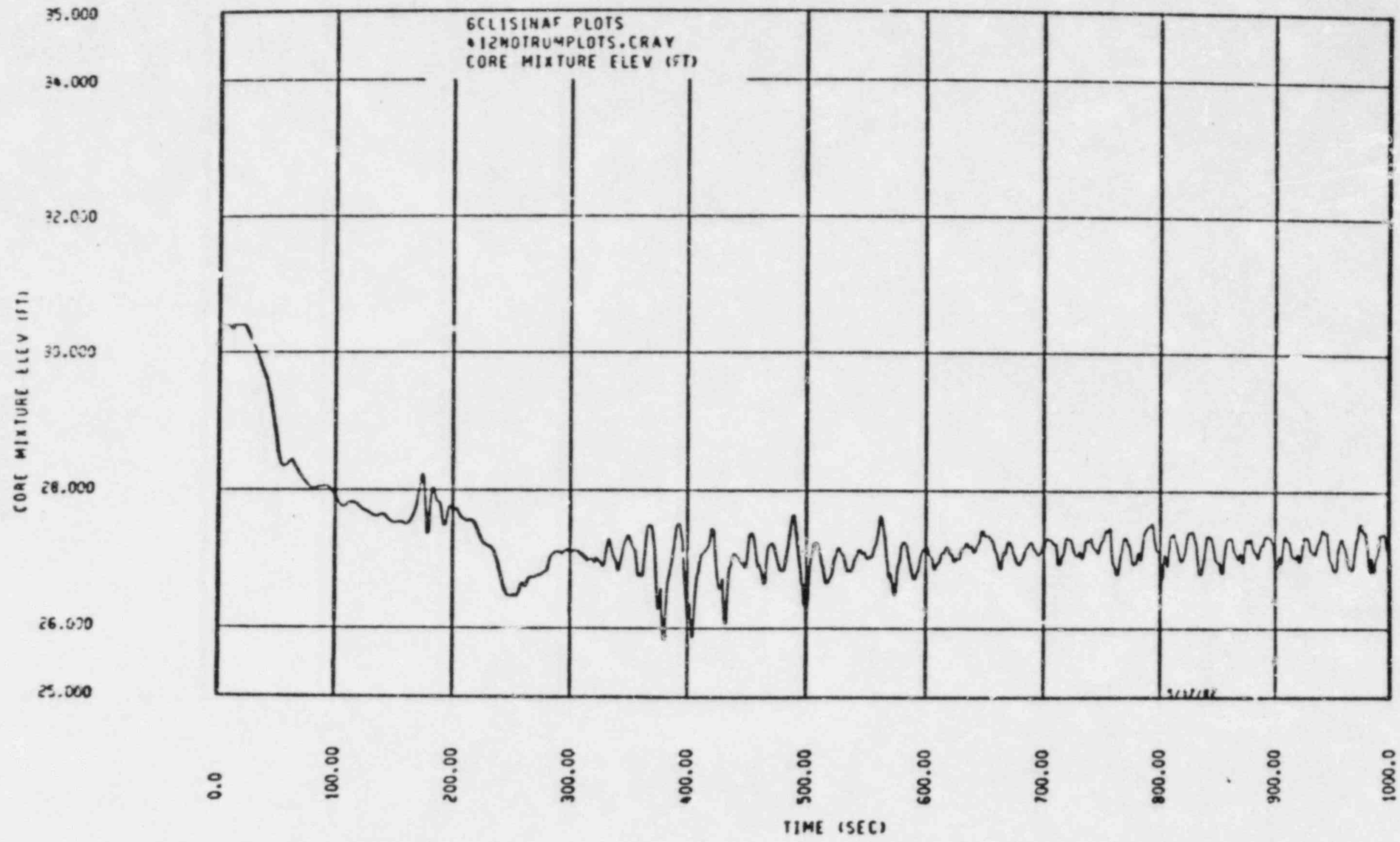

2CLIMPSINOAUXFEED
412NOTRUMPLOTS.CRAY
TOTAL BREAK FLOW (LB/SEC)
TOTAL BREAK FLOW (LB/SEC)
200.00
180.00
160.00
140.00
120.00
100.00
1000.0
1100.0
1200.0
1300.0
1400.0
1500.0
1600.0
1700.0
1800.0
1900.0
2000.0
TIME (SEC)

CASE B

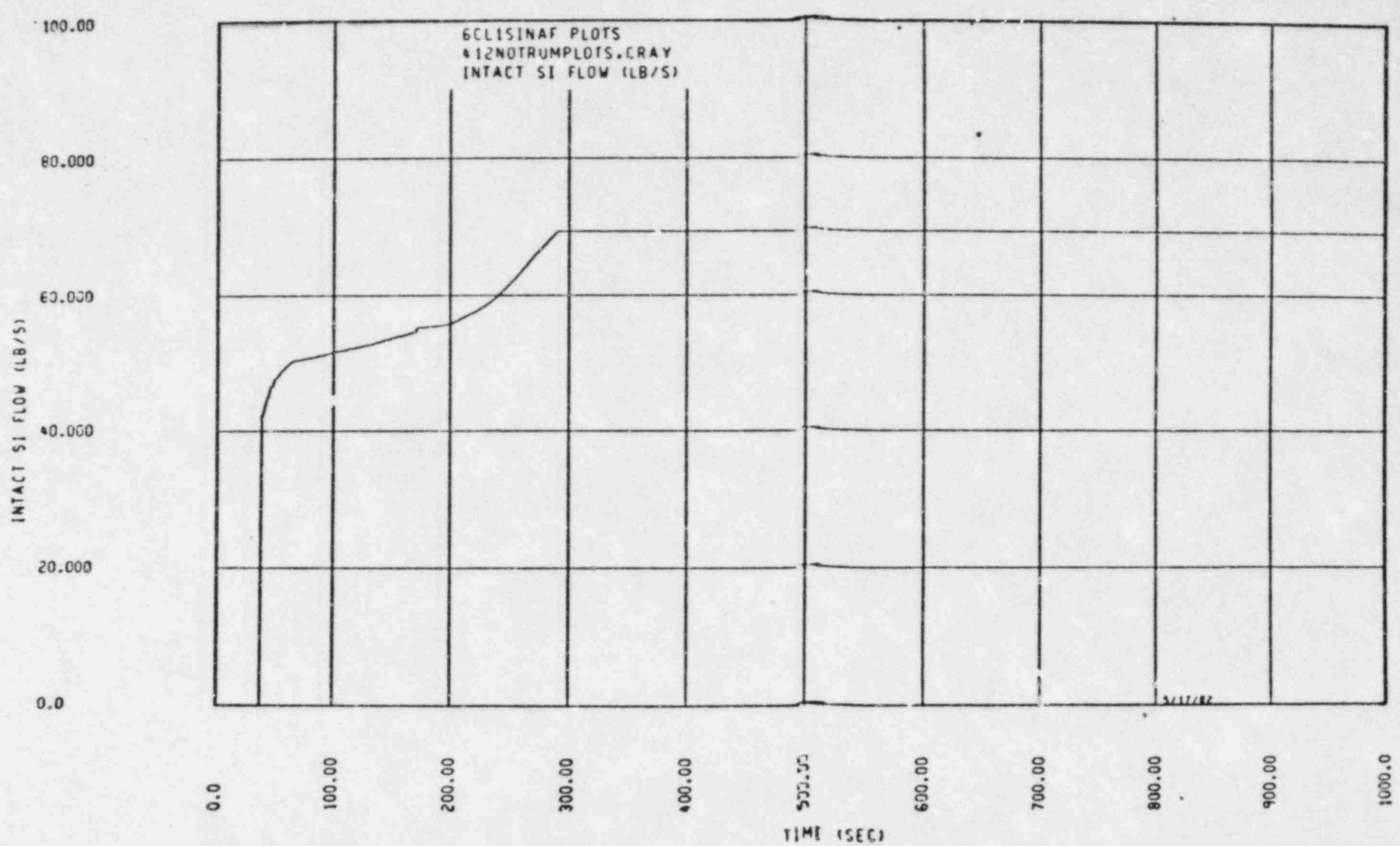
6-INCH COLD LEG BREAK PLOTS

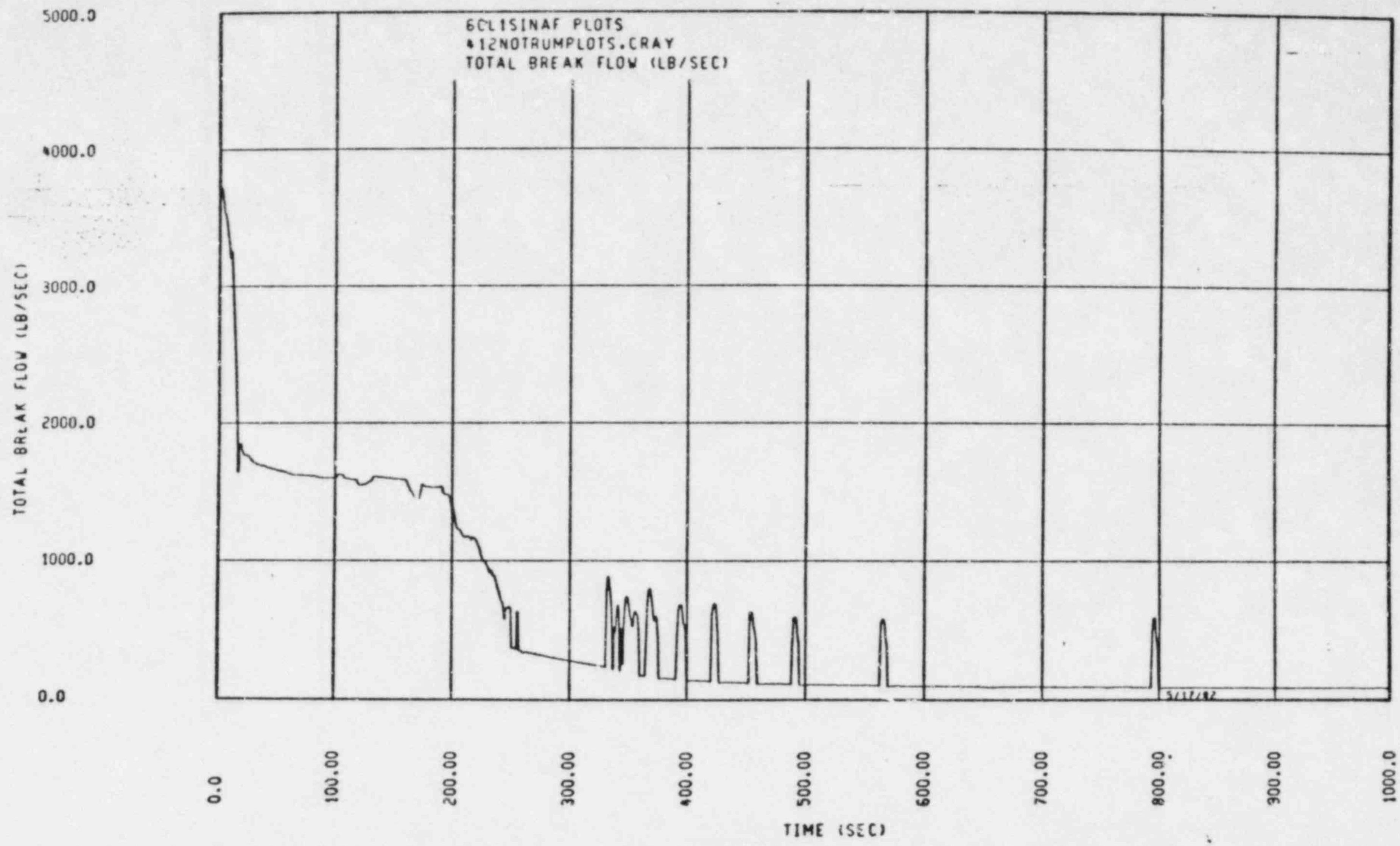


6CLISINAF PLOTS
412NOTRUMPLOTS.CRAY
CORE PRESSURE (PSIA)
2500.0
2000.0
1500.0
1000.0
500.00
0.0
CORE PRESSURE (PSIA)
0.0
100.00
200.00
300.00
400.00
500.00
600.00
700.00
800.00
900.00
1000.0
TIME (SEC)



6CLISINAF PLOTS
*12MOTRUMPLOTS.CRAY
CORE MIXTURE ELEV (FT)
CORE MIXTURE ELEV (FT)
35.000
34.000
32.000
30.000
28.000
26.000
25.000
0.0
100.00
200.00
300.00
400.00
500.00
600.00
700.00
800.00
900.00
1000.0
TIME (SEC)


6CLISINAF PLOTS
412NOTRUMPLOTS.CRAY
INTACT SI FLOW (LB/S)
100.00
80.000
60.000
40.000
20.000
0.0
INTACT SI FLOW (LB/S)
0.0
100.00
200.00
300.00
400.00
500.00
600.00
700.00
800.00
900.00
1000.0
TIME (SEC)


6CL1SINAF PLOTS
412NOTRUMPLOTS.CRAY
TOTAL BREAK FLOW (LB/SEC)
TOTAL BREAK FLOW (LB/SEC)
5000.0
4000.0
3000.0
2000.0
1000.0
0.0
0.0
100.00
200.00
300.00
400.00
500.00
600.00
700.00
800.00
900.00
1000.0
TIME (SEC)

Question 720.4: What is the justification for assuming that the operator can utilize secondary depressurization and LPI (OA-1) for safety injection during sequences which result in high RCS pressure? Previous PRAs have always assumed that HPSI is required in these sequences and that core melt would result if HPSI were unavailable.

Include in your reply an explicit consideration of the use of secondary depressurization and LPI for the instrument tube LOCA initiating event.

Response: A formal technical response to this question has already been provided on November 9, 1983 in Reference 1.

Question 720.5:	What is the justification for assuming that the operator can utilize bleed and feed cooling (OA-3, OA-7) for those sequences where auxiliary feedwater is unavailable? If bleed and feed is a viable technique, why is the operator required to provide additional bleed for small LOCA events? Wouldn't the break flow itself be sufficient?
Response:	A formal technical response to this question has already been provided on November 9, 1983 in Reference 1.

Question 720.6: What is the justification for assuming that the operator can eliminate the need for recirculation if he utilizes controlled primary depressurization (OA-2)? Even though this action would help to extend injection by preserving RWST inventory, it seems that recirculation would still be required eventually. Also, it appears that buildup in containment pressure is totally ignored in this scenario. Wouldn't CSR be required at a minimum to prevent containment overpressurization?

Include in your reply an explicit consideration of the use of controlled primary depressurization and the need for containment spray recirculation for the instrument tube LOCA initiating event.

Additionally, there are two ways that the operator could make an error in the use of this action which could result in significant injection cooling and subsequent core melt. First, in situations where this action is correct, he could perform it "too well". Second, he could perform this action in situations where it is inappropriate. That is, he could misinterpret plant conditions. It appears that these possibilities are not considered in event trees analysis. Are these possibilities treated in any way? If so, how are they treated? If not, is there some justification as to why they are not significant?

Response: A formal technical response to this question has already been provided on November 9, 1983 in Reference 1. A Supplemental technical response to this question was also provided on January 10, 1984 in Reference 2.

Question 720.7: What is the reasoning behind the use of the main condenser for removing steam from the steam generators while auxiliary feedwater is in use? In previous PRAs it was always assumed that in these cases steam was removed through the secondary steam relief (atmospheric dump) valves. It does not seem reasonable to use the condenser since there would be no place for the condensate to go, resulting in condenser isolation eventually.

Response: A formal technical response to this question has already been provided on November 9, 1983 in Reference 1.

Question 720.8: For SGTR events, what is the basis for assuming that the use of secondary depressurization (OA-4) or direct primary depressurization (OA-5) is sufficient to terminate the event? While it makes sense that these methods will eventually reduce primary pressure low enough to stop flow out the break will they do it rapidly enough to prevent uncovery of the core and allow for the use of only auxiliary feedwater to prevent core melt?

Response: A formal technical response to this question has already been provided on November 9, 1983 in Reference 1.

Question 720.9: For SGTR events, why is it deemed that HP-2 and AF-2 are sufficient to prevent a core melt? It seems that just allowing these systems to actuate and run would not result in ultimate success. Due to the nature of a SGTR, some operator action would be required to prevent pumping all the RWST water out the break and out of containment. What operator action is required during this sequence to prevent core melt?

Response: A formal technical response to this question has already been provided on November 9, 1983 in Reference 1.

Question 720.10 Why isn't PCS included as an event on the transient event trees which do not involve secondary system failures? Most other PRAs have concluded that for certain events, the PCS would be available to remove decay heat down to hot shutdown levels. There would seem to be nothing preventing this at Millstone. Please provide the following information to aid us in our demonstration of the viability of this cooling mode:

- the signals which result in shutdown of the PCS and specifically how the shutdown occurs (what equipment is tripped first)

- what causes the MSIVs to close and what conditions are required to re-open them

Response: A formal technical response to this question has already been provided on January 10, 1984 in Reference 2.

Question 720.11: For the spurious SI transient event tree, why does event OA-7 appear as a conditional event? This event includes in it the unavailability of HPI, which by implication is zero for this initiator. Also, credit is given for shutdown of HPI (OA-6) when AFWS is available, but no consideration is given to the operator performing this action when AFWS is unavailable (i.e., misinterpreting his situation) and thus causing insufficient cooling to be available. This incorrect action is also not treated on other event trees where the operator could believe he has a spurious SI and erroneously terminate HPI. Is there some justification for not treating these possibilities?

Response: A formal technical response to this question has already been provided on November 9, 1983 in Reference 1.

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

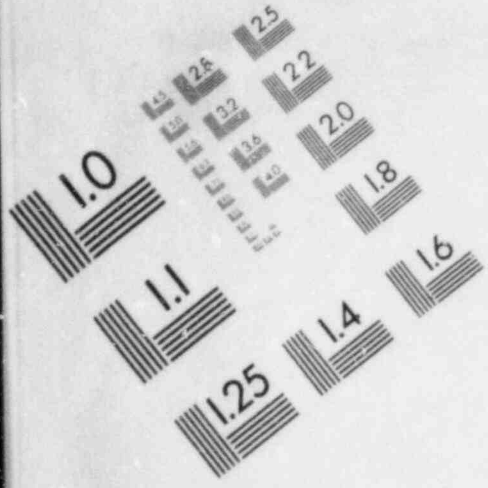
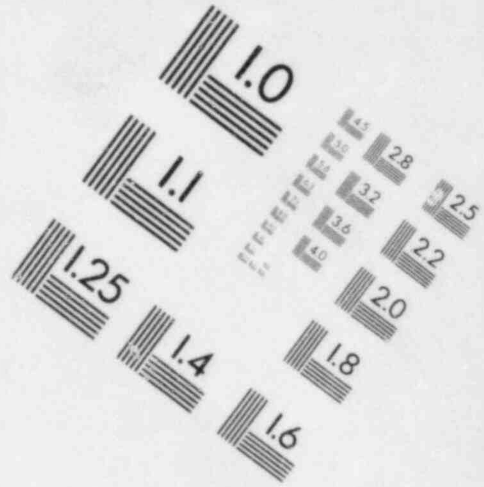
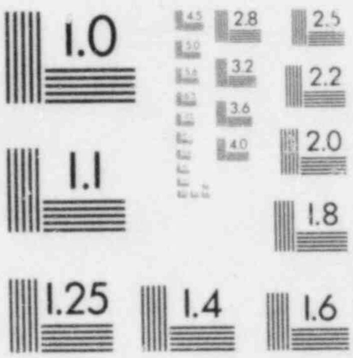


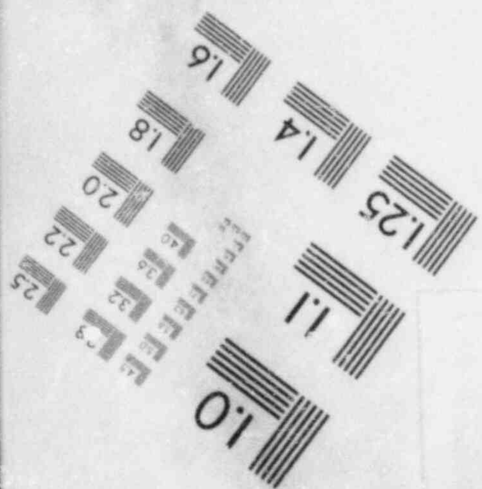
IMAGE EVALUATION TEST TARGET (MT-3)

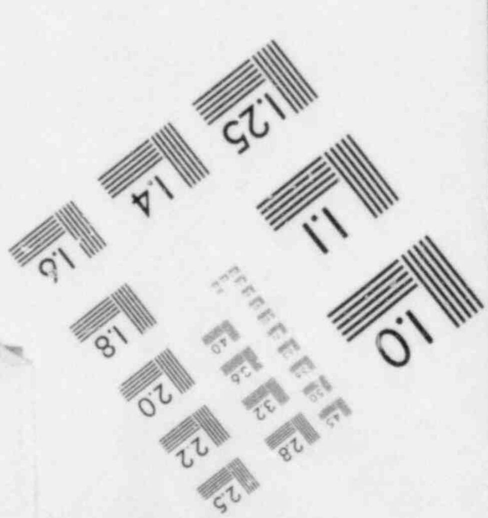




150mm

6″





Question 720.12: For the Loss of a DC bus event tree, the text states that bleed and feed is unavailable, however the tree shows a decision point for this event. Further, the support state tables imply that support state 2 dominates for this event and they show that OA-7 can be available for this event (a value which is not unity is given for this action). While it appears, based on the success criteria for OA-7, that the text is correct, the contradictory information is confusing. What is the correct assumption regarding the availability of OA-7 for this event and was the analysis handled in a correct and consistent manner?

Response: A formal technical response to this question has already been provided on November 9, 1983 in Reference 1.

Question 720.13: Provide justification for assuming that the operator actions of emergency boration and bleed and feed cooling can accomplish the dual function of subcriticality and core cooling.

Response: First, this dual operator action function of emergency boration and bleed and feed cooling is only assumed if the initial power level of the ATWS event is less than 25% or the moderator temperature coefficient is less than -5 pcm/°F. Under these limitations, the expected plant conditions following the ATWS event would allow this dual function to be accomplished. WCAP 8330, Westinghouse Anticipated Transients Without Trip Analysis, and WCAP 9715, PORV Sensitivity Study for LOFW-LOCA Analysis demonstrate the feasibility of emergency boration and bleed and feed cooling. In addition, Revision 1, Function Response Guideline FR-S.1 provides guidance for obtaining subcriticality given an ATWS event. The operator is advised to initiate emergency boration by aligning a charging path and opening PORVs as necessary.

Furthermore, a sensitivity analysis was performed to determine the effect of assuming that all ATWS events compounded by loss of auxiliary feedwater would result in core melt. This sensitivity analysis showed no increase in internal core melt frequency.

Question 720.14: For steamline break events, does the operator controlling HPI flow (OA-6) make any meaningful difference in the sequence progression? It appears not to, and thus it is questionable if it should be included. If this action is called for in a procedure, it is more of a concern that the operator may perform this action by mistake and cause a core melt (see Question 720.11). Why is this event included on the steamline break trees?

Response: A formal technical response to this question has already been provided on November 9, 1983 in Reference 1.

Question 720.15: For steamline break events, why is the failure of main steam line isolation assumed to result in the failure of auxiliary feedwater? The need for main steam isolation has only been considered in other PRAs as a method for isolating steam generators and the containment, but has never been assumed to have a direct effect on the availability of any safety systems. This seems to be an extremely conservative assumption. At worst, it seems that one could assume that MSI failure would effect the availability of the turbine driven AFW pump, but nothing more. What is the justification for making this assumption?

Response: A formal technical response to this question has already been provided on November 9, 1983 in Reference 1.

Question 720.16: The values for independent and common cause failure of the two diesel generators are significantly lower than the values obtained from numerous other sources. The apparent reason for most of this difference is the recent start data obtained on Fairbanks-Morse units as described in Appendix 2-E. In order to assist in the evaluation of this data and the extent to which it should be considered in deriving the Millstone 3 failures rates, please provide the following information;

a. To what extent were the units "prepped" before each test, including pre-warming, pre-lubing, pre-inspection, and component checking (air starters, lube pumps, cooling system, etc)?

b. Appendix 2-E describes the diesel units used for the tests (and also employed at the Millstone 3 plant) as having designs which are the result of an extensive reliability improvement program. Please provide details of the design improvements which have been provided for these diesels, and their impact on reliability.

c. What was the time interval between tests?

d. The 300 tests at Millstone 3 were, according to page 2-E-6, performed under conditions which rigorously stressed the diesels under numerous load conditions. Please describe these conditions. Also, were the other tests listed for other units performed under the same conditions? (The entire 1839 starts were apparently used to establish the failure rates.)

e. Were the tests performed under "fast start" (approx. 10 sec.) conditions which would exist for an actual demand?

Response: A formal technical response to this question has already been provided on January 10, 1984 in Reference 2.

Question 720.19: Sect. 4.7.1 indicates that containment isolation failure is assessed at 10^{-4} failures per demand but very little information is given to support the assessment. Please provide details in support of this value, including number, size, type of valves, actuation logic (does failure rate include actuation failures?), maximum temperature and pressure capability, etc.

Response: Millstone-3 employs a subatmospheric containment which would alarm in the control room under normal operation if a significant loss of integrity occurred prior to any accident condition (e.g. hence no significant undetected bypass paths such as open air purge lines left open could exist). For a subatmospheric containment a value of 10^{-4}/demand following an accident is reasonably conservative.

Millstone-3 Containment Isolation Features

Details of the Millstone-3 Containment Isolation System are given in Section 6.2.4 of the Millstone-3 FSAR. Table 6.2-65 of the FSAR gives information related to each fluid piping penetration including size, isolation valve actuation, closure time, leak testing, and so forth. A thorough discussion of the ultimate pressure capability of piping and electrical penetrations is given in Section 5.1 of Appendix 4-F to the MP-3 PSS. As indicated in Table 9-1 of Appendix 4-F, the limiting penetrations have a mean ultimate pressure capability of 128 psig. Details concerning the treatment of thermal loads are given in Section 6 of Appendix 4-F. As mentioned in Section 4.4.1.1 of the PSS, the limiting component from the viewpoint of temperature capability was assumed to be the seals of electrical penetration assemblies (EPA). A screening value of 400°F (on the liner wall temperature) was chosen to conservatively represent the lower bound for overtemperature failure.

Since publication of the PSS, additional information related to the potential for containment leak paths from EPAs under severe accident conditions have been made available (Ref. 1). Because of the double seal design of EPAs and "because the massive concrete walls (of large, dry type PWR containments) act as heat sinks, the outside of EPAs can be kept well below the temperature limits of organic seals." Additional confirmatory research is underway. Hence, assumptions concerning overtemperature failure in the PSS are probably conservative.

Normal Operation

The MP-3 containment is deisgned to be subatmospheric, operating in a range of 9.11 to 12.34 psia. Since the containment is a subatmospheric design with operation of the plant in this state guaranteed by the plant Technical Specifications, no unisolated paths would exist during plant operation other than systems which are known to be operating or required to be open.

Reliability of Containment Isolation in Subatmospheric Containments

Availability of containment systems to meet leakage specifications has recently been investigated by M. Weinstein of American Nuclear Insurers (Ref. 2, 3). Weinstein has compiled data from 1961 through 1980 under the categories of certain and probable long-term integrity failures. For PWRs as a class, the availability of leakage integrity defined as

$$ALI = 1 - \frac{\text{total duration of integrity failure}}{\text{total time since initial criticality}}$$

is 0.95. Via personal communications with Weinstein (Ref. 4) on 7/28/82 and 7/30/82 it was pointed out that for the reasons stated above, PWR plants with subatmospheric containments have "very high ALIs." The only known incident resembling an isolation failure was at Surry in 1981, where leaking electrical penetrations resulted in only a failure of an integrated leak rate test (1.5 times allowable). However, it should be noted, this leakage rate relates to only a fraction of a volume percent per day of leakage and not gross containment failure as is considered in PRA type evaluations. This is significantly less than the 200 volume percent per day leakage rate defined as containment isolation failure in WASH-1400 (Ref 5, Appendix VIII, p. VIII -19)

Therefore, from a risk perspective, there have been no known gross containment isolation failures in subatmospheric PWRs. A screening value of 10^{-4} failures per demand was therefore used. This value is consistent although somewhat lower than the values used in WASH-1400. To have any measurable impact on public risk, the value of containment isolation failure would have to increase by several orders of magnitude above the 10^{-4} value.

REFERENCES

1) W. SEBRELL, "The Potential for Containment Leak Paths Through Electrical Penetration Assemblies Under Severe Accident Conditions," NUREG/3234, SAND 83-0538, SANDIA NATIONAL LABORATORIES, July 1983.

2) M. WEINSTEIN, "Primary Containment Leakage Integrity: Availability and Review of Failure Experience, "Nuclear Safety, Vol. 21, No. 5, pp. 618-632.

3) M. WEINSTEIN, "Integrity Failure Experiences with Reactor Containments," Proceedings of the Workshop on Containment Integrity, NUREG/CP-0033, SAND 82-1659, SANDIA NATIONAL LABORATORIES, October 1982.

4) M. Weinstein, personal communication, 8/4/82.

(5) "Reactor Safety Study: An Assessment of Accident Risks in U. S. Commercial Nuclear Power Plants," WASH-1400, U. S. Nuclear Regulatory Commission, October, 1975.

REFERENCES

1) M. Berman, et. al., "Analysis of Hydrogen Mitigation for Degraded Core Accidents in the Sequoyah Nuclear Power Plant," NUREG/CR-1762, SAND80-2714, Sandia National Laboratories, March 1981.

2) J. C. Cummings, et. al., "Review of the Grand Gulf Hydrogen Igniter System," NUREG/CR-2530, SAND82-0218, Sandia National Laboratories, March 1983.

3) J.H.S. LEE, "Flame Acceleration Mechanisms in Closed Vessels, "Proceedings of the Workshop on the Impact of Hydrogen on Water Reactor Safety, January 1981, NUREG/CR-2017, SAND81-0661, September 1981.

4) J.H.S. LEE, I. O. MOEN, "The Mechanism of Transition from Deflagration to Detonation on Vapor Cloud Explosions," Prog. Energy Combustion Sci, 6, pp. 359-389.

Question 720.20: Pg. 4.7-6 of the PSS states that the Millstone 3 containment "is an open volume with no regularly spaced objects to generate strong turbulence." This assessment is used to argue that hydrogen detonation is not credible. Based on a tour of the Millstone 3 unit, just the opposite impression was obtained regarding objects in the containment, i.e. there appeared to be many objects of various size, some regularly spaced, especially in the lower regions of the containment where the hydrogen is expected to be released. Please provide further discussion on this matter.

Response: The statement concerning the relative openness of the Millstone - 3 containment must be taken in the context of the compartmentalization present in other reactor containments, and of the stringent requirements on confinement for hydrogen detonations to occur.

The Millstone - 3 containment is quite open with respect to ice condenser or BWR Mark III reactor containment designs, for example: Figures 4.1-5 and 4.1-6 of the MP-3 PSS show the potential release paths of hydrogen resulting from degraded core accidents at MP-3. Essentially all of the hydrogen released via the pressurizer relief tank, the lower reactor cavity, the steam generator cubicle and the pressurizer cubicle would be vented into the upper containment region above the operating platform. The vent openings have dimensions on the order of meters to many meters. In References 1 and 2, it was found that, although hydrogen detonations could not be completely ruled out, at least on a qualitative basis hydrogen detonations resulting from degraded core accidents would be very unlikely for Sequoyah and Grand Gulf. Because of the open containment design at MP-3, the likelihood would be even lower based on geometrical considerations.

The reference to "no regularly spaced objects to generate strong turbulence" relates to deflagration-to-detonation transition. Lee (References 3 and 4) has identified several flame acceleration mechanisms capable of causing such a transition. One example is that of flame propogation past obstacles causing shear flow, which leads to both fine-scale turbulence and larger-scale flame folding. It is possible for turbulent flames to escalate because of the cumulative effects of obstacles repeated in a regular pattern. This observation is related to small-scale tests performed at McGill University using spiral coils or repeated circular orifice plates inserted as turbulence generators in long circular tubes. There are clearly no such equivalent obstacles in a repeated pattern in the Millstone-3 containment, and so this mechanism of hdyrogen detonation is deemed non-credible.

Question 720.21: There appears to be an inconsistent and somewhat confusing discussion at various locations in the report with respect to the operability of the recirculation spray system without previous operation of the quench spray. Page 2.2.7-1 states that recirculation spray failure was assumed if quench spray failed. However, on Pg. 4.4-27, recirculation spray only cases are considered for sequences AEC", ALC", SEC", SLC", and TEC." Furthermore, it is stated that the accumulator water would be available for these sequences when recirculation spray is actuated. Please explain how the accumulator water gets to the sump in time for recirculation spray actuation for the small break and transient sequences? Also, please provide further justification for sufficient sump water inventory for small break and transient sequences when much of this water may remain in the primary system (and the quench tank for the transient sequences) when the recirculation spray system is actuated. On page 4.4-15, the recirculation spray is considered operable for T sequences in the absence of quench spray. Please explain the basis for this assumption.

Response: The plant systems analysis (Section 2.0 of the MP-3 PSS) and the containment analysis (Section 4.0) were conducted separately in parallel paths. When the plant event trees were constructed, an assumption was made concerning the availability of recirculation spray given the failure of quench spray. For accident sequences (either small or large LOCAs) with emergency coolant injection available, it was clear that quench spray would not be required for recirculation spray. There would be adequate sump water. For accident sequences (transients, small, and large LOCAs) with no emergency coolant injection, it was not clear at the time about the dependency of recirculation spray on quench spray. Therefore, the assumption was made in Section 2.2.7 that for accidents leading to early core melt, recirculation spray would not be available if quench spray were not available. One will note that damage states AEC", SEC", and TEC" are not defined anywhere in Section 2.0 nor in the plant matrix, M (see Table 2.4.1-6). Instead, AE, SE, and TE states were assigned.

Later, when the detailed containment response calculations were completed, it was determined that sufficient sump water would be available to operate the recirculation sprays independently of the success or failure of quench sprays. The only exception to this was the incore instrument tube rupture denoted by S', where recirculation spray is dependent on quench spray. This sequence was treated separately.

Therefore, the initial assumption made in Section 2.0 turned out to be conservative, although the overall impact on core melt frequency is zero and the impact on the risk estimates is small.

With regard to the availability of sump water when the recirculation spray is called upon, one needs to determine the sump water inventory for the accident sequences of interest (AEC", ALC", SEC", SLC", TEC"). Information concerning the timing of recirculation spray actuation is found in Table 4.4.2-1 under "spray on." Information on the sump water mass at the time of recirculation spray is taken from the COCOCLASS9 computer printouts. These are summarized below in Table 1.

TABLE 1 SUMP INVENTORY

ACCIDENT SEQUENCE	RECIRC. SPRAY ON (SEC)	SUMP MASS (LBM)
AEC"	290	6.7×10^5
ALC"	290	6.7×10^5
SEC"	1120	2.6×10^5
SLC"	1190	2.8×10^5
TEC"	15800	2.2×10^5

These calculated masses must now be compared to the amount of water required to fill the recirculation spray system piping as well as the amount of water required in the sump to meet net positive suction head requirements. Information from Reference 1 indicates that the volume of water in the recirc lines is 5,300 gal, and that a minimum of 1658 gal is needed in the sump to meet NPSH requirements for the design basis accident. This gives a total requirement of approximately 7000 gal or 5.6×10^4 LBM. Therefore, for all the C" sequences, there will be adequate sump water to run the recirculation pumps (a small mass will be in transit through the atmosphere as a spray).

With regard to the accumulator water and how it reaches the sump before recirculation spray, there is an apparent misunderstanding about what is meant in the MP-3 PSS. On page 4.4-27 it is stated: "The water inventory in the containment sump area would be limited to the reactor coolant system volume and the accumulator water volume." What is meant is that, in the analysis which was performed concerning spillover of water from the sump to the reactor cavity, at most, the only water available would be from the RCS and from accumulators. (Water from the accumulators is discharged into the primary system when the pressure drops below 600 psia. For small LOCAs and transients, this would be after recirculation spray actuation). Nowhere is it stated or meant to be implied that accumulator water would be discharged directly to the sump before recirculation spray actuation for small LOCAs and transients.

With regard to the pressurizer relief tank, it should be noted that the discharge of steam through the safety relief valves for transient-induced accident sequences was modeled as if the tank were not present. This is standard. The presence of the tank has two effects. First, with the tank's volume of 1800 ft^3 half filled with water, a good deal of the steam released via the pressurizer safeties would be condensed, hence delaying containment pressurization and spray actuation. Upon reaching the 100 psig setpoint, the relief tank rupture disk would open discharging much of the contents into the sump. Some water would remain in the tank, but the overall impact on sump water inventory would be small.

In summary, detailed analyses have shown that recirculation spray is not dependent on quench spray. However, in the plant analysis a conservative assumption was made that for early core melts (all LOCAs and transients), there was such a dependency. In either case, the impact on public risk is very low because of the unlikelyhood of the C" sequences.

REFERENCE

1.) Telephone Memorandum, D. A. Dube (NUSCO) and M. Donahue (S&W), December 12, 1982.

Question 720.22: Why is the impact on unavailability of test and maintenance on the motor operated valves MOV34A and MOV34B not modeled in the Quench Spray System Fault Tree?

Response: At the time the Millstone-3 P.S.S. was initiated specific test requirements for motor operated valves MOV34A and MOV34B had not yet been established. Cycling these valves could potentially lead to loss of subatmospheric containment integrity. Specific procedures to test these valves (which impact containment integrity) are being developed and will be incorporated into P.S.S. models before startup. The values in question are normally closed, as shown in Figure 2.3.3.9.1-1 of the PSS, and receive automatic signals to open on indication of accident conditions. Failure of this auto actuation logic to open the valves was modeled in the Quench Spray System fault tree.

The fault tree model thus assumes the two MOVs are not maintained during power operation.

REFERENCES

1) M. Berman, et. al., "Analysis of Hydrogen Mitigation for Degraded Core Accidents in the Sequoyah Nuclear Power Plant," NUREG/CR-1762, SAND80-2714, Sandia National Laboratories, March 1981.

2) J. C. Cummings, et. al., "Review of the Grand Gulf Hydrogen Igniter System," NUREG/CR-2530, SAND82-0218, Sandia National Laboratories, March 1983.

3) J.H.S. LEE, "Flame Acceleration Mechanisms in Closed Vessels, "Proceedings of the Workshop on the Impact of Hydrogen on Water Reactor Safety, January 1981, NUREG/CR-2017, SAND81-0661, September 1981.

4) J.H.S. LEE, I. O. MOEN, "The Mechanism of Transition from Deflagration to Detonation on Vapor Cloud Explosions," Prog. Energy Combustion Sci, 6, pp. 359-389.

Question 720.23: Was a quantitative analysis performed as a basis for excluding freezing of RWST and quench spray lines as common cause failures in the RWST?

Response: Common cause failure of the RWST and quench spray lines were excluded on the basis of the following considerations:

- Low RWST temperature is an alarmed parameter on the main control board. Such an alarm would occur long before freezing of the RWST took place. The operator would take appropriate action by responding to this alarm as outlined in the control room annunciator procedures.

- Redundant heat tracing exists on exposed quench spray lines that could be affected by freezing. The heat tracing is monitored for proper operation by operators on a regular basis as part of their rounds procedures.

Question 720.24: In the fault tree for the Main Electrical System, failure of the circuit breaker between the generator and the emergency bus is modeled in extreme detail. How do the results of the fault tree analysis compare to available data for this circuit breaker? What is the difference between this breaker and the one listed on page 2-A-18?

Response: The circuit breaker between the diesel generator and the emergency bus is identical to the breaker listed on page 2-A-18 of Appendix 2-A. The protective relaying and external interlocks, unique to Millstone, were modeled in detail to show the effect on the circuit breaker.

The "breaker fails to close on demand" failure rate, shown on page 2-A-18, was used to quantify one of the circuit breaker faults in the Main Electrical System fault tree. This is shown as base event 0063 under OR gate 0008.

Question 720.25: Closure of the circuit breaker between the emergency generator and emergency bus requires that the trip coil in this breaker be energized. This is done by a trip contact that can be closed either manually or automatically. In the fault tree for this system (Fig. 2.3.3.1-3), failure of this trip contact requires failure of both the manual and automatic trip contact. However, the automatic trip contact requires an actuation signal from the EGLS for operation. In addition, the EGLS is modeled as failed when there is no power on the 120 vital AC from the corresdpcnding power train, which is modeled as failed when there is no power on the corresponding emergency AC bus. (i.e. 34C or 34D). But if there is no power on the corresponding emergency bus, this circuit breaker would not be called upon. Why is credit taken for the availability of this automatic signal?

Response: Closure of the emergency generator output breaker is not dependent on the trip coil in that breaker being energized. Such a condition would, in fact, prevent closure of the breaker. Closure of the emergency generator output breaker is, however, dependent on the emergency bus being deenergized. One of the conditions that assures that an emergency generator is not loaded onto an energized bus is that the bus tie breaker to the emergency bus is open. This is apparently the trip contact failure logic that is referred to in this question. This failure is a negligible contributor to loss of emergency power to the emergency bus because of available multiple inputs to the tie breaker trip coil circuit, any one of which is capable of energizing the trip coil and opening the tie breaker.

Availability of the EGLS is addressed in the response to Question 720.73a, which points out that the unavailability of vital AC during the 10 second plus interval without any AC is bounded by the case with AC available.

Question 720.26: No data is provided on the failure rates of components on the normal station service transformer and reserve station service transformer feeds. How were the unavailabilities that are used in the main electrical system fault tree for these components obtained?

Response: Faults on the normal station service transformer (NSST) feed are caused by any one of several component failures that can fail the normal electrical supply to emergency bus 34C. The normal supply starts at the low side of the main transformer and goes to bus 34C via the NSST as shown in Figure 2.3.3.1-1. Any one of the following failures will fault this NSST feed:

1. Failure of the main transformer
2. Failure of the NSST
3. Circuit breaker faults on the breaker between bus 34A and the NSST
4. Bus faults on 4.16 kv bus 34A
5. Faults on the tie breaker between buses 34A and 34C.

The unavailability of the NSST feed was determined by listing the above five faults under an OR gate. Component unavailabilities were determined by using the failure rates in Appendix 2A and a 24 hour mission time.

Faults on the reserve station service transformer (RSST) feed, which is the alternate supply to 34C, can be caused either by failure of the RSST or faults on the breaker between the RSST and bus 34C. Both of these faults were quantified to determine the RSST feed unavailability, using the same method that was just described for the NSST feed.

Question 720.27: The fault tree for vital 120 AC power models this system as failed when there is loss of power on the corresponding emergency bus (34C). Apparently, the model does not take credit for the batteries. Why?

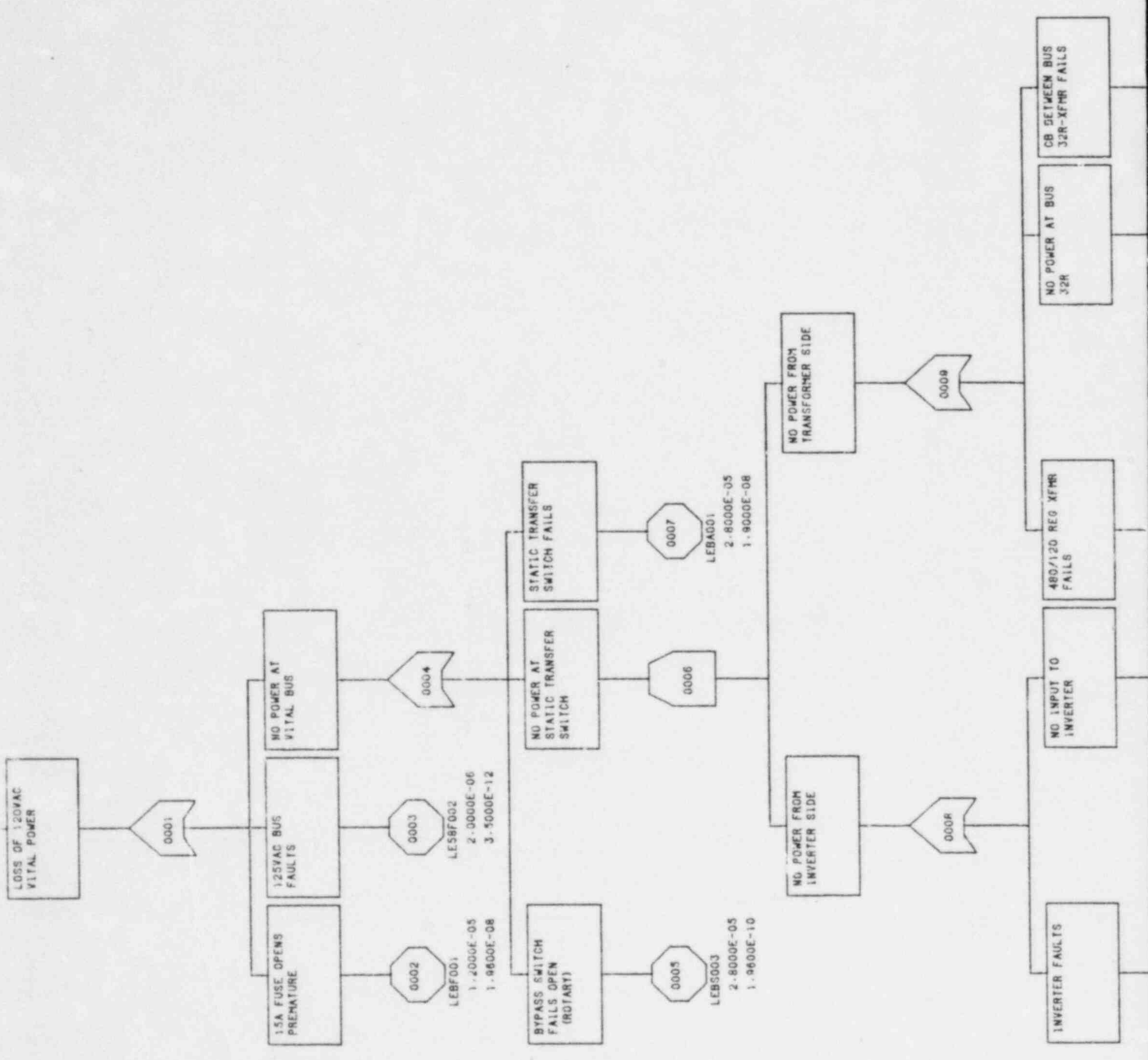
Response: A formal technical response to this question has already been provided on January 10, 1984 in Reference 2.

Question 720.28 The diagram of the vital 120 AC bus system in the PSS shows a change in connection between VIAC-1 and emergency buses 32T and 32R when compared to a similar diagram in the FSAR. Was this change a result of the PSS? How was this change carried out at the plant?

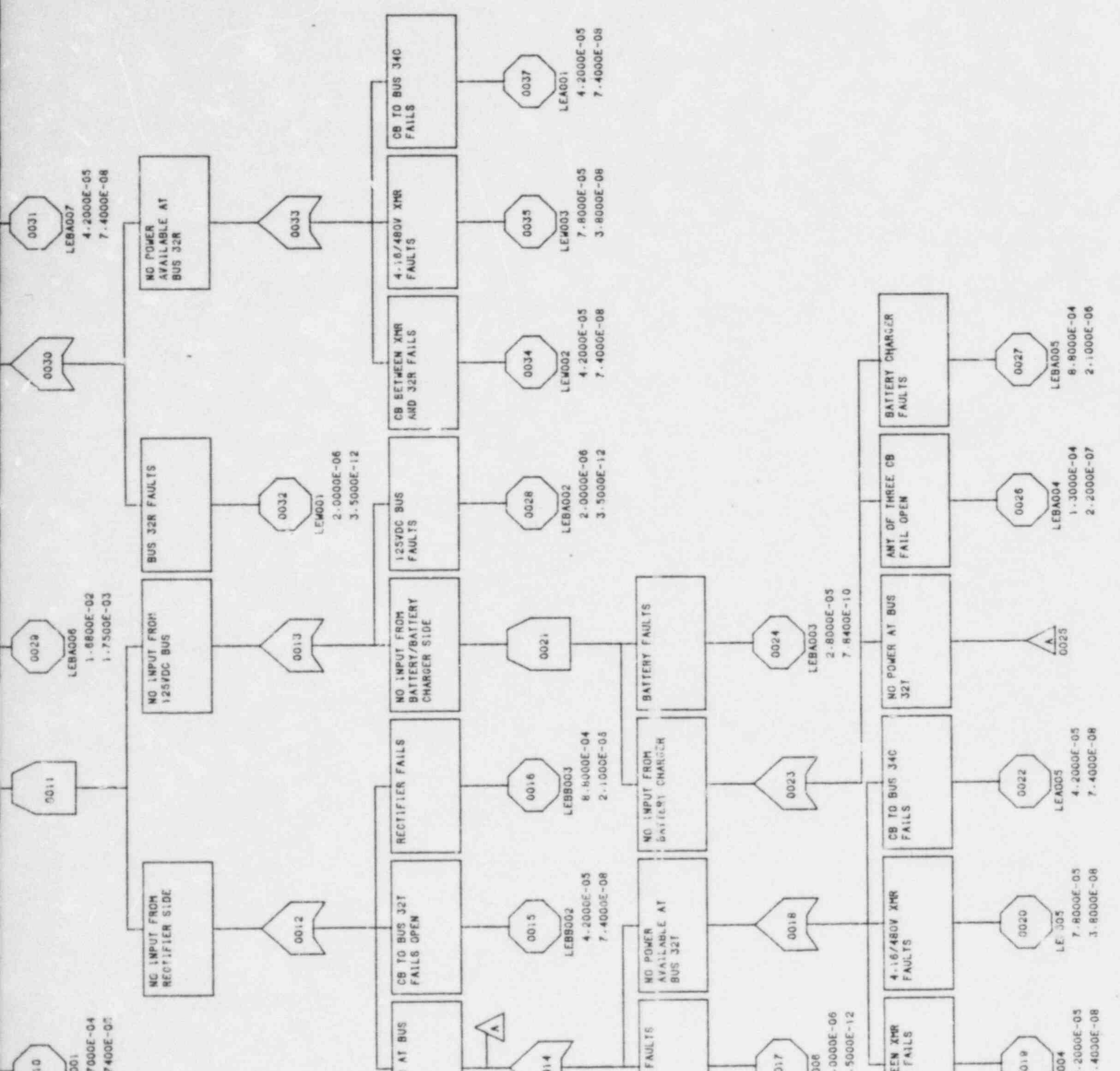
Response: The diagram of the 120V vital AC bus system shown in the FSAR is based on a design which has since been revised. The diagram for vital AC in the PSS reflects the current version. The change in design was carried out at some time prior to modeling the vital AC system for the PSS.

Question 720.29: In the fault tree for the vital 120 AC bus. the loss of power on bus 34C is modeled as a separate event (different lables, different unavailability) on two inputs to an AND gate. Why?

Response: The fault tree for the 120V Vital AC Bus System, shown as Figure 2.3.3.2-2, is an early version of the tree and was inadvertently used in the PSS (there were several revisions made to the vital AC tree). As noted in the response to Question 720.71, the vital AC power system was only quantified for those cases where AC power was available. A later, more correct version of the tree and WAM CUT quantification results are attached. There is no significant difference between the attatched results and those used in the PSS.



APERTURE
CARD



Also Available On
Aperture Card

8402080247-01

WAMCUT WITH 120VAC

CUT SETS FOR GATE G00001 WITH PROBABILITY .GE. 1.00E-08

1.	2.80E-05	LEBS003	
2.	2.80E-05	LEBA001	
3.	1.20E-05	LEBF001	
4.	2.00E-06	LE58F002	
5.	2.81E-08	LEBB001	LEW002
6.	5.23E-08	LEBB001	LEW003
7.	2.81E-08	LEA001	LEBB001
8.	1.13E-05	LEBA006	LEBB001
9.	2.81E-08	LEBA007	LEBB001

CUT SETS FOR GATE G00001 ORDERED BY PROBABILITY

1.	2.80E-05	LEBA001	
2.	2.80E-05	LEBS003	
3.	1.20E-05	LEBF001	
4.	1.13E-05	LEBA006	LEBB001
5.	2.00E-06	LE58F002	
6.	5.23E-08	LEBB001	LEW003
7.	2.81E-08	LEBA007	LEBB001
8.	2.81E-08	LEA001	LEBB001
9.	2.81E-06	LEBB001	LEW002

1ST MOMENT= 8.1393E-05 2ND MOMENT= 1.6201E-07 STD.DEV.= 3.9419E-04 95 PERCENT (CHEBYCHEV) = 1.3279E-03

CUT TOOK 8.850 SECS

TABLE 2.3.3.2-2
DOMINANT CUTSETS FOR
120V AC VITAL BUS SYSTEM

Question 720.30: In the quantification process for any accident sequence in which the support states, fault trees, and event trees are linked, what procedure was used to screen out logical inconsistencies in the model? For example, in the current model the potential exists for modeling the failure to transfer AC power to the emergency bus as being due, in part, to the unavailability of power on the same bus.

Response: The quantification process of accident sequences involved the linking of support states, fault trees and event trees. The generation of support states utilized event tree logic (Figure 2.2.1.3.5-1 of the MP3 PSS) to assure that the support states were logically correct. The fault trees utilized to quantify the support states were generated contingent on the previous systems in the support state model (see response to question 720.71) to assure that they were logically consistent. The fault trees utilized for the event tree quantification were generated based on being in support state one. Then, to account for the dependencies of the fault trees on the support states, the fault trees were quantified contingent on each specific support state. In the accident sequence quantification the support state and the system quantifications for that support state were combined. Technical reviews were performed for each step of this process to further assure that the support states, fault tree, event trees and their linkings were logically correct.

These technical reviews were performed by a three tiered technical review process respectively denoted as, Level I Review, Level II Review and Level III Review. Level I Review is the normal engineering Quality Assurance carried out by the organization which performs or uses original results of a certain portion of an analysis. Level II Review was the detailed technical review performed by NU personnel. This Level II Review included a thorough verification of the support states, fault trees, event trees and their linking in the quantification of accident sequences. Level III Review was the broad technical overview of the safety study performed by the Millstone-3 Probability Safety Study Review Board. Included in this Review Board activity was a detailed two week audit and critique of the quantification process utilized in the study by the staff of one of the Review Board members. Execution of this process provides reasonable assurance that significant logic errors such as those noted in the question do not exist.

Question 720.31: Was the equation relating MM intensity to magnitude listed on page 2 of Appendix 1-B used? If so justify using an m_b = 6.25 to represent a MM Intensity IX.

Response: The original equation relates MM Intensity to Richter Magnitude. In the later Dames & Moore study (dated October 1983), two methods of converting MMI to body-wave magnitude m_b were used. For MMI=VIII estimated values of m_b were 5.75 and 5.80 from these relations. For a zone which included a historical MMI = VIII earthquake, the best estimate of $m_{b,\ max}$ was chosen to be 0.5 magnitude units above the estimated maximum historical event, or in this case 6.3.

Question 720.32: What is the area of each source zone listed in Table 1 of Appendix 1-B? Has the activity rate listed in Table 1 been normalized to unit area? If so, justify assuming the same recurrence relationship for the Central New England zone (Dames and Moore Model) and the New England Maritime zone (Hadley and Devine Model).

Response: The seismogenic zones used in the later Dames & Moore study (October 1983), and the areas of the dominant zone, are listed in Table 2 of that study. The activity rate shown in that table is the total rate for the zone (not normalized by magnitude).

Question 720.33: What was assumed regarding the completeness of the historic catalog for each zone listed in Table 1 of Appendix I-B?

Response: The assumed times of completeness for each intensity level are given on page 16 of the later Dames & Moore report (October 1983).

Question 720.34: As discussed on page 5 of Appendix 1-B your recurrence relationship is in terms of magnitude, yet the values listed in Table 1 are in terms of intensity. What were the actual magnitude related recurrence values used? Have you directly converted "b" values using a magnitude to MM intensity relationship? How do your magnitude "b" values compare to other investigators for this region of the country.

Response: The October, 1983, Dames & Moore study (which supercedes the previous analysis) converts intensities to magnitudes using two separate mathematical relationships, and calculates b-values for each. The b-values are generally higher than those assumed by other investigators for the northeast, but are justified by the data.

Question 720.35: To have stated that uncertainty in b-value is conservatively assumed to have perfect negative correlation with uncertainty in the maximum body-wave magnitude. Where on the recurrence curves assumed,did you modify the "b" (i.e., at what magnitude did you pivot the curve)?

Response: In the October 1983 study, the recurrence relations were, in effect, pivoted around the value for $m_b = 4.5$.

Question 720.36: You have stated that for each hypothesized model, the maximum historical earthquake had an estimated MM Intensity of VIII and that MM Intensity IX, being equivalent to an m_b = 6.25, was used as the maximum event. The 1755 Cape Ann event is listed as m_{blg} = 6.0 by Street and LaCroix (1979). Justify using a median representation for the upper magnitude of 6.25 for any zones which includes Cape Ann, considering that your magnitude to intensity conversion implies that one intensity unit translates to an implicit increase in the maximum historic magnitude.

Response: There is a substantial uncertainty in the magnitude estimates of pre-instrumental events. Although one estimate of m_b for the 1755 Cap Ann earthquake may be 6.0, others are lower. In the October 1983 Dames and Moore study the value of $m_{b,max}$ = 5.8 for zones encompassing Cape Ann corresponds to the assumption that the 1755 event was of that magnitude and no larger earthquakes, are possible. A value of $m_{b,max}$ = 6.8 allows much larger earthquakes, as large as the 1886 Charleston event to occur in the vicinity of Cape Ann, which is an extreme position of the opposite kind. The best estimate value of 6.3 is justified in that it lies between the two extreme positions and represents an earthquake of one intensity unit higher than the maximum historical event.

Question 720.37: Were the average annual probabilities of exceedance in Table 4 calculated with any "b" or upper magnitude cutoff "Mu" uncertainty included?

Response: The final frequencies of exceedance (Table 5 in the October, 1983, study) include uncertainties in b-value and maximum magnitude.

Question 720.38: What would be the impact on the annual probabilities of exceedance assuming the Dames & Moore zonation, on Mu of 7.2 for Central New England zone, Nuttli attenuation & other values for the Dames and Moore model in Table 1 of Appendix 1-B constant?

Response: The effect of maximum magnitude is shown on Figure 22 of the October, 1983, study.

Question 720.39: You have stated in section 1.2.1, that for practical purposes, nearly coincident curves out of the 36 total, have been combined in order to obtain a manageable set of nine frequency of exceedance curves. Provide the staff with a plot of the 36 curves similar to Figure 1.2.1-1.

Response: The later study used 184 hazard analyses to synthesize results. The presentation of all hazard curves on a single plot would not be meaningful.

Question 720.40: Provide results at 0.70g and 0.80g for Table 4 in Appendix I-B. How sensitive are your core melt results to changes in the seismic hazard frequencies?

Response: Results for 0.7g and 0.8g are provided in the October, 1983, report by Dames & Moore.

Question 720.41: Figures 8 to 11 of Appendix 1-B do not show the variation in results for different attenuation, as stated in the report. However, Table 4 of the Appendix does show that probabilities of exceedance are similar for the two attenuation models used. How dependent is this result on the different factors of uncertainties you have used for the two models? How many standard deviations were assumed for each attenuation model? Would you reach a similar conclusion if the upper magnitude cutoff was significantly higher than you have assumed for the host zone?

Response: The sensitivity of results to attenuation is shown in Figure 19 of the October 1983 study. The same dispersion in predicted acceleration was used for all attenuation models in that study, with no truncation at any specified number of standard deviations. The sensitivity of results to attenuation is not generally a function of the upper-bound magnitude in the host zone.

Question 720.42: As stated in Appendix 2-1 (3.4.1), the "fifty percent design spectrum" for rock-founded structures contained in WASH-1255 was used to calculate the spectral shape factor (FRSS). What value of v/a did you use? Justify this value in light of the recommended value of v/a = 36 in/sec/g in NUREG-CR/0098. Would a change in the v/a value effect your estimate of the spectral shape variability?

Response: Site-specific response spectra were not available for the Millstone 3 site. The spectra chosen by SMA as representative broad band spectra for the site were derived from Reference 1. These spectra were generated from a data set of earthquakes having magnitudes between 5.3 and 6.3. This magnitude range is expected to contribute the most to the seismic risk. The spectra used are therefore considered more appropriate for definition of the Millstone 3 seismic fragilities than either the WASH-1255 or NUREG-CR/0098 spectra which include earthquakes of greater magnitudes.

Definition of the v/a ratio was necessary only to predict acceleration capacities for structure sliding-induced failure modes using a method described in the SMA report. The median v/a ratio for the earthquake data set used to define the representative broad band spectra in Reference 1 was not readily available. However, the peak ground velocity corresponding to a peak ground acceleration of 1.0g was estimated by scaling down the median spectrum from Reference 1 by the median velocity spectrum amplification factor from NUREG-CR/0098. The median five percent damped spectrum scaled to a 1g peak ground acceleration from Reference 1 is shown in Figure 720.42-1. Dividing the spectral values in the frequency range from 0.5 Hz to 2.5 Hz by the median velocity spectrum amplification factor for five percent damping of 1.65 from Table 3 of NUREG-CR/0098 results in the "equivalent" peak ground velocity, shown dashed in Figure 720.42-1. Because of the shape of the spectra used by SMA, the v/a ratio is not a constant value in the velocity amplified range (appropriate for structure sliding) such as occurs for the WASH-1255 spectra.

It is seen that the v/a value of 28 in/sec/g exceeds the "equivalent" v/a value by a factor ranging from 1.1 to 2.0. In the evaluation of sliding conducted by SMA, it was decided to use the 28 in/sec/g since the exact "equivalent" v/a ratio for a given structure is not readily attainable except by a nonlinear sliding analysis, and also to be consistent with WASH-1255. Use of the v/a value of 28 in/sec/g therefore results in slightly conservative estimates of the peak acceleration capacities for the structure sliding-induced failure modes.

Reference:

1. Bernreuter, D. L., "Seismic Hazard Analysis, Application of Methodology, Results, and Sensitivity Studies", NUREG/CR-1582, Vol. 4, Lawrence Livermore National Laboratory, October, 1981.

———————— Median response spectrum scaled to 1g, 5% damping
------------ "Equivalent" peak ground velocity
— — — v/a = 28 in/sec/g

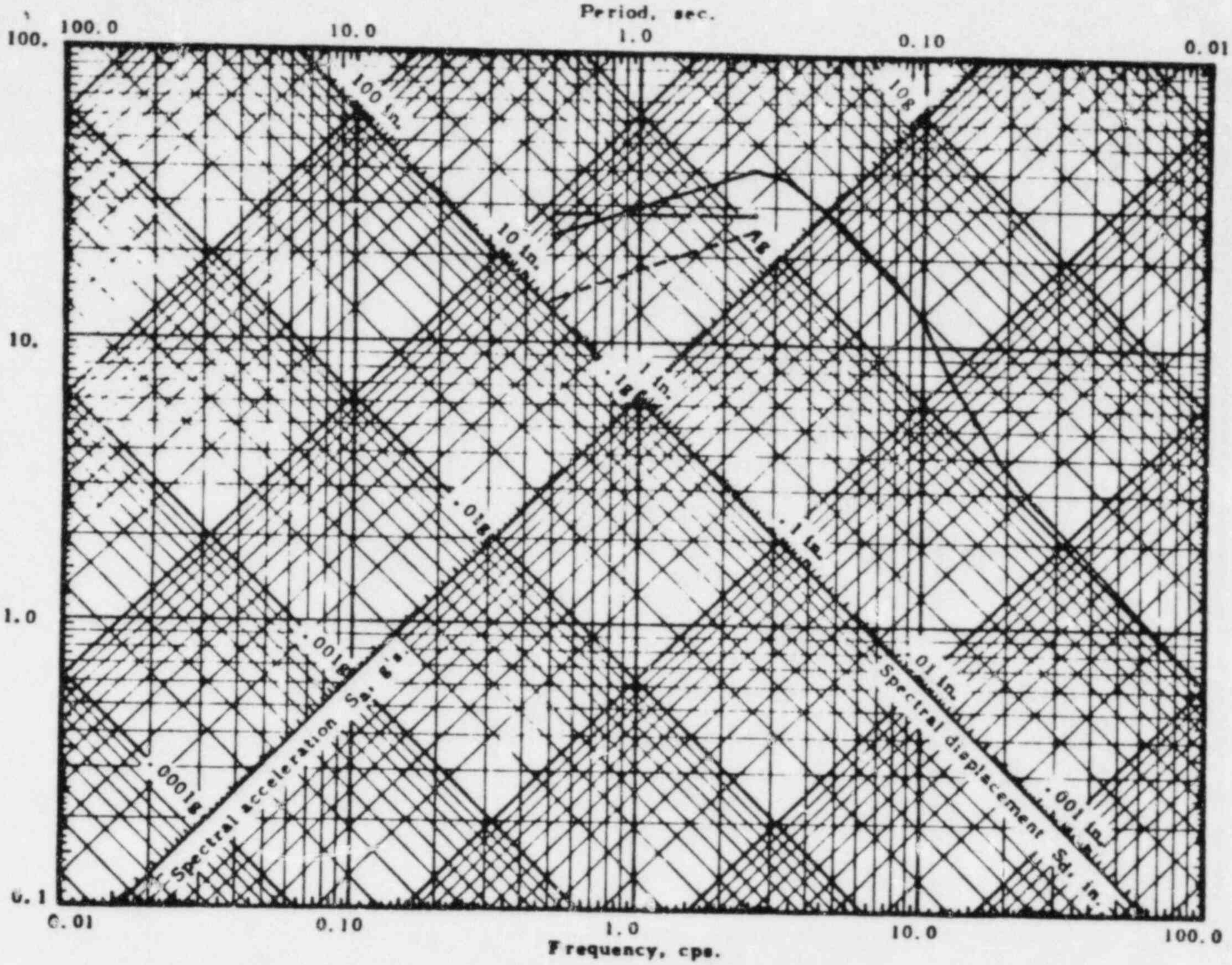


FIGURE 720.41-1. MEDIAN RESPONSE SPECTRUM AND "EQUIVALENT" PEAK GROUND VELOCITY

Question 720.43: As stated in Appendix 2-1 (3.6.1), the variability due to randomness of the spectra shape was a factor of 1.25, based upon WASH-1255, and variability in the uncertainty of the mean is assumed to be zero. Justify these values in light of the fact that the spectra shape variability is both damping and frequency dependent. For example, this factor would be about 1.5 for 5% damping and velocity amplification from WASH-1255. In addition uncertainty in the mean value could result from potential systematic uncertainty in the ground motion relationship and from your equation relating peak acceleration to sustained-based acceleration. How sensitive are your core melt results to increase in uncertainties associated with fragilities?

Response: The spectra chosen by SMA as representative broad band spectra for the Millstone 3 site were derived from Reference 1. The basis for the selection of these spectra for determination of the seismic fragilities is discussed in the SMA report and the response to Question 720.42. The determination of random variability associated with the spectral shape for a given structure damping and frequency was based upon dispersions reported in References 1 and 2. It was also recognized that there is uncertainty associated with use of the broad band spectra derived from Reference 1 as opposed to spectra for a single earthquake appropriate for the specific Millstone 3 site that could possibly be developed by further study. As noted in Section 4.1.4.1 of the SMA report, this uncertainty was estimated to be approximately 2/3 at the randomness.

There is also uncertainty in the structure response due to uncertainties in the structure damping and frequency. These uncertainties are accounted for separately in the variabilities associated with structure damping and modeling in SMA fragility report.

Reference:

1. Bernreuter, D. L., "Seismic Hazard Analysis, Application of Methodology, Results, and Sensitivity Studies", NUREG/CR-1582, Vol. 4, Lawrence Livermore National Laboratory, October, 1981.

2. Newmark, N. M., and W. J. Hall, "Developmemt of Criteria for Seismic Review of Selected Nuclear Power Plants", NUREG/CR-0098, May, 1978.

Question 720.44: As shown in Figure 7.5.1-2, 26 percent of the core melt frequency is attributable to the TE damage states. Loss of diesel start power is one component failure involved in the damage state. Section 3.4.6 of Appendix 2-1 states that amplification effects between the free field and the structure base (EGE) have been accounted for. Expand the discussion of how amplification has been taken into account in light of FSAR Figures 3.7B-63 and 3.7B-64 which show that peak acceleration is amplified by a factor of 1.7 for the EGE.

Response: The emergency generator enclosure (EGE) is founded on structural backfill and basal till overlying bedrock. The new structural fragilities for the EGE are based upon the results of the original design analysis. Seismic response of the EGE was generated by a finite element representation of the structure and the supporting soil media. The soil material properties were treated as being linearly viscoelastic. Nonlinear behavior of the soil properties was accounted for by the use of the Program SHAKE. The bottom boundary of the finite element model was established at bedrock while the side boundaries were represented as being energy transmitting.

Seismic input into the finite element model consisted of the design time-history, shown in FSAR Figure 3.7B-63, applied at the bedrock boundary. Due to combined response of the structure and the soil on which it is founded, the free-field bedrock motion is amplified. This amplification is directly accounted for in the time-history analyses generated using the finite element soil-structure model of the EGE. The degree of amplification is reflected by a comparison of the time-history records at bedrock, shown in FSAR Figure 3.7B-63, and at the base of the structure, shown in FSAR Figure 3.7B-64.

The treatment of soil amplification and soil-structure interaction effects by the original design analysis were judged to be approximately median-centered. Consequently, a median soil-structure interaction factor of unity was used by SMA in determining fragility values for the EGE.

Question 720.45: What were the bases that the project team used for selecting the seismic source zones used in the study? Were other zones considered, such as those of Algermissen et al (1982), or others (See Dames and Moore, Oct. 1983) which might increase the hazard results at the site? What data or map was used from Hadley and Devine (1974) to develop the Hadley and Devine seismic source zones?

Response: In the later study, eleven zonations (or variations on zonations) were examined and used for calculations, including those of Algermissen et al. (1982). The Tectonic Province Zones in the later study were based on sheet 3 of the Hadley and Devine (1974) study.

Question 720.46: What are the bases that the project team used for assigning the subjective probabilities for the seismic zones mentioned in the study?

Response: The basis for the subjective probabilities are described in the October 1983 Dames and Moore study in Section 3.9.

Question 720.47: How do the results of Algermissen et al. (1982) compare to those of Figure 1.2.1-1? Does the uncertainty in zonation and seismicity parameters that you have assumed reflect those of other investigators such as Algermissen et al. (1982) or TERA (NUREG/CR-1582 Vol. 3, August 1980)? Discuss differences which exist.

Response: The results of Algermissen et al. (1982) indicate higher hazard than the revised results for Millstone (October 1983) because of extremely conservative assumptions on zonation, b-value, and maximum magnitude. The range of assumptions on zonation and b-values in the revised study selects variations supported by the data available.

Question 720.48: Assumptions regarding leak path and behavior of locks, seals and O-rings are very much influenced by time-at-temperature. Also, some paths may be more prone to deterioration from high temperatures. Inflatable seals are generally exposed to local temperature effects such as in personnel airlocks for some plants. Discuss in more detail the leak paths, orientation, and critical penetration designs which can help mitigate severe accidents at Millstone 3.

Response: Section 6.2.6.2 of the Millstone-3 FSAR gives a brief discussion of primary containment penetrations whose deisgn incorporate seals, gaskets, or other similar compounds. In all instances, double seals are employed. While it is true that such seals are affected by time-at-temperature, the MP-3 PSS study (Section 4) indicated that in the vast majority of core melt accidents sequences, the highest temperatures experienced in containment would be below containment design temperature and the qualification temperature of the components.

In those sequences where higher temperatures are experienced, the large thermal mass of the containment walls itself would prevent the outer seal from heating up to the degradation temperature (typically 400°F or higher). Additional discussion is provided in the response to Question 720.19.

In the event of seal degradation, leakage from the containment might be expected to occur. Because of the small area for air flow, the leakage rate could be expected to be small, and certainly much below the leakage rate of 200 volume percent/day defined as failure in WASH-1400.

Section 6.2.3. of the FSAR describes the secondary containment at MP-3. It consists of the containment enclosure building and associated supplementary leak collection and release system (SLCRS). For containment leakage at a rate near the design leak rate of 0.2 volume percent per day, the SLCRS will remove most fission products with the exception of the Noble gases. Credit for SLCRS was not taken in the PSS for gross containment failure, but for leakage from seals the SLCRS would be very effective.

In summary, containment leakage from seals during degraded core accidents would in all likelihood be no different in magnitude than that assumed for the design basis accident. Any leakage which did occur would either be deposited within the enclosure building or be filtered out by the SLCRS. Any remaining Noble gases would be released in a controlled fashion via the Millstone-1 stack. The impact of such seal leakage on public risk, in any event, would be insignificant in comparison to the types of gross containment failure considered in the Millstone-3 P.S.S.

Question 720.49: We understand the doors to the Service Water (SW) Pump Room will normally be open. How will closure of these doors be assured during an internal or external flood? Describe the proposed emergency procedures or Technical Specification which will control these doors, or describe the ways you intend to prevent flooding in the service water pump house.

Response: The service water pump room water tight doors will be closed and administratively controlled during normal plant operations in such a fashion as to preclude flooding of one Service Water pump compartment leading to a consequential failure of the Service Water pumps in the adjacent compartments. This addresses protection against internal floods. It is important to further note that with offsite power available the plant can be safely shutdown with no service water available. This was documented in our response of January 10, 1984 in Reference 2.

When notified of an impending storm, high winds and/or high water levels by CONVEX, all pumphouse doors will be closed in accordance with an operating procedure. This addresses protection against external flooding.

Question 720.50: What is the capacity of the sump pumps in the SW pump house? Where do the floor drains empty to? Are the SW pump house sump pumps on the QA list? Are there check valves in the drain lines?

Response: No credit was assumed for the action of sump pumps or floor drains in the Millstone-3 P.S.S.

Question 720.51: If a Component Cooling Water pump or line were to rupture in the Service Water pumphouse, what would be the effect on the Service Water pumps (their doors are normally open) and their electrical equipment?

Response: The Service Water pumps are self-cooled and thus there are no Component Cooling Water pumps or lines located in the Service Water pumphouse.

The rupture of a single Service Water train is addressed in Section 1.2.4.3 of the Millstone - 3 P.S.S. Such a failure is assumed to fail both pumps in the compartment in question. The plant has committed to procedurally requiring the watertight doors to be closed while under normal operation in such a fashion as to preclude flooding of one Service Water pump compartment leading to a consequential failure of the Service Water pumps in the adjacent compartment.

As a further point, it should be noted that with offsite power available the plant can be safely shutdown even with no Service Water available. This was documented in our response of January 10, 1984 in Reference 2.

Question 720.52: We understand the analysis of local flooding due to runoff demonstrates only about a 0.3 inch margin below an unspecified door sill. We understand this runoff is not based on the local probable maximum precipitation. Provide a probabilistic analysis of runoff flooding based on the local probable maximum precipitation. Identify all entries which are vulnerable and identify all safety related equipment which could be affected by such flooding.

Response: A formal technical response to this question has already been provided on January 10, 1984 in Reference 2.

Question 720.53: Address Pressurized Thermal Shock in a probabilistic manner. Provide downcomer temperature versus frequency curves. Benchmark these curves based on Millstone 3 vessel characteristics.

Response: A formal technical response to this question has already been provided on January 10, 1984 in Reference 2.

Question 720.54: The development of the Boolean expression for the plant damage states considered in the seismic risk analysis (see Section 2, pages 2.5-9 to 2.5-18 of the Millstone 3 PSS) requires clarification. For example:

(a) There appear to be typographical errors which make reading the material difficult. For example, in the expression for SE on page 2.5-18, M_3 should have a bar over it, and 116 should also have a bar over it.

(b) On page 2.5-16, the expression for SE and TE do not appear to be mutually exclusive. They both contain the term

$$ATWS + \overline{M}_3 + QS + RS$$

(c) The method of inclusion of seismic-induced containment sliding in the seismic risk-analysis, as is discussed on the bottom of page 2.5-16, is unclear and may be incorrect. A simple, more direct approach would be to multiply the plant damage state descriptors by a Boolean variable Y, or a Boolean variable Y (equal to ' not Y'), where Y denotes containment failure by the sliding mode. Then the conditional probabiilty of the modified plant damage states should be calculated, for the various discrete values of the ground acceleration levels considered. In other words, one needs to calculate quantities like

$$P(SEY) = \int P(SEY|a)\, g(a)\, da$$

where g(a)da is the frequency of earthquakes with peak ground accelerations between a and a + da.

Response: NU is performing a revised Seismic Core Melt and Risk Analysis which will be documented in a formal amendment to the Millstone - 3 Probabilistic Safety Study. This is being performed because of errors in both seismic fragility calculations and seismic-induced failure definitions. This amendment will clarify the Seismic Core Melt and Risk Analysis currently in the PSS. The following responses apply to parts a, b, and c of Question 720.54, respectively.

(a) The mutually exclusive plant damage state Boolean expressions with respect to the contributing sequence are as follows.

For the SE plant damage state:

$$PSL(SE) = \{I3 \wedge (I2 \vee I13 \vee I21) \wedge \overline{AC} \wedge \overline{I6}\} \vee \{I12 \wedge (I2 \vee I13 \vee I20 \vee I21) \wedge \overline{I1} \wedge \overline{I16}\}$$

$$ATWS(SE) = (I10 \vee I18) \wedge \{I13 \vee I21 \vee [(I9 \vee SV) \wedge (I2 \vee I20)]\} \wedge \overline{I16} \wedge \overline{AC} \wedge \overline{PSL}$$

For the TE plant damage state:

$$M_1(TE) = I13 \vee I21 \vee (I9 \wedge I2) \wedge \overline{M}_2 \wedge \overline{M}_3$$

$$ATWS(TE) = (I10 \vee I18) \wedge \{I2 \vee [(I6 \vee OA8) \wedge I20]\} \wedge [\overline{I9} \wedge \overline{I13} \wedge \overline{I21} \wedge \overline{SV} \wedge \overline{I16}] \wedge \overline{PSL} \wedge \overline{AC}$$

For the SEC plant damage state:

$$PSL(SEC) = \{I3 \wedge (I9 \vee I15 \vee R2) \wedge I6\} \wedge \overline{QS} \wedge \overline{RS} \wedge \overline{I1}$$

$$ATWS(SEC) = [(I10 \vee I18) \wedge (I9 \vee SV)] \wedge \overline{QS} \wedge \overline{RS} \wedge \overline{PSL} \wedge \overline{AC}$$

For the TEC plant damage state:

$$M_1(TEC) = I9 \wedge (OA7 \vee I6) \wedge \overline{QS} \wedge \overline{RS} \wedge \overline{M}_3 \wedge \overline{M}_2$$

$$ATWS(TEC) = (I10 \vee I18) \wedge (I6 \vee OA8) \wedge \overline{I9} \wedge \overline{SV} \wedge \overline{QS} \wedge \overline{RS} \wedge \overline{PSL} \wedge \overline{AC}$$

(b) The expression on page 2.5-16 were not intended to be mutually exclusive. These expressions are general in nature and are only intended to show the type of seismic-induced core melts than could result in the various plant damage states. As discussed on page 2.5-16 and shown in Figure 2.2.7.22-1 an ATWS core melt sequence could result in either an SE or TE plant damage state depending upon the actual system failures for the sequence. Therefore the term

$$ATWS + \overline{M}_3 + QS + RS$$

appears in both the SE and TE plant damage states.

(c) The work being performed to revise the Seismic Core Melt and Risk Analysis includes recalculation and requantification of seismic-induced containment sliding. Based on the preliminary fragility analysis of this work the importance of containment sliding with respect to Seismic Risk will be greatly reduced.

Question 720.55: There appear to be some difficulties with the ATWS analysis, event tree 22.

We note:

(a) The success requirements for the auxiliary feedwater system have been assumed to be two motor driven pumps or one turbine-driven pump in other ATWS analyses (e.g., the Zion Probabilistic Safety Study).

(b) The human error probability of 1% for the failure to perform manual scram (RT-3) appears quite low for an action which must be performed in one minute.

(c) The failure of manual scram, given failure of automatic scram, should take into account not only the fact that manual scram may be impossible if there is a mechanical failure of the control rods to insert, but also that certain electrical reactor protection system failures (in particular, failure of the scram breakers) may also make manual scram impossible.

(d) The assumption that failure of the pressurizer safety valves to reclose results in a core melt sequence may be overly conservative. The report NUREG-0460 states on p. X-4 of Volume 2 that the TKQ sequences are now believed not to melt the core. If you are aware of any generic analyses applicable to Millstone-3 which verify this please reference them and modify your analysis accordingly.

(e) We note that for a sufficiently unfavorable moderator temperature coefficient one needs a pressurizer PORV to open as well as the three safety valves, in order to obtain adequate pressure relief in an ATWS from full power. This may not have a significant impact on the analysis unless the plant is operated for an appreciable fraction of the time with the PORVs blocked. Is it intended that the plant be operated with the PORVs blocked or unblocked?

Response: (a) Westinghouse has already acknowledged this non-conservative assumption in the ATWS event tree analysis during the comprehensive technical review. Subsequently, sensitivity analyses were performed to determine the possible impact on core melt frequency and risk. The unavailability at the auxiliary feedwater system used in the sensitivity was adjusted to be consistent with the Zion and Indian Point PSS analysis and representative of the unavailability if the success criteria was either two motor driven AFW pumps or one turbine driven pump supplying all four steam generators. The resulting increase in core melt frequency was determined to be less than .5% of the total internal core melt frequency or approximately 3×10^{-7}/yr. Based on this negligible increase, and the fact that all ATWS events were analyzed assuming a complete loss of normal feedwater, this error was considered to be insignificant. Furthermore, the resulting increase in core melt frequency did not occur in plant damage states which affect the risk of the plant.

(b) A human error probability of .01 was utilized for failure of the operator to perform manual reactor trip. This value was based on the fact that the conditions are easy to recognize and reactor trip is a major item of concern for the operator and is emphasized in training. The very first step of all emergency procedures is to confirm or initiate reactor trip. The operator would thus manually trip the reactor if it did not automatically trip. If this still does not result in reactor trip the operator follows the emergency procedure pertaining to ATWS. This procedure includes the operator action of manually driving in the control rods if both automatic and manual trip are unseccessful. This procedure also addresses emergency boration.

(c) Failure of manual scram, given failure of automatic scram, could be caused by mechanical failure of the control rods to insert or common cause failure of the reactor trip breakers. There are bypass breakers which will be utilized for testing of the reactor trip breakers during normal operation. Because of the testing of the reactor trip breakers and the awareness of the importance of the reactor trip breakers it was judged that the failure of manual scram would be dominated by mechanical failure of the control rods to insert.

(d) In the interest of analytical efficiency, the TKQ accident sequence was assumed to result in core melt. The PRA analysis team acknowledges that this assumption is overly conservative but the special treatment required to model this sequence (i.e., modeling high pressure safety injection, recirculation, etc.) does not provide a sufficient benefit. A sensitivity study was performed which shows less than a .5% decrease in core melt frequency would result if stuck open safety valve failures were recoverable.

(e) The PORV block valves will be open during normal operation unless PORV leakage warrants closure of the associated block valve. However, even if the PORV block valves were closed an applicable fraction of the time this would not have a significant impact on the analysis. This is due to both the limited time when the moderator temperature coefficient would be sufficiently unfavorable and the conservative value utilized in the quantification of failure of ATWS pressure relief. (See Section 2.2.3.4 of the MP3-PSS and response (d) above).

Question 720.56: What equation(s) were used to relate epicentral intensity of m_b? Clearly the equation given in Appendix 1-B was not used in all cases as implied.

Response: The equations used to relate epicentral intensity to m_b, in the revised study, are given as equations (1) and (2).

Question 720.57: Why were only 4 zonation models used - particularly when many other zonations exist which results in higher hazard at the Millstone site than obtained with the very limited choice of zonations used?

Response: The later study (October 1983) considers eleven zonation models, and reflects the range of those which are available.

Question 720.58: It is implied that your magnitude-recurrence model is in terms of m_b yet the values given in Table 1 of Appendix 1-B are in terms of intensity. Just when and how was the conversion made for use in the ground motion models in Section 5 of Appendix 1-B.

Response: In the revised study, historical intensities were converted to m_b using equations (1) and (2). Seismicity was represented using m_b exclusively; this is consistent with the attenuation relations used.

Question 720.59 McGuire's program makes use of truncated exponential model rather than the relation (Eq. 2) given in Appendix 1-B, Section 4. Just what recurrence model was used in the analysis? If the truncated exponential was in fact the model used, how well did it fit the historic data when m_b, max was at its minimum value?

Response: The truncated exponential model was used for magnitude. It fits the data adequately when $m_{b,max}$ is at its minimum value.

Question 720.60: What corrections were applied to the historic data to obtain the "a" and "b" values given in Table 1 of Appendix 1-B. How were the "a" and "b" values obtained from corrected data? What variation was used for the uncertainty in the estimates of the "a" values? If none was used, justify relative to the historic data set and corrections for incompleteness used.

Response: The method of "correcting" for incompleteness of historical data is described in section 4.1 of the revised report, as is the method of calculating "a" and "b" values. The number of events used to calculate activity rate implies that uncertainty in this parameter (from a statistical standpoint) is low. The completeness periods were chosen to give the highest rates possible; alternative completeness periods would have the effect of reducing the estimated rate of activity.

Question 720.61: Other credible experts (e.g. the set of experts used in the SEP study documented in NUREG/CR-1582, Vols 1-5 and in the latest USGS study by Algermissen et al.) have used, in additional to different zonations, different values for the seismicity parameters than used in your study. The work of these other credible experts results in higher hazard curves than obtained using your restricted set of models. Justify not including a reasonable sample of these models in your analysis.

Response: Other studies have assumed b-values, activity rates, and maximum magnitudes which are more conservative than those used in the Millstone studies. Where comparisons with parameter values from other studies are possible, they indicate that the values used in the Millstone studies are justified by the data, and values used in other studies are conservative.

Question 720.62: It is coefficient of the R term in Eq. (3) of Section 5 of Appendix 1-B correct? The value of 0.0032 appears more consistent with the use of the natural log rather than the base 10 indicated. If the value is correct, justify the use of such a large value for this attenuation coefficient.

Response: The attenuation equation used in the original study has been superceded in the later Dames & Moore study (October 1983). Equations (4), (6), (12), and (13) give the attenuation equations used in the revised study.

Question 720.63: Justify the use of only two very similar ground motion models (neither of which is based on EUS ground motion data) with similar values for the random uncertainty when what little recorded strong ground motion data for New England Region which exists is much higher than obtained from either of these two models.

Response: Four attenuation equations are used in the revised study, and they give a range of results consistent with uncertainty in ground motion estimation. The ground motion data from New England is not inconsistent with two of the attenuation functions used in the revised study.

Question 270.64: It is very difficult to interrelate Table 1.2.1-2, Fig. 1.2.1-1 and the results/discussion given in Appendix 1-B. Added explanation would be useful. For example, what sets of models led to the "bounding curves" What range existed for the full 36 hazard curves?

Response: A more complete discussion of the relationship between assumptions and results is given in section 6.0 of the revised report.

Question 720.65: In regards to the resistor inside the diesel-generator control cabinet which we examined during the plant visit on December 14th (the resistor was attached to the under-frequency relay), was the flexibility of the resistor mounting considered in the development of the fragility parameters for the diesel-generator control cabinet?

Response: NUSCO is presently evaluating the original qualification reports of said cabinet to verify the tested configuration. The results of this evaluation will be transmitted upon its completion.

Question 720.66: Provide the steam generator U-tube failure seismic stress report(s). The reference given in the PSS calculation is: Westinghouse letter NEU-4346, WMRA-133, NUSCO Millstone 3 Risk Assessment, Final Transmittal of Information for S & W Fragility Analyses, to B. L. Carlson from R. W. Hofer dated September 30, 1982.

Response: The requested data used in the Millstone - 3 P.S.S. is attached.

STEAM GENERATOR

The seismic analysis of the Millstone Unit #3 Model F steam generator was not completed until late in 1983. Consequently, at the time the PSS was performed, the seismic results for Korea Unit #2, which uses the Model F steam generator, were used.

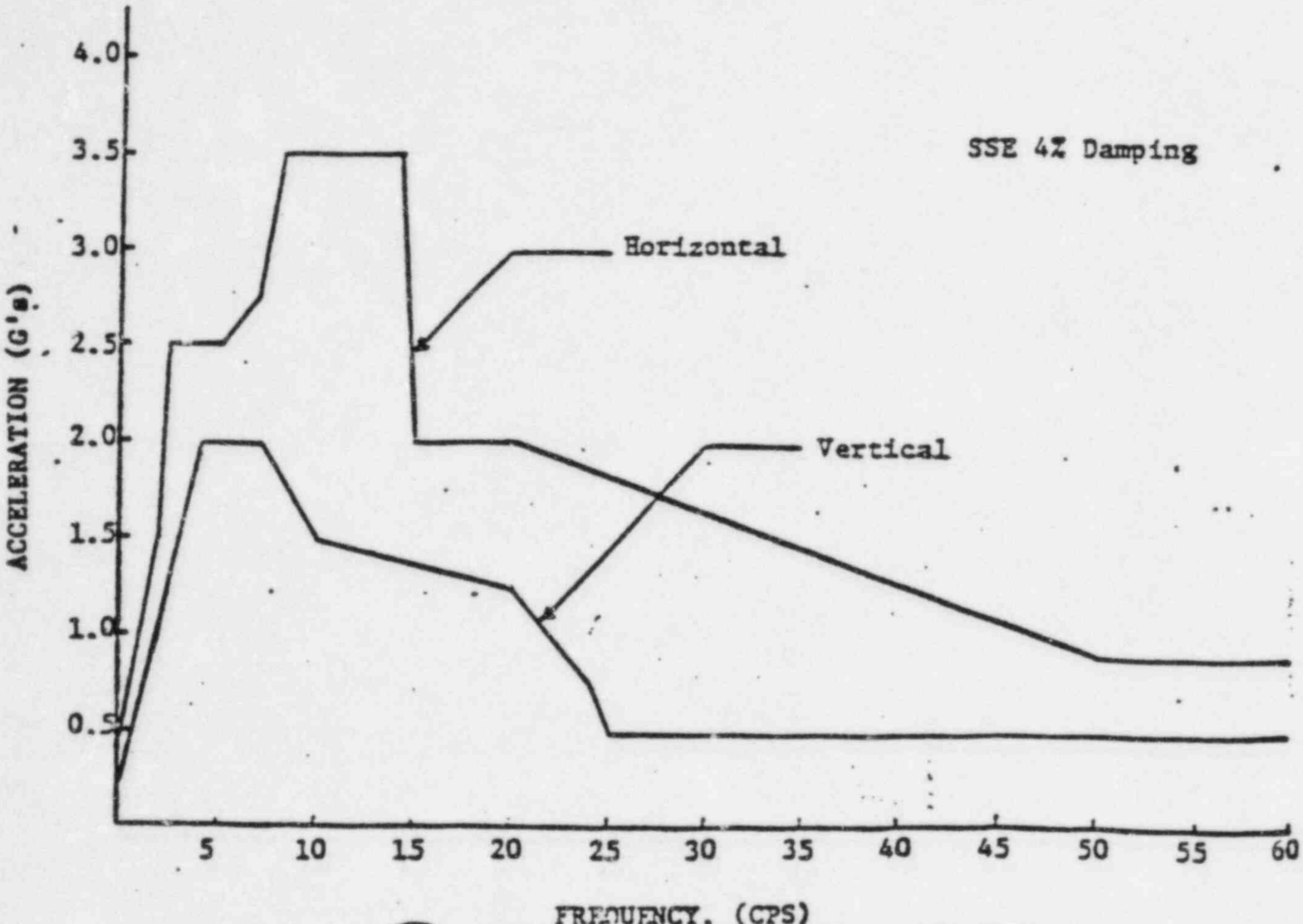
The steam generator U-tubes are the location of the highest seismic stress. The U-tubes are stressed to 34.1 ksi with a limit of 56.0 ksi. The seismic portion of this stress is 23.3 ksi.

Attached are the response spectra curves for Korea Unit #2 and generic Model F qualification. The loading for this condition is SSE + Normal. The stresses are from a 3-D analysis using 4% equipment damping.

The Millstone specific response spectra curves and the generic Model F qualification curves are attached for comparison.

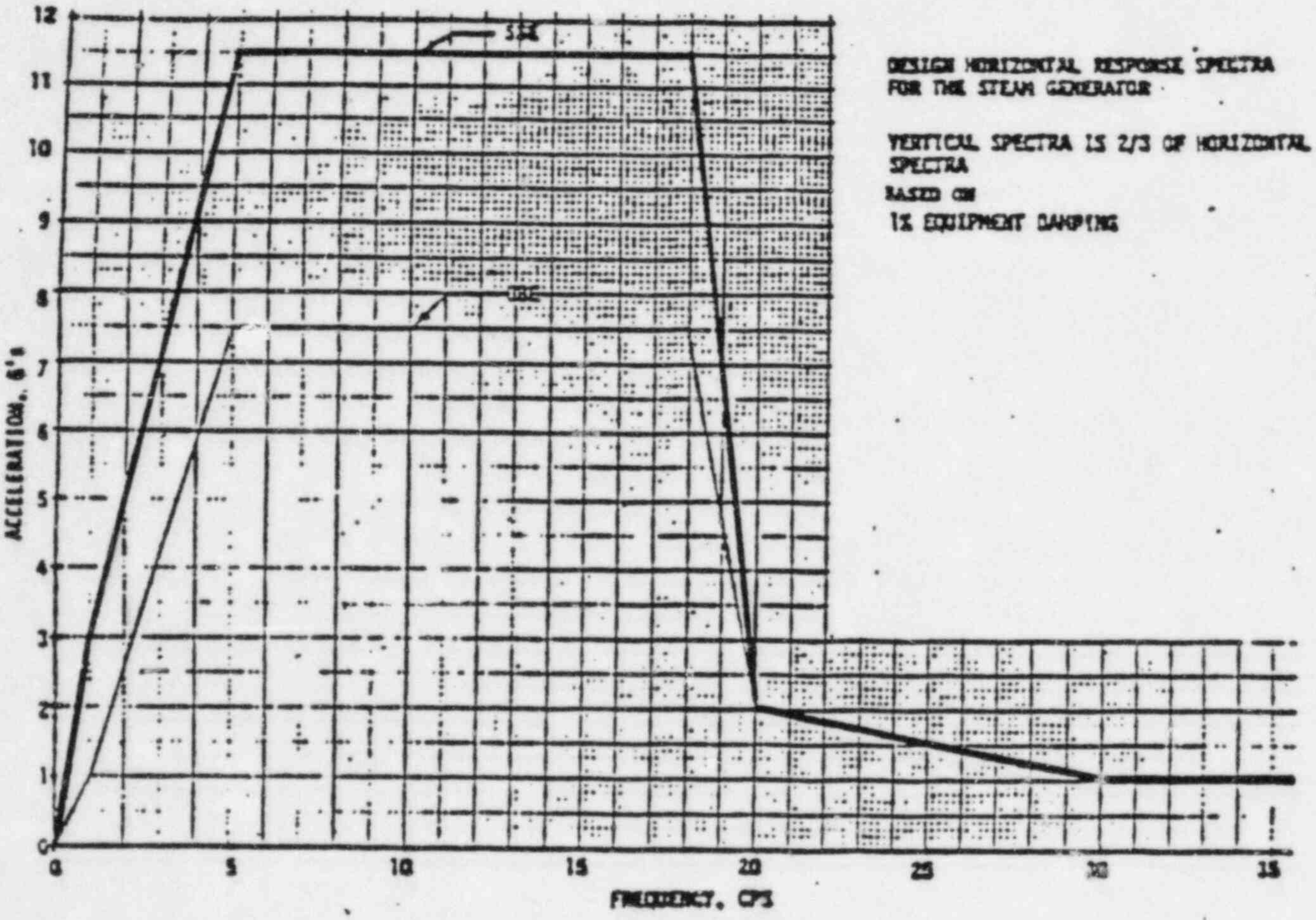
The steam generator U-tubes in the region of the U-bend are the location of the highest seismic stress. The U-tubes are stressed to 46.7 ksi with a limit of 75.0 ksi (the limit of $0.7S_u$-56.0 ksi reported in Reference 1) should have had a thin-wall tube factor of 1.34 applied for a total limit of 75.0 ksi). The seismic portion of this stress is 20.0 ksi. The loading condition for these stresses is SSE + 100% power operation. This was also a 3-D seismic analysis with 4% equipment damping.

MODEL F STEAM GENERATOR



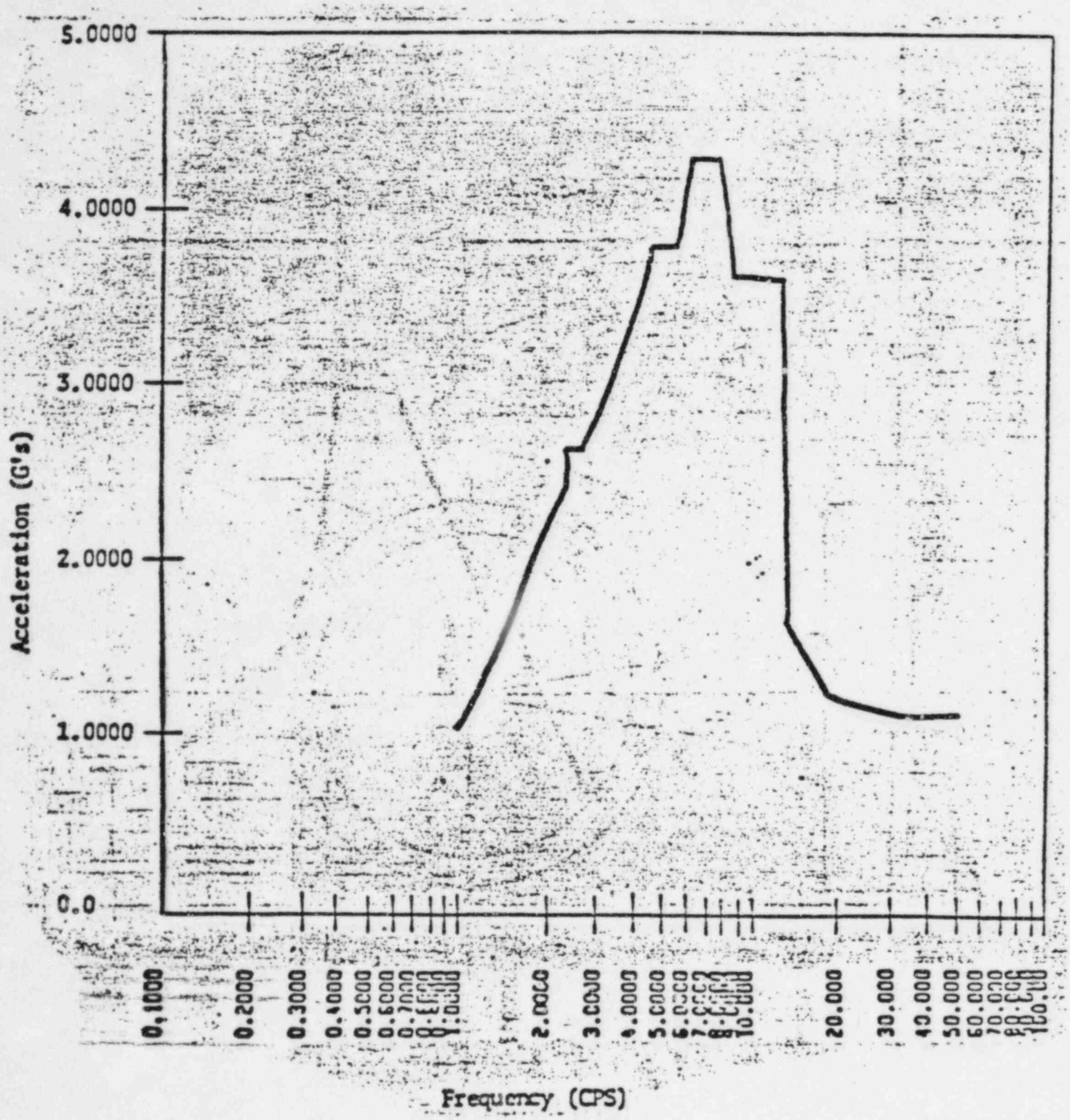
KPR Floor Response Spectra

KOREA UNIT #2



Model F Steam Generator Response Spectra

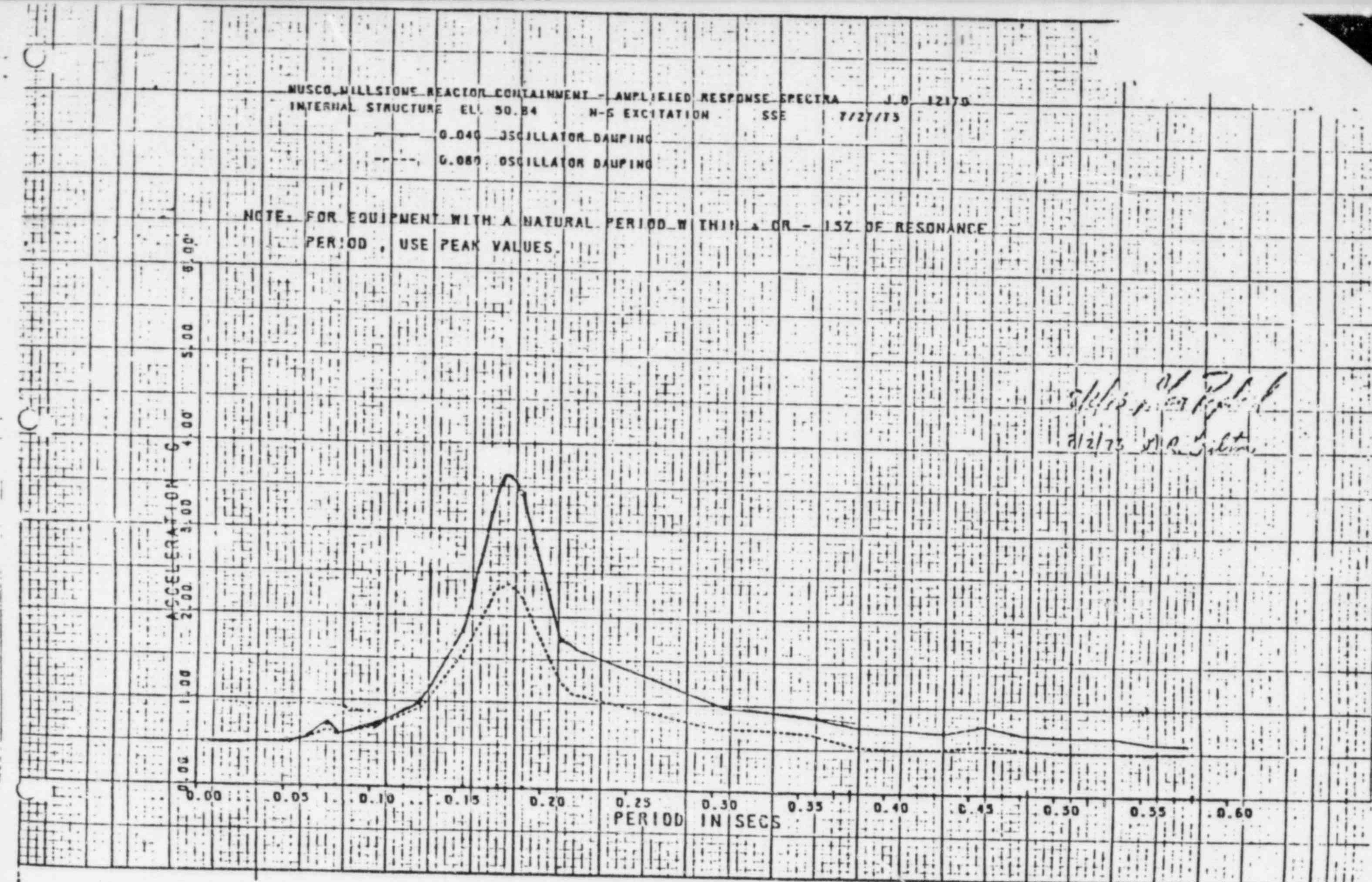
GENERIC MODEL F



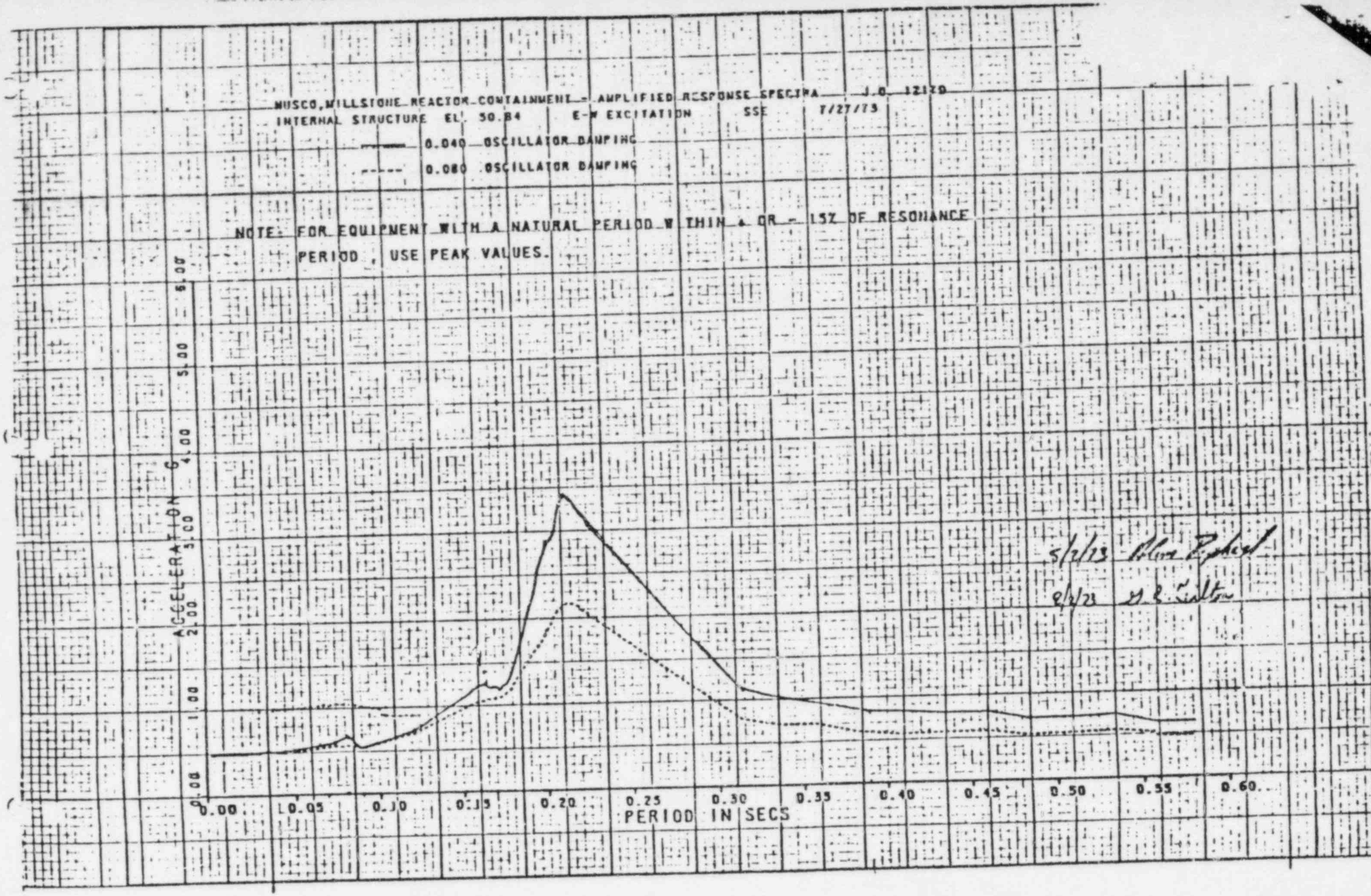
DESIGN HORIZONTAL SPECTRA FOR THE STEAM GENERATOR
VERTICAL SPECTRA IS 2/3 OF HORIZONTAL SPECTRA

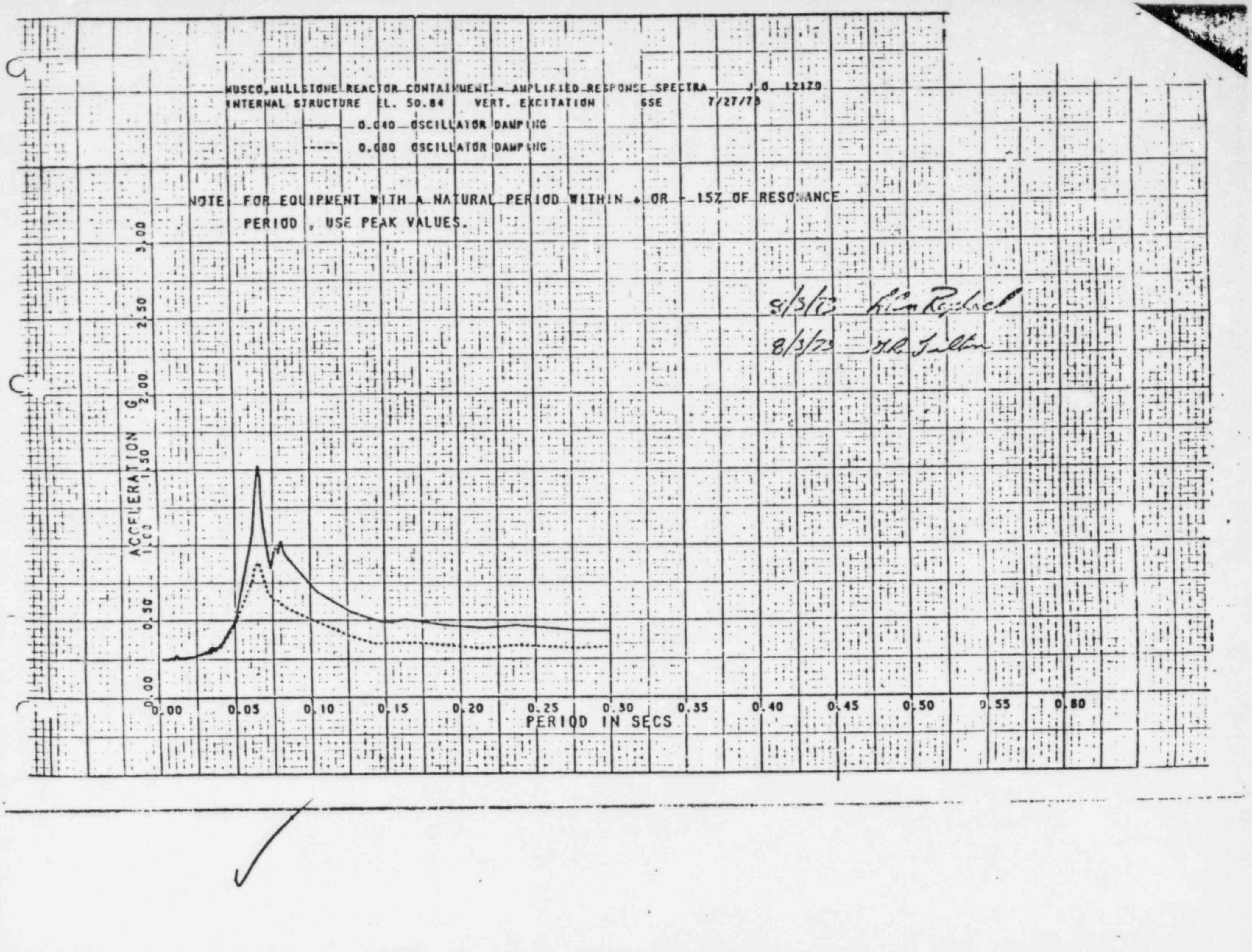
GENERIC MODEL F

4% EQUIPMENT DAMPING
SSE



NUSCO MILLSTONE REACTOR CONTAINMENT - AMPLIFIED RESPONSE SPECTRA J.O. 12179
INTERNAL STRUCTURE EL. 50.84 N-S EXCITATION SSE 7/27/73
0.040 OSCILLATOR DAMPING
0.080 OSCILLATOR DAMPING
NOTE. FOR EQUIPMENT WITH A NATURAL PERIOD WITHIN + OR - 15% OF RESONANCE PERIOD, USE PEAK VALUES.
ACCELERATION G
0.00
1.00
2.00
3.00
4.00
5.00
6.00
0.00
0.05
0.10
0.15
0.20
0.25
0.30
0.35
0.40
0.45
0.50
0.55
0.60
PERIOD IN SECS


NUSCO, MILLSTONE REACTOR CONTAINMENT - AMPLIFIED RESPONSE SPECTRA J.O. 12179
INTERNAL STRUCTURE EL. 50.84 E-W EXCITATION SSE 7/27/73
0.040 OSCILLATOR DAMPING
0.080 OSCILLATOR DAMPING
NOTE: FOR EQUIPMENT WITH A NATURAL PERIOD WITHIN + OR - 15% OF RESONANCE PERIOD , USE PEAK VALUES.
ACCELERATION G
0.00
1.00
2.00
3.00
4.00
5.00
6.00
0.00
0.05
0.10
0.15
0.20
0.25
0.30
0.35
0.40
0.45
0.50
0.55
0.60
PERIOD IN SECS
5/2/73
8/2/73


NUSCO, MILLSTONE REACTOR CONTAINMENT - AMPLIFIED RESPONSE SPECTRA J.O. 12170
INTERNAL STRUCTURE EL. 50.84 VERT. EXCITATION SSE 7/27/73
0.040 OSCILLATOR DAMPING
0.080 OSCILLATOR DAMPING
NOTE: FOR EQUIPMENT WITH A NATURAL PERIOD WITHIN + OR - 15% OF RESONANCE PERIOD, USE PEAK VALUES.
8/3/73
8/3/73
ACCELERATION G
0.00
0.50
1.00
1.50
2.00
2.50
3.00
0.00
0.05
0.10
0.15
0.20
0.25
0.30
0.35
0.40
0.45
0.50
0.55
0.60
PERIOD IN SECS

Question 720.67: Recent analyses (see reference below) has indicated that for high pressure core melt accidents, the steam generator tubes may become overheated to the failure point, releasing radionuclides to the secondary and possibly out the ADV's. This scenario does not seem to have been considered in depth in the PSS. Has NUSCO considered and analyzed this possibility? If so, can results be supplied?

Reference: Transient Analysis in a PWR, L. Winters, Engerleondezoek Centrum Nederland (ECN), July 1982.

Response: The Millstone - 3 P.S.S. considered in Appendix 4-J (Section 4-J.3) a number of high pressure core melt scenarios which could potentially lead to early containment failure including: in-vessel steam explosion and in-vessel steam spikes failing the steam generator U-tubes. Our technical assessment of the issue of hypothetical containment failure via early steam generator tube failures concentrated mainly on scenarios involving high pressure induced failures. Under such scenarios the reactor core would be in a highly degraded state but the vessel would not have failed yet (thus giving rise to high internal pressures). All other investigations in this area have assumed that reactor vessel failure <u>would have already occurred</u> prior to the time that temperatures high enough to lead to steam generator tube failure could exist.

The consideration of hypothetical temperature induced steam generator tube failure apparently originates from an obscure Dutch publication not readily available within the United States. NUSCO has made repeated attempts to obtain a copy of the referenced report without success. If a ledgible copy of this report can be made available to NUSCO, a thorough technical evaluation of this issue will be performed.

Question 720.68: The emergency diesel load sequencer strips loads from the safety buses upon receipt of a LOSP signal. The sequencer subsequently blocks manual reloading of the buses until the diesel-generator starts and the breaker closes. The sequencer then loads the safety buses and blocks manual trips. Has the possibility been considered that the load sequencer could fail after stripping loads and blocking manual loading of safety equipment? What would the recovery procedure(s) be for this failure? Can the operator override the manual trip block?

Response: The type of failure, described in Question 720.68, which could lead to the inability to start safety equipment is considered to be non-credible.

If the emergency generator load sequencer (EGLS) failed to load safety equipment after stripping loads and blocking manual starts, the operator would manually load ESF equipment as part of his emergency equipment start verification procedure. A total failure resulting in the inability to load equipment, both automatically and manually, would require a highly unlikely combination of selective failures and successes in the EGLS logic. This is shown by the following example. Initially, the manual start block (MSB) would have to be successful in blocking manual equipment starts. This would have to be followed by failure of safeguard sequencer start (SSS) signals to ESF equipment. Finally, the MSB would have to fail (after successful initial operation) to automatically terminate at forty seconds after the emergency generator was loaded on its bus. The sequencial occurrence of these events is considered to be non-credible.

The second part of Question 720.68 asks whether the operator can override the manual trip block. All MTB's can be overidden or reset by accomplishing two operator actions. First, each EGLs train must be individually reset. This action involves resetting the particular signal or each of the separate signals (if there are more than one) that caused actuation in the first place. Second, the equipment control switch must be placed to the reset position. Only after performing both actions can individual ESF equipment then be tripped.

Question 720.69: The vital 120 VAC buses are continuously monitored and displayed in the control room. An alarm is sounded in the control room on "change of state in the static transfer switch due to loss of inverter output." What is it that is actually sensed and alarmed? How is it sensed?

Response: Each vital AC bus has an inverter trouble alarm associated with it. The alarm sounds and lights up as "120 VAC INVERTER X TROUBLE" on main board #8 ("X" is the number of the inverter of which there are 6)

Any one of the following conditions will produce the alarm and light up the trouble message:

(1) Loss of the preferred AC input to the rectifier which connects to the inverter. This is sensed by an alarm relay.

(2) Opening of the vital DC battery breaker which is between the inverter and the vital DC bus. This is sensed by aux. breaker contact.

(3) Inverter <u>output</u> voltage low is sensed by an undervoltage relay.

(4) Transfer to the alternate AC Source is detected by an alarm relay.

Question 720.70: If offsite power is lost, onsite AC can only be provided if the batteries are available since the DGs require DC power to flash their field and perhaps close breakers. Your modeling of the vital 120 VAC does not consider station batteries. What is the effect on the reliability of vital 120 AC power of including station batteries in your modeling? In particular, we are interested in the modelling of the sequence:

(1) Loss of Offsite Power

(2) Batteries unavailable because of prior undetected fault
(DC supplies before Loss of Offsite Power by chargers)

(3) DGs cannot supply AC because DC unavailable

(4) Therefore, no DC or AC is available.

Response: A formal technical response to this question has already been provided on January 10, 1984 in Reference 2.

Question 720.71: In the support system analysis, why was vital 120 VAC evaluated as if offsite power was available for all cases - even when offsite power is not available?

Response: The support system analysis models vital AC power as part of the EGLS and ESF actuation systems. Thus, vital AC is not explicitly part of the support state model.

Vital AC power is only important to the EGLS and ESF actuation systems when a source of AC power is available to power the trains of ESF equipments that they actuate. As an example, refer to the support state model shown as Figure 2.2.1.3.5-1 in the Millstone 3 PSS. In support states 1-52, either offsite or emergency onsite AC power is available on both 4160V ESF buses. Failures in either the EGLS or ESF actuation trains will prevent the corresponding ESF equipment from operating. In support states 53-68, only one ESF bus is available so that there is no reason to ask whether the EGLS or ESF actuation systems worked for the train where AC power is not available. Furthermore, since vital AC is a support system to the above actuation systems there is no reason to ask whether it worked. As a result, vital AC is only quantified and credited for cases where AC power is available to operate ESF equipment.

Question 720.72: The fault tree for the main electrical system appears to contain a logical inconsistency concerning the circuit breaker between the diesel-generator and the corresponding emergency bus (due to no credit being given to the batteries). The breaker requires that a trip coil be energized by a trip contact that must be closed (manual or automatically). The fault tree for this system shows that failure of this trip contact requires failure of both the manual and automatic modes. The auto-trip contact requires a signal from the EGLS for operation. However, the EGLS is modeled as failed when there is no power on the 120 VAC vital bus from the corresponding power train, and this train is modeled as failed when there is no power on the corresponding emergency AC bus (34C or 34D). Thus, if power is available to operate the auto-trip coil, then the circuit breaker will not be called upon and if the circuit breaker is called upon, then the auto-trip coil will likely not be available.

Is this a simple logic error in the fault tree? If not, please provide an explanation of the logic for the system.

Response: The logic for the system is discussed in the responses to Q720.71 and Q720.73a and amplified in the response to Q720.25.

Question 720.73a: In the EGLS fault tree, the dependence of the single sequencers on the corresponding vital AC and vital DC systems does not appear to be correctly modeled. In particular, the fault tree does not, address the fact that following a loss of offsite the EGLS would be the primary initial support system and that for the first 10 to 40 seconds following this event, it would be functioning with AC power unavailable on bus 34C and 34D, i.e., it would be dependent on station batteries. Provide an explanation.

Response: In the response to Question 720.71, it was noted that the vital AC power system was only quantified for those cases where AC power was available. The following discussion will show why vital AC was not quantified and used for the brief period when there is no AC power at all, following a loss of offsite power.

After a loss of offsite power, the diesel generators start on their own and motors on the ESF buses are tripped via the EGLS. The EGLS must have vital AC power in order to perform its function. Prior to any loss of offsite, the vital AC system receives the feed on its vital buses from a preferred AC source. This source is from a 480V MCC which is first rectified to D.C. and then inverted to 120VAC before feeding the vital bus. When offsite is lost, there is an initial brief period when the preferred AC source is not available. During this time, the vital AC buses are powered by their associated vital D.C. batteries. Each battery is already in parallel with the preferred source on a common inverter, so that no change of state is required to allow the inverter to continue feeding the vital bus. At 10 seconds after receiving the start signal, both diesel generators are loaded on to their respective ESF buses. The 480V MCC's which provide the preferred vital AC source are immediately picked up because they are never shed from the bus. Thus, the vital AC system only has to function for a brief period of time without AC power.

When the vital AC system was modeled, the total time it was quantified for spanned 28 hours. This was based on a fault detection interval of 4 hours plus a mission time of 24 hours. The unavailability of this system over 28 hours, with AC available, is much greater than the unavailability of the system over 60 seconds with just vital DC available. This latter unavailability is approximately 0.5% of the total vital AC system unavailability used to quantify the EGLS fault tree. The unavailability of vital AC during the 10 second plus time interval without any AC is considered to be bounded by the case with AC available.

Question 720.73b: The unavailability of both EGLS cabinets is apparently dominated by common cause failures, but the common cause failure probability used in the analysis is based on the electrical portion of the reactor protection system (RPS) shown in NUREG-0460. Provide justification for the use of this number in place of a more rigorous evaluation, with due consideration for the major contribution of the EGLS to the latent cancer fatality risk.

Response: The two Emergency Generator Loading Sequencer (EGLS) cabinets are identical solid state digital systems which are powered from separate 120V AC vital buses. The input signals to one EGLS cabinet are independent from the input signals to the other cabinet. As discussed in Section 2.3.3.4.4, an automatic test sequence, performed at intervals of every 30 seconds, verifies all critical electrical paths of each EGLS. In the support state event tree, Figure 2.2.1.3.5-1, EGLS is only addressed if there is power available at 4160V AC buses 34C and 34D. Therefore, the common cause loss of power was addressed in Section 2.3.3.1. Also as discussed in Section 2.3.3.1.5 components such as wiring, circuit breakers, protective relays, etc. have negligible common cause failure rates as compared to their random failure rates. Therefore, common cause failure between the inputs to both EGLS cabinets would dominate common cause failure of both EGLS cabinets. As discussed in Section 2.3.3.3.5, the best estimate common cause failure probability was judged to be 1.5×10^{-5} per demand.