NSP

Northern States Power Company
Prairie Island Nuclear Generating Plant
1717 Wakonade Dr. East
Welch, Minnesota 55089

February 28, 2000

Generic Letter 88-20

U S Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket Nos. 50-282 License Nos. DPR-42
50-306 DPR-60

Response to Request for Additional Information Regarding Report NSPLMI-96001, Individual Plant Examination of External Events (IPEEE), Related to Generic Letter 88-20

In December 1996 and October 1998, we submitted, respectively, report NSPLMI-96001, PINGP Individual Plant Examination for External Events (IPEEE) and Revision 1 to that report, in response to Generic Letter 88-20. This letter responds to a Request for Additional Information (RAI) contained in a letter from the NRC dated May 20, 1999.

The RAI has three areas of concern: 1) Seismic, 2) High Winds, Floods, and Other Events, and 3) Fire. The responses to the questions related to areas of concern #2 and #3 were submitted by letter dated September 17, 1999. As discussed in that letter, additional seismic evaluation was necessary. That evaluation has been completed and our response to the seismic area of concern is attached to this letter.

In this letter we have made no new Nuclear Regulatory Commission commitments. Please contact Jack Leveille (651-388-1121, Ext. 4142) if you have any questions related to this letter.

Joel P. Sorensen

Joel P. Sorensen
Site General Manager
Prairie Island Nuclear Generating Plant

c: (see next page)

A011

c: Regional Administrator - Region III, NRC
Senior Resident Inspector, NRC
NRR Project Manager, NRC
J E Silberg

Attachments:

1. Affidavit
2. Response to RAI Seismic Questions

UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND NUCLEAR GENERATING PLANT   DOCKET NO. 50-282
50-306

GENERIC LETTER 88-20, INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS FOR SEVERE ACCIDENT VULNERABILITIES - 10 CFR 50.54(f)

Northern States Power Company, a Minnesota corporation, with this letter is submitting information requested by NRC Generic Letter 88-20.

This letter contains no restricted or other defense information.

NORTHERN STATES POWER COMPANY

BY Joel P. Sorensen
Joel P. Sorensen
Site General Manager
Prairie Island Nuclear Generating Plant

On this 28th day of February 2000 before me a notary public in and for said County, personally appeared Joel P. Sorensen, Site General Manager, Prairie Island Nuclear Generating Plant; and being first duly sworn acknowledged that he is authorized to execute this document on behalf of Northern States Power Company, that he knows the contents thereof, and that to the best of his knowledge, information, and belief the statements made in it are true and that it is not interposed for delay.

Jack Leveille

JACK ALAN LEVEILLE
NOTARY PUBLIC - MINNESOTA
My Comm. Expires Jan. 31, 2005

Response to RAI Seismic Questions

- **Prairie Island Seismic IPEEE RAI Responses** text (18 pages)
- **Table 1: Prairie Island Seismic Analysis** (2 pages)
- **Table 2a. Support to Frontline System Dependency Matrix (Unit 1)** (1 page for table and 1 page of notes)
- **Table 2b. Support to Frontline System Dependency Matrix (Unit 2)** (1 page for table and 1 page of notes)
- **Figure 1. Success Block Diagram** (1 page)
- **Figures 2 to 12, Simplified Flow Diagrams** (1 page each)
- **Figures 13 to 36, Building Response Spectra Plots** (1 page each)

PRAIRIE ISLAND SEISMIC IPEEE RAI RESPONSES

Introduction

Supplement 4 to Generic Letter 88-20 requested all licensees to perform an IPEEE to find plant specific vulnerabilities to severe accidents caused by external events. Section 4.1 of Supplement 4 addresses the seismic portion of the IPEEE. Four bins were identified into which US sites were classified based on seismic hazards analyses; SSE reduced scope, 0.3g focused scope, 0.3g full scope and 0.5g full scope. The Prairie Island plant was classified as a focused scope plant requiring the review for seismic vulnerabilities to be performed at a review level earthquake of 0.3g.

In its original response to Supplement 4 to Generic Letter 88-20, Prairie Island intended to perform a seismic PRA. However, a subsequent review of the seismic hazards analyses used to bin the US sites resulted in the NRC concluding that seismic hazards estimates were less than that perceived when issuing the original Generic Letter. Supplement 5 to Generic Letter 88-20 was issued modifying the requirements for focused scope plants. In this supplement, the need for evaluating the capacities of reactor internals and soil-related failures was eliminated. The staff further stated "Modifying the scope of the seismic IPEEE for focused-scope plants in this manner will make these evaluations equivalent to those for reduced scope plants with the additional evaluation of a few known weaker, but critical, components or items." Attachment 1 to Supplement 5 listed these additional known weaker components as relays, masonry and block walls, flat-bottom tanks and other items related to anchorage, physical interactions, building impact or pounding.

With this reduction in scope, Prairie Island modified its commitment to complete the Seismic IPEEE. Given the similarity with the requirements for reduced scope plants, a seismic margins assessment (SMA) was selected. Like the reduced scope plants, the SMA was completed at the SSE (0.12g for Prairie Island) and submitted with the evaluation of the remainder of the external events.

It is apparent from the RAIs (TAC NOS. M88663 and M88664) that the staff did not intend to imply in Supplement 5 of Generic Letter 88-20, that making the evaluations for focused scope plants equivalent to that for reduced scope plants meant the review level earthquake could be equivalent to that for reduced scope plants (the SSE). For this reason, the attached responses to the RAIs expand the scope of the seismic IPEEE for Prairie Island from our submittal (NSPLMI-96001, Appendix A, Revision 0) in that the RLE has been modified to 0.3g.

The evaluation at the 0.3g RLE concluded that all important safety functions could be accomplished following a seismic event. All components included in the SMA that support these functions are found to have HCLPFs greater than or equal to 0.3g with the exception of the component cooling water heat exchangers. The component cooling heat exchangers HCLPFs of 0.28g are very close to the 0.3g threshold and thus are considered to be adequate.

Responses to RAI

<u>NRC Request</u>

1. Although it is stated in the IPEEE submittal that the Prairie Island seismic margins assessment follows the guidance of EPRI NP-6041-SL, the procedures used in the

success path selection of structures, systems, and components are not consistent with those described in EPRI NP-6041-SL. Success path logic diagrams (SPLDs) were not provided in the IPEEE submittal; although equipment for important safety functions were identified and discussed, specific success paths that could bring the plant to a safe shutdown condition were not identified. It is not clear whether the selected equipment can provide two success paths with sufficient redundancy and diversity.

In addition, the six safety functions used in the Prairie Island IPEEE for system selection (discussed on page A-11, Reference 11) are not consistent with the four safety functions identified in EPRI NP-6041-SL: (1) reactivity control; (2) reactor coolant system pressure control; (3) reactor coolant system inventory control; and (4) decay heat removal.

Please provide information on success path development and system selection consistent with that described in EPRI NP-6041-SL. Please include in the discussion the development and identification of the success paths, the systems and equipment included in the Safe Shutdown Equipment List (SSEL) and their safety functions, and the isolation of systems that are excluded from the SSEL (for example, successful isolation of the condensate storage tanks when the water source of the auxiliary feedwater (AFW) pumps is switched).

<u>NSP Response</u>

Although success path logic diagrams (SPLDs) were not explicitly presented in the Seismic Margins analysis, the required safety functions and the systems and component needed to accomplish the critical safety functions (CSF) are discussed in detail. The four CSFs identified in the EPPI NP-6041-SL are properly addressed by the six CSF assessed in the Seismic Margins analysis. The correspondence between the CSFs and the systems credited are shown below.

Prairie Island CSF	EPRI NP-6041-SL CSF	Preferred Systems
Reactivity control.	Reactivity control	Reactor Protection System (RPS)
Reactor coolant pump (RCP) seal cooling (Reactor coolant system integrity)		Component Cooling (CCW)
Secondary heat removal	Decay heat removal and RCS pressure control	Auxiliary feedwater (AFW)
Short term inventory (injection)	Reactor coolant inventory control	Safety Injection (SI)
Long term inventory (recirculation)	Reactor coolant inventory control and decay heat removal	Safety Injection (SI) in the Recirculation mode with RHR providing suction pressure to SI pumps.
Containment pressure control	Decay heat removal	Steam Generators, RHR Heat Exchangers, Fan Cooler Units (FCU)

The six safety functions defined in the IPEEE Seismic analysis were derived from the Prairie Island Internal Events PRA. The development of the success paths for these functions began with identifying the systems available following a seismic event. The systems were to ensure that at least two trains are available to perform their functions. Because a loss of offsite power is postulated during a seismic event, all active components in the systems that could be available following a loss of offsite power event were placed in the SSEL. The logic models developed in the internal events PRA were used to identify these active components. The equipment list was also supplemented with some passive components (e.g., tanks, heat exchangers, panels, cabinets, and support structures) that are needed to bring the plant to a stable condition and maintain this condition for 72 hours.

The success path block diagrams for loss of offsite power and for a small LOCA are shown in Figure 1. Table 1 lists the CSFs credited and identifies the preferred and alternate systems supporting these CSFs. It also provides the figure numbers (Figures 2 though 13) for the attached simplified line diagram of each system. The trains of major equipment credited in the seismic margins assessment are highlighted in the line diagrams. The simplified flow diagrams developed for the Internal Events PRA serve as a substitute for success path block diagrams. The preferred systems have at least two independent trains, as shown in the system line diagrams.

Table 1 also lists required operator actions as well as important equipment required in the success paths. Isolation of systems to enable operation of systems in the success paths are discussed in Response to #3. Table 2a and 2b are dependency matrices listing the frontline trains and their support systems for units 1 and 2, respectively.

<u>NRC Request</u>

2. Prairie Island has been identified in NUREG-1407 as a plant belonging to the 0.3g focused-scope seismic margin assessment group; hence, the evaluation that was performed for the Prairie Island seismic IPEEE (using a reduced-scope at 0.12g pga) does not conform to the guidance in NUREG-1407 and Supplement 4 to Generic Letter (GL) 88-20.

   Please provide the following:

   a) a list of structures, systems, and components (including SSEL items and containment systems equipment) that did not screen at the 0.3g Review Level Earthquake (RLE).

      <u>NSP Response</u>

      The 11, 12, 21, and 22 Component Cooling Water Heat Exchangers were found to have High Confidence of a Low Probability of Failure (HCLPF) capacities of 0.28g. The seismic evaluation of these heat exchangers is summarized in the response to Item 2b. All other structures, systems, and components on the SSEL and containment systems equipment were concluded to have HCLPF capacities greater than or equal to the 0.3g Review Level Earthquake (RLE).

NRC Request

2.b) the basis for the disposition of each item that did not screen at the 0.3g RLE, including the results of new calculations for seismic capacities.

NSP Response

The seismic evaluation of the 11, 12, 21, and 22 Component Cooling Water Heat Exchangers was performed following the guidelines of EPRI NP-6041-SL. The heat exchanger HCLPF capacity of 0.28g was found to be controlled by the anchorage. The seismic evaluation of the 11, 12, 21, and 22 Component Cooling heat exchangers is summarized as follows.

Data on the heat exchanger configuration, construction, and anchorage were obtained from structure and component drawings as well as field walkdown.

Seismic responses in the longitudinal, transverse, and vertical directions were determined. The heat exchanger was modeled as an equivalent single-degree-of-freedom system to determine its seismic response in each of the three orthogonal directions. Conservative lower bound fundamental frequencies were determined considering mass and stiffness properties of the heat exchanger and its supports. 4% damping was considered to be median-centered following EPRI NP-6041-SL guidance. Fundamental mode spectral accelerations were obtained from the applicable Auxiliary Building RLE floor spectra. Overall seismic loads were determined as the product of the total heat exchanger mass and the fundamental mode spectral acceleration.

The overall seismic loads due to responses in the three orthogonal directions were distributed to the heat exchanger anchor bolts, including consideration of bolt holes slotted in the longitudinal direction at one of the two support saddles. Anchor bolt shear and tension demands due to the seismic responses in the three orthogonal directions were combined by 100-40-40 as permitted by EPRI NP-6041-SL. Anchor bolt seismic demands were combined with demands due to gravity load. Net anchor bolt tension does not occur because of the relatively high weight and relatively low center of gravity of the heat exchanger.

Anchor bolt shear capacities were determined following ACI 349-97 Appendix B provisions. The Component Cooling Water Heat Exchanger HCLPF capacity of 0.28g was obtained by comparison of anchor bolt shear capacity versus shear demand.

Since the HCLPF capacity of 0.28g is very nearly the RLE of 0.3 g and more than double the SSE of 0.12g, modifications to increase the HCLPF capacity are not considered necessary nor cost effective.

NRC Request

2.c) an evaluation (at 0.3g RLE) of masonry/block walls that may influence the performance of success path components.

<u>NSP Response</u>

The seismic evaluation of concrete block walls was performed following the guidelines of EPRI NP-6041-SL, "A Methodology for Assessment of Nuclear Power Plant Safety Margin." All concrete block walls were found to have HCLPF capacities greater than the 0.3g RLE. The seismic evaluation of concrete block walls is summarized as follows.

Essentially all safety-related concrete block walls were encompassed in the SMA. This approach is conservative since it probably includes concrete block walls whose failure does not affect components on the IPEEE SSEL. A limited number of safety-related block walls were screened out based on wall-specific review confirming that their failure does not affect components on the SSEL.

Data on concrete block wall configurations and constructions were obtained from structural and architectural drawings, Inspection and Enforcement (I & E) Bulletin 80-11 submittals, and field walkdown. The Prairie Island concrete block walls typically have vertical reinforcing bars and horizontal joint reinforcement. Some of the concrete block walls were strengthened as a result of I & E Bulletin 80-11. Past experience has indicated that such walls typically have significant seismic capacities.

The concrete block walls were segregated into groups having different boundary conditions. Bounding case concrete block walls for each group were selected for detailed, wall-specific evaluation based on review of parameters significantly affecting seismic capacity, including wall weight, span, span-to-thickness ratio, reinforcement, openings, and building elevation. Concrete block walls not subjected to detailed evaluation are considered to have seismic capacities higher than the bounding cases.

Seismic evaluations of the bounding case concrete block walls for out-of-plane seismic loads due to the 0.3g RLE were performed following EPRI NP-6041-SL guidelines, supplemented by Appendix A to the USNRC Standard Review Plan (SRP) Section 3.8.4 and ACI 530-95/ASCE 5-95/TMS 402-95.

Out-of-plane wall fundamental frequencies were obtained by closed-form solutions for the given wall configurations, constructions, and boundary conditions. Median-centered damping was considered to be 6% based on EPRI NP-6041-SL. Fundamental mode spectral accelerations were obtained from the applicable RLE floor spectra.

Out-of-plane wall moment capacities, which are typically controlling, were determined following the criteria noted above. Capacities for out-of-plane moment about the horizontal axis which consider deformed vertical reinforcing bars were increased by a conservative inelastic energy absorption factor, F_μ, of 1.25 rather than determined by the more rigorous approach in EPRI NP-6041-SL Appendix R. An inelastic energy absorption factor of 1.0 was conservatively assigned to other concrete block wall failure modes.

HCLPF capacities for the bounding case concrete block walls were obtained from comparisons of their seismic demands due to the 0.3g RLE with their seismic capacities. All bounding case concrete block walls were found to have HCLPF capacities greater than the 0.3g RLE.

NRC Request

2.d) an evaluation (at 0.3g RLE) of flat-bottomed tanks, as requested in NUREG-1407 and GL 88-20 for focused-scope plants. Address both tank failures themselves as well as potential flooding concerns resulting from tank failures.

NSP Response

The only flat-bottomed tanks in the IPEEE SSEL are the Refueling Water Storage Tanks (RWSTs). The RWSTs were judged to have HCLPF capacities greater than the 0.3g RLE. The basis for this screening is provided in Response 2f below.

Flat-bottomed tanks not included in the IPEEE SSEL were reviewed. No tanks whose seismic-induced failure could lead to flooding of essential components on the IPEEE SSEL were identified.

NRC Request

2/e) the comparisons of the design basis ground spectrum and in-structure response spectra (IRS) to the IPEEE 0.3g pga RLE ground spectrum and in-structure response spectra. If scaling is used, describe the scaling method. If new IRS are generated, describe the analyses performed to generate all significant RLE IRS.

NSP Response

Ground Response Spectra

In accordance with NUREG-1407, RLE ground motion was defined to be a median NUREG/CR-0098 ground response spectrum anchored to a peak horizontal ground acceleration of 0.3g. The Prairie Island Safe Shutdown Earthquake (SSE) has a peak horizontal ground acceleration of 0.12g. The 5% damped horizontal RLE ground response spectrum is compared to the 0%, 0.5%, 2%, and 5% damped SSE ground response spectra in Figure 13. Vertical ground response spectra for the 0.3g RLE and SSE are both two-thirds of the horizontal ground response spectra.

Reactor-Auxiliary-Turbine Building In-Structure Response Spectra

New in-structure response spectra (IRS) for the Reactor-Auxiliary-Turbine Building were generated for the 0.3g RLE. The methodology used for generation of these new IRS is summarized as follows.

Two horizontal and one vertical artificial ground acceleration time-histories whose response spectra match the 5% damped RLE ground response spectra were generated. EPRI NP-6041-SL recommendations on spectra matching were satisfied. Statistical independence of the three time-histories was verified. The power spectral density functions were verified to match an applicable target.

The model of the Reactor-Auxiliary-Turbine Building structure was developed from the structure model used in the original seismic design analysis. This model consists of multiple lumped mass sticks representing the different portions of the building, including the Containment Vessel, Shield Building, Reactor Support Structure, Auxiliary Building, Auxiliary Building Fuel Tank Area, and Turbine Building. Horizontal springs between the Auxiliary Building, Auxiliary Building Fuel Tank Area, and Turbine Building represent their physical connectivity. 5% material damping was assigned the structure model. This introduces slight conservatism in the analysis, since EPRI NP-6041-SL recommends damping values of 7% or greater for steel and concrete structures beyond or just below yield.

The best estimate low strain soil shear moduli and material damping was based on available site-specific geotechnical data. Best estimate soil shear moduli and material damping values compatible with strains due to the 0.3g RLE were calculated using EQE's version of Computer Program SHAKE91 in conjunction with generic strain degradation curves. This soil case is designated as the "best estimate high strain" soil case in following discussion. Following EPRI NP-6041-SL recommendations, lower and upper bound high strain soil shear moduli were obtained by scaling the best estimate high strain soil shear moduli by factors of 2/3 and 1.5, respectively. An additional upper bound soil case was defined to be 90% of the best estimate low strain shear moduli. This latter soil profile is also recommended by EPRI NP-6041-SL to ensure that uncertainties in soil properties are enveloped at higher ground motion levels.

Foundation impedance functions were calculated by Computer Program CLASSI for the best estimate high strain and 90% of best estimate low strain cases. The foundation under the Reactor-Auxiliary-Turbine Building was modeled as a rigid mat consistent with the original design seismic analysis. Foundation embedment was considered to be minimal and was neglected. Foundation impedance functions for the lower and upper bound high strain soil cases were obtained by scaling the impedances for the best estimate high strain soil case by ratios of shear moduli.

Soil-structure interaction (SSI) analyses utilizing the elements described above were generated using Computer Program CLASSI. The three statistically independent artificial ground motion time-histories were input concurrently. Analyses were performed for each of the following four soil cases: Best estimate high strain, lower bound high strain, upper bound high strain, and 90% of best estimate low strain. IRS at selected building locations were calculated for each of the soil cases. The envelopes of IRS for the four cases were obtained and utilized in seismic assessments of components on the SSEL.

5% damped Reactor-Auxiliary-Turbine Building RLE IRS are compared to 5%

damped SSE IRS used for resolution of USI A-46 at selected building locations containing essential equipment in Figures 14 to 25. Only horizontal IRS are compared since vertical motions typically do not have significant impact on the IPEEE seismic evaluations.

Peak spectral accelerations for the SSE IRS typically occur due to a mode having a frequency of about 2.7 Hz. Review of the original seismic design analysis indicates that this is a rigid body mode consisting of the relatively rigid structure translating on flexible soil. Radiation damping due to SSI effects, which should be considerable for this structure, were conservatively neglected by the original seismic design analysis. Spectral accelerations for the RLE IRS at frequencies in the range of 3 Hz are much less than those for the SSE IRS. This difference results from the more rigorous treatment of SSI effects using current analytical methods.

Screenhouse In-Structure Response Spectra

The Screenhouse is surrounded on three sides by soil. Most of the Screenhouse is embedded to a depth of about 40 feet below grade, with only a single 20 foot high story above grade. SSEL components in the Screenhouse with the potential to control the plant HCLPF capacity are located at or below grade. Past experience in the analysis of similar structures demonstrates that the median-centered IRS should not exceed the ground response spectra at grade. Screenhouse IRS at or below Elevation 695' were consequently judged to be equal to the free-field 0.3g RLE ground response spectra at grade. Screenhouse IRS above Elevation 695' were conservatively estimated to be 1.5 times the free-field 0.3g RLE ground response spectra at grade.

Examination of the Reactor-Auxiliary-Turbine Building RLE IRS confirms that application of the free-field 0.3g RLE ground response spectra at and below grade of the Screenhouse is reasonable. As shown in Figures 18, 19, 22, 23, 28, and 29, the Reactor-Auxiliary-Turbine Building RLE IRS at Elevations 697.5' and 695', which correspond to the top of base mat, are less than the free-field 0.3g RLE ground response spectra at frequencies greater than about 3 Hz which typically corresponds to fundamental frequencies of components vulnerable to seismic effects. Exceedances of the free-field ground response spectra at frequencies between about 1.5 Hz and 3 Hz are minimal.

The 5% damped horizontal Screenhouse RLE IRS at grade is compared to the SSE IRS used for resolution of USI A-46 in Figure 36. The SSE IRS exhibits peak amplification between frequencies of about 5 Hz to 8 Hz. This peak probably corresponds to the fundamental mode determined by the original seismic design analysis model. Similar to the Reactor-Auxiliary-Turbine Building, the SSE IRS exceeds the RLE IRS in this frequency range because radiation damping due to SSI effects was conservatively neglected by the original seismic design analysis model.

NRC Request

2.f) the seismic evaluation for the refueling water storage tank (RWST).

NSP Response

The Refueling Water Storage Tanks (RWSTs) are located in the Auxiliary Building. They are constructed of A240 Type 304 stainless steel. The RWSTs are 26 feet in diameter by 75 feet high with wall thicknesses ranging from 3/16 to 0.28 inches. The tank bottoms are supported on a two foot thick concrete pad poured on the Auxiliary Building base mat. The tank walls are backed by and in contact with cylindrical concrete walls ranging in thickness from 1'-6" to 2'-0". These concrete walls are substantially reinforced, with reinforcement ratios on the order of 1% or greater. They are constructed integral with the Auxiliary Building concrete floor and roof slabs at Elevations 715'-0", 735'-0", 755'-0", and 775'-0". Reinforcing dowels that extend into the concrete slabs are hooked into the cylindrical walls backing the RWSTs.

Review of the RWST and Auxiliary Building drawings did not reveal any significant seismic vulnerabilities. The seismic capacities of the RWSTs are considered to be controlled by the seismic capacity of the enclosing concrete structure. The Auxiliary Building structure was judged to be seismically rugged with a HCLPF capacity greater than 0.3g PGA following the guidance of EPRI NP-6041-SL. It was concluded that the RWSTs consequently have HCLPF capacities greater than 0.3g PGA.

Supplement No. 5 to GL 88-20 correctly notes that earthquake experience data and analytical evaluations have demonstrated that flat-bottom tanks with poor anchorage are vulnerable to failure due to earthquake ground motion. Such data and evaluations are applicable to free-standing flat-bottomed steel tanks and not the concrete-backed Prairie Island RWSTs. Quantitative evaluation of the RWSTs is not considered necessary. The qualitative evaluation described above is considered to be consistent with the intent of NUREG-1407 and conforms to the use of expert judgement expressed by Supplement Nos. 4 and 5 to GL 88-20.

NRC Request

3. Non-seismic failures and human actions are not specifically discussed in the IPEEE submittal. For non-seismic failures and human actions, NUREG-1407 states that "Success paths are chosen on a screening criterion applied to non-seismic failures and needed human actions. It is important that the failure modes and human actions are clearly identified and have low enough probabilities to not affect the seismic margins evaluation." Since specific success paths were not identified in the IPEEE (see Question 1) discussions of non-seismic failures and human actions, and their impact on the selection and reliability of the success paths, were not provided.

   Please discuss these issues in accordance with Section 3.2.5.8 of NUREG-1407 and Section 3 of EPRI-NP-6041-SL.

   NSP Response

   The success paths were chosen based on screening criterion applied to non-seismic *failures and human actions. Table 1 provides the critical safety functions and the systems that supplied these functions. Also included are the operator actions required for each function, the time in which the action must be completed, and the location in the plant in which the action must take place. Table 1 also lists the non-seismic equipment important to the success paths. The human errors and non-seismic failures were determined to have sufficiently low failure probabilities that the success paths that they support would be considered to be reliable.*

Important Human Actions

<u>Switch suction of AFW from CST to cooling water</u>
This action would be taken following the assumed failure of the non-seismically qualified condensate storage tank (CST), the normal suction source for the auxiliary feedwater pumps. The auxiliary feedwater pumps are assumed to trip on low suction pressure following loss of suction from the CSTs. Upon CST low level and subsequent pump trip, both indicated by alarms in the control room, the operator would realize that the normal suction source from the CSTs is not available. He would step through the "Earthquake" procedure AB-3 to align an alternate suction source and realize that only cooling water is available since other systems require offsite power, and may otherwise be unable to provide makeup through the failed CST. After cooling water is aligned to the pump suction, the operator proceeds to reset the low suction pressure trips and restart the AFW pumps. The operator may have to reset and open the trip throttle valves for the turbine driven (TD) AFW pumps before starting them. The IPEEE walkdown identified the potential for earthquake-induced vibration to trip these valves. If this is the case, the operators have to reset and reopen the valves in the AFW pump room prior to starting the associated pumps. This task is fairly simple and is procedurized [1]. Furthermore, indicators are available in the control room to signal the pump has tripped due to either low pressure or pump lockout. After aligning the pump cooling water suction supply and restarting the pumps, the operators will eventually have to be concerned about long term cooling water operation given limited supply in the screenhouse safeguards bay. They will have to reduce cooling water demand to below the capacity of the emergency intake line (see cooling water load management section below).

The operator has up to 43 minutes to diagnose and perform the switchover properly before the steam generators become dry [2]. This assumes that the AFW pumps trip immediately after the seismic event although it may take a little time to drain the CST. Forty-three minutes is sufficient time for the operator to properly diagnose the event and perform the alignment given that alarms are readily available in the control room and that critical actions are performed in the control room. The exception is resetting and opening trip throttle valves, which is done in the AFW pump room [1,8,9,10].

<u>Resetting the TD AFW pump trip and throttle valve (CV-31059, CV-31060)</u>
As discussed above, the vibration from the earthquake may cause the trip mechanism to trip and close the TD AFW trip throttle valve thus resulting in tripping of the pump. The trip would result in an alarm in the control room indicating pump lockout. Upon receipt of the alarm, the operators are directed by procedure to reset the trip and restart the pump. Resetting the trip must be performed in the AFW pump room where the valves are located (valves are part of the TD AFW pumps). The three step procedure is called out in Reference 1. The operator has roughly 43 minutes to reset the trip and restart the pump before the SGs become dry (see timing in the above operator action). As noted

above, this action may be done in concert with the action to align cooling water supply to the AFW pump suction thus shortening the time allowed to reset the valve. However, the procedure is fairly simple and considered to be completed within the allowable time frame. The path from the control room to the valve location was walked down. No potential obstructions were identified as a result of an earthquake that could prevent the operators from reaching the valves and completing the restoration activity. Therefore, it is highly likely that this action can be completed successfully.

Cooling water load management
This action is taken to reduce cooling water flow to within the capacity of the emergency intake line (EIL). The emergency intake line is assumed to be the only long term source of water from the Mississippi River to the safeguards bay following a design basis earthquake. It has limited flow capacity (approximately 12,171 GPM based on minimum submergence) and would not be able to keep up with the postulated system's flow demand (29,750 GPM) following a seismic event. The post trip cooling water flow is based on two diesel-driven safeguards cooling water pumps operating. Upon receipt of the seismic annunciator in the control room , the operators enter Abnormal Procedure AB-3 and immediately monitor the water level in the safeguards bay (LI-41011, LI-41017 and LI-41503). If the level is decreasing rapidly, an indication of loss of normal supply from the river, they will proceed to reduce cooling water flow to avoid pumping out the safeguards bay. The reduction in flow is accomplished by isolating cooling flow to the Turbine Building loads for both units, the fan coil units (8 FCUs, 4 in each unit) and a component cooling water heat exchanger train in each unit. When the system total flow demand is less than 13,000 GPM, the number of cooling water pumps taking suction from the safeguards bay is reduced to one. The system flow is closely monitored and adjustments are made according to heat removal demands. All the preceding actions discussed can be performed in the control room and are anticipated to be completed within 15 minutes after receiving the alarm in the control room and subsequent lowering of safeguards bay level. The latest time to complete the actions is dependent on the initial inventory in the safeguards bay and the water remaining in the intake canal. An evaluation of the intake canal capacity shows that the canal is able to support the safeguards function of the cooling water system (USAR Page 10.4-7). The volume in the intake canal provides approximately 4.8 hours for a flow demand of 31,750 GPM (assuming additional 2,000 GPM cooling water flow from the diesel fire pump) after plant shutdown. After depletion of the intake canal, the EIL would be the sole supply of water to the cooling water pumps. The operators will have more than enough time to correctly diagnose and perform the task. In addition, indicators are available in the control room to allow the operators to properly diagnosis the event. The operators are also trained on the procedure. Consequently, this action is considered to be highly reliable.

Open 4.16 KV safeguards switchgear room doors
This action is taken in response to loss of cooling to the Unit 1 4.16KV safeguards switchgear rooms (Bus 15 and 16). Loss of room cooling could occur as a result of failure of the unit coolers or loss of chilled water supply to the unit coolers (Chilled Water is not a part of the safe shutdown list and is conservatively assumed not available). The limiting component in the 4.16KV switchgear room is the Load Sequencer, which has a maximum qualified temperature of 104 deg F. A room heat-up calculation [3] demonstrated that at least two hours is required for the temperature in the switchgear room to reach 104 deg F on loss of cooling to the rooms. The load sequencer performs its function at the initiation of the event, i.e., the seismic event that causes a loss of

offsite power. The remaining components in the room will operate satisfactorily in the elevated temperature of 120 deg F [3]. The operator would enter procedure C37.11 AOP 1, "Loss of Safeguards Chilled Water," following receipt of control room water chiller trip annunciator in the in control room. The procedure instructs the operator to open the switchgear room door to the Turbine Building to provide for heat removal through natural circulation (or install fans if necessary). It is demonstrated in the room heat-up calculation that the temperature would stabilize below 120 deg F with the door opened. There is a high likelihood that the operator would open the switchgear room door well within 2 hours following the loss of chilled water event given the availability of signals in the control room and the procedural guidance.

Initiate containment sump recirculation (high head recirc)
This action is required in the event that the seismic event also causes a small LOCA. Following a small LOCA, the SI initiates to makeup to the RCS from the RWST. As the inventory in the RWST decreases to 33%, an annunciator in the control room alerts the operator to place the system in recirculation mode. This is to ensure continuous makeup to the RCS prior to RWST depletion. After the signal to switch to recirculation, the operator will have approximately one hour after the annunciation of low RWST level to perform the switchover before the RWST becomes empty (based on [4], a small LOCA event, it would take a total of 4.1 hours after SI initiation to deplete the RWST). The operator performs a series of actions that include opening the sump to RHR pump suction valve, starting the idle RHR pump, closing the SI to RWST suction valve, opening supply RHR to SI valve, and starting the idle SI pump. A majority of the actions are performed in the control room with some performed locally. These actions are explicitly specified in several EOPs. The operators are also trained to perform these actions on the simulator. The operator has more than enough time to perform this alignment either in the control room or locally, and moreover, the equipment (i.e., pumps and valves) is determined to be available following RLE of 0.3g.

NRC Request

4. The Cooling Water system is very important for Prairie Island. In addition to providing the cooling water source for both equipment cooling and heat removal (directly or indirectly through component cooling water (CCW) and Safeguards Chilled Water), it also provides an alternate water supply to the AFW system (but represents the only AFW source for the IPEEE). It consists of five pumps shared by the two units. Only three of the five pumps (two diesel-driven and one motor-driven) will be available following a loss of offsite power, and all of them were found in the IPEEE and the A-46 program to have anchorage and shaft stability problems. As a result of the A-46 program finding, the two diesel-driven pumps were classified by the Seismic Qualifications Utilities Group (SQUG) as outliers and the problem will be resolved with the closure of the A-46 program. On the other hand, no action is planned for the motor-driven pump. The motor-driven pump was subsequently removed from the equipment list for the IPEEE, because, according to the Updated Safety Analysis Report (USAR), the cooling needs for both units can be met by the operation of one diesel-driven pump. Consequently, all the cooling needs for both Prairie Units will be provided by the two diesel-driven Cooling Water pumps.

   According to the IPEEE submittal, the normal water supply for the Cooling Water system is from the circulating water pump bays in the Screenhouse, and an emergency intake pipe is used if the normal path from the Mississippi River through the outer Screenhouse is blocked or if Lock/Dam # 3 fails. Because of the limited capacity of the emergency intake pipe, operator actions to reduce the cooling water loads is required.

Please provide discussions of the following:

a) the seismic capacity of the diesel-driven Cooling Water pumps including the potential impact of losing both pumps in a seismic event.

NSP Response

The 12 and 22 diesel-driven Cooling Water Pumps were classified as A46 outliers because anchor bolt edge distances do not meet the least acceptable values specified by the Generic Implementation Procedure (GIP) and the vertical shaft length exceeds the maximum length in the GIP bounding spectrum caveat. The pump anchorage, shaft, and column were evaluated for the 0.3g RLE following the guidelines of EPRI NP-6041-SL. Pump anchorage, shaft and column HCLPF capacities were determined to be greater than the 0.3g RLE. The seismic evaluation of the 12 and 22 diesel-driven Cooling Water Pumps is summarized as follows.

Data on the pumps and their anchorage were obtained from the component and structural drawings, the original seismic design analysis, and field walkdown. The pump motor/gear assembly is located above the pump base plate. The pump column assembly consists of a 20" diameter pipe column, 5" diameter shaft tube, and 3.44" diameter shaft. The shaft is laterally supported within the shaft tube by bearings spaced at intervals discussed in more detail below. The impeller is located at the end of the shaft. The shaft tube is supported against the column. The column is attached to the pump base plate. The bowl assembly is located at the end of the column.

Seismic responses in the North-South, East-West, and vertical directions were determined. Seismic input to the pumps was considered to be the free-field ground motion for the 0.3g RLE as noted in Response 2e above. 3% damping was considered to be median-centered following EPRI NP-6041-SL.

In the two horizontal directions, modal responses associated with the pump column assembly and the motor/gear assembly below and above the pump base plate, respectively, were included. These responses were considered to be decoupled because of the significant difference between the modal frequencies. Pump column assembly horizontal seismic loads were obtained by scaling the original design seismic analysis results to the 3% damped spectral acceleration for the 0.3g RLE at the column assembly fundamental frequency calculated by the original design seismic analysis. The motor/gear assembly was found to be rigid and its responses were based on the floor zero period acceleration. Responses in the column assembly and motor/gear assembly modes were combined by square-root-of-the-sum-of-the-squares (SRSS).

For vertical response, the pump was modeled as a single-degree-of-freedom system including flexibility of the base plate. Pump vertical seismic loads were *obtained as the product of the total pump mass and the 3% damped spectral acceleration for the 0.3g RLE at the fundamental vertical frequency.*

Anchor bolt seismic shear and tensile forces due to reactions from the base plate onto the floor slab for the three orthogonal responses were determined. The anchor bolts are located in close proximity to the edge of the rectangular opening in the floor slab for the pump. Consequently, horizontal base plate reactions were distributed to only the four bolts loaded in shear away from the opening edge. Shear resistance provided by the other eight bolts was conservatively neglected. This approach is considered to be appropriate since unrestrained horizontal translation of the pump cannot occur unless the four bolts loaded away from the free edge fail.

Anchor bolt shear and tensile demands due to the three orthogonal responses were combined by SRSS following EPRI NP-6041-SL. Net anchor bolt demands including reduction in tensile demands due to pump dead load were determined. Anchor bolt shear, tension, and shear-tension interaction capacities were determined following ACI 349-97 Appendix B provisions. Anchor bolt HCLPF capacities well in excess of the 0.3g RLE were obtained.

The pump column was evaluated for combined moment and axial load, which was considered to be controlling. The pump column moment capacity was based on the AISC allowable stress factored by 1.7 following EPRI NP-6041-SL guidelines. This capacity was increased by a conservative inelastic energy absorption factor of 1.25 since bending failure of the column is ductile. The pump column was found to have a HCLPF capacity well in excess of the 0.3g RLE.

The pump drawings indicate that bearings between the shaft and the shaft tube have a maximum spacing of about 10 feet with the lowest bearing located about 7 feet from the end of the shaft. The spacings of these bearings are considered to be sufficient to constrain the shaft displacements to tolerable levels. The shaft HCLPF capacity is judged to be greater than the 0.3g RLE.

<u>NRC Request</u>

4.b) the overall cooling loads of the Cooling Water system for the selected success paths and the ability of the Cooling Water system to meet these requirements (based on one pump for both units) including the effect of the loss of the normal water supply path.

<u>NSP Response</u>

The initial system flow following plant trip is about two and a half times the capacity of the EIL. This is based on both diesel-driven pumps starting and running after a loss of offsite power. In order to avoid pumping out the screenhouse pump bay, the system flow has to be reduced below EIL's capacity. The operators are instructed by plant abnormal procedures to reduce system flow by isolating flow to non-critical cooling loads (see "Cooling Water Load Management" in Response #3 above). Once flow is reduced, one of two diesel-

driven pumps would be secured. Operator action to reduce system flow and timing for the action are discussed in Response #3 above.

The cooling water requirements for safe shutdown of both units can be adequately supplied by one of the three vertical cooling water pumps with suction from the emergency intake line (EIL). Although the EIL has a flow capacity of approximately 12,171 GPM, it is able to satisfy the cooling water demands needed to maintain safe shutdown condition. The flow capacity is based on actual flow test conducted at normal river levels and adjusted to the condition of minimum EIL submergence of 4.5 feet (i.e., failure of Lock and Dam #3 downstream of the screenhouse intake structure). The equipment required for safe shutdown and their flow rates are listed below. The total system flow rate would be higher than the required flowrate given that not all the essential loads would be isolated per AB-3 "Earthquake" procedure. Moreover, loads that can not be isolated from the control room also contribute to the total system flowrate. A thermal-hydraulic model was generated to calculate the flow demands of the system under these conditions [5]. The model also assumes a crack in each non-safety related cooling water pipe off the main header, which is considered to be conservative. This would increase the total system flow demands. Under this scenario, the model calculates a total system flow demand of 10,643 GPM. This model did not account for the potential flow to a second EDG. However, it is reasonable to assume that the flow to the EDGs are comparable. Accounting for the additional EDG would yield a total flow of approximately 11,889 GPM, which is still less than the capacity of the EIL of 12,171 GPM

Cooling Water Loads for Safe Shutdown	*Cooling water Load (gpm) demands [5] see Note*	*EIL Capacity*	*Comment*
Unit 1 emergency diesel generators (2 EDGs)	*2,492 (assuming flow through 2nd EDG is same as the 1st EDG)*		*Unit 2 diesel generators do not require cooling water, as they are self cooled (air-cooled). Although only one EDG is required for safe shutdown, both EDGs will start up upon loss of offsite power. Therefore, an additional 1,246 GPM is included for the second EDG*
Auxiliary feedwater pumps (1 per unit)	*440*		*Alignment of cooling water to suction of the AFW pumps is required following failure of the non-seismically qualified condensate storage tanks. A discussion on the required actions and time available to perform the actions are discussed in Response #3.*
Component cooling Hx (1 per unit)	*4,869*		*Procedure AB-3 "Earthquakes" instructs operators to secure one component cooling water train for each unit as part of*

			reducing cooling water flow demand. Component Cooling Hxs' seismic capacities are discussed in Response #2b.
Containment Fan Coil Units (1 per unit, 450 GPM ea.)	*900*		*Operators are instructed to maintain throttled flow to one FCU per unit.*
Control Rm Chiller (1 for both units)	*623*		
Loads that are not isolated plus postulated cracks in non-safety related piping	*2,565*		
Total	*11,889 (accounting for 2nd EDG)*	*12,171 (based on EIL flow test [6])*	*EIL flow is greater then required flow after isolation cooling flow management is completed.*

Note: The flowrates are based on Reference 5 and success criterion of the system modeled in the PRA.

The EIL was judged to have a HCLPF capacity greater than the 0.3g RLE. Such buried piping typically does not fail at ground motion levels of 0.3g or less unless significant soil movements occur. USAR Section 10 describes construction of the emergency cooling water intake pipe, which included excavation of the natural soil, placement of the pipe in non-liquefiable backfill material, and use of flexible expansion joints at the screenhouse and intake crib. Such features are expected to prevent soil movements sufficient to cause pipe failure at ground motions less than the 0.3g RLE.

<u>NRC Request</u>

4.c) system alignment and isolation in case of loss of the normal water supply path; operator actions required for system alignment and isolation, and coordination between the operators of the two units, if any; and whether seismic failure of components not included in the SSEL would have an adverse effect on the operators' ability to isolate non-essential cooling water loads.

<u>NSP Response</u>

Following a seismic event and loss of normal cooling water supply from the Mississippi River, the operator would reduce system flow to below the capacity of the emergency intake line by isolating non-essential cooling water loads. The actions are discussed in Response #3. Failure of equipment not included in the SSEL does not have any adverse impact on the operator's ability to isolate non-essential cooling water loads. The following valves and support components are required to function to isolate cooling flow to the non-essential loads.

Component	*Support*	*Comment*
CL to cool nonessential Turbine Building equipment	*BUS 111 – 4.16KV BUS 15 – EDG 1*	*Normally open. Valves MV-32031(33) auto*

MV-32031 (unit 1) *MV-32033 (unit 2)*	*BUS 221 – 4.16KV BUS 26 – EDG 6*	*close on U1 or 2 Train A(B) SI and Low Loop A(B)CL header pressure..*
FCUs Isolation Valves		
MV-32132, 32138	*BUS 112 – 4.16KV BUS 15 – EDG 1*	*Normally open. Manually closed from Control Room*
MV-32135, 32141	*BUS 122 – 4.16 BUS 16 – EDG 2*	*Normally open. Manually closed from Control Room*
MV-32147, 32153	*BUS 212 – 4.16 KV BUS 25 – EDG 5*	*Normally open. Manually closed from Control Room*
MV-32150, 32156	*BUS 222 – 4.16 KV BUS 26 – EDG 6*	*Normally open. Manually closed from Control Room*
11or 12 CC Hx CL inlet MV-32145 or 32146, respectively	*BUS 111 – 4.16KV BUS 15 –EDG 1,* *BUS 121 – 4.16KV BUS 16- EDG 2*	*Normally closed. Open corresponding pump start. Manually close from the control room.*
21 or 22 CC Hx CL inlet MV-32160 or 32161, respectively	*BUS 211 – 4.16KV BUS 25 – EDG 5* *BUS 221 – 4.16 KV BUS 26 –EDG 6*	*Normally closed. Open corresponding pump start. Manually close from control room*

Isolation of cooling water loads in both units is coordinated by the Shift Supervisor as he goes through the procedure with the unit 1 and 2 control room operators. The control room operators isolate the loads to their respective units while monitoring the flow demands of the critical cooling water loads. As discussed in Response #3, Cooling Water Load Management, the operators have more than adequate time to coordinate their actions such that long term cooling to the critical safe shutdown equipment is preserved.

<u>NRC Request</u>

5. Both diesel-driven Cooling Water pumps would be lost in Burn Sequence 69 as stated in the evaluation of seismic-induced fires (page B-77 of the submittal). It is argued in the submittal that this is not a problem because the remaining motor-driven pump can provide sufficient cooling water supply for both units. However, this is not consistent with the seismic assessment portion of the IPEEE in that the motor-driven pump is not included in the equipment list (or not available in a seismic margin earthquake) because of anchorage and shaft stability problem. Burn Sequence 69 will therefore result in the loss of all Cooling Water pumps, and consequently, the loss of nearly all the safety systems required to bring the plant to a safe shutdown condition.

   Please resolve this apparent inconsistency.

<u>NSP Response</u>

The IPEEE Fire analysis assumes the control cable for the diesel-driven cooling water pumps are routed through Fire Area 29 (FA 29). Fire Area 29 opens to FA 69 where the fire is postulated after a seismic event. Fire Area 69 is among the fire areas contained in Burn Sequence 69.

Subsequent to the fire IPEEE, cable tracing revealed that, unlike what was assumed previously, only control cables for the 12 diesel driven pump and the control and power cables for the 11 safeguards screenhouse ventilation fan run through FA 29. The corresponding cables for the 22 diesel driven pump and 21 safeguards screenhouse ventilation fan runs through FA 30, which is separated from FA 29 by a 3-hour fire barrier [7]. It is also not open to FA 69. Both diesel driven cooling water pumps were found to have HCLPF capacities well in excess of 0.3g as discussed in Response 4a. Per this investigation, a seismically induced fire in FA 69 would only result in failure of the 12 diesel driven cooling water pump to automatically start and 11 safeguards screenhouse ventilation fan to run. The remaining 22 diesel driven cooling water pump and the 21 safeguards screenhouse ventilation fan would start and continue to run supplying cooling water to the essential plant loads necessary for safe shutdown.

In addition, evaluation of the 121 motor-driven cooling water pump shows that it has a HCLPF capacity well in excess of the 0.3g RLE. The 121 motor-driven Cooling Water Pump is constructed and anchored similar to the 12 and 22 diesel-driven Cooling Water Pumps. Seismic evaluation of the 121 motor-driven Cooling Water Pump was performed similar to the seismic evaluation of the diesel-driven Cooling Water Pumps, which is described in Response 4a above. As the 121 CL pump has no power or control dependencies in FA 69, it too would be available to provide flow were a seismically induced fire to occur in this area.

References

1. Plant Procedure C28.1 "Auxiliary Feedwater System" Revision 3.
2. Prairie Island Calculation File V.SPA.92.006 "MAAP 014/92", May 1993
3. Engineering Calculation ENG-ME-278, "Loss of Safeguards Chilled Water Room Heatup Calculation," 6/12/96
4. Prairie Island Calculation File V.SPA.93.004 "MAAP 013/92", May 1993
5. Engineering Calculation ENG-ME-310, "Emergency Intake Line; Post Seismic Minimum Flow Requirement" 3-17-97.
6. Work Order WO 9901268, "Measure Flow Through The Emergency Intake Line", Completed 3-15-99.
7. NSP Prairie Island Calculation File V.SMN.00.001, "Seismic IPEEE RAI Information"
8. Abnormal Procedure AB-3 "Earthquakes," Revision 16, May 1999
9. Abnormal Operating Procedure C28.1 AOP2, "Loss of condensate supply to auxiliary feedwater pump suction," Revision 1.
10. Abnormal Operating Procedure C28.1 AOP4 "Restarting an AFWP after low suction/discharge pressure trip," Revision 2.
11. Prairie Island Nuclear Generating Plant Individual Plant Examination of External Events (IPEEE), Revision 1. Licensing and Management Issues Department, Northern States Power Company, NSPLMI-96001, September 1998.

Table 1: Prairie Island Seismic Analysis (Page 1)

Functions System	Figure	Success Paths		Operator Actions			Random Failures		Capacity		Comments
	Simplified line Diagrams	Preferred	Alternate	Action	Timing	Location	Equipment	Probability	Component	HCLPF ≥ 0.3g	
Reactivity Control RPS	2	X		Automatic (fail safe)	-	-	RPS/Control Rods	1.5E-5	-		Diverse Scram (AMSAC was replaced by Diverse Scram system) is not credited. The potential for a failure to trip coincident with an earthquake is considered to be of low potential.
Diverse Scram			X								
Secondary Cooling AFW	3	X		Align Cooling water supply to suction of AFW pumps Reset TD AFW pump trip throttle valve	43 minutes (SG dryout (MAAP case MPP014/92) [2] -	Control room AFW pump room	TD AFW pump FTR	4.18E-2	-	Yes	The TDAFW pump trip and throttle valve is susceptible tripping closed during a seismic event. The operator would have to reset the trip locally in the AFW pump room.
RCP Seal Cooling CCW	4	X					See CCW below	-	-	See CCW below	
CVCS			X				Charging pump FTR	2.34E-3			CVCS is not credited to provide cooling flow to the seals because of availability of CCW. CCW automatically provides cooling flow to the RCP thermal barrier upon loss of Charging flow to the seals. CCW is also required to provide cooling in the RHR Hx.
Short term Injection Safety Injection (HP inject)	5	X					MOVs FTO	4.3E-3	-	Yes	
Long term makeup (recirc) SI recirc. from sump	5,6	X		Transfer to recirculation	4.1 hours [4]	Control room/Auxiliary Building	MOVs FTO	4.3E-3	-	Yes	
RHR recirc. from sump			X								LP RHR recirc. is not credited given primary system pressure is would be above the discharge head of the RHR pumps during small LOCAs and LOOP events. No credit is taken for RCS cooldown, which involves auxiliary spray and secondary depressurization. Secondary depressurization with SG PORVs will be unavailable due to loss of the non-seismic qualified Instrument Air system. Long term makeup is performed by SI recirculation which piggybacks on RHR taking suction from the containment sump.
Containment Pressure Control SI recirc with RHR Hx and SGs	5,6	X					MOVs FTO	4.3E-3	-	Yes	Containment Spray is not credited due to the adequacy of SGs and RHR hx and a FCU in removing decay heat.
RHR recirc from sump		-	X								
Fan Coil Units (FCU)	7	X								Yes	
Containment spray			X								

Table 1: Prairie Island Seismic Analysis (Page 2)

Functions System	Figure	Success Paths		Operator Actions			Random Failures		Capacity		Comments
Support systems											
Offsite AC											Offsite power assumed to fail on seismic event.
Onsite AC	8	X		EDG auto start after LOOP			EDGs failure to run	1.10E-2	-	Yes	
125VDC (Div I&II)	,9, 10	X						1.2E-5	-	Yes	
120V Instrument AC	11	X					Battery chargers FTRE	5.5E-6	-	Yes	
							Fuses	2.7E-2	-	Yes	
Cooling Water	12	X		CL load Management	4.8 hours (USAR page 10.4-7)	Control room	Diesel driven CL pump FTR				
Component Cooling Water	4	X						4.3E-3	CC Hx	0.28 g	
SafeguardsChilled Water		X		Open Bus room doors	2 hours [3]	4.16KV bus rooms (Turbine Building)	MOV FTO				Although loss of the Safeguards Chilled Water leads to loss of cooling to the 4KV bus rooms, the operator can manually open the bus room door to the turbine build to allow for natural air circulation. See discussion on Response #3. The impact to other rooms served by Chilled Water is determined to be minimal given the type of equipment in those rooms or the long time heat up time to the critical equipment temperature. Therefore, it is not credited in the analysis.

Table 2a. Support to Frontline System Dependency Matrix (Unit 1)

Support System Train – Unit 1	**Reactivity Control**		**Emergency Diesel**		**Seal Cooling**		**Secondary Cooling**		**Short-term Makeup**		**Long Term & DHR**	
	RPS		EDG		Comp Cooling		AFW		Safety Injection		SI Recirculation & RHR	
	A	B	A	B	A	B	A (TDAFW)	B (MDAFW)	A	B	A	B
4.16KV Bus 13	(1)											
4.16KV Bus 14		(1)										
4.16KV Bus 15					X(2)				X(2)		X(2)	
4.16KV Bus 16						X(2)		P(8)		X(2)		X(2)
480V MCC 1K1					X(3)				X(3)		X(3)	
480V MCC 1KA2						X(4)				X(4)		X(4)
480V MCC 1K2						X(5)						
480V MCC 1A1							P(6)					
480V MCC 1A2								P(7)				
125VDC DP 11	(1)		X(2)		X(2)				X(2)		X(2)	
125VDC DP 12		(1)		X(2)		X(2)		P(8)		X(2)		X(2)
125VDC DP 15	(1)											
125VDC DP 16		(1)										
Cooling Water A			X(9)		D(9)		X(10)				D(9)	
Cooling Water B				X(9)		D(9)		X(10)				D(9)
CC Water A					D(9)						D(9)	
CC Water B						D(9)						D(9)
Main Steam Loop A							P(11)					
Main Steam Loop B							P(11)					

Notes
X = Complete Dependence
P = Partial Dependence
D = Delay Dependence

Notes for Table 2a

1. The success criteria for RPS is successful unit trip (Subcriticality). Loss of any support system either causes system success (trip) or provides a half-trip due to loss of one train of analog protection circuitry. Therefore, loss of the asscociated 125VDC trains or loss of power to the MG sets results in system success. A seismically induced loss of offsite power would result in loss of power to normal buses 13 and 14, which in turn results in loss of power to the MG sets and thus reactor trip.
2. One train of SI, RHR, CS and CC will be lost upon failure of 4.16KV essential bus 15 or 16. However, if these failures are associated with components that supply power to the buses (i.e., EDGs) and not the buses themselves, the buses 15 and 16 can be cross-tied to Unit 2 4.16KV buses 25 and 26, respectively. One train of SI, RHR and CS is also lost if either of the 125VDC trains fail to supply control power to close the pump breakers. Local operation of the breaker is possible but is not considered. One train of EDG's is lost if its 125VDC supply train fails to provide starting control power.
3. 480VAC essential MCC 1K1 provides motive power to much of Safeguards Train A. Power is provided to the bus from 4.16KV switchgear 15 through 480VAC bus 111. The following are essential loads powered from MCC 1K1.
   - RHR train A suction from containment sump B (MV-32075, MV-32077)
   - CC to RHR heat exchanger valves (MV-32093)
   - SI test line A valve to RWST (MV-32202)
   - SI train A suction valve from RHR pump 11 (MV-32206)
   - SI RWST suction valve (MV-32079)
   - Cooling Water to Component cooling heat exchanger valves (MV-32120, MV-32145)

4. 480VAC essential MCC 1KA2 provides motive power to much of Safeguards Train B. Power is provide to the bus from 4.16KV switchgear 16 through 480VAC bus 121. The following are essential loads powered from MCC 1KA2.
   - RHR train B suction from containment sump B (MV-32076, MV-32078)
   - SI test line B valve to the RWST (MV-32203)
   - SI train B suction valve from RHR pump 12 (MV-32207)
   - SI RWST suction valve (MV-32080)
   - Cooling Water to Component cooling heat exchanger valve (MV-32121)

5. 480VAC essential MCC 1K2 provides motive power to Cooling Water to Component cooling heat exchanger valve MV-32146 and Component Cooling to RHR Heat Exchanger B valve MV-32094. Power is provide to the bus from 4.16KV switchgear 16 through 480VAC bus 121.
6. MCC 1A1 provides motive power to 11 TD AFW pump CL suction valve MV-32025. Loss of power will prevent the remote opening of the valve. Operator can manually open the valve locally, however.
7. MCC 1A2 provides motive power to 12 MD AFW pump CL suction valve MV-32027. Loss of power will prevent the remote opening of the valve. Operator can manually open the valve locally, however.
8. 4.16KV emergency bus 16 supplies power to AFW motor driven pump 12 (train B). Makeup to unit 1 SGs can be supplied form the MD pump in the Unit 2 AFW system through a system cross-tie. Unit 2 AFW MD pump 21 motive power is supplied from unit 2 emergency bus 25. The 12 and 21 MD driven pump breakers control are supplied from unit 1 and 2 DC sources, respectively.
9. Cooling water (CL) provides the heat sink for the CC heat exchangers. With failure of Train A(B) CL, Train A(B) CC is assumed to fail eventually causing loss corresponding SI and RHR trains for decay hat removal function. Loss of CC alone will have the same result. Loss of CL A(B) also result in failure of the corresponding Unit 1 EDGs due to overheating.
10. Cooling water normally is a backup suction source to the CSTs for long term secondary cooling using AFW. During a seismic event, however, it becomes a primary suction source for AFW as the CSTs are assumed to fail because they are not seismically qualified.
11. Steam to the AFW turbine for pump 11 is provided from MS loops A and B. Failure of either Main Steam loop does not necessarily leads to loss of AFW train A.

Table 2b. Support to Frontline System Dependency Matrix (Unit 2)

Support System Train - Unit 2	**Reactivity Control**		**Emergency Diesel**		**Seal Cooling**		**Secondary Cooling**		**Short-term Makeup**		**Long Term & DHR**	
	RPS		EDG		Comp Cooling		AFW		Safety Injection		SI Recirculation & RHR	
	A	B	A	B	A	B	A (MD AFW)	B (TDAF W)	A	B	A	B
4.16KV Bus 23	(1)											
4.16KV Bus 24		(1)										
4.16KV Bus 25					X(2)		P(8)		X(2)		X(2)	
4.16KV Bus 26						X(2)				X(2)		X(2)
480V MCC 2K1					X(3)				X(3)		X(3)	
480V MCC 2KA2						X(4)				X(4)		X(4)
480V MCC 2K2						X(5)						
480V MCC 2A1							P(6)					
480V MCC 2A2								P(7)				
125VDC DP 21	(1)		X(2)		X(2)		P(8)		X(2)		X(2)	
125VDC DP 22		(1)		X(2)		X(2)				X(2)		X(2)
125VDC DP 25	(1)											
125VDC DP 26		(1)										
125VDC DP 27			X(2)									
125VDC DP 28				X(2)								
Cooling Water A					D(9)		X(10)				D(9)	
Cooling Water B						D(9)		X(10)				D(9)
CC Water A					D(9)						D(9)	
CC Water B						D(9)						D(9)
Main Steam Loop A								P(11)				
Main Steam Loop B								P(11)				

Notes
X = Complete Dependence
P = Partial Dependence
D = Delay Dependence

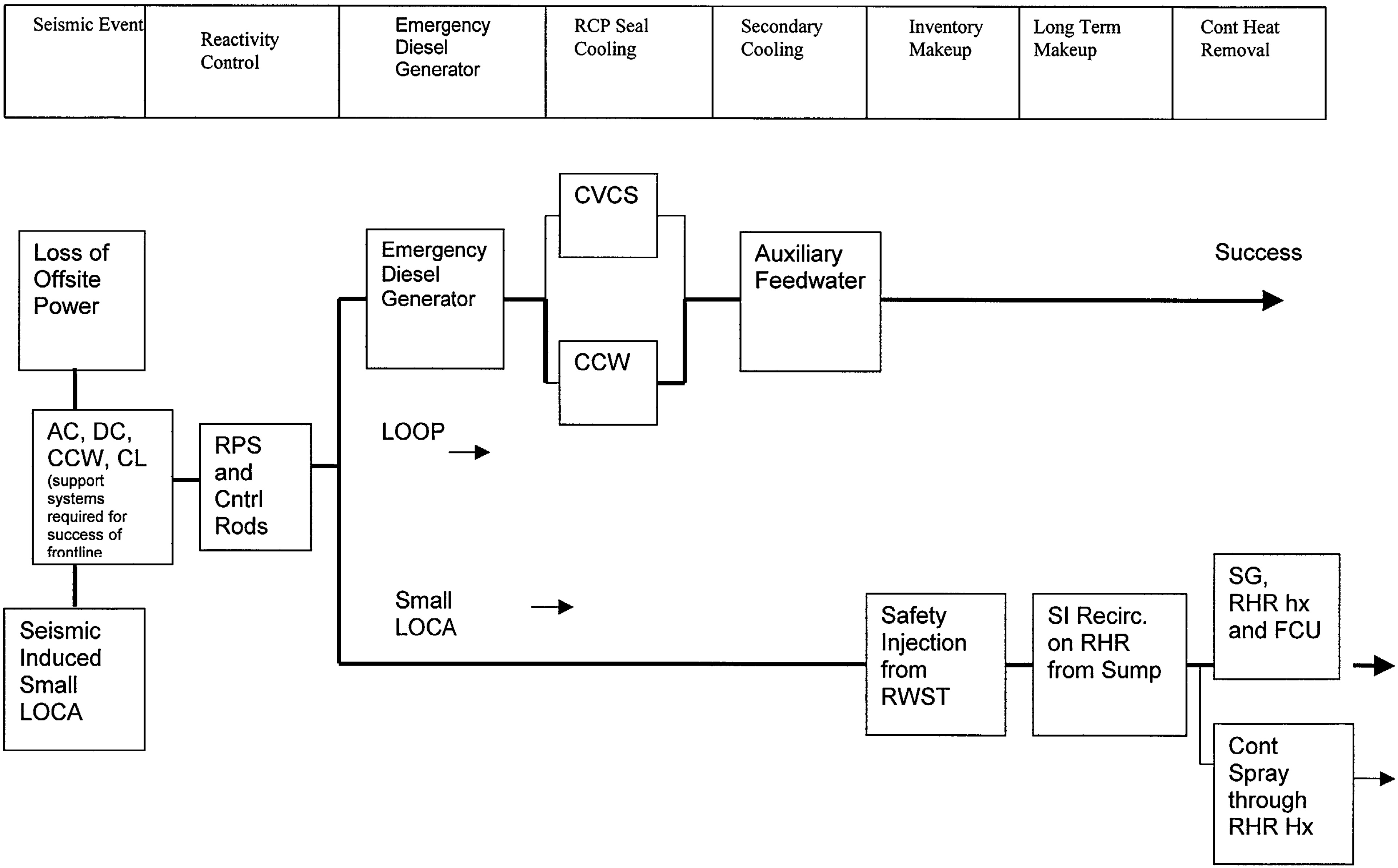
Notes for Table 2b

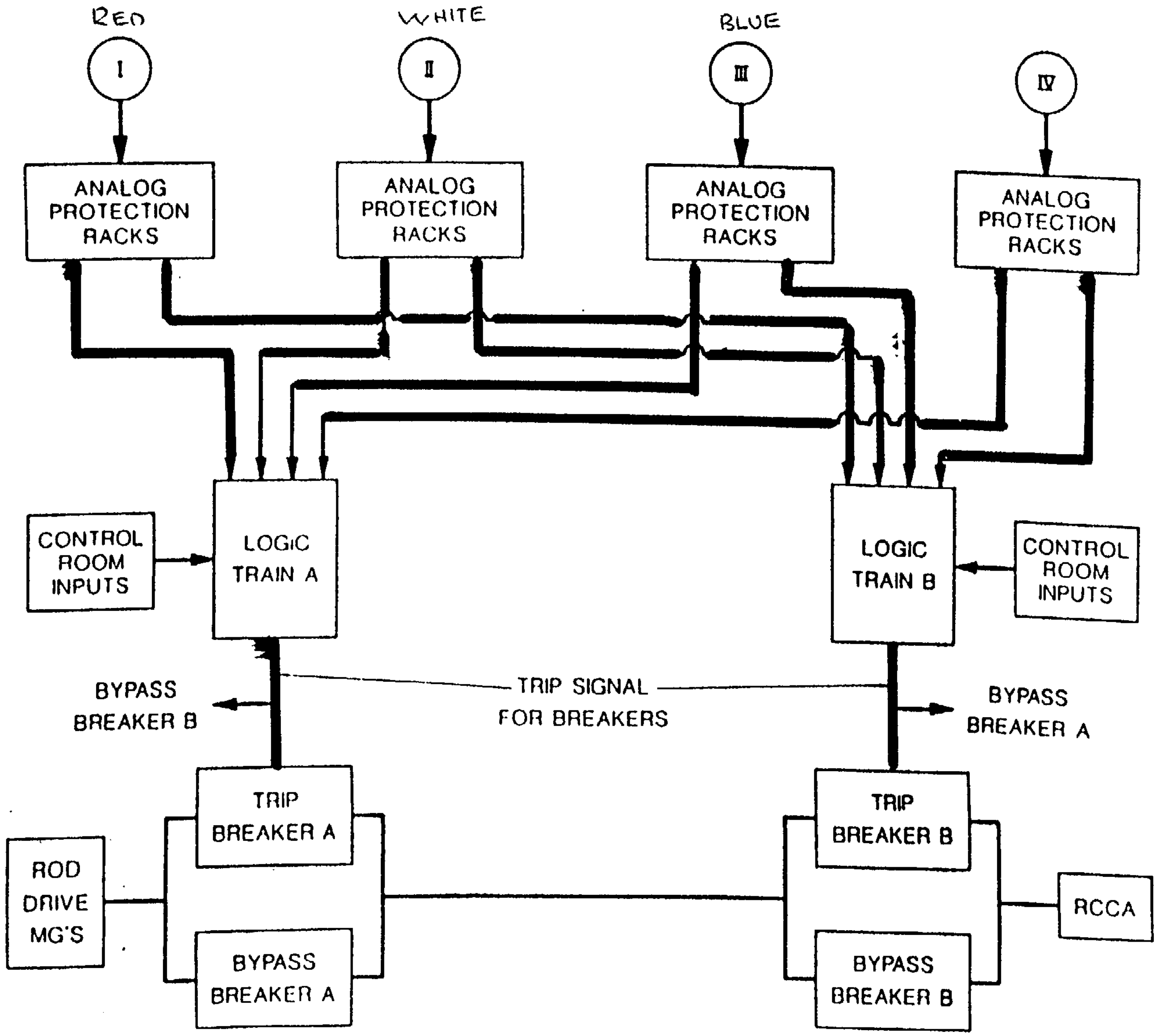
1. The success criteria for RPS is successful unit trip (Subcriticality). Loss of any support system either causes system success (trip) or provides a half-trip due to loss of one train of analog protection circuitry. Therefore, loss of the asscociated 125VDC trains or loss of power to the MG sets results in system success. A seismically induced loss of offsite power would result in loss of power to normal buses 23 and 24, which in turn results in loss of power to the MG sets and thus reactor trip.
2. One train of SI, RHR, CS and CC will be lost upon failure of 4.16KV essential bus 25 or 26. However, if these failures are associated with components that supply power to the buses (i.e., EDGs) and not the buses themselves, the buses 25 and 26 can be cross-tied to Unit 1 4.16KV buses 15 and 16, respectively. One train of SI, RHR and CS is also lost if either of the 125VDC trains fail to supply control power to close the pump breakers. Local operation of the breaker is possible but is not considered. One train of EDGs is lost if its 125VDC supply train fails to provide starting control power.
3. 480VAC essential MCC 2K1 provides motive power to much of Safeguards Train A. Power is provided to the bus from 4.16KV switchgear 25 through 480VAC bus 211. The following are essential loads powered from MCC 2K1.
   - RHR train A suction from containment sump B (MV-32178, MV-32180)
   - CC to RHR heat exchanger valves (MV-32128)
   - SI test line A valve to RWST (MV-32204)
   - SI train A suction valve from RHR pump 21 (MV-32208)
   - SI RWST suction valve (MV-32182)
   - Cooling Water to Component cooling heat exchanger valves (MV-32122, MV-32160)

4. 480VAC essential MCC 2KA2 provides motive power to much of Safeguards Train B. Power is provided to the bus from 4.16KV switchgear 26 through 480VAC bus 221. The following are essential loads powered from MCC 2KA2.
   - RHR train B suction from containment sump B (MV-32179, MV-32181)
   - SI test line B valve to the RWST (MV-32205)
   - SI train B suction valve from RHR pump 22 (MV-32209)
   - SI RWST suction valve (MV-32183)
   - Cooling Water to Component cooling heat exchanger valve (MV-32123)

5. 480VAC essential MCC 2K2 provides motive power to Cooling Water to Component Cooling heat exchanger valve MV-32161 and CC to RHR Heat Exchanger B valve MV-32129. Power is provided to the bus from 4.16KV switchgear 26 through 480VAC bus 221.
6. MCC 2A1 provides motive power to 21 MD AFW pump CL suction valve MV-32026. Loss of power will prevent the remote opening of the valve. Operator can manually open the valve locally, however.
7. MCC 2A2 provides motive power to 22 TD AFW pump CL suction valve MV-32030. Loss of power will prevent the remote opening of the valve. Operator can manually open the valve locally, however.
8. 4.16KV emergency bus 25 supplies power to AFW motor driven pump 21(train A). Makeup to unit 2 SGs can be supplied from the MD pump in the Unit 1 AFW system through a system cross-tie. Unit 1 AFW MD pump 12 motive power is supplied from unit 1 emergency bus 16. The 12 and 21 MD driven pump breakers control are supplied from unit 1 and 2 DC sources, respectively.
9. Cooling water (CL) provides the heat sink for the CC heat exchangers. With failure of Train A(B) CL, Train A(B) CC is assumed to fail eventually causing loss of corresponding SI & RHR trains for decay heat removal function.
10. Cooling water normally is a backup suction source to the CST for long term secondary cooling using AFW. During a seismic event, however, it becomes a primary suction source for AFW since the CST is assumed to fail as a result of earthquake as it is not seismically designed.
11. Steam to the AFW turbine for pump 22 is provided from MS loops A and B. Failure of either Main Steam loop does not necessarily lead to loss of AFW train B.

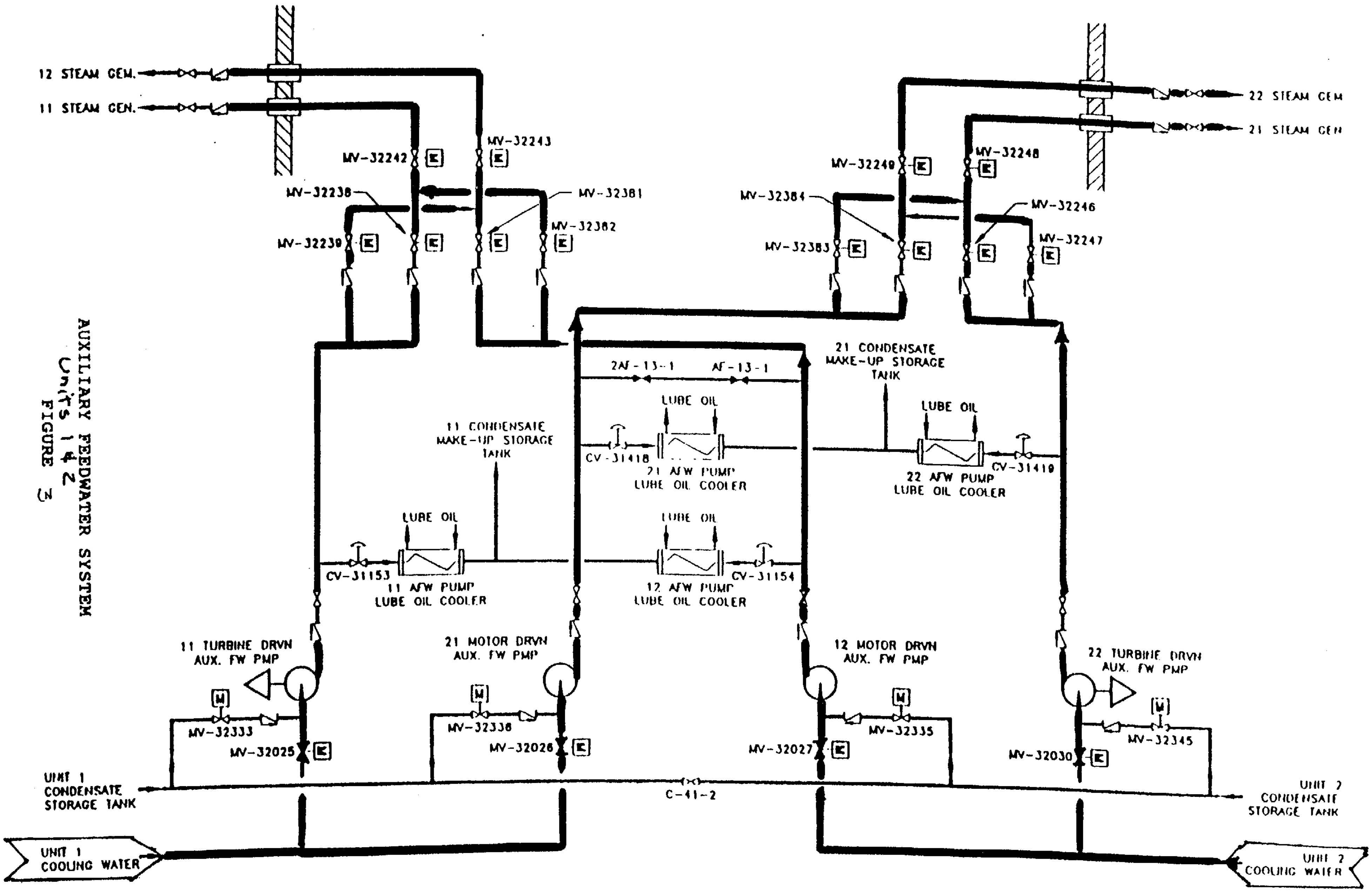
Figure 1. Success Block Diagram



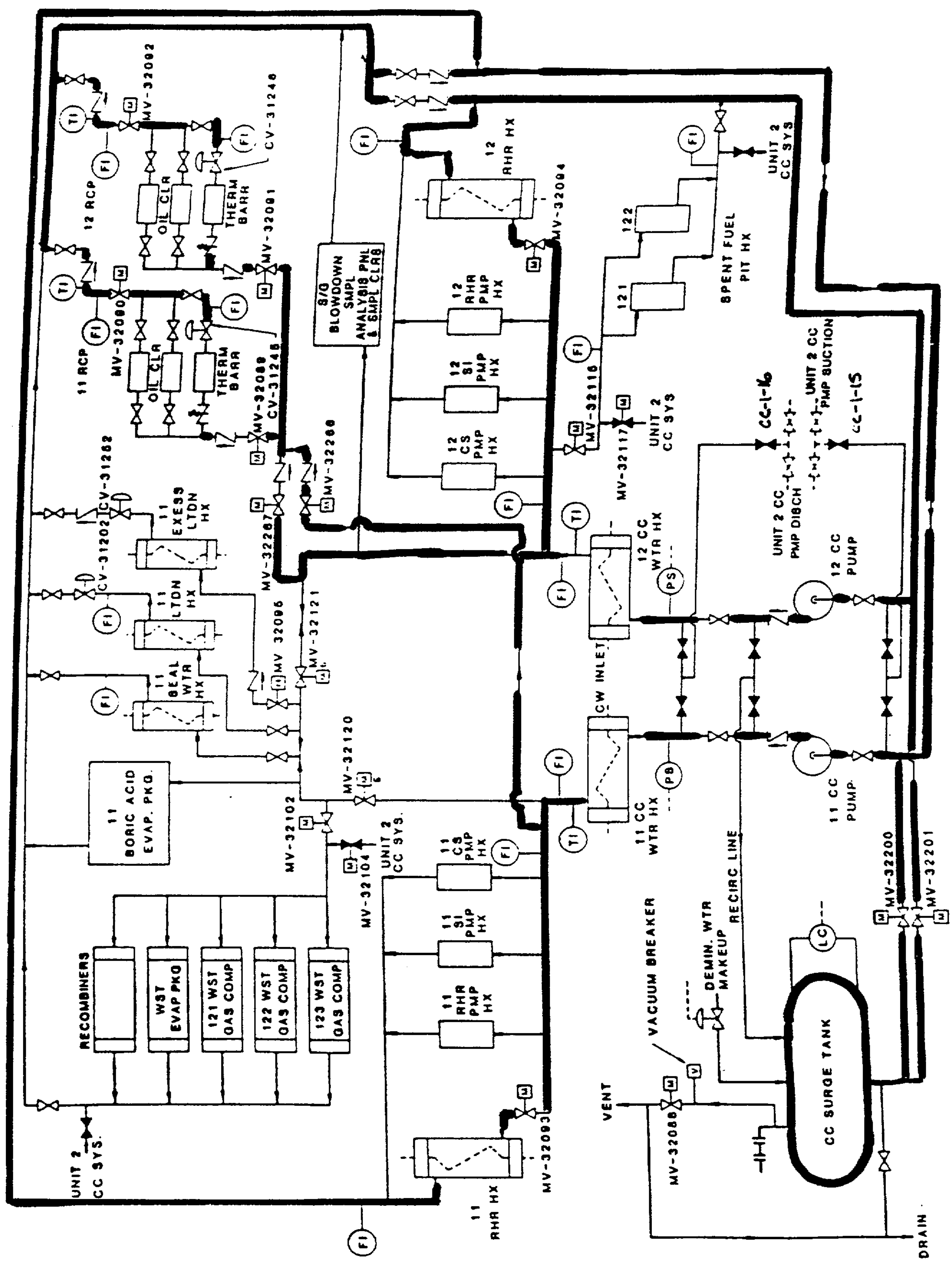


SIMPLIFIED REACTOR PROTECTION SYSTEM

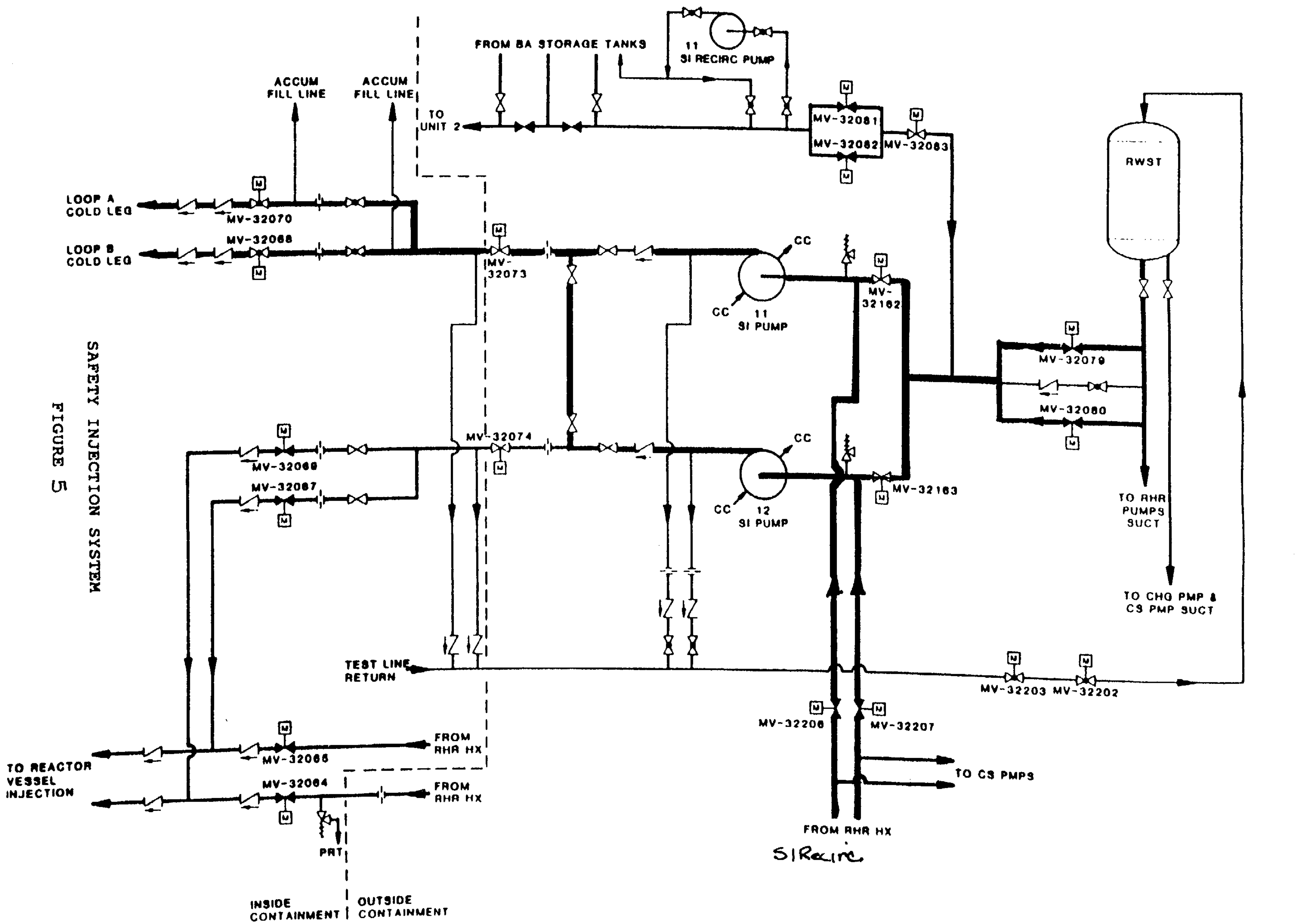
FIGURE 2



AUXILIARY FEEDWATER SYSTEM
Units 1 & 2
FIGURE 2

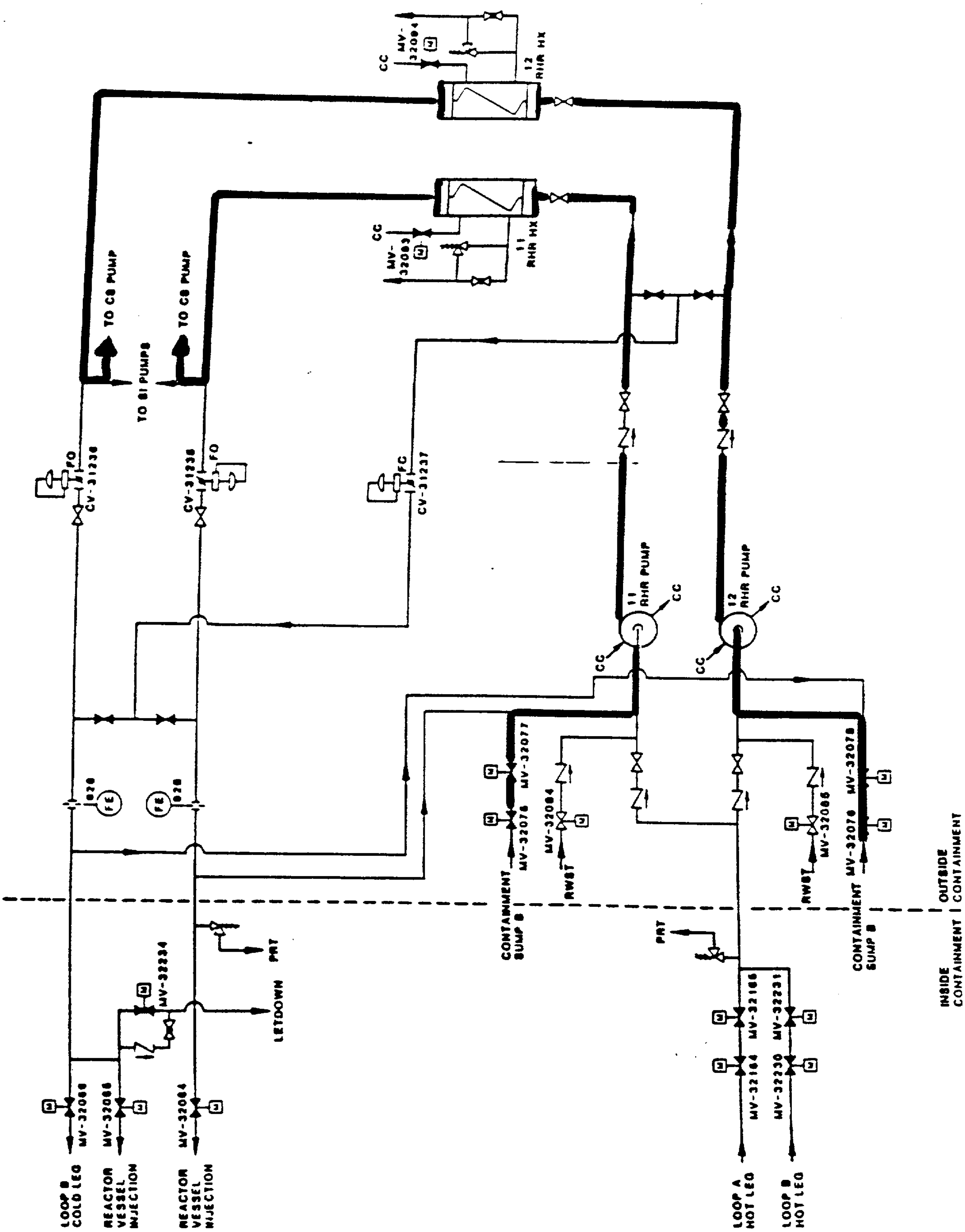


UNIT 1 COMPONENT COOLING WATER SYSTEM
(TYPICAL -RCP SEAL COOLING & COOLING TO RHR HXs)
FIGURE 4



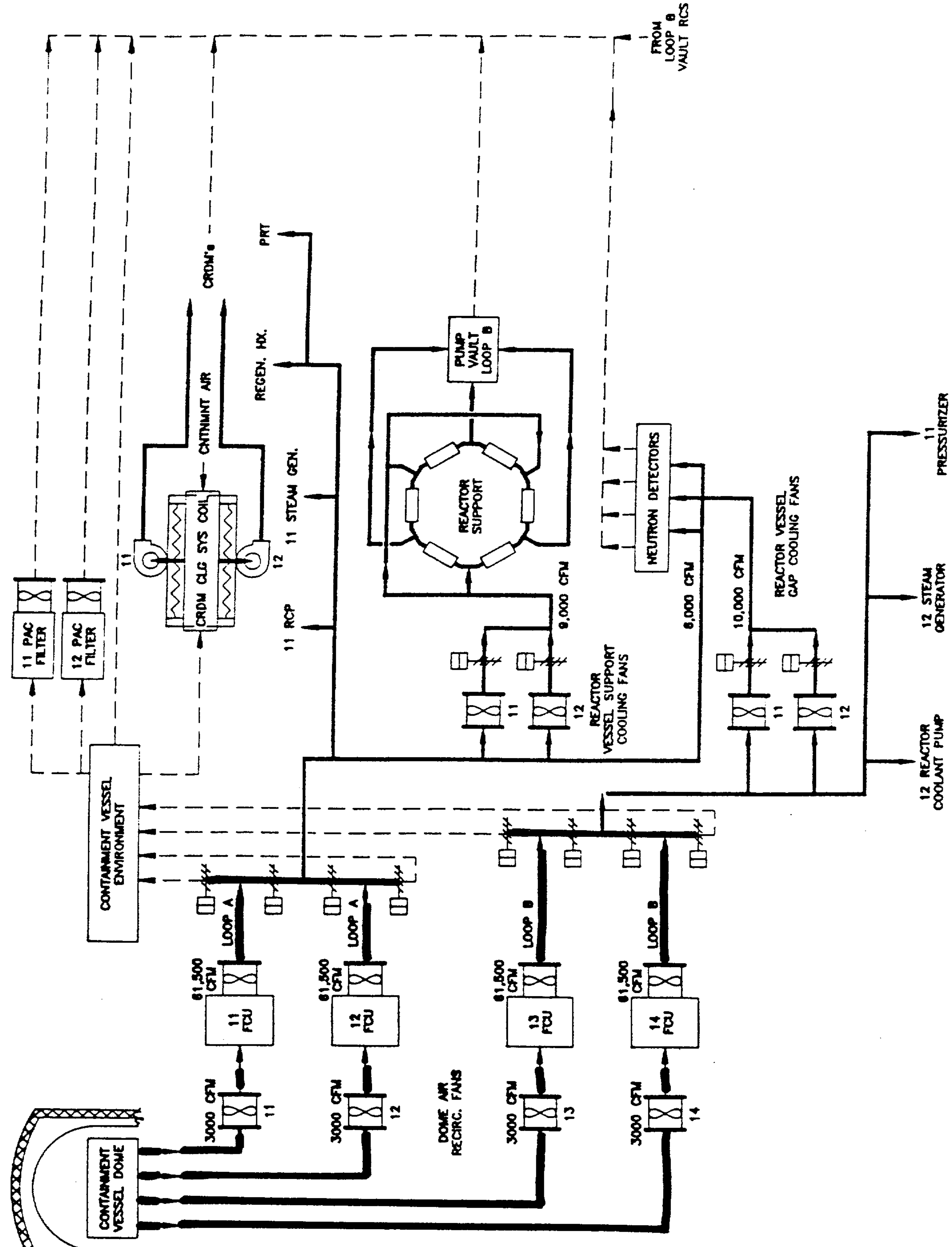
SAFETY INJECTION SYSTEM

FIGURE 5



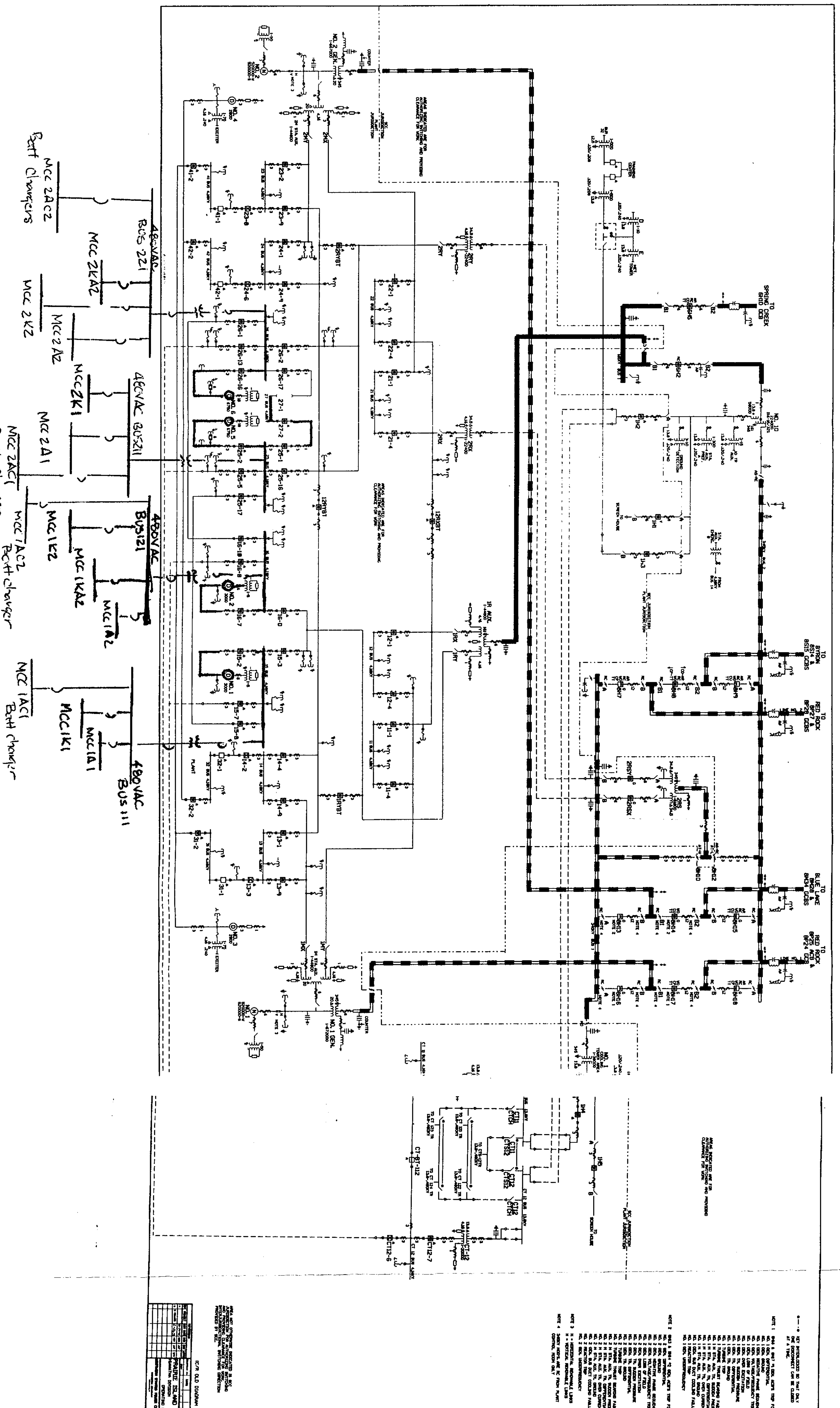
RESIDUAL HEAT REMOVAL SYSTEM (SI RECIRCULATION)

FIGURE 6



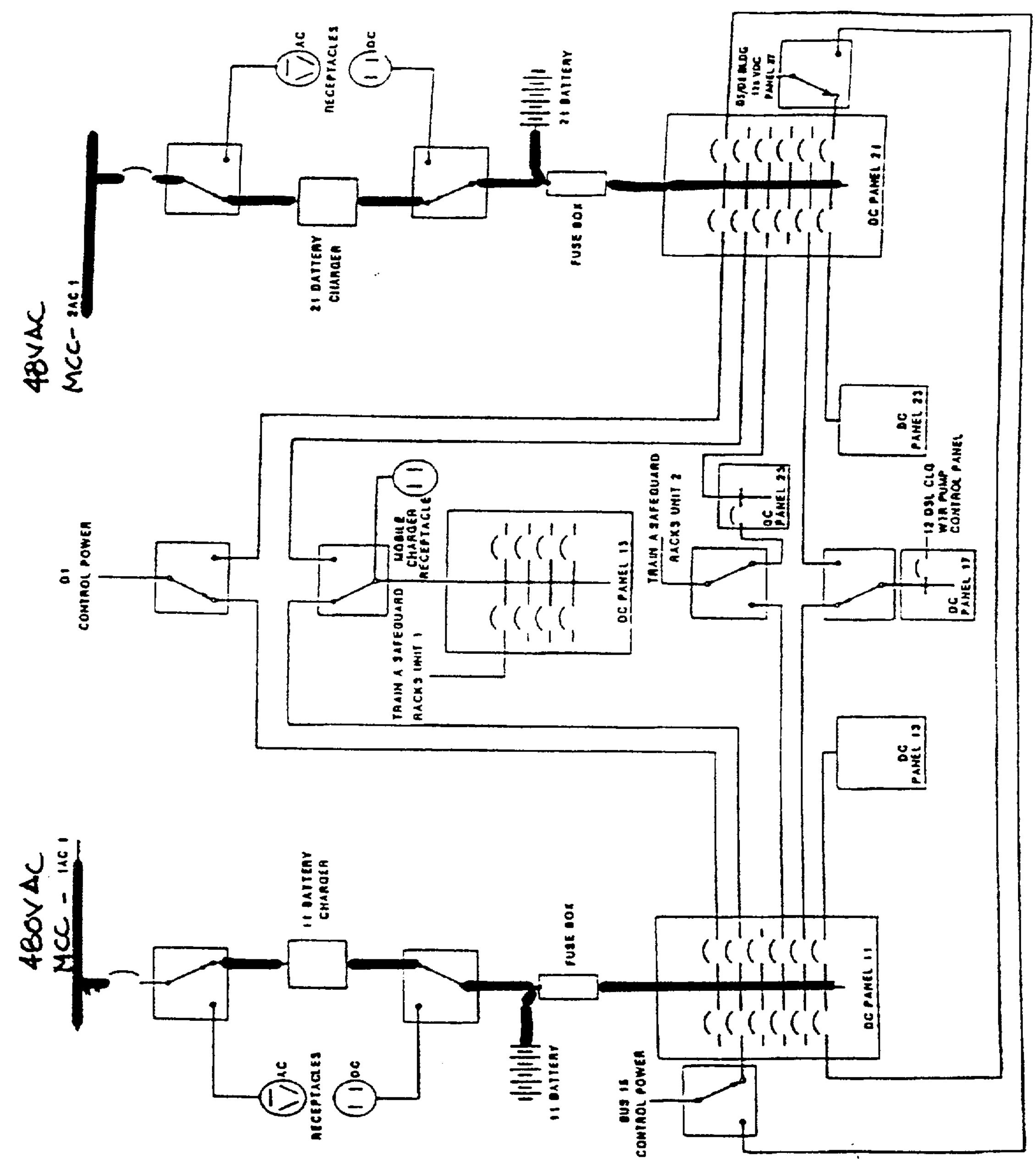
CONTAINMENT VESSEL AIR HANDLING SYSTEM

FIGURE 7



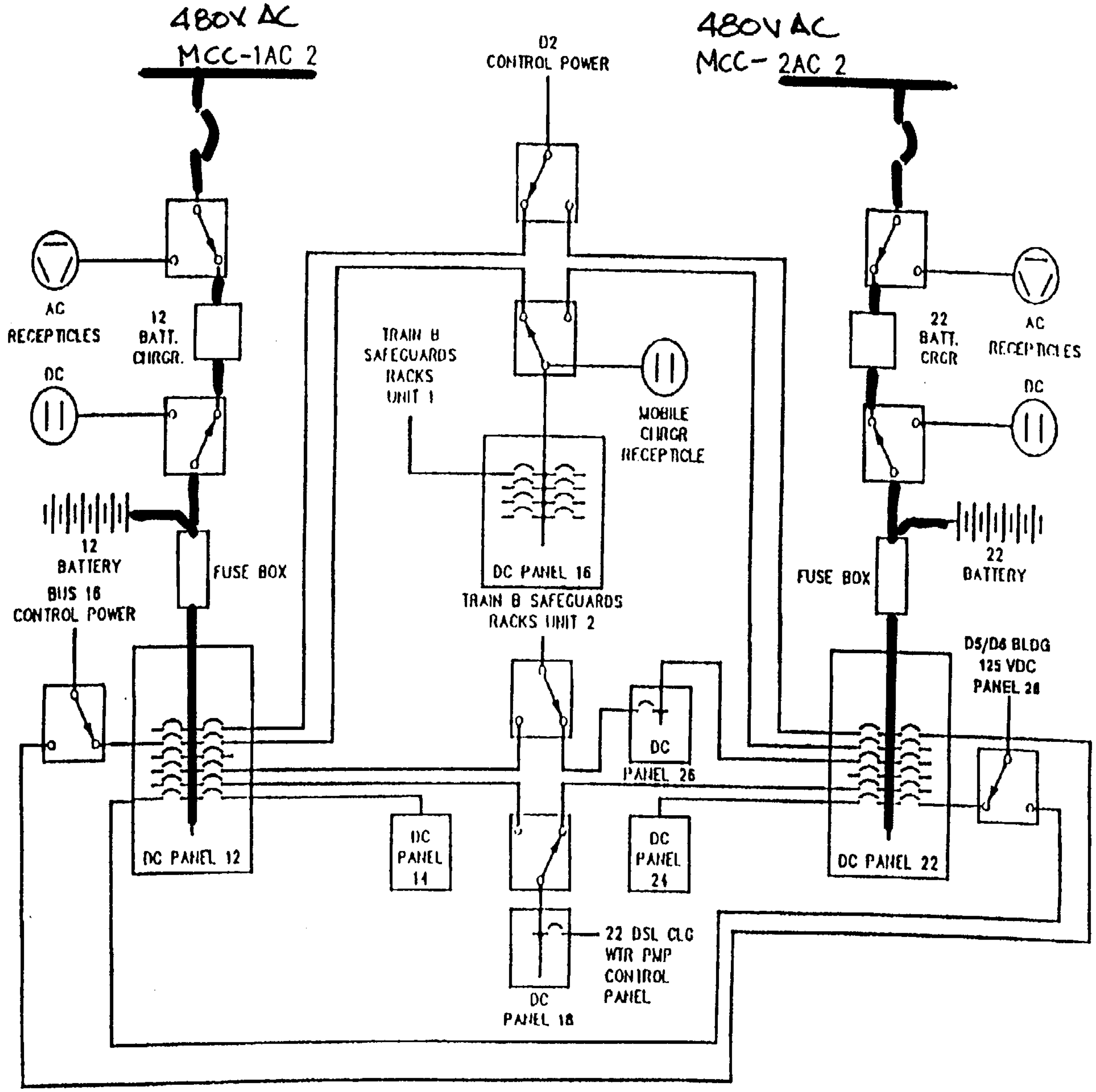
AC Power Distribution

Figure 8



TRAIN A DC POWER

FIGURE 9



TRAIN B DC POWER

FIGURE 10

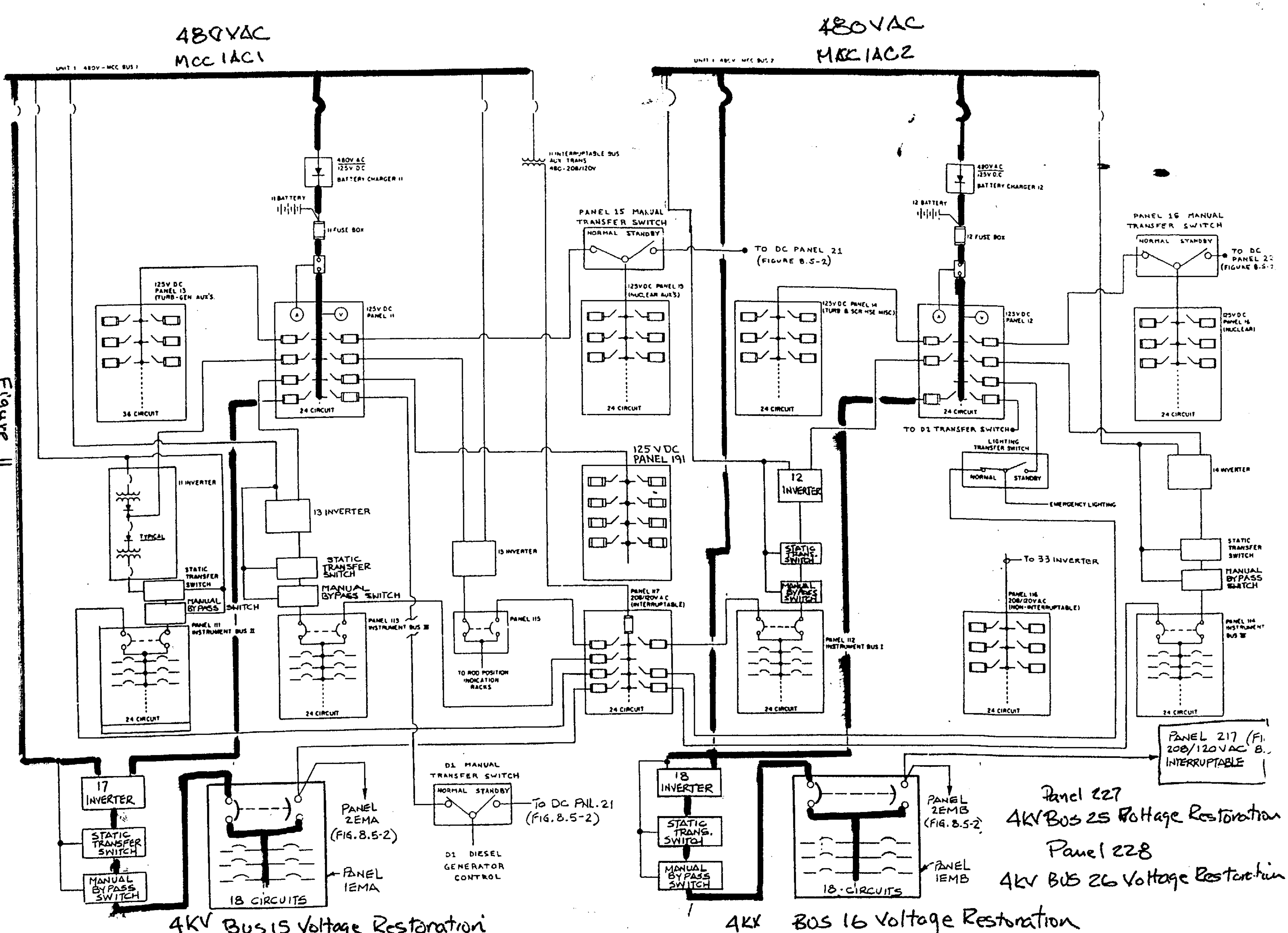
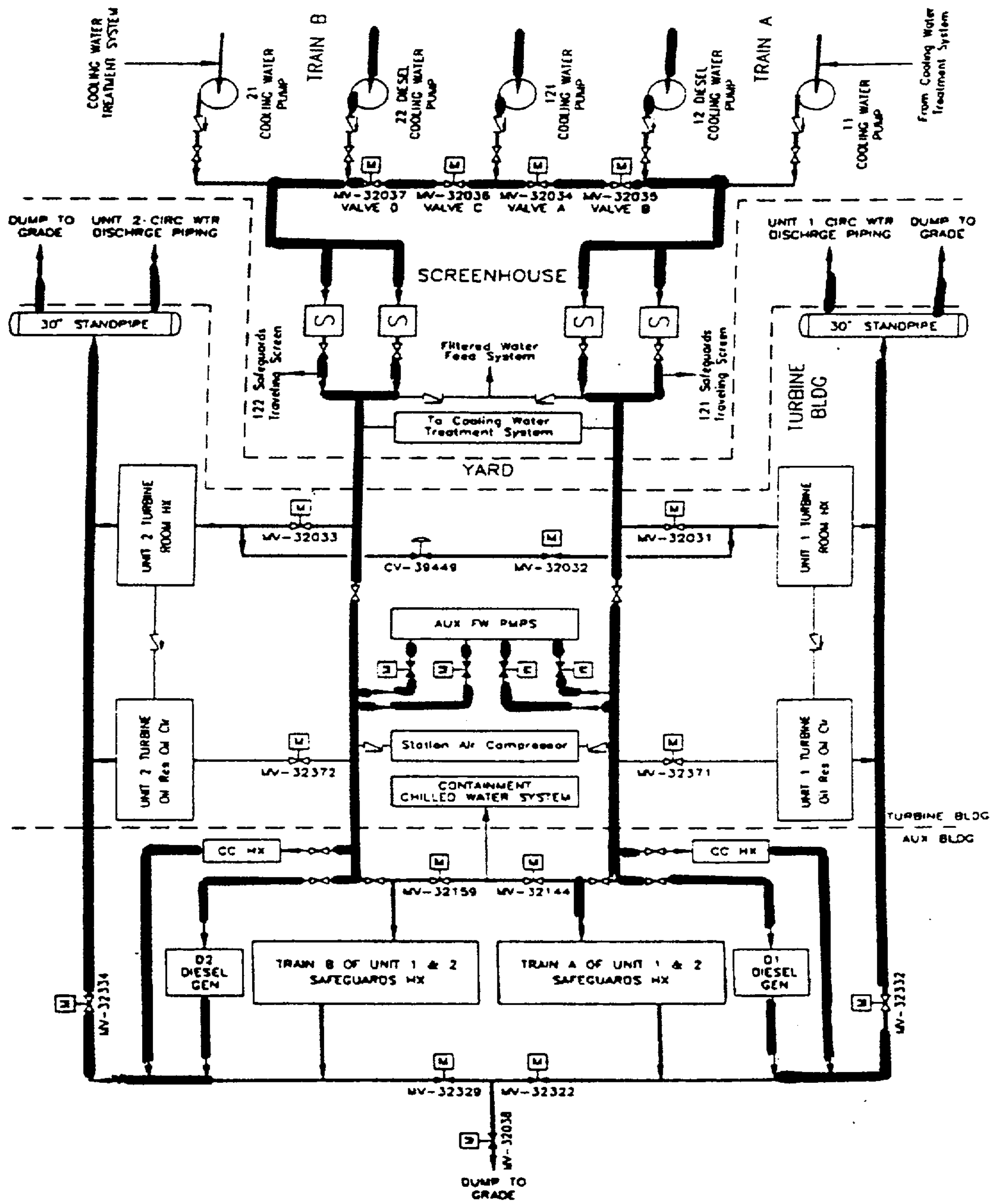


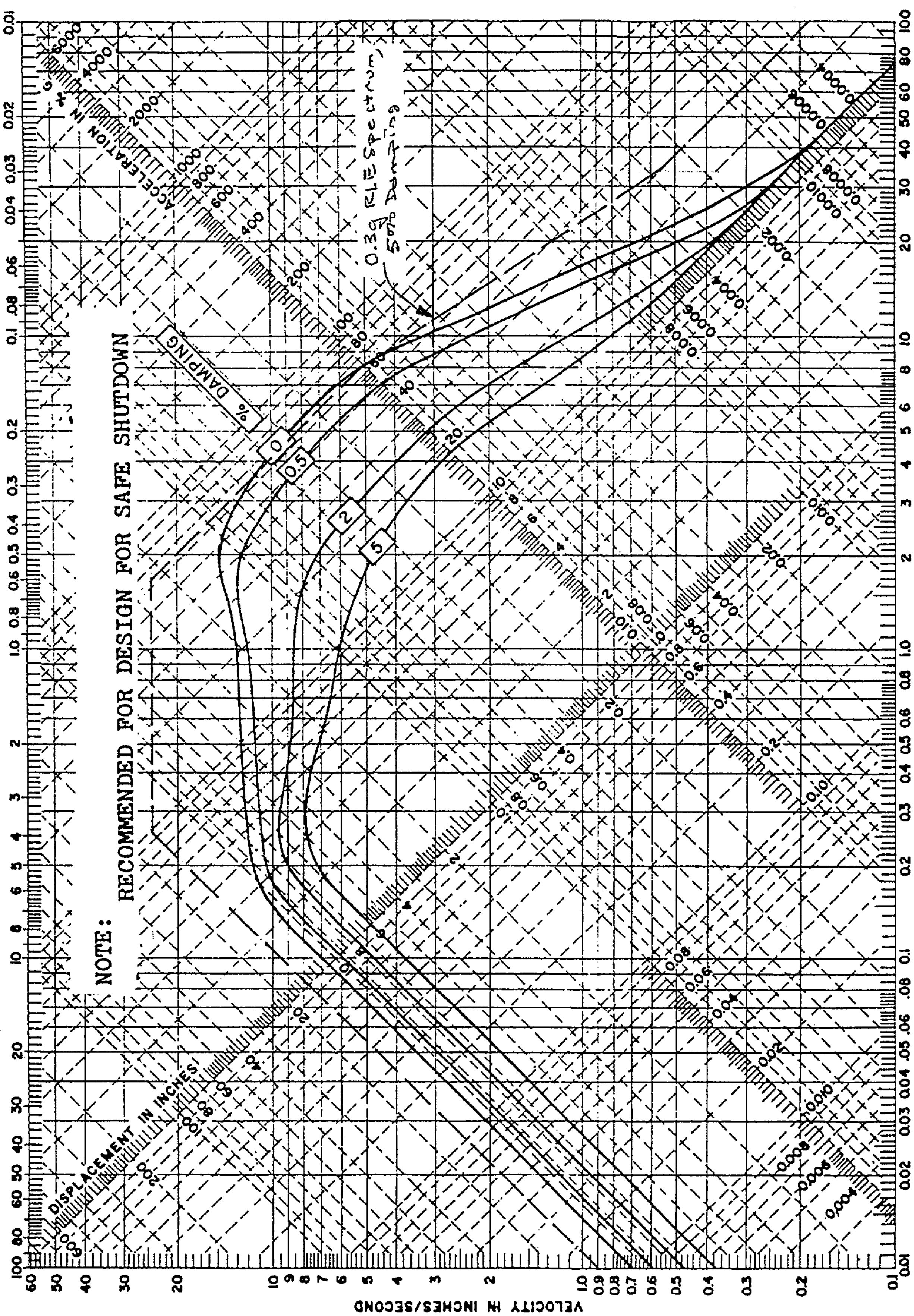
Figure 11

125V DC 120 V AC INSTRUMENT SUPPLY - UNIT 1



COOLING WATER SYSTEM

FIGURE 12

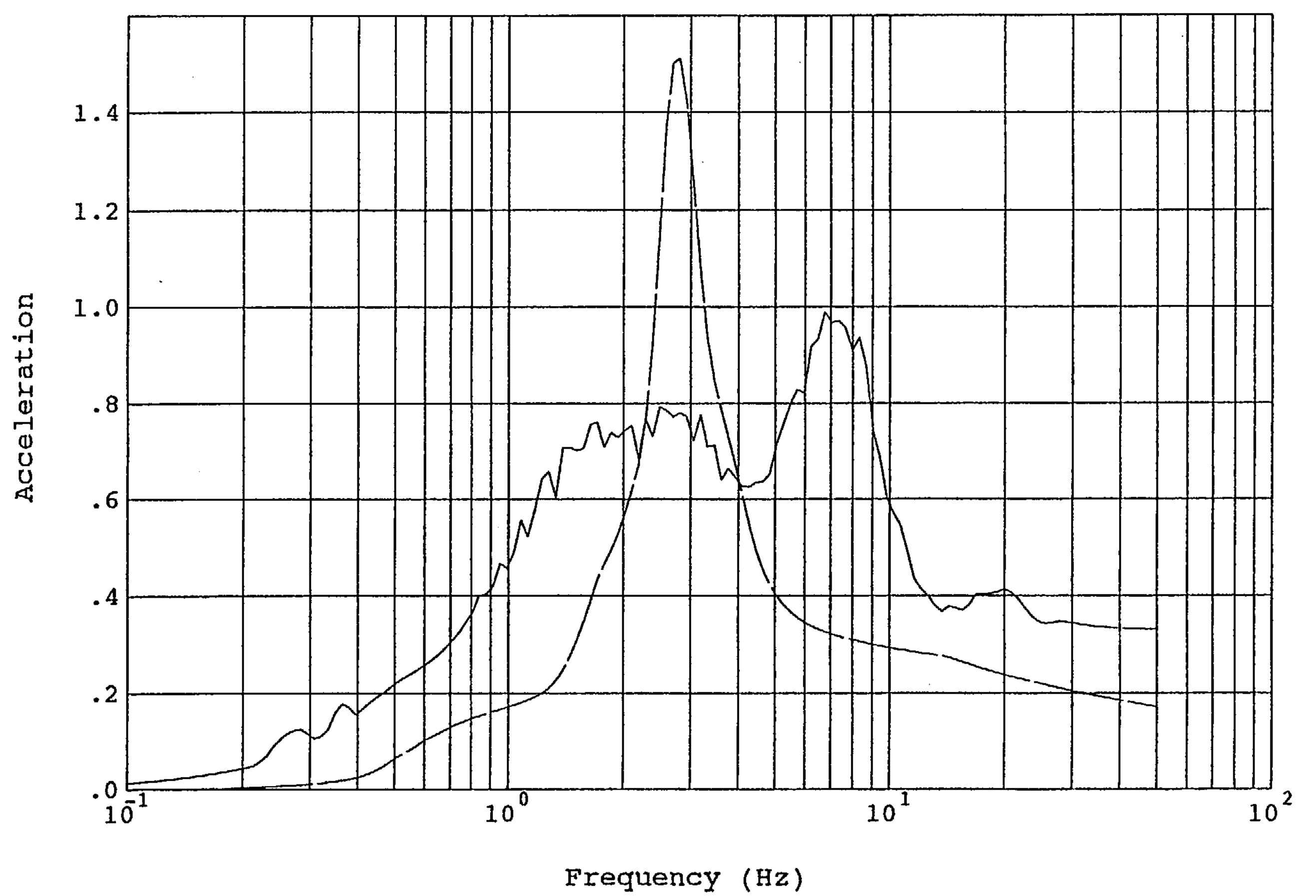


RECOMMENDED RESPONSE SPECTRA

MAXIMUM CREDIBLE EARTHQUAKE - 12% ACCELERATION DAMES & MOORE

PLATE H 4.6 (Rev.1)

Figure 13



Legend:

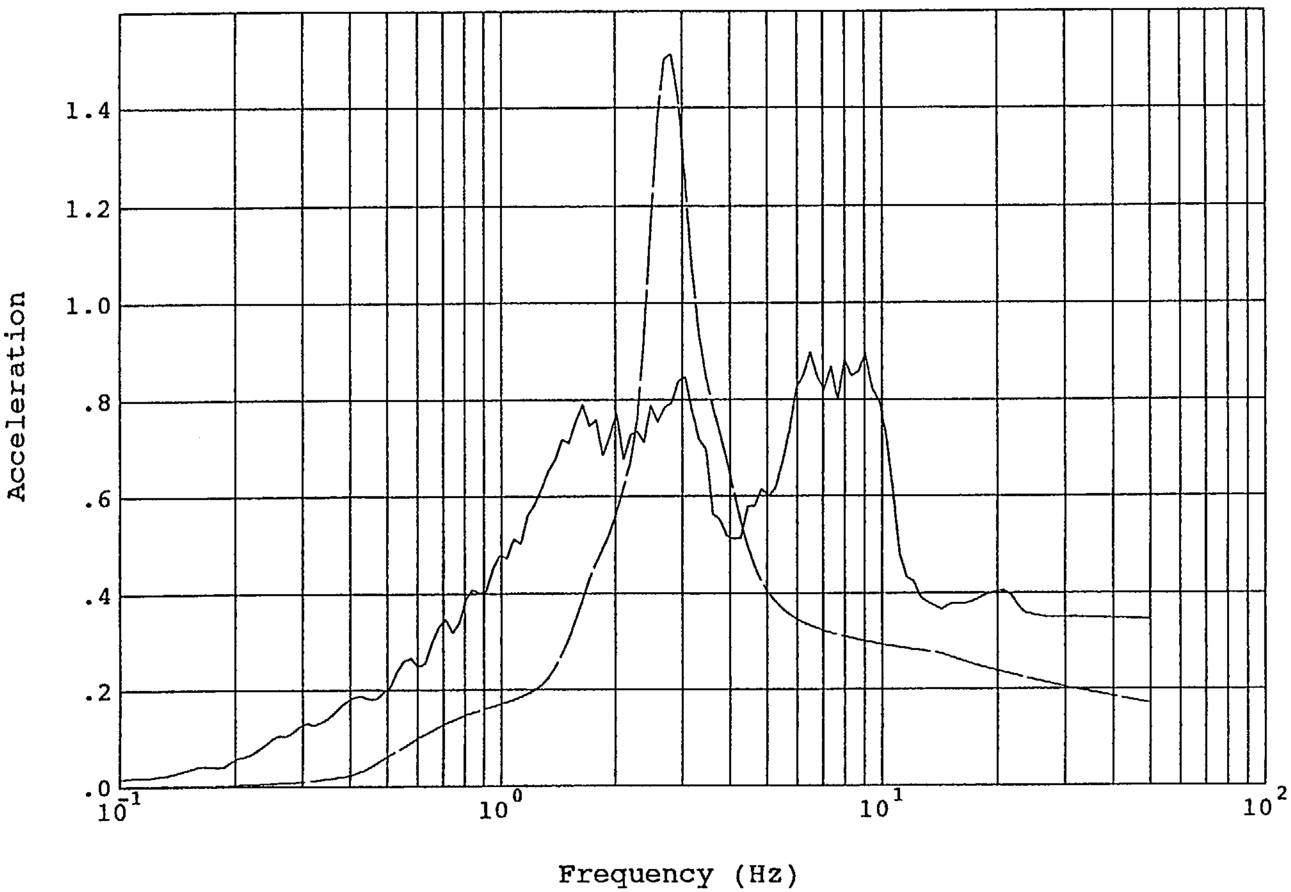
SMA 0.3g RLE	————————
SSE	——— ———

Notes:

5% Spectral Damping
Accelerations in g's

250800 C005: Praire Island Nuclear Generating Plant Spectra
Auxiliary Building Elev. 735 X (E-W) Direction

Figure 14



Legend:

SMA 0.3g RLE ————

SSE — — — —

Notes:

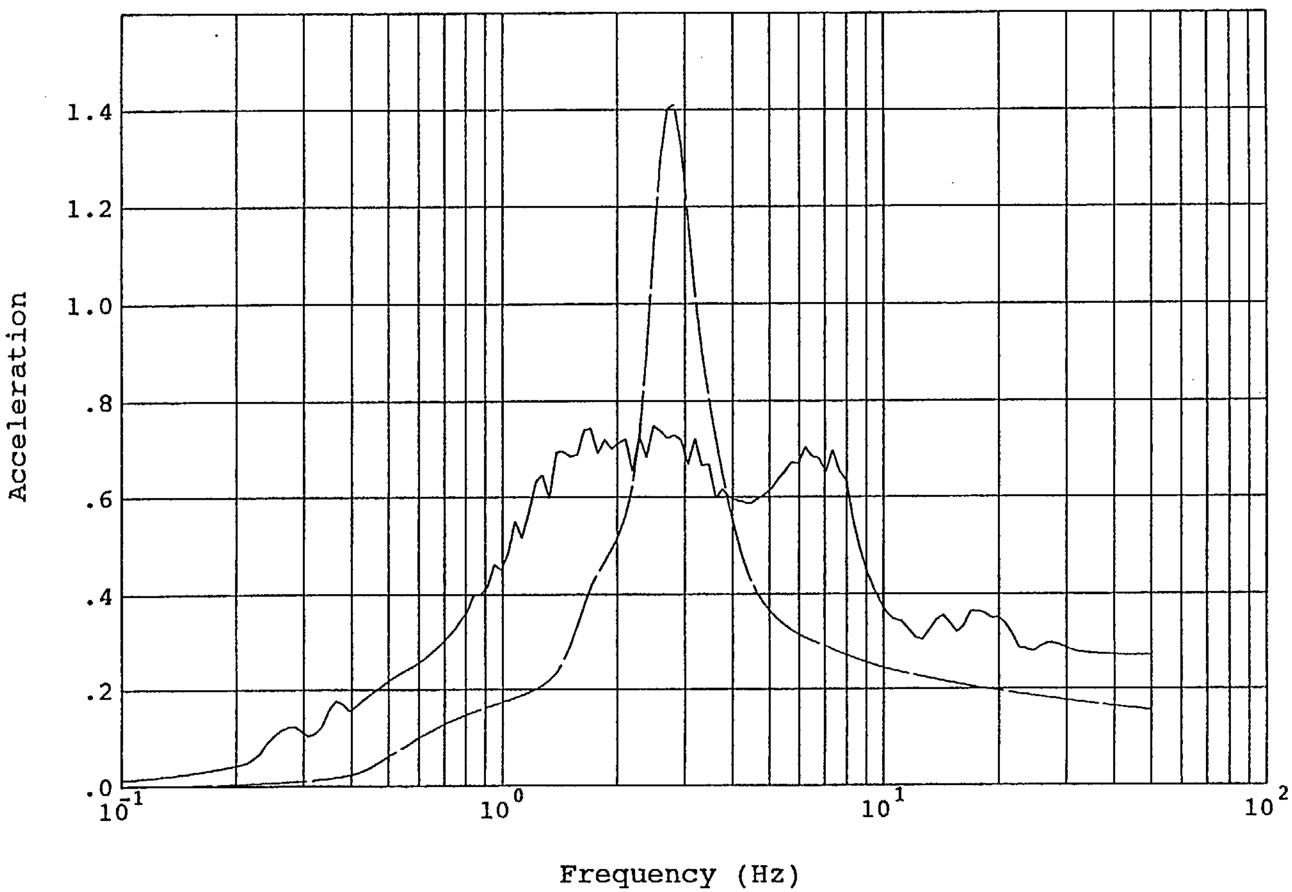
5% Spectral Damping

Accelerations in g's

250800 C005: Praire Island Nuclear Generating Plant Spectra
Auxiliary Building Elev. 735 Y (N-S) Direction

Figure 15

p. 110



Legend:

SMA 0.3g RLE ————————

SSE ——— ———

Notes:

5% Spectral Damping

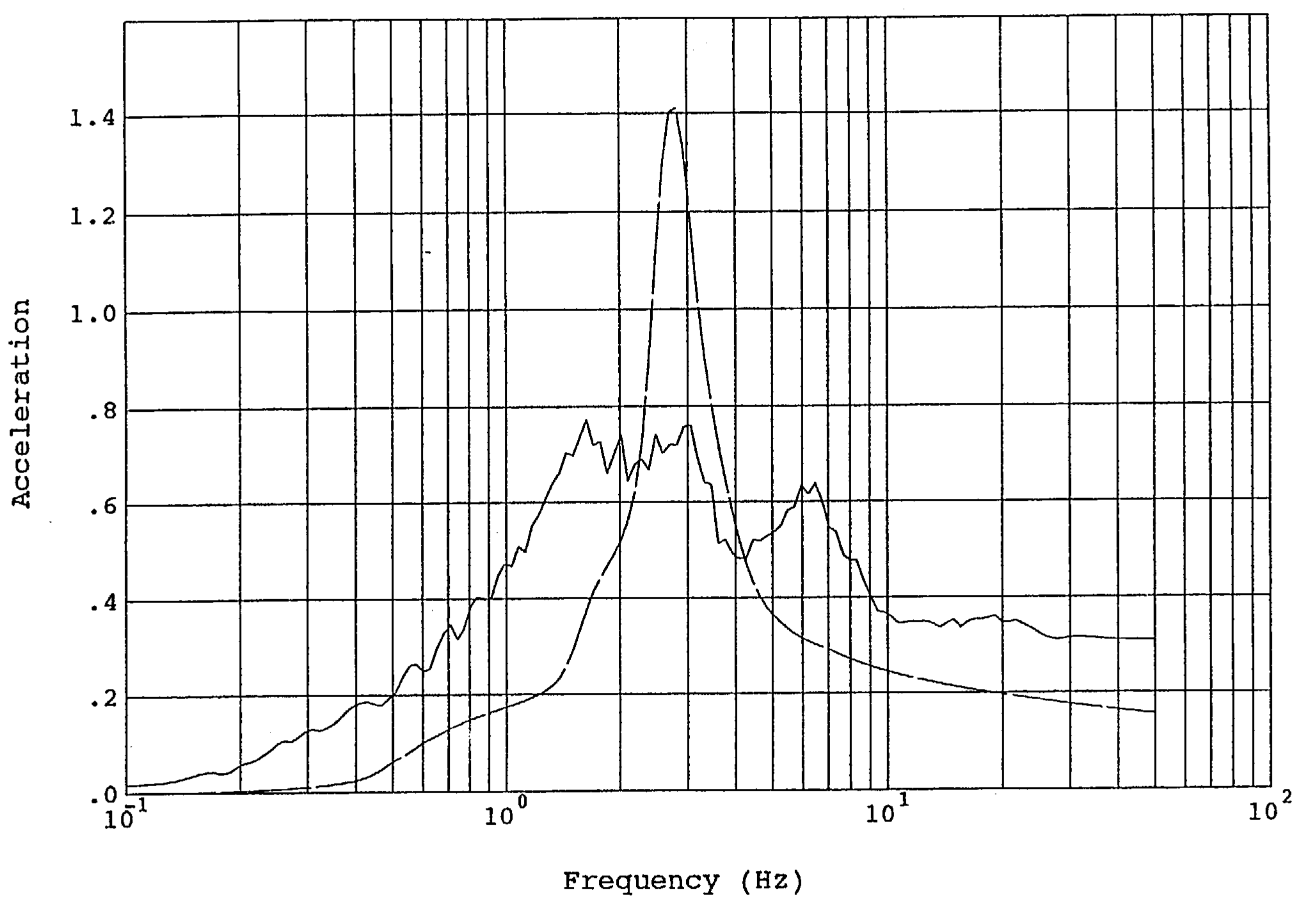
Accelerations in g's

250800 C005: Praire Island Nuclear Generating Plant Spectra
Auxiliary Building Elev. 715 X (E-W) Direction

Figure 16

RSPLT INTEL386 VERS 1.2P COMPILED 08/04/93 09:30:31 EXECUTED 12/09/99 12:12:24 aux39x.ps

p. 111



Legend:

SMA 0.3g RLE ————————

SSE ——— ———

Notes:

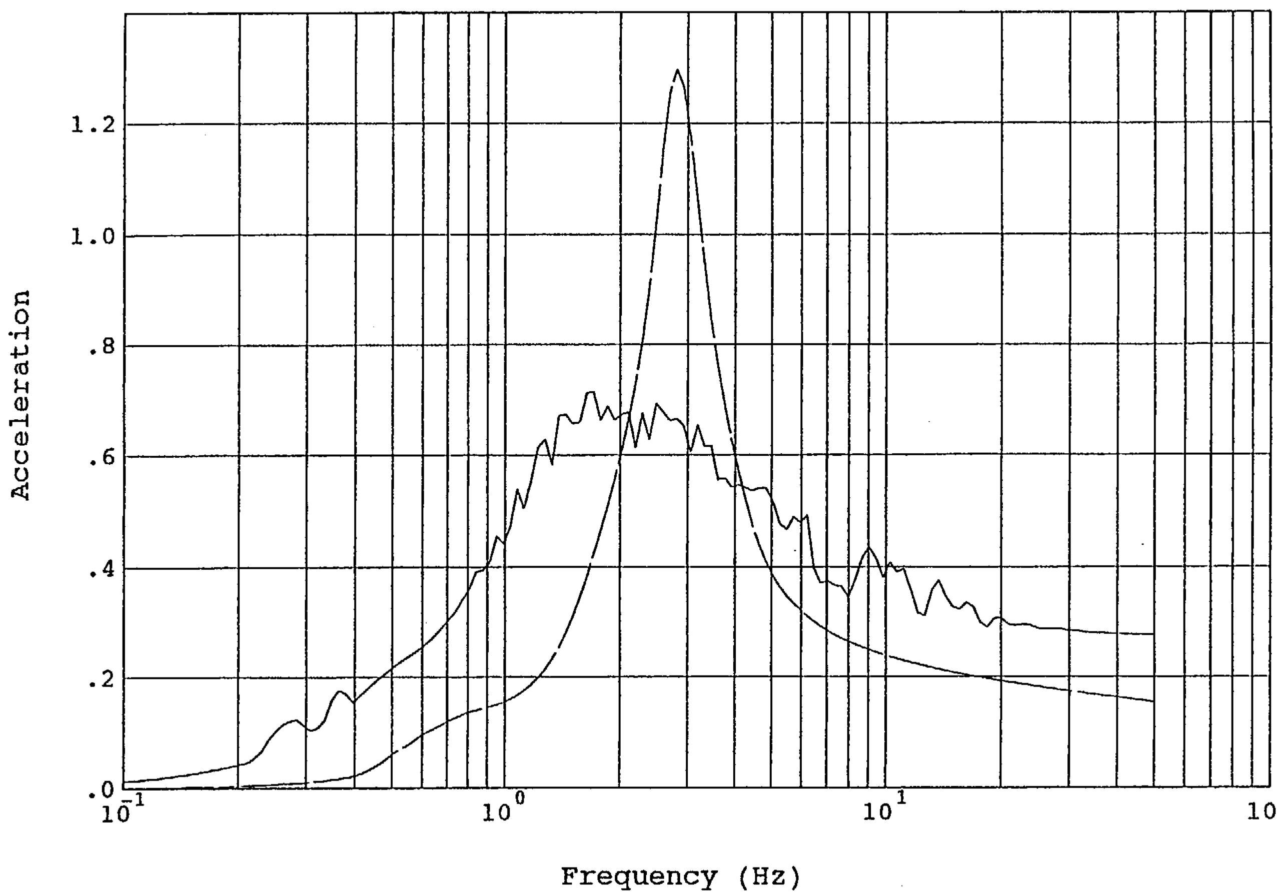
5% Spectral Damping

Accelerations in g's

250800 C005: Praire Island Nuclear Generating Plant Spectra
Auxiliary Building Elev. 715 Y (N-S) Direction

Figure 17

RSPLT INTEL386 VERS 1.2P COMPILED 08/04/93 09:30:31 EXECUTED 12/09/99 12:12:24 aux39y.ps



Legend:

SMA 0.3g RLE ____________

SSE ______ ______

Notes:

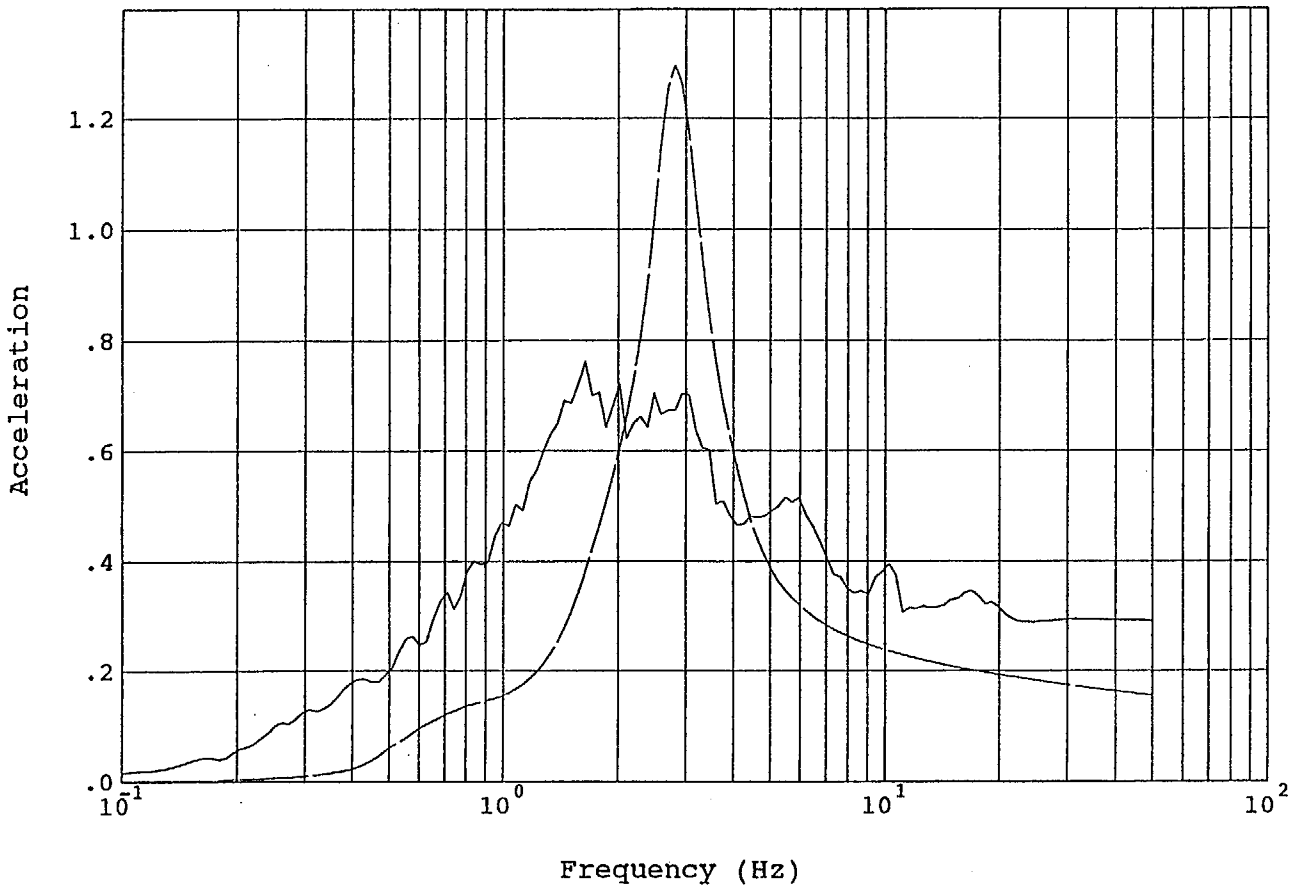
5% Spectral Damping

Accelerations in g's

250800 C005: Praire Island Nuclear Generating Plant Spectra
Auxiliary Bldg. 695' X (E-W) Direction

Figure 18

RSPLT INTEL386 VERS 1.2P COMPILED 08/04/93 09:30:31 EXECUTED 12/10/99 05:49:55 aux12x.ps



Legend:

SMA 0.3g RLE ————————

SSE ———— ————

Notes:

5% Spectral Damping

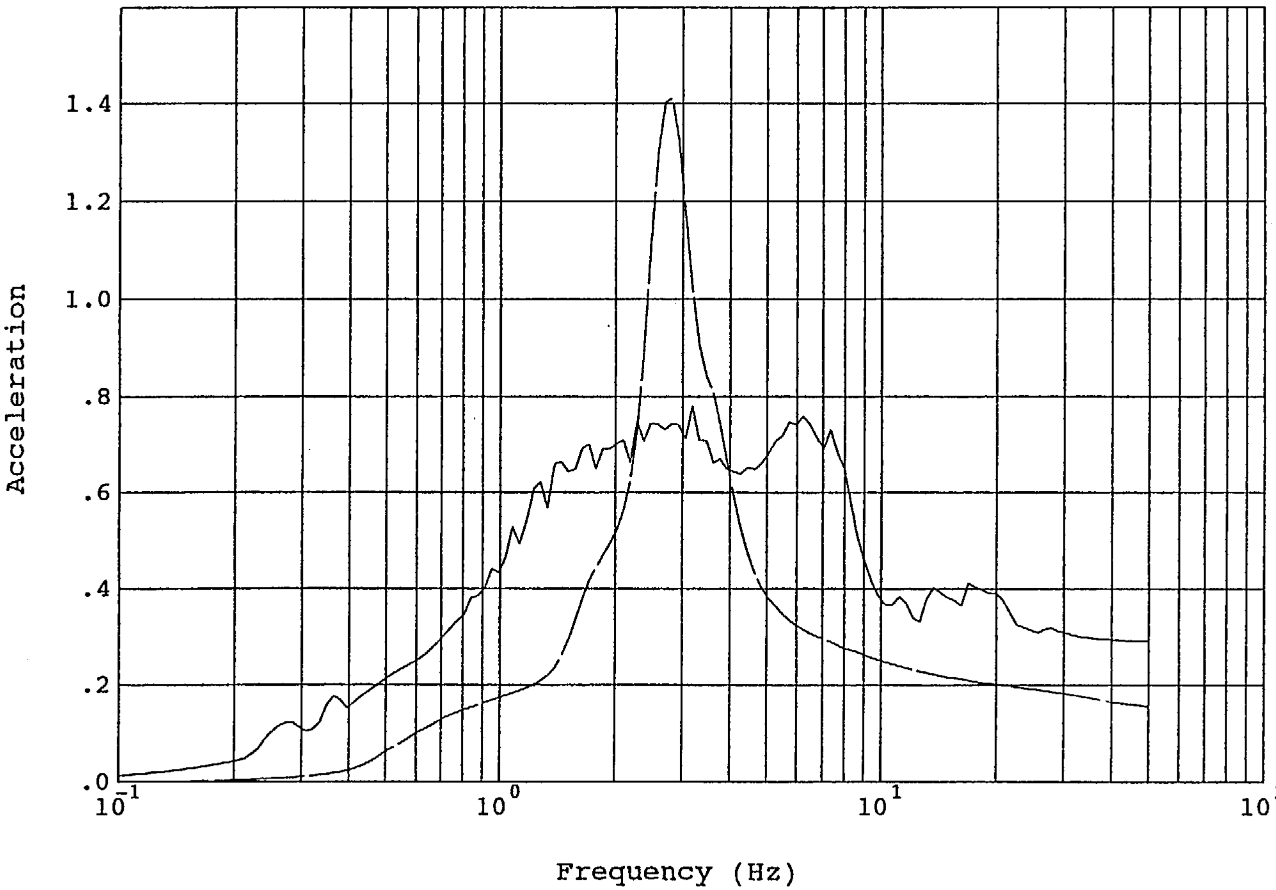
Accelerations in g's

250800 C005: Praire Island Nuclear Generating Plant Spectra
Auxiliary Bldg. 695' Y (N-S) Direction

Figure 19

RSPLT INTEL386 VERS 1.2P COMPILED 08/04/93 09:30:31 EXECUTED 12/10/99 05:49:55 aux12y.ps

p.114



Legend:

SMA 0.3g RLE ____________

SSE ______ ______

Notes:

5% Spectral Damping

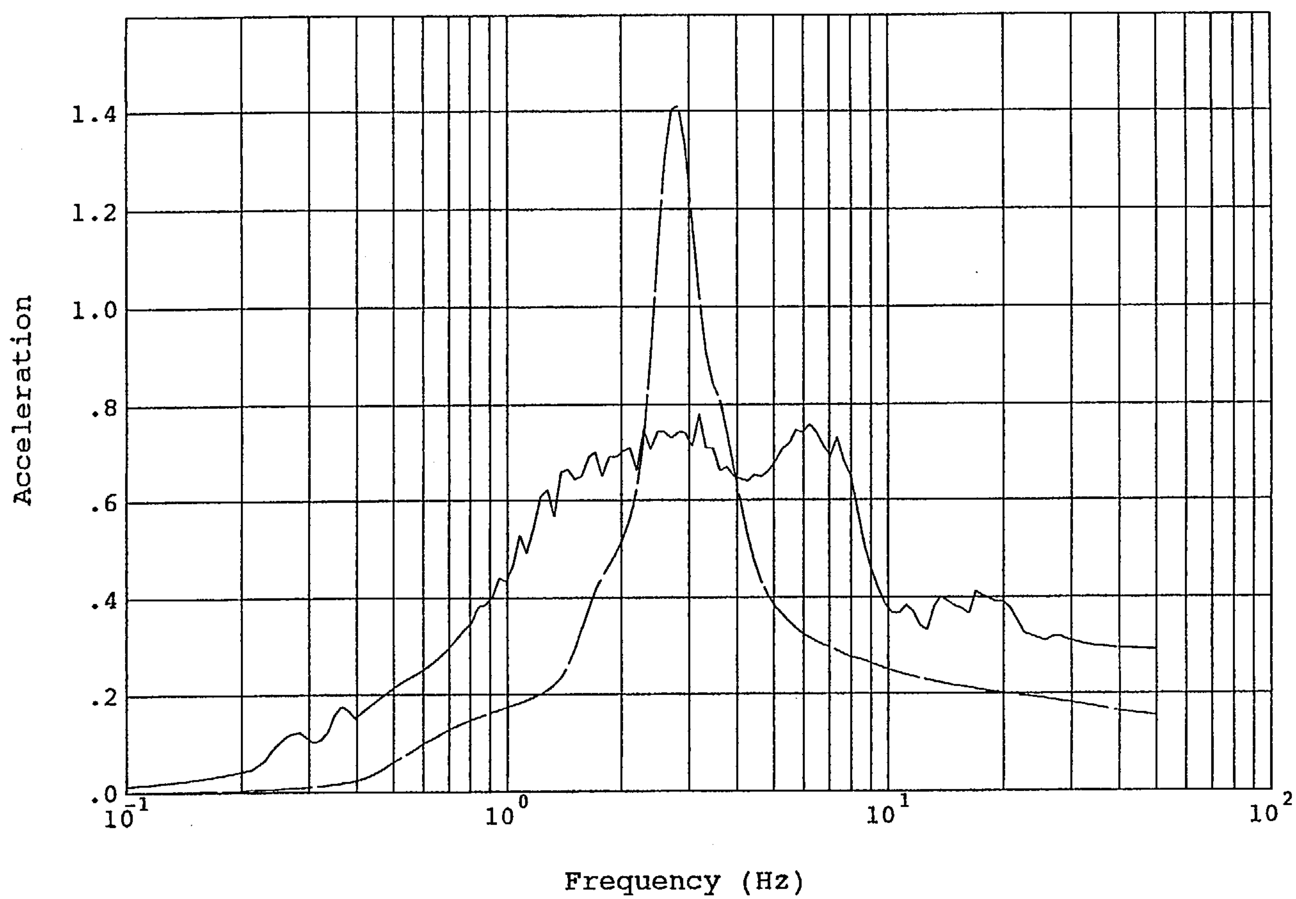
Accelerations in g's

250800 C005: Praire Island Nuclear Generating Plant Spectra
Turbine Building Elev. 715 X (E-W) Direction

Figure 20

RSPLT INTEL386 VERS 1.2P COMPILED 08/04/93 09:30:31 EXECUTED 12/09/99 12:12:24 turb21x.ps

p. 115



Legend:

SMA 0.3g RLE

SSE

Notes:

5% Spectral Damping

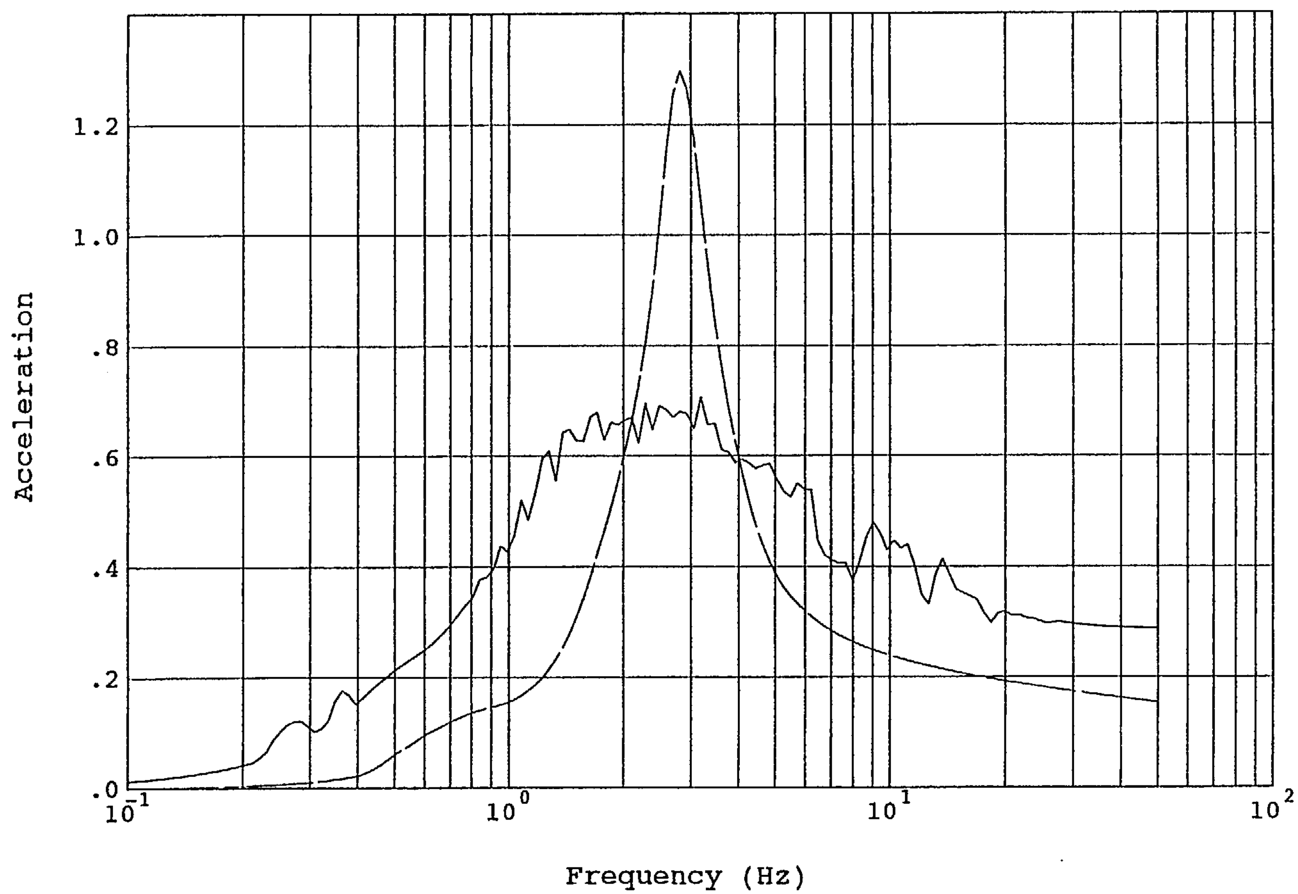
Accelerations in g's

250800 C005: Praire Island Nuclear Generating Plant Spectra
Turbine Building Elev. 715 Y (N-S) Direction

Figure 21

RSPLT INTEL386 VERS 1.2P COMPILED 08/04/93 09:30:31 EXECUTED 12/09/99 12:12:24 turb21y.ps

p.116



Legend:

SMA 0.3g RLE ————————

SSE ———— ————

Notes:

5% Spectral Damping

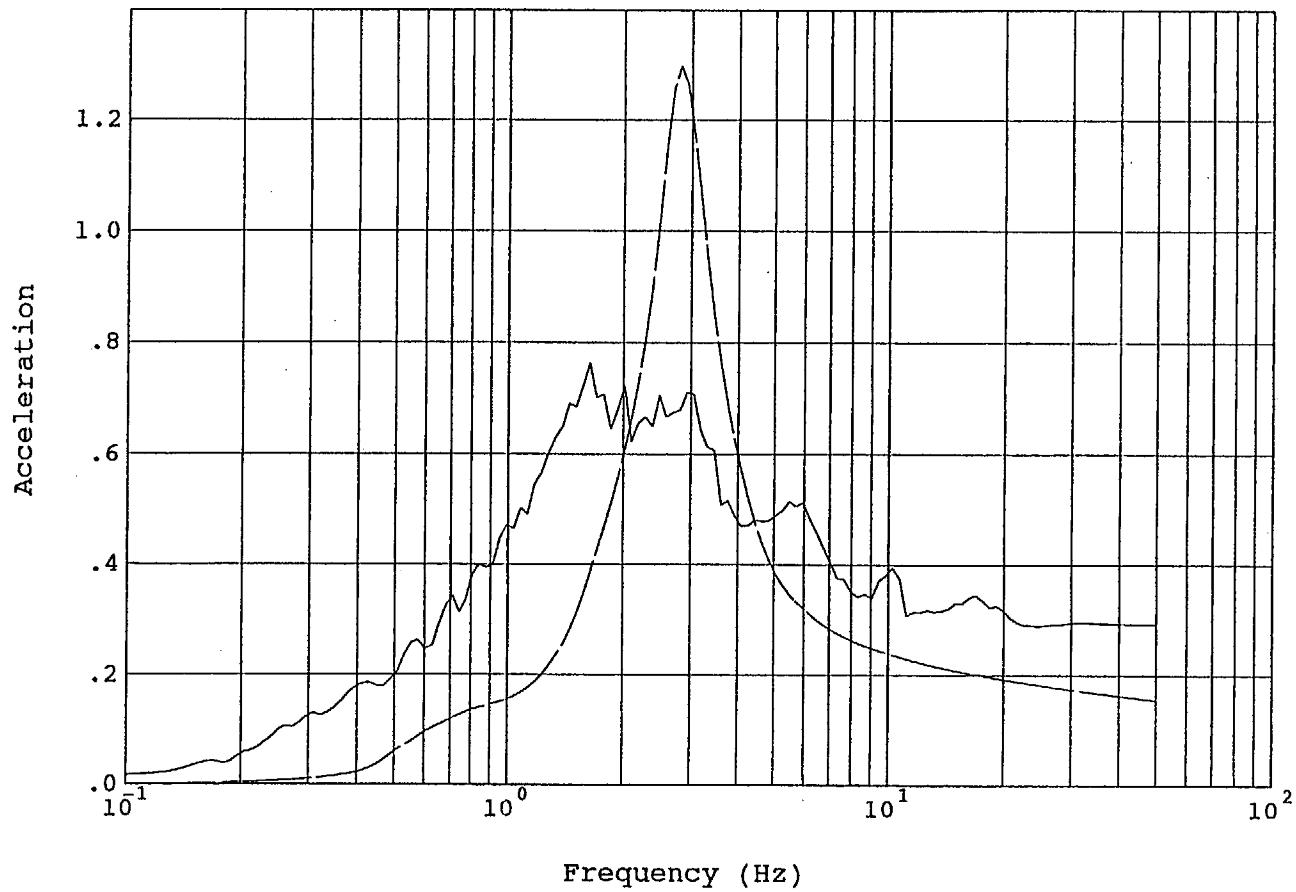
Accelerations in g's

250800 C005: Praire Island Nuclear Generating Plant Spectra
Turbine Bldg. 695' X (E-W) Direction

Figure 22

RSPLT INTEL386 VERS 1.2P COMPILED 08/04/93 09:30:31 EXECUTED 12/10/99 05:49:55 aux22x.ps

p. 117



Legend:

SMA 0.3g RLE ____________

SSE ______ ______

Notes:

5% Spectral Damping

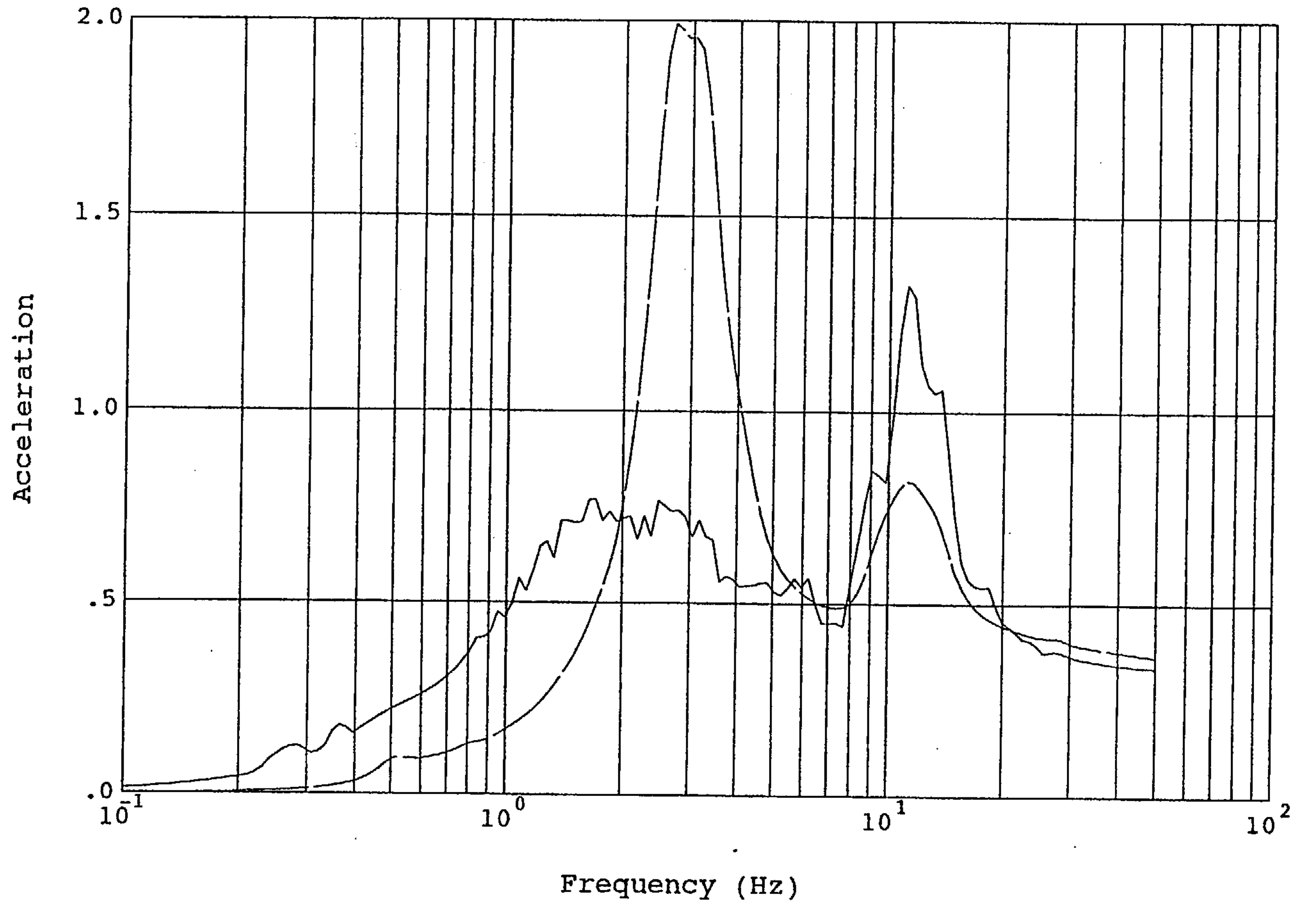
Accelerations in g's

250800 C005: Praire Island Nuclear Generating Plant Spectra
Turbine Bldg. 695' Y (N-S) Direction

Figure 23

RSPLT INTEL386 VERS 1.2P COMPILED 08/04/93 09:30:31 EXECUTED 12/10/99 05:49:55 aux22y.ps

p.118



Legend:

SMA 0.3g RLE ————————

SSE ———— ————

Notes:

5% Spectral Damping

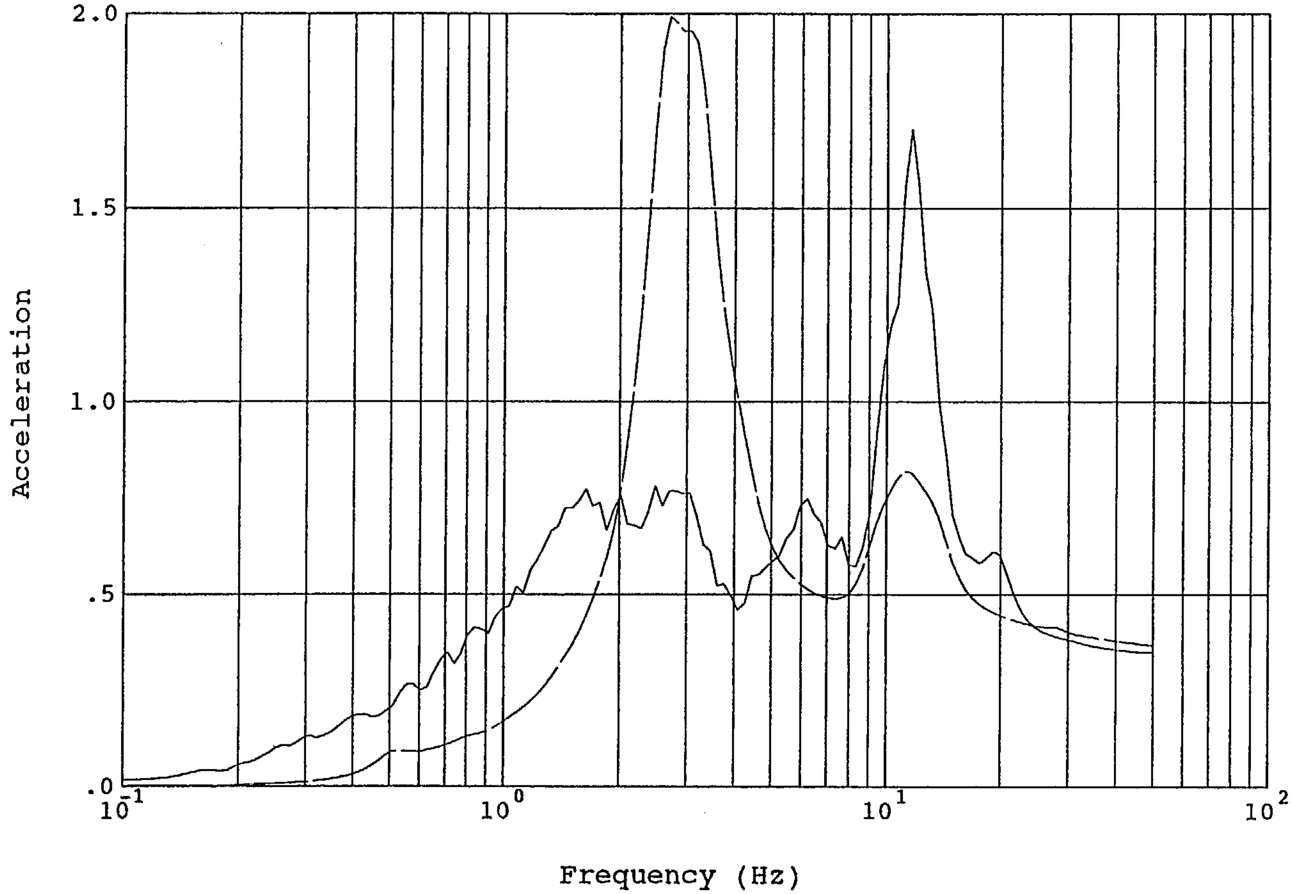
Accelerations in g's

250800 C005: Praire Island Nuclear Generating Plant Spectra
Unit 1 Reactor Support Elev. 755 X (E-W) Direction

Figure 24

RSPLT INTEL386 VERS 1.2P COMPILED 08/04/93 09:30:31 EXECUTED 12/10/99 09:52:51 u1rs22x.ps

p. 119



Legend:

SMA 0.3g RLE ____________

SSE ______ ______

Notes:

5% Spectral Damping

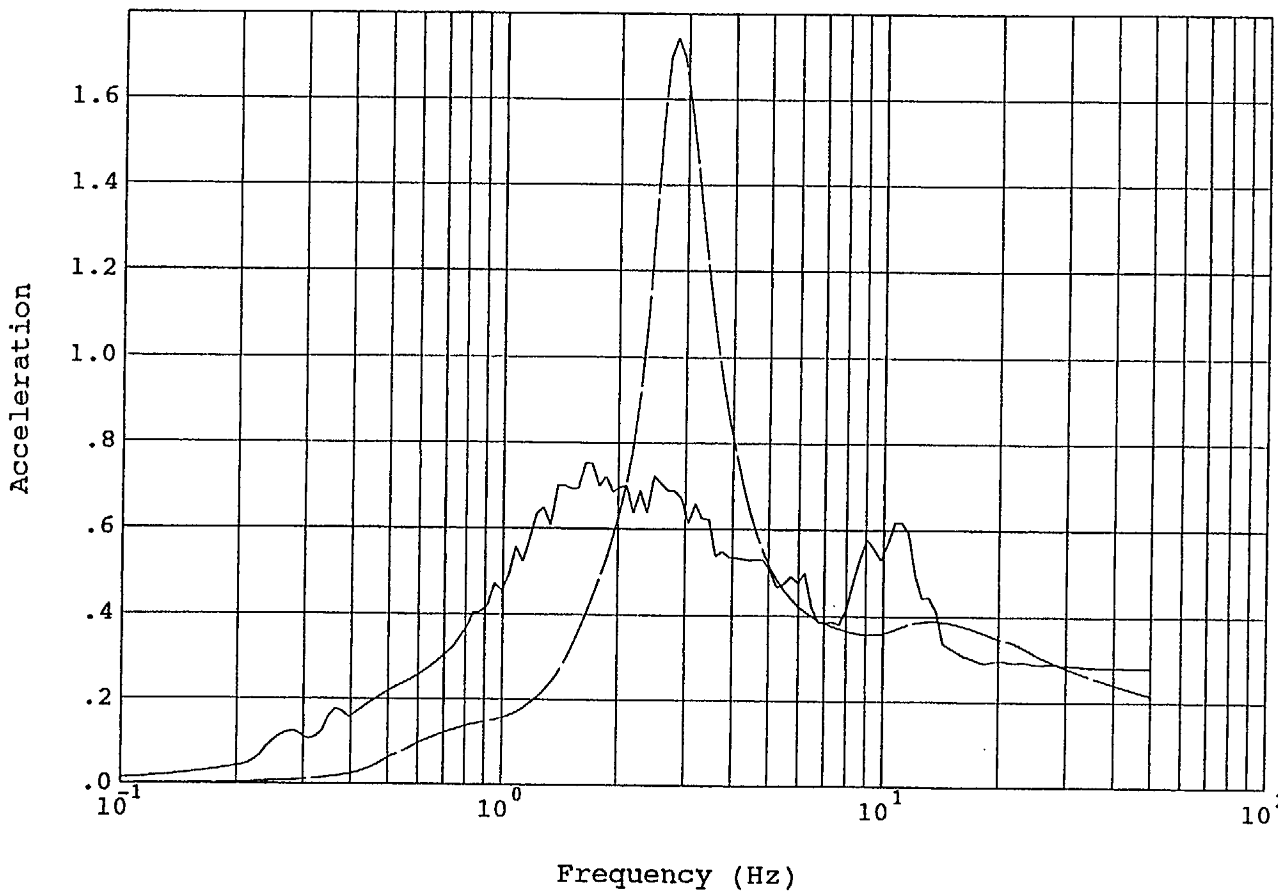
Accelerations in g's

RSPLT INTEL386 VERS 1.2P COMPILED 08/04/93 09:30:31 EXECUTED 12/10/99 09:52:51 ulrs22y.ps

p. 120

250800 C005: Praire Island Nuclear Generating Plant Spectra
Unit 1 Reactor Support Elev. 755 Y (N-S) Direction

Figure 25



Legend:

SMA 0.3g RLE ————————

SSE ——— ———

Notes:

5% Spectral Damping

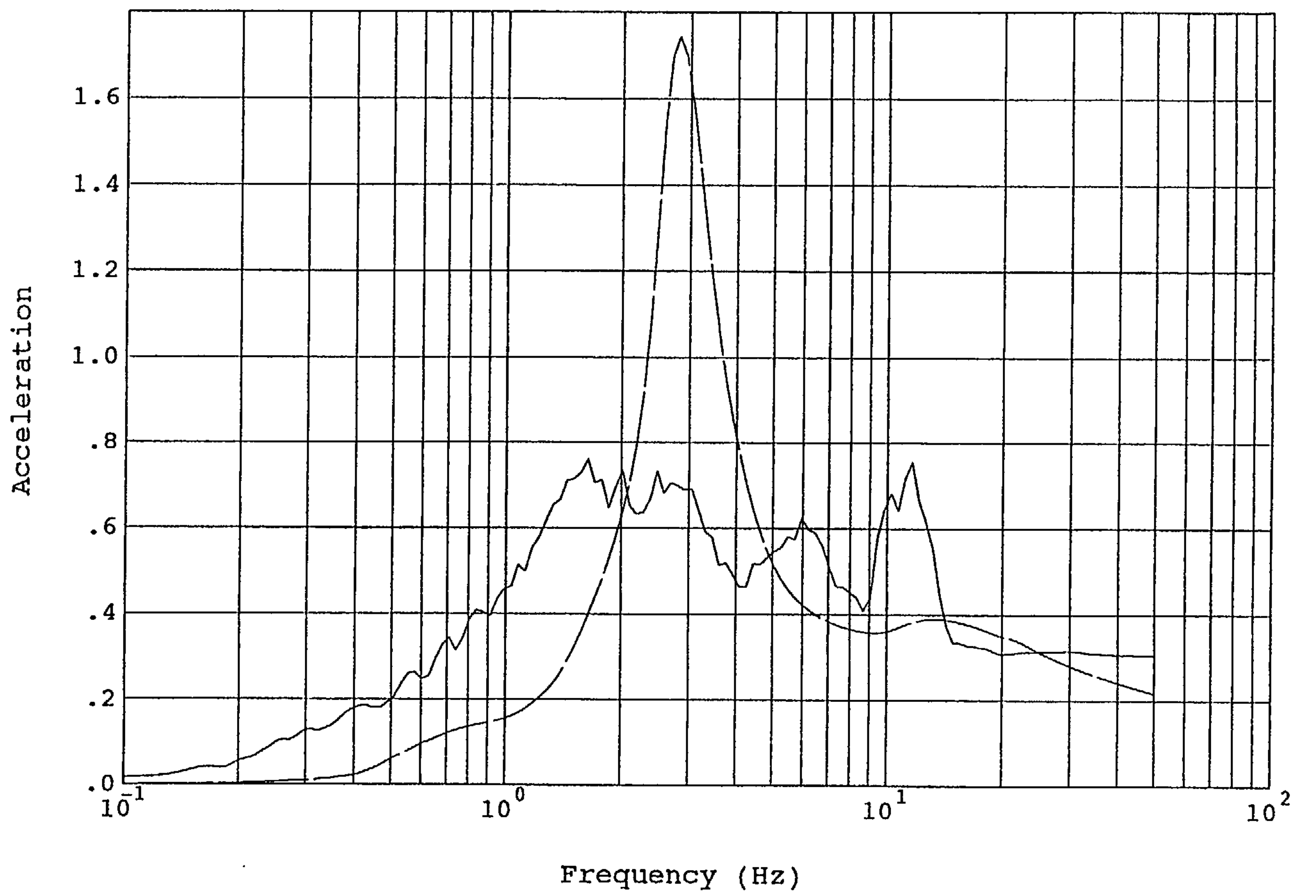
Accelerations in g's

250800 C005: Praire Island Nuclear Generating Plant Spectra
Unit 1 Reactor Support Elev. 711.5 X (E-W) Direction

Figure 26

RSPLT INTEL386 VERS 1.2P COMPILED 08/04/93 09:30:31 EXECUTED 12/10/99 09:52:51 u1rs24x.ps

p. 121



Legend:

SMA 0.3g RLE ————————

SSE ———— ————

Notes:

5% Spectral Damping

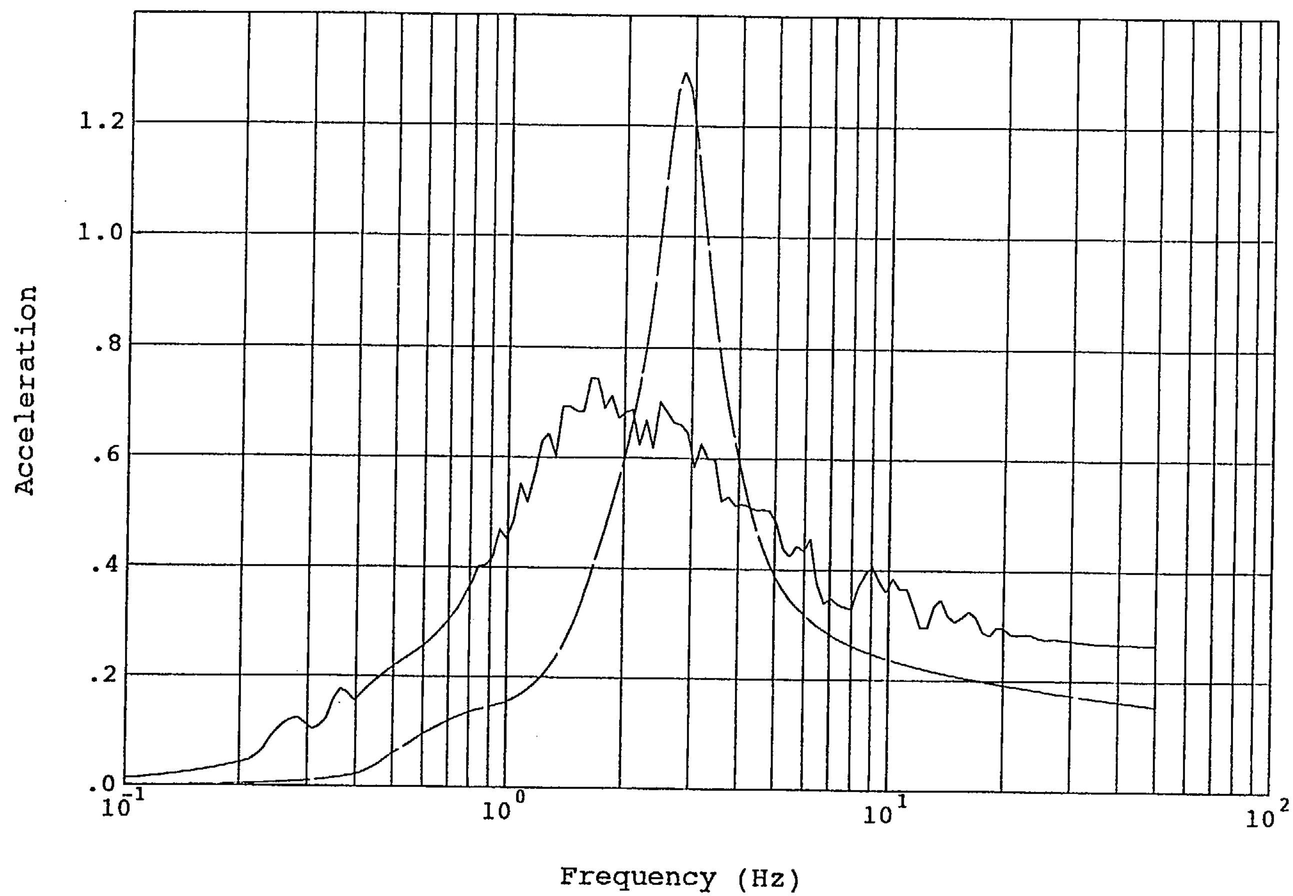
Accelerations in g's

250800 C005: Praire Island Nuclear Generating Plant Spectra
Unit 1 Reactor Support Elev. 711.5 Y (N-S) Direction

Figure 27

RSPLT INTEL386 VERS 1.2P COMPILED 08/04/93 09:30:31 EXECUTED 12/10/99 09:52:51 u1rs24y.ps

p.122



Legend:

SMA 0.3g RLE ————————

SSE — — — — — — —

Notes:

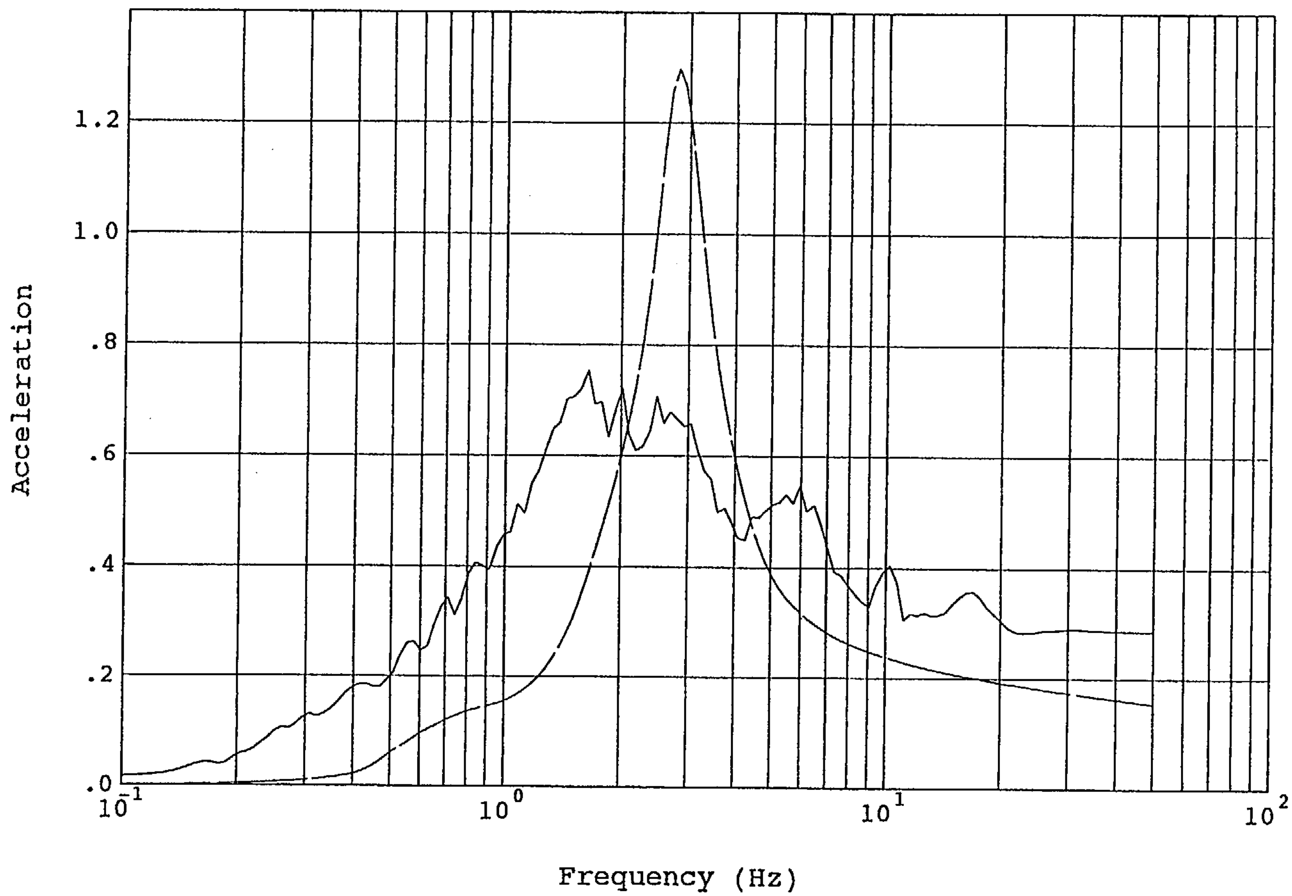
5% Spectral Damping

Accelerations in g's

250800 C005: Praire Island Nuclear Generating Plant Spectra
Unit 1 Reactor Support Elev. 697.5X (E-W) Direction

Figure 28

RSPLT INTEL386 VERS 1.2P COMPILED 08/04/93 09:30:31 EXECUTED 12/10/99 05:49:55 u1_9x.ps



Legend:

SMA 0.3g RLE ————————

SSE —— ——

Notes:

5% Spectral Damping

Accelerations in g's

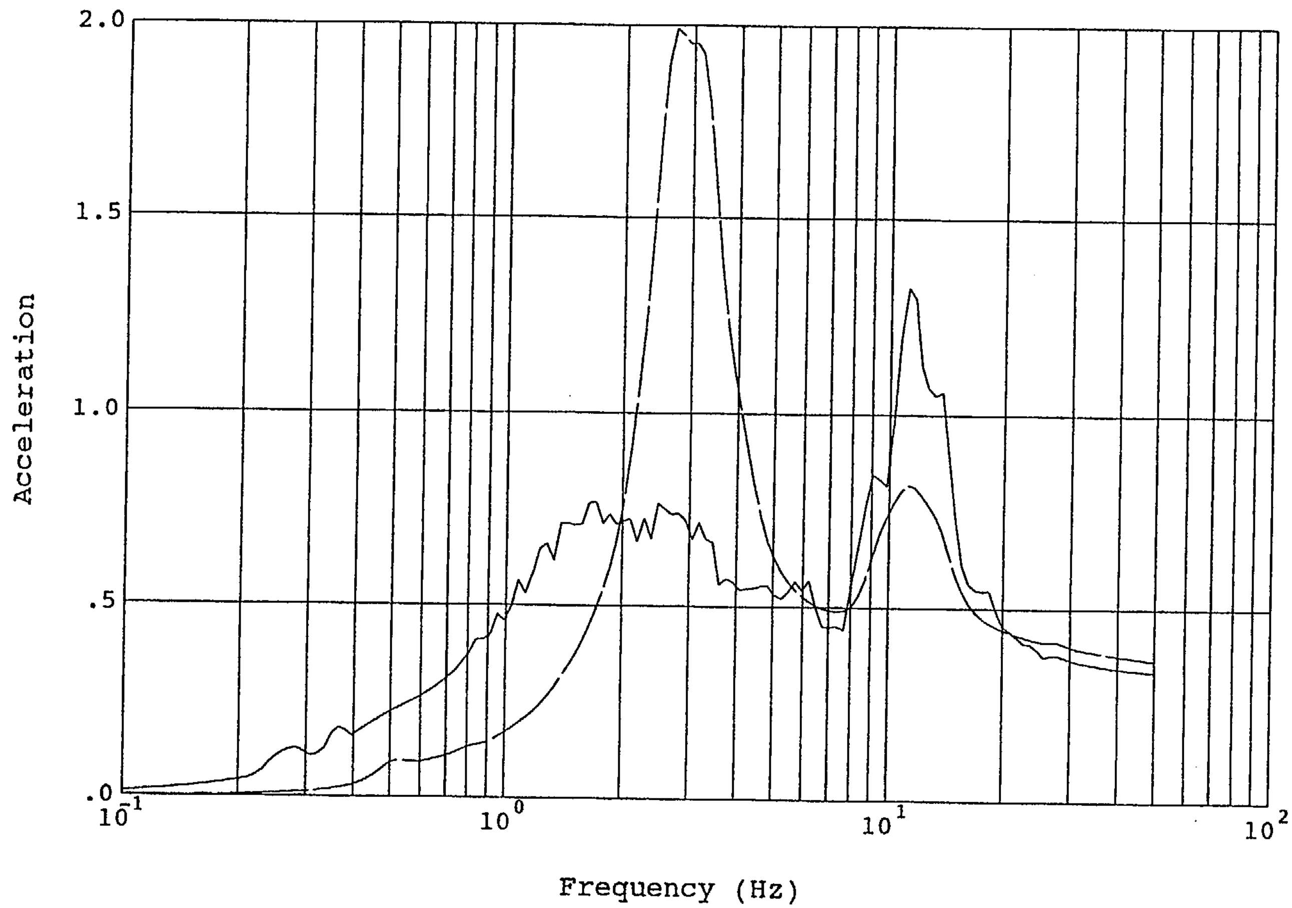
250800 C005: Praire Island Nuclear Generating Plant Spectra
Unit 1 Reactor Support Elev. 697.5Y (N-S) Direction

Figure 29

p.124

RSPLT INTEL386 VERS 1.2P COMPILED 08/04/93 09:30:31 EXECUTED 12/10/99 05:49:55 u1_9y.ps

RSPLT INTEL386 VERS 1.2P COMPILED 08/04/93 09:30:31 EXECUTED 12/10/99 09:52:51 u2rs22x.ps



Legend:

SMA 0.3g RLE ————————

SSE ———— ————

Notes:

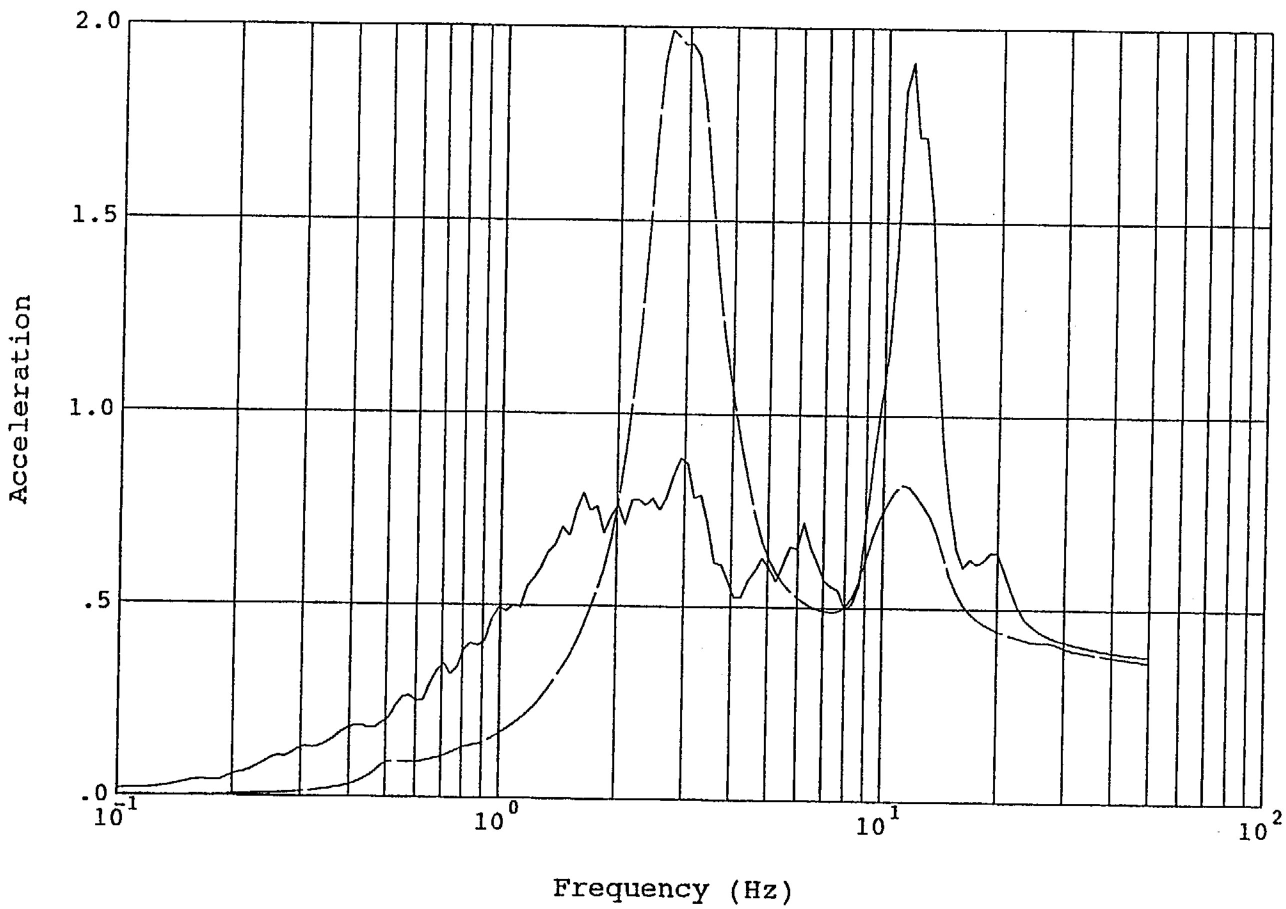
5% Spectral Damping

Accelerations in g's

250800 C005: Praire Island Nuclear Generating Plant Spectra
Unit 2 Reactor Support Elev. 755 X (E-W) Direction

Figure 30

p 125



Legend:

SMA 0.3g RLE ————

SSE — — —

Notes:

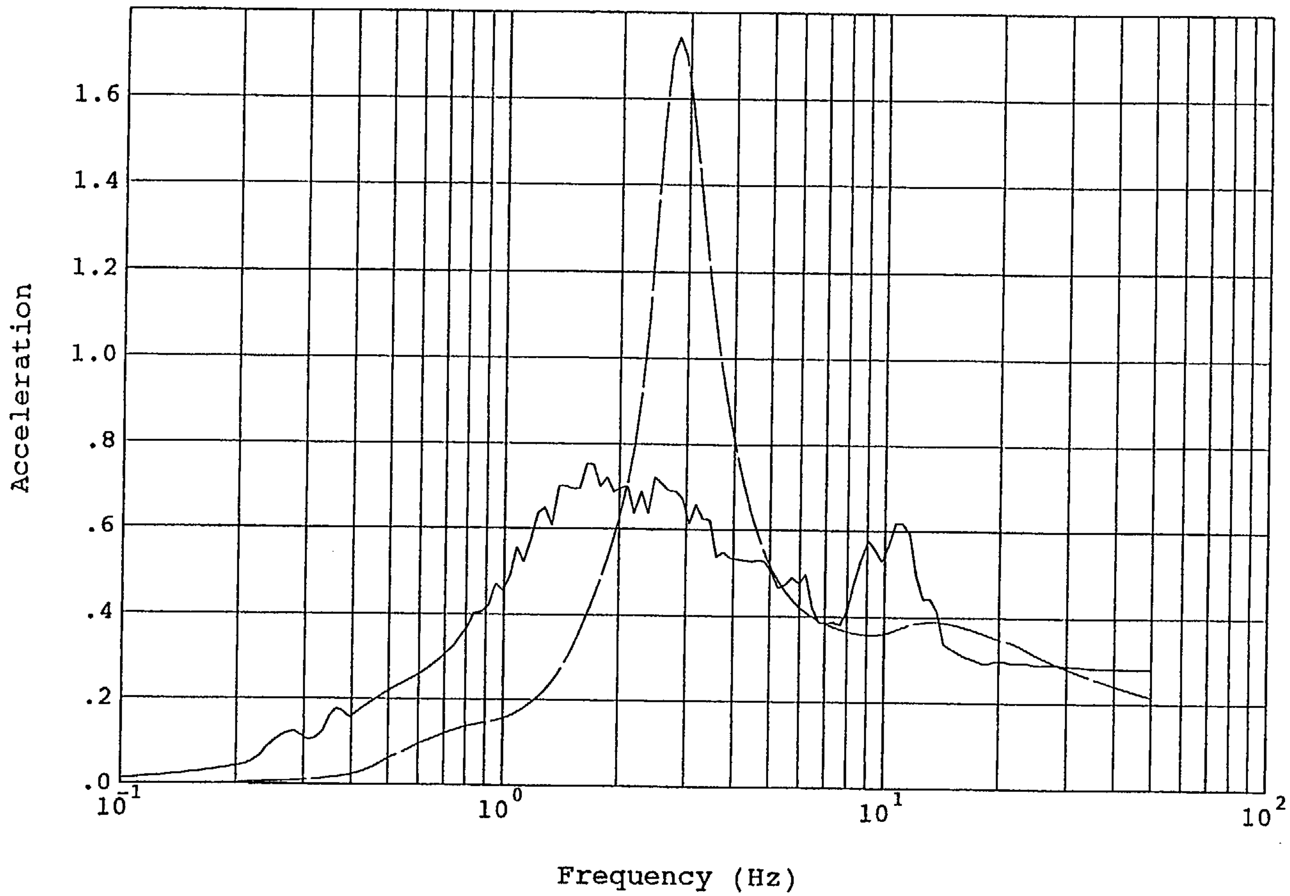
5% Spectral Damping

Accelerations in g's

p. 126

250800 C005: Praire Island Nuclear Generating Plant Spectra
Unit 2 Reactor Support Elev. 755 Y (N-S) Direction

Figure 31



Legend:

SMA 0.3g RLE ————————

SSE — — — — — —

Notes:

5% Spectral Damping

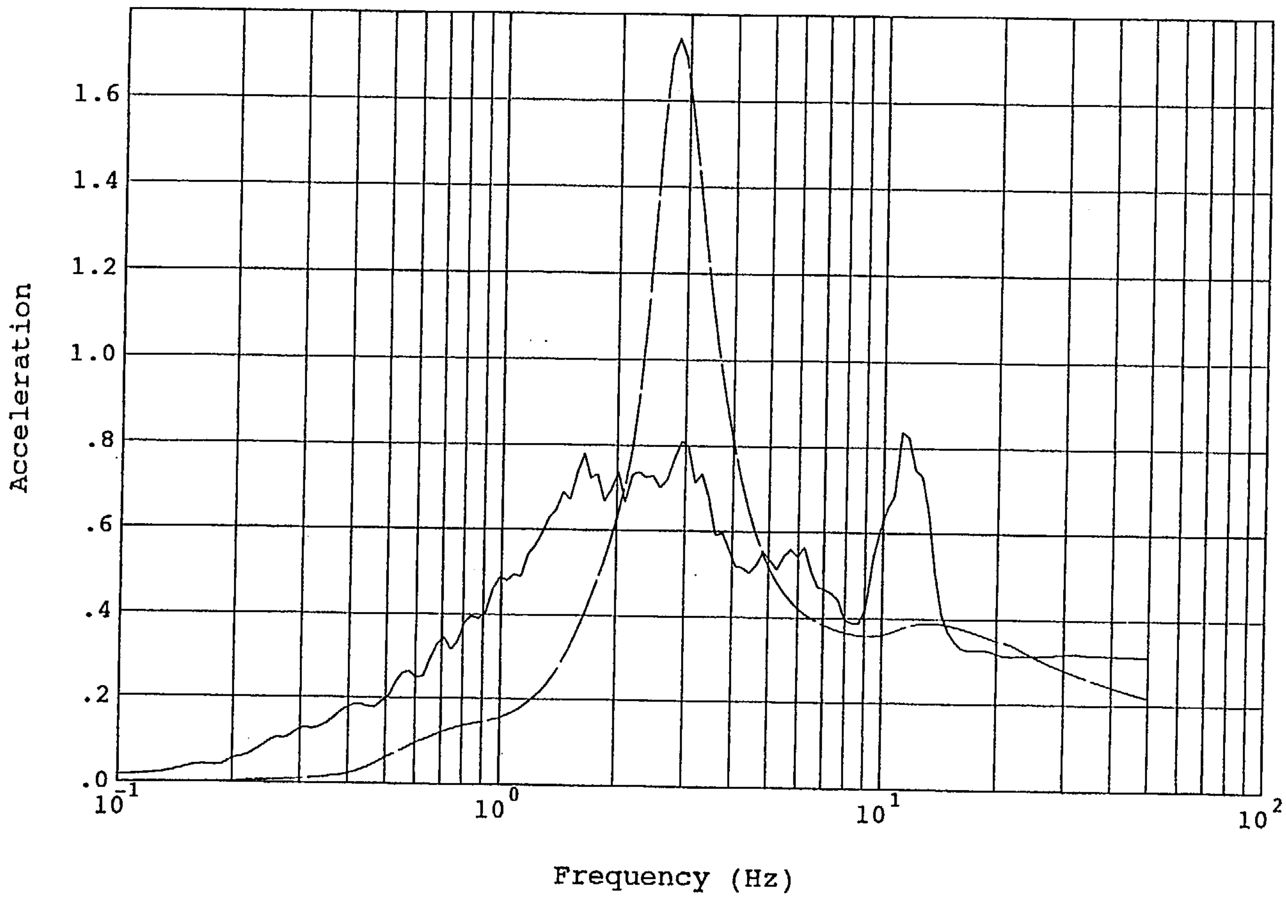
Accelerations in g's

250800 C005: Praire Island Nuclear Generating Plant Spectra
Unit 2 Reactor Support Elev. 711.5 X (E-W) Direction

Figure 32

RSPLT INTEL386 VERS 1.2P COMPILED 08/04/93 09:30:31 EXECUTED 12/10/99 09:52:51 u2rs24x.ps

p.127



Legend:

SMA 0.3g RLE ————————

SSE ———— ————

Notes:

5% Spectral Damping

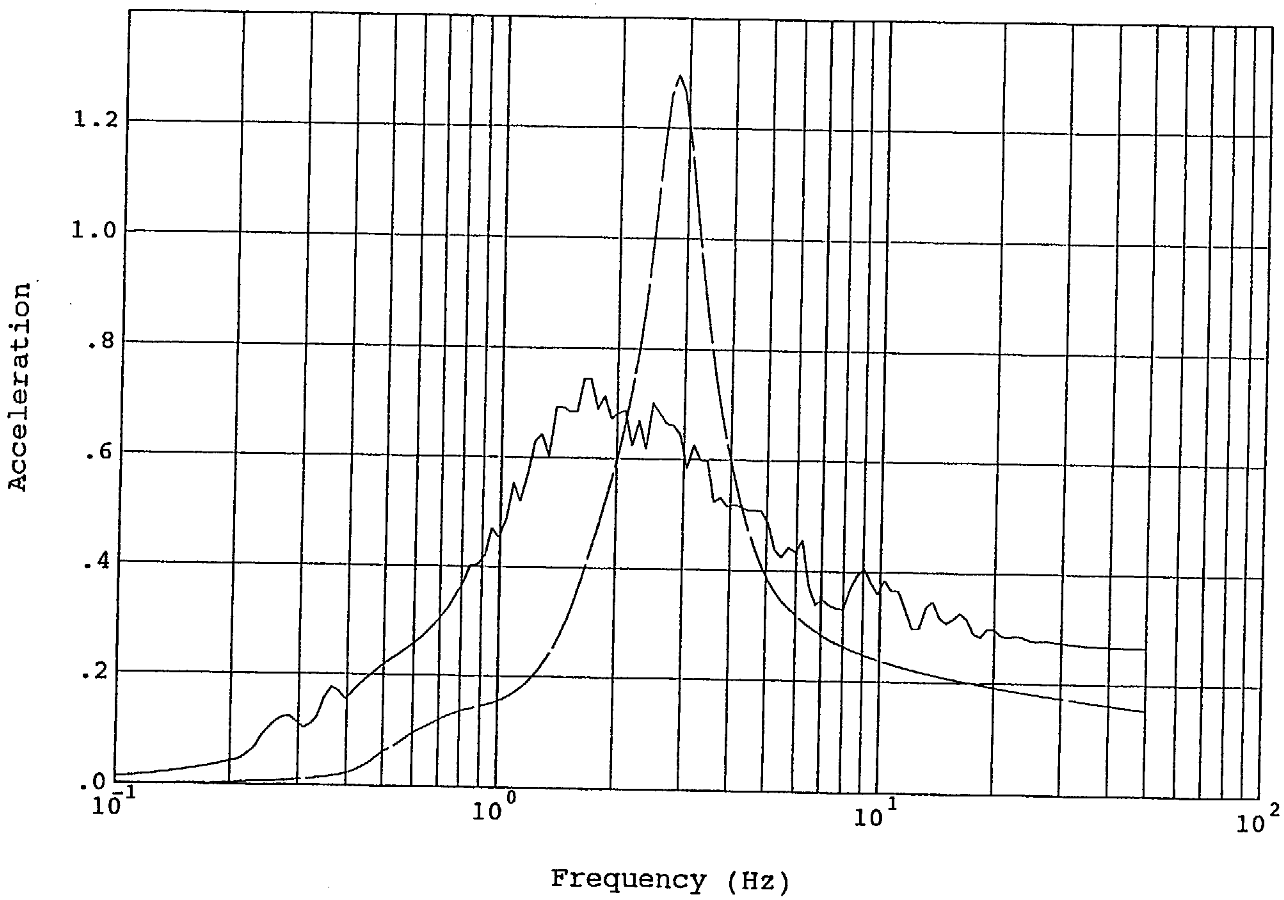
Accelerations in g's

250800 C005: Praire Island Nuclear Generating Plant Spectra
Unit 2 Reactor Support Elev. 711.5 Y (N-S) Direction

Figure 33

RSPLT INTEL386 VERS 1.2P COMPILED 08/04/93 09:30:31 EXECUTED 12/10/99 09:52:51 u2rs24y.ps

p. 128



Legend:

SMA 0.3g RLE ——————

SSE —— ——

Notes:

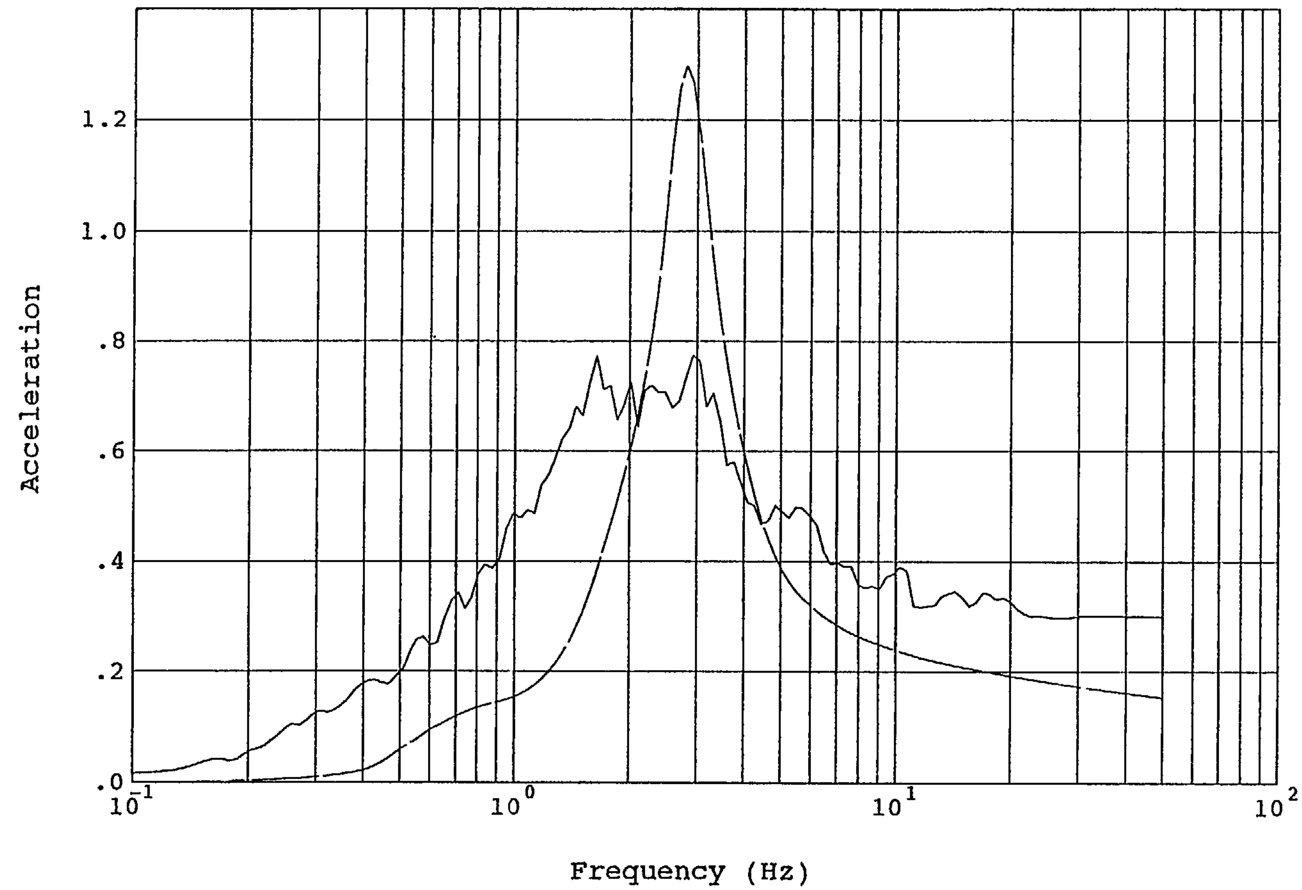
5% Spectral Damping

Accelerations in g's

250800 C005: Praire Island Nuclear Generating Plant Spectra
Unit 2 Reactor Support Elev.697.5X (E-W) Direction

Figure 34

RSPLT INTEL386 VERS 1.2P COMPILED 08/04/93 09:30:31 EXECUTED 12/10/99 05:49:55 u2_9x.ps



Legend:

SMA 0.3g RLE ____________

SSE ______ ______

Notes:

5% Spectral Damping

Accelerations in g's

250800 C005: Praire Island Nuclear Generating Plant Spectra
Unit 2 Reactor Support Elev.697.5 Y (N-S) Direction

Figure 35

RSPLT INTEL386 VERS 1.2P COMPILED 08/04/93 09:30:31 EXECUTED 12/10/99 05:49:55 u2_9y.ps

p. 130

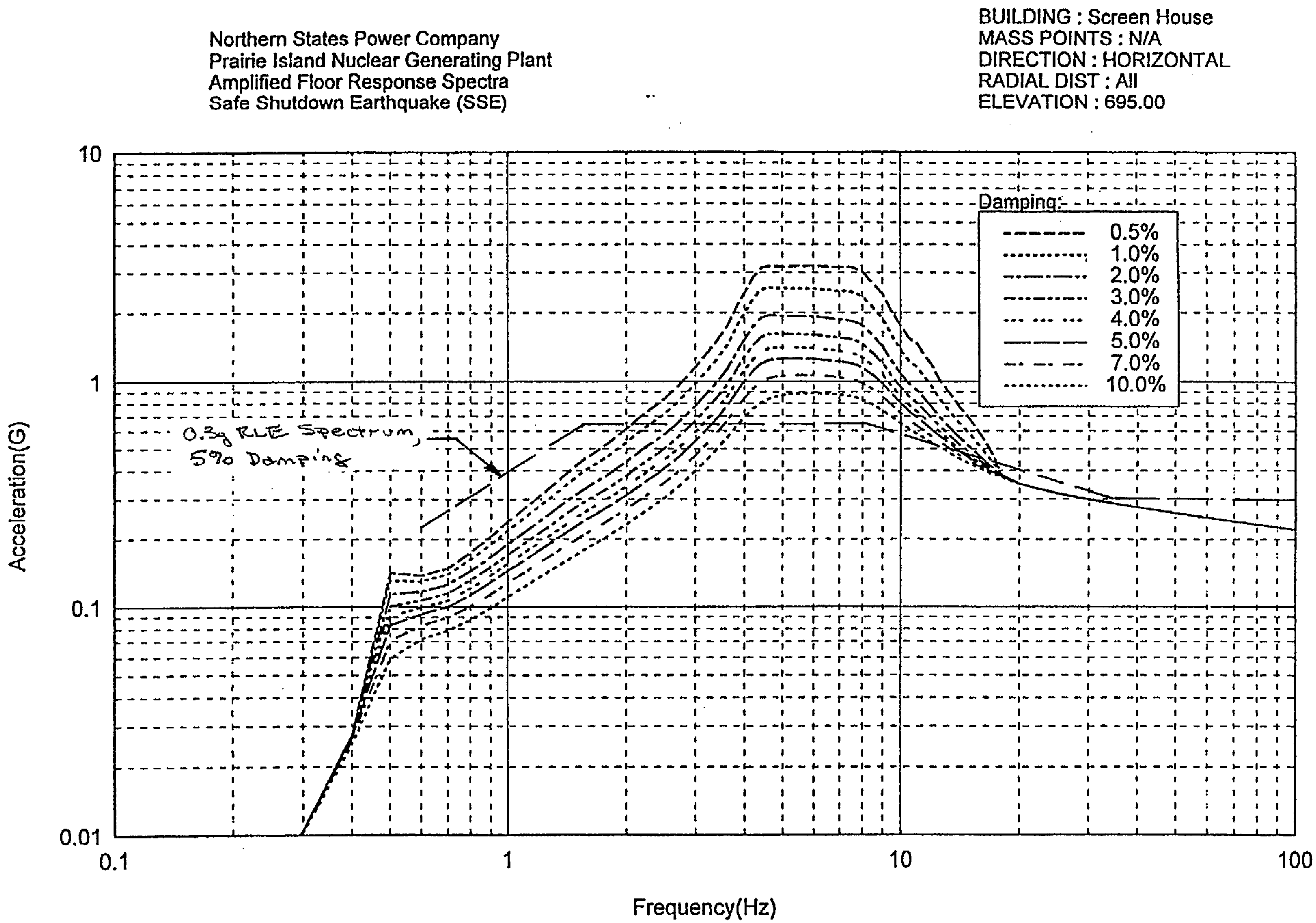

Northern States Power Company
Prairie Island Nuclear Generating Plant
Amplified Floor Response Spectra
Safe Shutdown Earthquake (SSE)
BUILDING : Screen House
MASS POINTS : N/A
DIRECTION : HORIZONTAL
RADIAL DIST : All
ELEVATION : 695.00
Damping:
0.5%
1.0%
2.0%
3.0%
4.0%
5.0%
7.0%
10.0%
0.3g RLE Spectrum, 5% Damping
Acceleration(G)
Frequency(Hz)
10
1
0.1
0.01
0.1
1
10
100

Figure 36