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10 CFR 50.54 (f)

ComEd

March 30, 2000

PSLTR: #00-0068

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Dresden Nuclear Power Station, Units 2 and 3
Facility Operating License Nos. DPR-19 and DPR-25
NRC Docket Nos. 50-237 and 50-249

Subject: Request for Additional Information Regarding Individual Plant Examination of External Events

Reference:

(1) Letter from J. M. Heffley (ComEd) to U. S. NRC, "Request for Additional Information Regarding Individual Plant Examination of External Events," dated September 30, 1999

(2) Letter from J. M. Heffley (ComEd) to U. S. NRC, "Request for Additional Information Regarding Individual Plant Examination of External Events," dated February 15, 1999

(3) Letter from U.S. NRC to O. D. Kingsley (ComEd), "Request for Additional Information Regarding the Individual Plant Examination of External Events (IPEEE) for Dresden Nuclear Power Station, Units 2 and 3 (TAC NOS. M83616 and M83617)," dated December 14, 1998

(4) Letter from J. M. Heffley (ComEd) to U. S. NRC, "Final Report – Individual Plant Examination of External Events (IPEEE) Generic Letter 88-20, Supplement 4," dated December 30, 1997

The purpose of this letter is to provide the Commonwealth Edison (ComEd) Company, Dresden Nuclear Power Station response to the Request for Additional Information regarding our submittal of the Individual Plant Examination of External Events (IPEEE). Reference 3 requested that we provide additional information to the NRC regarding four (4) seismic and twelve (12) fire questions regarding our submittal in Reference 4.

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In References 1 and 2, we stated that we would provide our answers for both the fire and seismic issues concurrent with our submittal of a new Fire Risk Model. In Reference 2, we also stated that we would provide a schedule for completing the work associated with the seismic issues. This information was provided to the NRC in Reference 1.

The revised fire risk models resulted in a lower Core Damage Frequency (CDF) value than previously reported for each unit due to the application of less conservatism in the models. The upgraded fire analysis produced significantly different results as compared to the original Fire IPEEE. The much lower calculated CDF contribution due to postulated fire events and the distribution of the risk contributors amongst the fire compartments evidence these differences. A review of the upgraded analysis results provides risk information that is considered to be a much more accurate characterization of the risk from fire at Dresden station. Summary CDF values are shown:

Unit	Previous CDF/Rxyr	Current CDF/Rxyr
2	2.5E-04	1.69E-05
3	2.8E-04	3.08E-05

Attachment 1 to this letter contains the revised portions of the IPEEE. Attachment 2 contains our responses to the four (4) seismic questions and Attachment 3 contains the responses to the twelve (12) fire related questions. Fire Question number 9 in Attachment 3 regarding control and instrumentation functions requires additional evaluation to completely formulate the response. Upon its completion, this information will be provided under a separate submittal currently scheduled for July 31, 2000. Attachment 4 contains the SQUG/IPEEE relay screening and evaluation tabulation. Attachment 5 contains the ground and in-structure spectra utilized for the IPEEE. Attachment 6 contains the Safe Shutdown Earthquake (SSE) design basis ground and in-structure response spectra.

Should you have any additional questions regarding this letter, please contact Mr. Dale Ambler, Regulatory Assurance Manager, at (815) 942-2920 extension 3800.

Respectfully,

For Preston Swafford
Russell Fisher ([illegible])

Preston Swafford
Site Vice President
Dresden Nuclear Power Station

ATTACHMENT 1	Revision to the Individual Plant Examination of External Events for Dresden Nuclear Power Station Units 2 and 3
ATTACHMENT 2	Response to Seismic Questions
ATTACHMENT 3	Response to Fire Questions
ATTACHMENT 4	SQUG/IPEEE Relay Screening and Evaluation Tabulation
ATTACHMENT 5	Ground and In-structure Spectra for Dresden Nuclear Power Station Units 2 and 3
ATTACHMENT 6	Safe Shutdown Earthquake (SSE) Design Basis Ground and In-structure Response Spectra for Dresden Nuclear Power Station Units 2 and 3

cc: Regional Administrator, Region III
NRC Senior Resident Inspector, Dresden Nuclear Power Station

Attachment 1

Revision to the Individual Plant Examination of External Events for Dresden Nuclear Power Station Units 2 and 3

- Section 1.0, EXECUTIVE SUMMARY
- Section 3.0, SEISMIC EVENT ASSESSMENT
- Section 4.0, INTERNAL FIRE EVALUATION

1.0 EXECUTIVE SUMMARY

1.1 Background and Objectives

The NRC issued its policy on Severe Reactor Accidents Regarding Future Designs and Existing Plants in 1985, which concluded that existing nuclear power plants pose no undue risk to the public health and safety and that there is no present basis for immediate action on any regulatory requirements for these plants. However, the Commission recognized, based on NRC and industry experience with plant specific probabilistic risk assessments (PRAs), that systematic examinations are beneficial in identifying plant specific vulnerabilities to severe accidents.

As part of the closure process for the Severe Accident Program, the NRC issued Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities 10CFR50.54(f)", [1.1] on November 23, 1988, formally requesting that each licensee conduct an Individual Plant Examination (IPE) for internally initiated events, including internal flooding. Dresden Station formally submitted the IPE report to the NRC in January, 1993 and submitted a revised IPE report in June, 1996 [1.3].

On June 28, 1991, the NRC issued Generic Letter 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10CFR50.54(f)" [1.2], which first introduced the requirements for external event phenomena. The objectives of the IPE and the IPEEE are similar:

1) to develop an appreciation of severe accident behavior;
2) to understand the most likely severe accident sequences that could occur under full power operations;
3) to gain a qualitative understanding of the overall likelihood of core damage and fission product releases; and
4) if necessary, reduce the overall likelihood of core damage and fission product releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

This report updates the Dresden Station IPEEE report that was formally transmitted to the NRC in December 1997. This report meets the objectives and requirements of Generic Letter 88-20, Supplement 4 [1.2], and provides a summary of the methodologies, results and conclusions of the Dresden IPEEE.

1.2 Plant Familiarization

Dresden Units 2 and 3 are similar generating units which include two boiling water reactor (BWR) nuclear steam supply systems (NSSS) and turbine-generators furnished by General Electric Company (GE).

Each nuclear steam supply system is designed for a power output of 2,527 MWt which is the license application rating. The equivalent approximate design net electrical output of each unit is 809 MWe.

Dresden Station is located in northeastern Illinois, specifically in the northeast quarter of the Morris quadrangle (as designated by the United States Geological Survey) near the town of Morris in the county of Grundy (Goose Lake Township). The north and east boundaries are formed by the Illinois and Kankakee Rivers. The size of the site is 953 acres plus a 1275-acre cooling lake. The lake, which was formed by constructing an impervious earth-fill dike, is connected to the intake and discharge flumes of Units 2 and 3 by two canals (one intake and one discharge); each canal is about 11,000 feet long. Units 2 and 3 were completed and went into commercial service in June 1970 and November, 1971, respectively.

1.3 Overall Methodology

ComEd followed the guidance for performing the IPEEE provided in NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities" [1.4].

The seismic IPEEE is a Seismic Margins Assessment (SMA) based on the guidance provided in the EPRI NP-6041 "Seismic Margins Assessment Methodology" [1.6].

The fire risk evaluation was performed using the EPRI Fire-Induced Vulnerability Evaluation (FIVE) methodology [1.5]. Additionally, the EPRI Fire PRA Implementation Guide [1.7] was used to augment the FIVE methodology.

The Dresden IPEEE utilized, wherever possible, information from the Dresden Updated Final Safety Analysis Report (UFSAR) and IPE effort [1.3]. This included information on the as-built, as-operated plant. Additional walkdowns were performed and documented to provide additional information required for completion of the seismic, fire and other external events analyses.

The methodology for identifying other external events is consistent with the approach described in NUREG-1407 [1.4] and is based on a thorough review of the UFSAR.

Each of the analyses (seismic, fire and other external events) received detailed reviews by plant staff and consultants. An independent external review was performed for the SMA including a review of the walkdowns and the evaluations. An independent external review was also performed for the internal fire assessment. Detailed reviews of the fire PRA modeling techniques, assumptions and data were also performed. An internal independent review was performed for the assessment of other external events.

Details of the specific methodologies are contained in Section 2.3 and the seismic, fire and other external events sections of this report.

1.4 Summary of Major Findings

This section summarizes the results and conclusions of the Dresden IPEEE. Details of the results and conclusions are presented in Sections 3, 4, 5, 7 and 8.

1.4.1 Seismic Study Summary

There were no significant seismic concerns identified as a result of the Seismic Margins Assessment (SMA). As a general observation, some electrical equipment anchorages have limited anchorage margin. This condition was also noted during the Systematic Evaluation Program (SEP) and led to Information Notice 80-21 that identified marginally anchored (or unanchored) equipment assemblies in older plants. Many of the equipment anchorages at Dresden were modified (improved) during the SEP. The IPEEE Seismic Margin effort identified other equipment not reviewed or considered in the SEP as having limited seismic anchorage capacity. That equipment has already been improved (anchorage upgrades completed) or is scheduled for design improvements.

The following items were found to have a seismic capacity of less than 0.30g peak ground acceleration (PGA):

Item No.	Capacity (PGA)	Description	Notes
1	0.202g	Buses - D03-8303B-----M05, D02-8302B----M05, , D03-8303A---M05, and Distribution Panel D03-83125---P06	Anchorage controls capacity.
2	0.27g	Distribution Panel D02-83125----P06 and Bus D02-8302A---M05	Anchorage controls capacity.
3	0.20g	Condensate Storage Tanks - D00-3303-A----T05, D00-3303-B----T05	Tank buckling controls capacity.
4	0.26g	Diesel Fuel Oil Storage Day Tank D00-5202-T05	Block wall controls capacity.
5	0.27g	Battery Charger - D02-8300-2A---B05	Anchorage controls capacity.
6	0.27g	Distribution Panels - D02-9802-A & B---P06	Anchorage controls capacity.
7	0.27g	Switchgear - D02-7328---S35 & D02-7329----S35	Anchorage controls capacity.
8	0.27g	Bus #2A-1 - D02-8302A1----P06	Anchorage controls capacity.
9	0.27g	125V DC/TB Battery Bus #2 D02-83125-2-P06	Anchorage controls capacity.
10	0.27g	125V DC/Battery Charger #2 D02-8300-2---B05	Anchorage controls capacity.
11	0.28g	125V DC Battery Charger - D03-8300-3A---B05	Anchorage controls capacity.

12	0.28g	Unit 2&3 Torus Suppression Chambers	Torus shell stress controls capacity.
13	0.29g	Motor Control Centers D02-83250-----M05 & D02-7826-4---M05	Anchorage controls capacity.
14	0.29g	Bus #2B-1 - D02-8302B-1---P06	Anchorage controls capacity.
15	0.29g	125V DC/TB Res Bus #2 D02-83125-1---P06	Anchorage controls capacity.

In December 1997, Items 1 and 2 were previously found to have a capacity less than 0.2g PGA. The new capacities shown above are based on additional evaluations that included any proposed modifications. Items that were found to have a capacity greater than or equal to 0.3g PGA based on these evaluations have been removed from the above table.

The above items meet or exceed the design basis requirement of 0.2g PGA, thereby meeting Dresden's intention to ensure that all IPEEE components have a seismic capacity that complies with design basis requirements [1.8]. Items 1 and 3 are the most limiting conditions and are controlled by anchorage capacity or tank buckling. However, based on experiences with actual industrial facilities in moderate to severe earthquakes, it is concluded that the Dresden plant possesses reasonable margin with respect to its design basis earthquake, and safe shutdown capability will not be lost.

As a result of this study, no programmatic issues or weak links were identified among buildings and distribution systems that include piping, cable trays, or relays. Issues that have been identified are associated with limited capacities of some equipment anchorages, seismic interaction, and in one instance, tank buckling (Condensate Storage Tanks). No single class of equipment was singled out as being particularly worse than another with respect to base anchorage.

1.4.2 Internal Fires

The following is a summary of the fire-induced core damage results for Dresden Units 2 and 3. These results include analysis findings from fire modeling of single fire compartments and the Control Room analysis. Since the multi-compartment analysis screened all scenarios, there is no multi-compartment contribution to the CDF presented below.

The Core Damage Frequencies (CDFs) resulting from these analyses for Dresden Station are:

Unit 2 CDF/Rxyr	Unit 3 CDF/Rxyr
1.69E-05	2.97E-05

The upgraded fire analysis produced significantly different results as compared to the original Fire IPEEE. These differences are evidenced by the much lower calculated CDF contribution due to postulated fire events and the distribution of the risk contributors amongst the fire compartments. A review of the upgraded analysis results provides risk insights that are considered to be a much more accurate characterization of the Dresden station.

Figure 1-1 shows the distribution of these results among the plant fire areas for each unit. Fire area TB-V includes the Main Control Room and Auxiliary Electric Equipment Room, and fire area TB-I encompasses the Unit 2 Turbine Building. Fire area TB-III emcompasses the Unit 3 Turbine Building, including the Cable Tunnel.

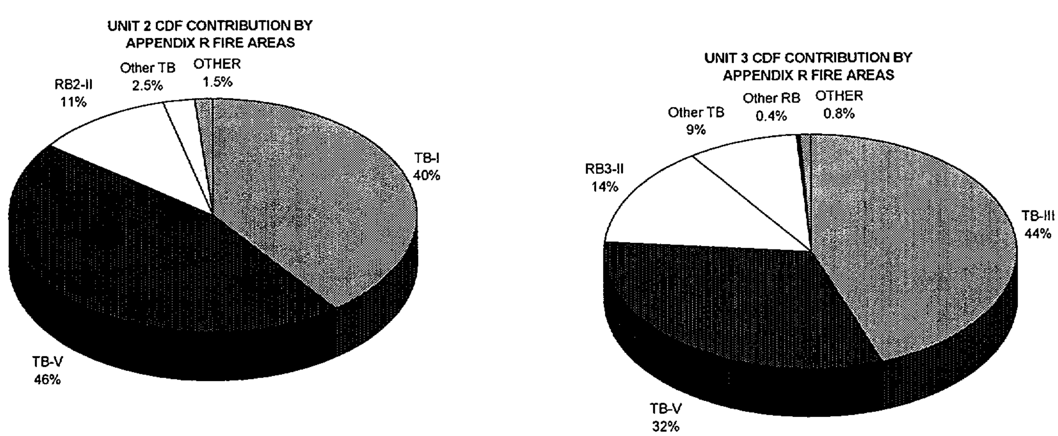

Figure 1-1. Distribution of CDF by Plant Location

Insights

The upgraded analysis highlighted the following nine insights related to fire risk:

1. The calculated CDF contribution due to postulated fire events is consistent with other Boiling Water Reactor (BWR) plants.
2. A large oil fire involving Unit 2 Reactor Feedwater Pump C or a fire involving MCC 26-1 is a dominant contributor to the Unit 2 CDF. This is because of the location of the cables needed for the Unit 2 DC power system. The Unit 3 DC power feed to one train of the Unit 2 DC system as well as the Unit 2 AC power cable to the battery charger for the redundant DC train are exposed to a common hazard. Although the circuits are located in separate trays, they are stacked vertically. The occurrence of a postulated large fire event requires an operator action to either align the spare battery charger or to connect the spare Unit 2 battery bank.
3. For both Unit 2 and Unit 3, four of the top ten scenarios involve a loss of the Isolation Condenser due to loss of DC power to the valves. These scenarios contribute approximately one-third to the CDF of each unit. Although this failure is recoverable through local manual opening the valves, this action is not proceduralized in the EOPs. Therefore, no credit is taken for this action in the upgraded analysis. If this action were to be credited, however, the overall Unit 2 and Unit 3 CDF values would be reduced by approximately a third.
4. Excluding the control room severe fire, the dominant core damage sequence is loss of decay heat removal. Opportunities to recover decay heat removal functions will have significant risk reduction potential.
5. Postulated fires in the Main Control Room represents the largest risk contributor. The bounding Main Control Room fire event which forces abandonment is the single largest risk contributor. While the risk importance of the Main Control Room is consistent with other plant IPEEE studies, the significant fraction of the risk contribution by this area is not typical. Further review of the analysis results indicate that the large fraction is not the result of an unusually high CDF contribution, but instead the somewhat lower CDF contribution by other postulated fire events. A review of the analysis results shows that the Isolation Condenser and its ability to support decay heat removal needs represents an additional level in defense in depth that is not available at other types of BWR plants.
6. The configuration of the ADS system at Dresden is such that there are unprotected (no fire wrapping) circuits located outside of the auxiliary electric equipment and main control rooms whose fire induced failure could cause the spurious actuation of ADS valves. The spurious opening of the ADS valve(s) for this postulated case would not be precluded by the ADS inhibit switch. However, these cables are routed in such a fashion that they are not exposed to any significant external fire threat and were not a dominant risk contributor.
7. The most risk significant control room fire scenario which did not require the control room to be abandoned involves a postulated fire in panel 902-8/903-8. Such a fire results in a loss of offsite power and Division II of the onsite AC power distribution system (Buses 24/24-1, 34/34-1).

8. The lower damage threshold for non-IEEE 383 qualified cables limited the effectiveness of installed automatic fire suppression systems. Although a specific sensitivity study was not performed, it is expected that the results of the dominant risk contributor (Reactor Feed Pump oil fire) would be reduced if IEEE 383 qualified cables had been installed.

9. The fire risk analysis for Units 2 and 3 identified three specific asymmetries in the design which had a risk impact and contributed to Unit 3 having a higher CDF than Unit 2. The routing of the DC power circuits in the Unit 2 RFP area resulted in a dominant risk contributor. The design of the equivalent circuits in Unit 3 were such that the redundant DC circuits were not co-located. However, the one DC circuit which was routed through the Unit 3 RFP area is exposed to fires originating at MCC 36-1, the air compressors, or any of the three RFPs. While the CCDP for these scenarios for Unit 3 is lower than for Unit 2, the Unit 3 configuration requires the use of a fire ignition frequency that is higher by more than a factor of 3.

   The location of a DC distribution panel in Unit 3 is located such that a postulated fire could affect circuits in an overhead cable tray. The equivalent bus in Unit 2 is located in a separate subcompartment.

   The routing of the cables in Unit 3 to the Main Control Room uses an underground cable tunnel. No such feature exists for the Unit 2 cable routing. As such, the tunnel effectively represents an extension of the auxiliary electric equipment room and Main Control Room with respect to potential fire risk consequences.

1.4.3 Other External Events Summary

There were no other external events identified that have any significant impact on the core damage frequency at Dresden. Guidance from NUREG-1407 [1.4] and Generic Letter 88-20, Supplement 4 [1.2] as well as NRC SEP evaluations were used in screening initiators.

1.5 References

1.1 NRC Generic Letter 88-20, Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR 50.54(f), November 1988.

1.2 NRC Generic Letter 88-20, Supplement 4, Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f), June 1991.

1.3 Dresden Nuclear Generating Station Units 2 and 3 Individual Plant Examination Submittal Report, Revision 1, June 1996

1.4 NUREG-1407, Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities, June 1991.

1.5 Fire-Induced Vulnerability Evaluation (FIVE) Methodology Plant Screening Guide, Professional Loss Control, EPRI TR-100370, April 1992

1.6 Jack R. Benjamin & Associates, et. al., A Methodology for Assessment of Nuclear Power Plant Seismic Margin (Rev 1), EPRI NP-6041-SL, August 1991.

1.7 J. Parkinson, et. al., Fire PRA Implementation Guide, EPRI TR-105928, December 1995.

1.8 ComEd Letter JMHLTR #99-0016, from J.M. Heffley (ComEd) to USNRC, “Request for Additional Information Regarding Individual Plant Examination of External Events,” dated February 15, 1999.

3.0 SEISMIC EVENT ASSESSMENT

3.1 Methodology

3.1.1 Overall Approach

NUREG-1407 [3.1] specifies that Dresden is a 0.3g focused scope plant. As such, the objective of the seismic margin assessment (SMA) is to assign a seismic capacity to each plant component in terms of the peak ground acceleration (PGA) and to compare the capacity to the seismic margin earthquake (SME). The assessed components include the structures, equipment, and distribution systems identified through the systems analysis presented in Section 3.2.

First, a component was evaluated based on the screening criteria presented in NP-6041 [3.5]. Thus, a component was assigned a seismic capacity based on three seismic levels: 0.8g, 0.8g to 1.2g, or >1.2g, expressed in terms of 5% damped peak *spectral* acceleration. As is common practice, seismic levels were converted to peak *ground* accelerations: 0.3g, 0.3g to 0.5g, or >0.5g. Per NUREG-1407 [3.1], Section 3.2.2 and Table 3.1, the Seismic Margin Earthquake (SME) for Dresden is a NUREG/CR-0098 [3.2] median rock or soil spectrum anchored at 0.3g.

Components were not assigned a PGA greater than 0.5g, i.e., the NP-6041 criteria [3.5] for the third earthquake level was not applied. If a component met the requirements for the second earthquake level, it was assigned a capacity of 0.5g. If it could only meet the requirements for the first level, it was assigned a capacity of 0.3g. If a component could not meet the requirements for the first level, a capacity was calculated.

The evaluation of major structures was based primarily on a review of the design bases, augmented by a walkdown to identify any anomalous conditions. Seismic capacities were calculated for masonry block walls, either by scaling existing IEB 80-11 calculations [3.9], or by specific calculation [3.10].

The evaluation of mechanical and electrical equipment relied heavily on the USI A-46 walk downs [3.7]. Equipment that met A-46 requirements (i.e. was not an outlier) was assigned an *equipment* seismic capacity of 0.3g or 0.5g, depending on the criteria in NP-6041 [3.5]. If the equipment was an A-46 outlier, a seismic capacity was calculated. If the equipment had anchorage that was not judged robust by the walkdown team, the A-46 anchorage evaluation was scaled to obtain an *anchorage* seismic capacity. During the walk downs, any masonry block walls adjacent to the equipment were noted. The final seismic capacity assigned to the equipment was the minimum of the equipment capacity, the anchorage capacity, and the capacity of any adjacent block walls.

IPEEE equipment which was not A-46 equipment was evaluated identically to A-46 equipment.

Distribution systems included piping, electrical raceways, and ductwork. The seismic capacity of the raceways was based on the A-46 raceway evaluations. Piping and ductwork was evaluated based on a review of the design bases, augmented by walkdowns.

Dresden Station previously submitted the Seismic IPEEE results to the NRC in December, 1997. The report identified a number of components that were found to have a seismic capacity less than 0.3g PGA. At the time of the original submittal, items with capacities less than 0.2g were considered potential design basis issues. New capacities for such items were determined based on additional evaluations that included proposed modifications, if applicable. The new capacities are included in this report.

3.1.2 Seismic Review Team

The Seismic Review Team for this effort was essentially the same as the Unresolved Safety Issue A-46 Seismic Review Team.

The station walkdowns were conducted from June 12-16, August 23-24, September 6-8, and October 2-4, 1995; and February 19-21, March 4-8, April 11-12, 1996, and November 10-11, 1997. The seismic capability engineers for the Dresden walkdown were Dr. J. D. Stevenson, G. G. Thomas, S. Anagnostis, P. A .Gazda and W. Djordjevic of S&A, Dr. R. P. Kennedy of RPK Structural Mechanics, Inc. and Messrs. B. M. Lory, R. Janowiak, T. Loch, and F. Polak of ComEd. All have been Seismic Qualification Utility Group (SQUG) trained and certified, and the majority have also had the EPRI IPEEE Add-On training. Their resumes, and SQUG and IPEEE Add-on training Walkdown Course Completion Certificates, are provided in Appendix A.

An independent evaluation and peer review of the walkdown process was performed by Mr. Harry Johnson of Programmatic Solutions during April 11-12, 1996. The review included an assessment of the walkdown and analyses by audit and sampling to identify any gross errors. Mr. Johnson personally conducted two days of walkdowns to ascertain completeness and correctness of the IPEEE walkdowns. His review included comparing completed SEWS with equipment previously inspected by the SRTs. Mr. Johnson also reviewed the documentation packages the SRTs used to determine equipment design details that could not be readily determined by walkdown. Mr. Johnson concluded that the IPEEE walkdowns were being conducted competently and the findings made were appropriate. Appendix B provides documentation of Mr. Johnson's peer review.

3.1.3 Plant Seismic Design Basis

3.1.3.1 Description of Input Motions

The input motions used to create the seismic design of Dresden are based on the Housner-type Ground Response Spectrum (GRS) and the north-south component earthquake record of El Centro of May 18, 1940. The Dresden design basis Safe Shutdown Earthquake (SSE) ground spectra are smoothed Housner-type spectra. The design basis In-Structure Response Spectra (ISRS) were generated using a time-history method of analysis. The El Centro 1940 earthquake N-S component, anchored to 0.10g, was used to generate the ISRS for the Dresden Operating Basis Earthquake (OBE). For SSE design, the spectral values were obtained by doubling the OBE spectra. The OBE is defined in the horizontal direction by the Housner-type GRS scaled to 0.10g peak ground acceleration (PGA) and ISRS developed from the El Centro earthquake time history scaled to 0.10g. The OBE in the vertical direction is defined by 2/3 of the Housner-type GRS with a resulting PGA of 0.067g. The SSE is defined by multiplying the OBE acceleration by a factor of 2, resulting in a horizontal direction GRS PGA of 0.20g.

3.1.3.2 Description of Dynamic Modeling and Selection of Key Modeling Parameters

The Dresden Nuclear Station is made up of the following Seismic Class 1 structures:

1. Drywell Containment Structures & Internals including Reactor Pressure Vessel
2. Reactor Building
3. Suppression Chamber
4. Control Room
5. 310 Foot Chimney

Safe shutdown equipment is located in structures 1 through 4 listed above. The seismic structural response was determined for the structures above. The buildings are founded on bedrock so any soil-structure interaction effects were considered negligible.

The mathematical model of the above Seismic Class 1 structures was constructed in terms of lumped masses and stiffness coefficients on a fixed base. Damping values used were based upon the evaluation of the materials and mode shapes. The damping values used are as shown in the table below:

Dresden Design Basis Damping Values

Structure/Component Type	Damping
Welded Assemblies	1%
Steel Frame Structures	2%
Bolted & Riveted Assemblies	2%
Reinforced Concrete Structures	5%
Vital Piping Systems	0.5%

3.1.3.3 In-Structure Response Spectra

The horizontal response spectra curves are based on a lumped mass, fixed base model for the Reactor-Turbine building. The vertical in-structure response constant acceleration or seismic coefficient is defined by 2/3 of the horizontal Housner-type GRS PGA (OBE - 0.067g, SSE - 0.133g).

Additional in-structure response spectra at additional damping values have been developed for other programs such as the review of masonry walls (IE Bulletin 80-11). As described in the GIP [3.14], the use of 5 percent damped in-structure response acceleration curves are allowed for characterizing seismic demand for equipment.

The NRC staff reviewed the original and subsequent modeling performed by ComEd and its contractors and determined that the building modeling was adequate. The NRC staff concluded that the resulting in-structure spectra could be utilized as conservative design spectra as defined in the GIP [3.14] as opposed to realistic median centered ISRS.

3.1.4 Hydrodynamic Loads

In BWRs, events resulting in the discharge of steam into the suppression pool can cause hydrodynamically induced structural vibrations. The suppression pool at Dresden is a toroidal steel structure anchored to the Reactor Building's foundation mat, which rests directly on bedrock. Eight vent pipes form the major connection between the suppression chamber and the drywell; expansion joints are provided in these vent pipes to allow for differential movement. As a result of this configuration, vibrations due to hydrodynamic loads are not considered in the Seismic Margins Assessment, except for the evaluation of the suppression chamber itself.

3.1.5 Summary of Results

The results are included in Tables 3.1, 3.2, and 3.3. The following items were found to have a seismic capacity of less than 0.3g PGA, however the capacities meet or exceed the design basis requirement of 0.2g PGA. Items 1 and 3 are the most limiting conditions and are controlled by anchorage capacity or tank buckling. Based on experiences with actual industrial facilities in moderate to severe earthquakes, it is concluded that the Dresden plant possesses reasonable margin with respect to its design basis earthquake, and safe shutdown capability will not be lost.

Item No.	Capacity (PGA)	Description	Notes
1	0.202g	Buses - D03-8303B-----M05, D02-8302B-----M05, D03-8303A---M05, and D03-83125----P06	Anchorage controls capacity.
2	0.27g	Distribution Panel - D02-8302A---M05 and D02-83125----P06	Anchorage controls capacity.
3	0.20g	Condensate Storage Tanks - D00-3303-A----T05, D00-3303-B----T05	Tank buckling controls capacity.
4	0.26g	Diesel Fuel Oil Storage Day Tank D00-5202-T05	Block wall controls capacity.
5	0.27g	Battery Charger - D02-8300-2A---B05	Anchorage controls capacity.
6	0.27g	Distribution Panels - D02-9802-A & B---P06	Anchorage controls capacity.

Item No.	Capacity (PGA)	Description	Notes
7	0.27g	Switchgear - D02-7328---S35 & D02-7329----S35	Anchorage controls capacity.
8	0.27g	Bus #2A-1 - D02-8302A1----P06	Anchorage controls capacity.
9	0.27g	125V DC/TB Battery Bus #2 D02-83125-2-P06	Anchorage controls capacity.
10	0.27g	125V DC/Battery Charger #2 D02-8300-2---B05	Anchorage controls capacity.
11	0.28g	125V DC Battery Charger - D03-8300-3A---B05	Anchorage controls capacity.
12	0.28g	Unit 2&3 Torus Suppression Chambers	Torus shell stress controls capacity.
13	0.29g	Motor Control Centers D02-83250-----M05 & D02-7826-4---M05	Anchorage controls capacity.
14	0.29g	Bus #2B-1 - D02-8302B-1---P06	Anchorage controls capacity.
15	0.29g	125V DC/TB Res Bus #2 D02-83125-1---P06	Anchorage controls capacity.

In December of 1997, Items 1 and 2 were previously found to have a capacity less than the design basis requirement of 0.2g. The new capacities shown above are based on additional evaluations that included any proposed modifications. Items that were found to have a capacity greater than or equal to 0.3g PGA based on these evaluations have been removed from the above table.

3.2 *Dresden Success Path Equipment List (SPEL)*

3.2.1 Introduction

The composite Seismic Individual Plant Examination for External Events (IPEEE) Success Path Equipment List (SPEL) contains all mechanical and electrical equipment (excluding relays) needed to achieve and maintain safe shutdown conditions at Dresden Station. The equipment listed contains those components that are needed to support the IPEEE Safe Shutdown Functions as well as those components needed for IPEEE containment performance.
The equipment was chosen based on the requirements of the Seismic Qualification Utility Group (SQUG) Generic Implementation Procedure (GIP) [3.14], Supplement 1 to GL 87-02 [3.20], and NP-6041 [3.5].

Components have been identified that are in the primary and backup shutdown paths and have been grouped together by both function and system. Components have also been listed in the order of the flowpath. The primary and backup safe shutdown paths are identified in the color-coded P&IDs on file with ComEd.

3.2.2 Purpose

The purpose of the Seismic IPEEE SPEL is to identify all mechanical and electrical equipment (excluding relays) that is needed to achieve and maintain safe shutdown conditions at Dresden Station. Based upon a review of the operating procedures, the Operations Department identified the systems to be utilized for the primary and backup safe shutdown paths to accomplish the safe shutdown functions identified in the SQUG GIP [3.14] and NP-6041 [3.5]. This list was generated to provide essential information related to the components associated with safe shutdown functions.

3.2.3 Basis For Selection

Dresden Station followed the SQUG GIP [3.14], NP-6041 [3.5], and Supplement 1 to GL 87-02 [3.20] to evaluate the seismic adequacy of the mechanical and electrical equipment needed to bring the station to a safe shutdown condition following an SSE in accordance with NUREG-1407 [3.1]. Per section 3.3.3 of the GIP, "safe shutdown is defined as bringing the plant to a hot shutdown condition and maintaining it there for a minimum of 72 hours following an earthquake. Some plants may not have sufficient water inventory to stay in hot shutdown for three days, while other plants may prefer to be brought to a cold shutdown condition during this period of time instead of staying in the hot shutdown condition."

The four safe shutdown functions that must be accomplished to achieve hot safe shutdown are as follows:

1. Reactor Reactivity Control
2. Reactor Coolant Pressure Control

3. Reactor Coolant Inventory Control
4. Decay Heat Removal

The systems/functional paths chosen are to meet the following criteria:

1. Achieve and maintain the plant in a hot shutdown condition for 72 hours following the SSE.
2. A LOOP is considered to have occurred coincident with the SSE and lasts for the first 72 hours.
3. No other design basis event is considered to occur other than the SSE.
4. HPCI is assumed to be capable of controlling a 1 inch LOCA.

REACTOR REACTIVITY CONTROL

To achieve reactor reactivity control under this safe shutdown path, the Control Rod Drive (CRD) System was chosen. This system was chosen for the following reasons:

The CRD System is a safety-related system.
The control rods are designed to be driven in automatically upon a loss of power.
The CRD System will actuate automatically.
The CRD System is designed to fail in the safe position, that is, with the control rods in the fully inserted position.
The CRD System is a Class I system.
The CRD System major equipment is located in the Reactor Building on elevation 517' 6".
No major support systems are required for CRD system operation.

The CRD System is used as both the primary and backup system for the Reactor Reactivity Control function. This is because each of the 177 hydraulic control units (HCUs) per unit are completely separate and independent. Therefore, the system is designed to accommodate a single failure, and still accomplish safe shutdown.

REACTOR COOLANT PRESSURE CONTROL

To achieve reactor coolant pressure control under this safe shutdown path, the Target Rock Safety Relief Valve and the Electromatic Relief Valves (ERVs) were chosen as the primary path. These valves are sized to rapidly remove steam generated from the reactor upon closure of the turbine stop valves and coincident with failure of the turbine bypass system. The blowdown from each relief valve is routed through a separate line below the torus water line. The ERVs require control power from the 125 V Direct Current system. The Target Rock Safety Relief Valve requires the support of Drywell Pneumatic System while in the Relief Mode, but does not require any support systems while in the Safety Mode. The backup system for this function is the reactor safety valves. These valves are sized to protect the pressure vessel against overpressurization during a failure of the reactor relief valves. These valves are balanced, spring-loaded-type safety valves which discharge directly to the drywell airspace.

The advantages of these relief and safety valves are as follows:

These valves are part of a safety-related system.

These valves will actuate automatically.
These valves are part of a Class I system.
These valves are located on the second floor of the drywell (elevation 537' 0").

REACTOR COOLANT INVENTORY CONTROL

To achieve reactor coolant inventory control under this safe shutdown path, the High Pressure Coolant Injection (HPCI) system was chosen as the primary path. The HPCI system consists of a steam driven pump that can take suction from either the suppression pool or the condensate storage tank and pump water into the reactor vessel. The steam that runs the turbine comes from the reactor and is exhausted to the suppression pool. The HPCI system is designed to pump makeup water to the reactor at a rate of 5,600 gpm. The 125V DC system provides control power to the HPCI system while the 250V DC system provides power to MOVs and HPCI auxiliaries. The advantages of the HPCI system are as follows:

The HPCI system is a safety-related system.
The HPCI system will actuate automatically.
The HPCI system is a Class I system.
The HPCI system major equipment is located in the HPCI rooms which are in the Reactor Building at elevation 476' 6".

The backup systems for this function are the Automatic Depressurization System (ADS) and Division I of the Low Pressure Coolant Injection (LPCI) system. ADS will depressurize the reactor vessel so that the LPCI system could be initiated for reactor coolant inventory control. Although Division II of LPCI is the preferred path according to emergency procedures, Division I was chosen since the HPCI system is powered by Division II. By selecting Division I of LPCI, an added safety feature of the SSEL/ SPEL is provided for the reactor coolant inventory control function. The advantages of the LPCI and ADS systems are as follows:

The LPCI and ADS systems are safety-related systems.
The LPCI and ADS systems will actuate automatically.
The LPCI and ADS systems are a Class I system.
Only one division of LPCI is required.
The ADS system major equipment is located on the second floor of the drywell.
The LPCI system major equipment is located in the corner rooms which are in the Reactor Building at elevation 476' 6" (Pumps and Valves).

DECAY HEAT REMOVAL

To achieve decay heat removal under this safe shutdown path, Division II of the Low Pressure Coolant Injection (LPCI) system and the Containment Cooling Service Water (CCSW) system was chosen as the primary path. The HPCI turbine takes steam from the reactor and exhausts it to the suppression pool. The LPCI system would then be aligned in the torus cooling mode. This alignment includes the LPCI pumps taking suction from the suppression pool, routing the water through the LPCI heat exchanger, and injecting the water back into the suppression pool. CCSW provides the cooling water to the LPCI heat exchanger and takes suction from the cribhouse bay. The backup systems for this function are the ADS valves and Division I of the LPCI and CCSW systems. The ADS valves discharge steam from the reactor to the suppression pool. The LPCI and CCSW systems would be aligned the same as Division II stated above.

The advantages of the LPCI and CCSW systems are as follows:

The LPCI and CCSW systems are safety-related systems.
The LPCI system will actuate automatically.
The LPCI and CCSW systems are Class I systems.
Only one division of LPCI is required.
The CCSW system is located in the Turbine Building at elevation 495' 0".
The LPCI system major equipment is located in the corner rooms which are in the Reactor Building at elevation 476' 6".

3.2.4 Methodology

The methodology for preparation of the Seismic Individual Plant Examination for External Events (IPEEE) Success Path Equipment List (SPEL), given in Table 3.2 is as follows:

1. The following systems have been chosen by the Dresden Operations Department to satisfy the four safe shutdown functions for primary and backup shutdown paths.

Safe Shutdown Function	Primary Shutdown Path	Backup Shutdown Path
Reactor Reactivity Control	Control Rod Drive System	Control Rod Drive System
Reactor Coolant Pressure Control	Automatic Depressurization (Target Rock and ERVs)	Automatic Depressurization (Reactor Safety Valves)
Reactor Coolant Inventory Control	High Pressure Coolant Inj.	Automatic Depressurization and Low Pressure Coolant Injection (Division I)
Decay Heat Removal *	High Pressure Coolant Inj. Low Pressure Coolant Inj. (Division II) and Cont. Cooling Service Water (Division II)	Automatic Depressurization Low Pressure Coolant Inj. (Division I) and Cont. Cooling Service Water (Division I)

* The Unit 2 Emergency Diesel Generator cooling water system is included in the success path for Isolation Condenser makeup to mitigate the consequences of a Dresden dam failure.

2. Color-coded P&IDs were prepared to trace the primary and backup shutdown paths that include the above systems and the components that are used to perform the safe shutdown function. Components of systems that support the above systems are also color-coded on the P&IDs. The color coding scheme is as follows:

Red	- Primary Path (Train #1)
Green	- Secondary Path (Train # 2)
Yellow	- Path Common to Both Primary and Secondary Path (Train # 3)
Blue	- Optional Path or Equipment

There are some components that are listed more than once. This is because some systems are used for more than one function. In addition, a component may be used as a primary path for one function, and a secondary path for another function. Components were repeated to ensure the list for each individual function is complete. If a component is listed once as a primary path (Division I) and again as a secondary path (Division II), then the component was color coded yellow since it is common path (Division III), used for both primary and secondary paths.

3. Using the SQUG SSEM software, a list was generated that contains all equipment/components needed for safe shutdown of the plant and to maintain containment performance. This list includes all components that are needed to support the equipment/components used for safe shutdown or containment performance. This list also includes all components that are inherently rugged or not possessing any active safety function.

4. For each piece of equipment/component identified in the list, all of the appropriate information fields were completed. The following is a description of the information that is included in the IPEEE SPEL.

FIELD	FIELD NAME	DESCRIPTION OF FIELD
1	Line Number	A unique number used to identify the order in which the equipment was selected and what function it supports. Sequential numbers are assigned as components are selected from a starting point, working towards the end of a functional path. X1000 Series: Reactor Reactivity Control X2000 Series: Reactor Coolant Pressure Control X3000 Series: Reactor Coolant Inventory Control X4000 Series: Decay Heat Removal X5000 Series: Auxiliary and Support Systems X7000 Series: Containment Performance X8000 Series: Electrical Systems X9000 Series: Racks and Panels X identifies if the component is specific to Dresden Units 2 or 3 shutdown path, or if the component is common to both units. X = 0 (Common for both units) X = 2 (Unit 2 path) X = 3 (Unit 3 path)

2	Train	1 = Primary success path component 2 = Backup success path component 3 = Component is shared by both success paths OP = Optional path
3	Equipment Class	Identify the appropriate equipment classification. Classifications for A-46 plant equipment are obtained from table 3-1 of the SQUG Generic Implementation Procedure (GIP). If a component is inherently rugged, it was identified with an "R" in this field space. These components change state from the normal to desired state. If the component was included for completeness or information purposes only, an "I" was inserted in this field.
4	Mark Number	The unique component or equipment identification number. This number was taken from the Dresden Master Equipment List (MEL) whenever possible.
5	System/Equipment Description	Identifies the system that component is a part of. Also provides a brief description of the function of the equipment or component. The Rule of the Box mother component is identified here if applicable. (ROB - _)
6	Drawing/ Rev/ Zone	Identifies the Piping and Instrumentation Diagram (P&ID) or electrical single line diagram where the component can be found. Identifies the revision of the drawing. Identifies the coordinates of the diagram where the equipment or component can be found.
7	Building	Identifies the building where the equipment or component is located.
8	Floor Elevation	Identifies the floor elevation within the plant where the equipment or component is located.
9	Room or Row/Col	Identifies where the component is located in the plant by room or by row and column found on general arrangement drawings.

10	Sort	Identifies what type of evaluation is required using one of the following codes: S = Seismic R = Relay SR = Seismic and Relay B = Component falls within the "Rule of the Box" and a seismic review separate from that of it's "mother" component is not required BR = "Rule of the Box" component that also requires a relay review. Components with the following codes do not require any type of seismic or relay review and are included in the list for completeness only. N/A = Not Applicable (inherently rugged component)
11	Notes	The following notes have been used in this list: 1 = Actuation of component from normal to desired state is not required until after 30 seconds. 2 = Actuation of component from normal to desired state may not occur spuriously. 3 = Typical of 177 (Used with Control Rod Drive Unit components) 4 = To close the tie breakers between SWGR 28 and 29 (38 and 39 for unit 3), one of the main feed breakers to SWGR 28 (38) or 29 (39) should be open. 5 = Component contains "essential" relays for safe shutdown 6 = Component requires an expansion anchor bolt tightness check
12	Normal State	This field contains one of the following descriptions: OPEN = Equipment is normally open

CLOSED	=	Equipment is normally closed
VENT	=	A three-way valve in the vented position
ON	=	Equipment is on and normally operating (i.e., a pump)
OFF	=	Equipment is off and normally not operating
OPERABLE	=	Equipment normally changes state from open to closed or from closed to open
ENERGIZED	=	Equipment is on and normally operating (i.e., required electrical bus is energized)
DEENER-GIZED	=	Equipment is off and normally not operating
N/A	=	Not Applicable

FIELD	FIELD NAME	DESCRIPTION OF FIELD
13	Desired State	This field contains one of the following descriptions: OPEN = Equipment is desired to be open CLOSED = Equipment is desired to be closed VENT = A three-way valve in the vented position ON = Equipment is on and now operating (i.e., a pump) OFF = Equipment is off and now not operating OPERABLE = Equipment normally changes state from open to closed or from closed to open ENERGIZED = Equipment is on and operating (i.e., required electrical bus is energized) DEENER-GIZED = Equipment is off and now not operating N/A = Not Applicable
14	Power Required	Determines whether an external source of power is required to operate or control the equipment or component.
15	Support System Drawing/ Rev	Identifies the drawing number and revision which shows the systems or components required to support the operation of the equipment or components being evaluated.
16	Required Interconnections/ Supporting Components	Identifies the name of each system or component supporting the equipment identified being evaluated. Identify the motive and control power if required. (Motive Power - ____) (Control Power - ____)
17	Regulatory Issue	I - Component evaluated per Generic Letter 88-20

3.2.5 Peer Review

A Peer Review of the Safe Shutdown and Success Path Lists was performed by Mr. Robert J. Budnitz, president of Future Resources Associates, Inc, and is provided in Appendix D.

3.3 Analysis of Structure Response

3.3.1 Seismic Margin Earthquake Selection

Per NUREG-1407 [3.1], Section 3.2.2 and Table 3.1, the Seismic Margin Earthquake (SME) for Dresden is a NUREG/CR-0098 [3.2] median rock or soil spectrum anchored at 0.3g. The buildings at Dresden are founded on bedrock [3.3, Section 3], so the CR-0098 rock spectrum applies.

The SME is derived as follows:

The peak ground acceleration is defined to be 0.3g.

Per Reference 2, Section 7.2:

v/a	= 36 in/sec/g, therefore peak velocity	= 0.3 x 36	= 10.8 in/s
ad/v^2	= 6.0, therefore peak displacement	= $(6.0 \times 10.8^2)/(0.3 \times 386)$	= 6.04 in

Using the values from Reference 3.2, Table 3, the median, 5% damped, response spectrum peak values are:

acceleration	= 2.12 x 0.3	= 0.636g
velocity	= 1.65 x 10.8	= 17.8 in/s
displacement	= 1.39 x 6.04	= 8.40 in

The frequency control points occur where the displacement and velocity match, and where the velocity and acceleration match:

f_1	$= v / 2\pi d$	$= (17.8) / (2\pi \times 8.40)$	= 0.34 Hz
f_2	$= a / 2\pi v$	$= (0.636 \times 386) / (2\pi \times 17.8)$	= 2.2 Hz

Consistent with Reference 3.2, Figure 3, the peak acceleration value of 0.636g is linearly interpolated (on a log-log plot) starting at 8 Hz down to the peak ground acceleration value of 0.3g at 33 Hz.

The response spectrum values for the horizontal direction were reduced by reduction factors as recommended in Reference 3.5 [pg 4-6] for the basemat size and horizontal spatial variation in ground motion. The reduction factors used are as follows:

<u>Frequency</u>	<u>Reduction Factor</u>
5 & lower	1.0
10	(1-0.1 x 210/150) = 0.86
25 & larger	(1-0.2 x 210/150) = 0.72

For frequencies between the values given above, linear interpolation of the reduction factor on a log-log plot was used.

The vertical response spectral amplitudes are taken as 2/3 of the corresponding unreduced values in the horizontal direction across the entire frequency range.

3.3.2 Development of Seismic Margin Earthquake In-Structure Demand

SME floor response spectra were generated by Sargent & Lundy Engineers [3.18]. The original horizontal models were planer models (2-dimensional) in the north-south and east west directions. Given asymmetry in one of the compass directions, a 3-dimensional model was developed to include the eccentricity between the stiffness center and mass center, torsional rigidity and torsional mass properties of the buildings at each floor level.

A vertical seismic model was developed for the SMA study. Appropriate areas for steel and concrete columns in the Reactor and Turbine Buildings were developed to represent vertical building stiffness. It was also determined that slabs 18" thick or greater were vertically rigid; however, 12" thick slabs were compliant and were included as slab panel oscillators at each floor elevation.

For SME evaluations, NP-6041 recommends a structural damping range from 5% to 10% for structural stresses from 1/2 yield stress to near yield (Reference 3.5, Table 4-1). A damping value of 7% was selected for the SME evaluations. This increase in damping from the original design value is reasonable considering the increase in earthquake level and is in the middle of the range recommended in NP-6041. However, the damping used for the drywell and reactor vessel were conservatively selected as 3% and 1%, respectively.

The resulting SME floor response spectra can be found in Reference 3.18.

3.4 Evaluation of Seismic Capacities of Components and Plant

3.4.1 Civil Structures

Containment

The containment (drywell) is a freestanding steel structure in the shape of an inverted light bulb, surrounded by a reinforced concrete biological shield wall. There is a 2" air gap between the drywell and the biological shield walls. The base of the drywell sits on a massive reinforced concrete pedestal rising up from the Reactor Building foundation. Concrete was poured both inside and around the drywell, so the drywell is essentially integral with the reactor building foundation. Additional load transfer between the base of the drywell and the foundation is provided by an array of steel studs projecting from the bottom of the drywell into the concrete.

The drywell was assigned a seismic capacity of 0.5g PGA. Per Table 2-3 of NP-6041 [3.5], the Dresden drywell meets the requirement for the second earthquake level - the steel pressure boundary is keyed to the base mat to prevent slipping.

Suppression Chamber (Torus)

The suppression chamber is a toroidal steel structure supported by sixteen (16) vertical saddles sitting on the reactor building base mat. The saddle base plates are free to slide to allow for thermal expansion. A sway rod assembly at the outside columns provides lateral support for the suppression chamber. The seismic sway rods consist of 3.5" diameter sway rods and 3.75" diameter turnbuckles to provide restraint for movement along the torus centerline resulting from lateral loads acting on the suppression chamber. The sway rods are joined to the 1.5" thick wing plate at the top of the support columns by 4" diameter pins. The lower ends of the sway rods are joined to 2" thick seismic tie plate at the column base.

Per Table 2-3 of NP-6041 [3.5], Mark I tori require evaluation for any earthquake exceeding the design basis.

The Dresden torus was evaluated under the Mark I Containment program [3.8]. This evaluation included normal operating loads (dead weight, pressure, temperature), seismic loads (OBE and DBE), and hydrodynamic loads (SRV discharge, pool swell, condensation oscillation, and chugging), including those for large-break and intermediate-break LOCAs. Numerous load cases were considered; results are presented only for the controlling load combination for each component evaluated. The controlling load combinations almost always include one of the major hydrodynamic loads - pool swell, condensation oscillation, or chugging due to large or intermediate break LOCAs.

Appendix K of NP-6041 [3.5] discusses hydrodynamic loads as they apply to the Seismic Margins Assessment. Per this appendix, the simultaneous occurrence of an earthquake and an intermediate or large break LOCA is not credible. A simultaneous earthquake and a small break LOCA is considered credible, but the only hydrodynamic load associated with a small break LOCA is chugging, and that will occur after the earthquake has ended. The only hydrodynamic load that can credibly be expected to occur simultaneously with the earthquake is SRV discharge, and these loads can be combined by the square root of the sum of the squares (SRSS'd) with the earthquake loads.

The torus evaluation [3.8] does not provide results for individual load cases, only for the controlling load cases. It is the SRT's experience that seismic loads are not usually a major load case in torus evaluations. Based on this, it was decided to base the torus' seismic capacity on the Reference 3.8 results for the torus components most directly affected by the seismic loads - the sway bar seismic restraint system and the torus shell adjacent to the sway bar restraint system. According to Reference 3.8, the governing load case is Load case 2b per Table 6.2-4. The sway bar itself clearly has significant margin, so the primary + secondary stresses in the torus shell adjacent to the sway bar coupler plate are in control. The maximum ratio between the 5% damped SME and the SSE statically applied load of 0.5g is 1.27. Subtracting the stresses due to other loads per Table 6.1-1 and ratioing the resulting seismic stresses by 1.27, adding back in the stresses due to other loads and ratioing the allowable stress by the inverse of the stress interaction ratio, the seismic capacity based on the SME PGA is therefore:

$0.3g \times 1/[(49.0)(1.27)+12.3]\ 61.34/0.88 = 0.28g$.

Note that this value is conservative because (1) the load case includes other loads which probably include chugging, and (2) the individual loads were summed, not SRSS'd.

Reactor Building

The Reactor Building is a reinforced concrete structure from the foundation up to the refueling floor. All required equipment and plant systems are below the refueling floor. Per Section 3 of the UFSAR [3.3], the Reactor Building is a Class I structure and is designed for a 0.20g Safe Shutdown Earthquake (SSE).

The Reactor Building was assigned a seismic capacity of 0.3g PGA. Per Table 2-3 of NP-6041 [3.5], the Dresden Reactor Building meets the requirement for the first earthquake level - it is a reinforced concrete frame designed for an SSE of 0.1g or greater.

Turbine Building Complex

The entire Turbine Building complex is a reinforced concrete structure except for the turbine hall superstructure. Note that this area does not house SMA components.

Although the Turbine Building is a Class II structure, for seismic design the entire Turbine Building complex was included in a single structural model with the Reactor Building and evaluated as a Class 1 structure (see Section 3 of the UFSAR, Reference 3.3). Based on the results of a dynamic analysis of this model, the Class I structures were designed for the 0.20g design basis earthquake and a 0.10g operating basis earthquake.

The Turbine Building complex was assigned a seismic capacity of 0.3g PGA. Per Table 2-3 of NP-6041 [3.5], the areas of the complex housing SMA components meet the requirement for the first earthquake level - they are reinforced concrete frames designed for an SSE of 0.1g or greater.

3.4.2 Other Structures and Structural Issues

There is no potential for impact between the Reactor Building and the Turbine Building complex because they are connected at numerous elevations.

The 310-foot main stack, which is a Class I structure, is designed for statically applied SSE seismic coefficients. The stack is essentially identical to the Quad Cities stack in height, diameter and site conditions. The Quad Cities stack was dynamically analyzed to the Golden gate park earthquake normalized to 0.24g design basis earthquake (SSE) and a 0.12g operating basis earthquake and was assigned a seismic capacity of 0.3g PGA. Per Table 2-3 of NP-6041 [3.5] and by comparison to the Quad Cities chimney, the Dresden chimney meets the requirement for the first earthquake level designed for an SSE of 0.1g or greater.

The Cribhouse is a Class II structure with masonry walls above grade and reinforced concrete below grade. The concrete structure of the Cribhouse would not be affected by tornado or earthquakes. Per the walkdown review of Drs. R. P. Kennedy and J. D. Stevenson, and Mr. W. Djordjevic, they concur with this assessment and adjudge the Cribhouse to be screened out with respect to the 0.3g RLE seismic margins earthquake.

All buildings were designed to the 1963 Editions of ACI 318 and AISC as well as the 1964 Edition of the Uniform Building Code (UBC).

It should be noted that the seismic design of Dresden Unit 2 was examined by the NRC under the Systematic Evaluation Program (SEP), Topic III-6. The NRC evaluated the capability of Dresden Unit 2 to withstand a safe shutdown earthquake (SSE) by sampling review and confirmatory analysis. The NRC concluded that the majority of "safety related structures and structural elements of the Dresden 2 facility are adequately designed to resist the postulated seismic event."

All structures including the Cribhouse were screened using the first column in Table 2-3 in EPRI Report NP-6041. The screening approach utilizes the experience gained in performing seismic margin assessments (SMAs) to screen components out at the RLE level of 0.3g, PGA.

All caveats of Table 2-3 were dispositioned including the concrete containment requirements, separations between structures, reinforcement detailing, and penetrations including associated requisite piping flexibility.

The control room ceiling is a T-bar system supported by threaded rods which are suspended from a light metal strut gridwork. The gridwork is, in turn, attached to structural steel by beam clamps. The support system is adjudged seismically adequate and is assigned a seismic capacity of 0.3g PGA.

Dresden Station has a dike surrounding the cooling lake and a dam on the Illinois river. The potential failure of these structures is discussed in Section 2.4.4.2 of the UFSAR. The NRC safety evaluation for SEP Topic II-4.E concluded that the plant is designed so that it can be safely shut down in the event of failure of the Dresden dam and loss of the pool impounded by it. Part of the basis for this conclusion was that there is enough water impounded in the intake and discharge canals below their high point elevations to allow a safe shutdown of Dresden Units 2 and 3. Based on the SEP evaluation, the failure of the dam or dike will not impact the ability to safely shut down Units 2 and 3.

To mitigate the consequences of a dam failure, the Unit 2 Emergency Diesel Generator cooling water system is a seismically verified source of Isolation Condenser makeup for providing decay heat removal.

Reference 13 removed the evaluation of soil-related failures from the scope of the seismic IPEEE for focused scope plants. In any event, Dresden is a rock site, so there are no significant soil-related issues.

3.4.3 Masonry Block Walls

During the USI A-46/SMA equipment walkdowns, any block walls that were judged a potential interaction hazard were noted. This resulted in the list of block walls shown in Table 3.1. The majority of these walls had received a design-basis evaluation as part of the 80-11 Program [3.9, 3.10, 3.11, 3.12]. Two walls noted during the walk downs were not part of the 80-11 Program (in the Unit 2/3 ventilation room). These walls were installed after the 80-11 Program and are, in general, better engineered than the 80-11 walls. The block wall seismic capacities, in terms of the PGA, are summarized in Table 3.1.

The 80-11 walls are constructed of unreinforced, hollow and solid block, face-bedded masonry units. All the walls are single wythe; most are 12" thick, and some are 8" thick. The mortar compressive strength (m_0) used in the evaluation was 750 psi. The 80-11 evaluation consisted of a response spectrum dynamic analysis using the 2% damped DBE floor response spectra. The controlling stress in the evaluation was almost always the tensile stress normal to the face bed. The allowable value used for this stress was $1.67 \times 0.5 \times \sqrt{m_0} = 23$ psi.

A seismic capacity (HCLPF) value was first calculated by scaling the 80-11 evaluations. The scaling included two factors:

First, a factor to scale the floor response spectra. The floor response factor was the ratio, at the fundamental frequency of the wall, of the 2% damped DBE floor response spectrum to the 7% damped SME floor response spectra. The use of 7% damping is based on the same argument presented in Section 3.3.2 of this report for using 7% damping for concrete structures to calculate the SME floor response spectra.

Second, a factor to scale the allowable stress. The allowable stress factor was the ratio between the 80-11 allowable stress of 23 psi, and an SMA allowable stress of 32.3 psi. The 32.3 psi was based on multiplying by 1.7 the design allowable stress of 19 psi in Table 6.3.1.1 of ACI 530-92 [3.15] for Type N Portland cement mortar, normal to the bed joints, hollow and ungrouted units.

The scaling showed that the majority of the walls have a capacity greater than the 0.3g PGA SME, but a few of the walls indicated lower capacities, and were investigated more closely:

Two masonry walls were found that were not part of the 80-11 effort in the 2/3 Ventilation room. These walls were installed after the 80-11 program, are fully grouted and/or reinforced, and are generally stronger than the 80-11 walls which have been seismically analyzed. Per Appendix A of NP-6041, externally reinforced walls with external rolled sections do not require reinvestigation for earthquakes less than 0.3g, so these walls are screened out for the RLE level (0.3g).

3.4.4 Mechanical and Electrical Equipment

3.4.4.1 Introduction

The seismic margins assessment of the equipment considered three factors:

- Seismic capacity of the equipment.
- Seismic capacity of the equipment anchorage.
- Seismic capacity of adjacent masonry block walls.

The seismic capacity of each component is expressed in terms of the peak ground acceleration (PGA). The overall seismic capacity of each item of equipment is the minimum of the three capacities identified above.

Table 3.2 summarizes the equipment assessment. Outliers include items of equipment that do not meet A-46 requirements, whether on the A-46 Safe Shutdown Equipment List (SSEL) or not. That is, specific non A-46 equipment is analyzed in accordance with the Generic Implementation Procedure (GIP) [3.14] as part of the IPEEE Program to account for potential small-break accidents concurrent with the SME.

3.4.4.2 Equipment Seismic Capacity

The assessment of IPEEE electrical and mechanical equipment was based on the Generic Implementation Procedure (GIP) [3.14] that was used to implement the USI A-46 resolution in accordance with Generic Letter (GL) 87-02 [3.19]. The assessment of each item of equipment was documented on a Screening Evaluation Worksheet (SEWS), which can be found in the ComEd document control system.

The seismic capacity for the equipment is based on Table 2-4 of NP-6041 [3.5]. Except for atmospheric storage tanks and equipment supported on vibration isolators, an item of equipment that passes a GIP evaluation satisfies the requirements for the first earthquake level in Table 2-4. For a number of equipment classes (e.g. horizontal pumps), a component that passes a GIP evaluation also satisfies the requirements for the second earthquake level in Table 2-4. Except for atmospheric storage tanks and equipment supported on vibration isolators, if an item of equipment passed the GIP evaluation, then it was assigned a seismic capacity of either 0.3g PGA (first earthquake level), or - if Table 2.4 does not require further evaluation - 0.5g PGA (second earthquake level). Note that all classes of equipment, except passive valves, would require further evaluation to meet the requirements of the third earthquake level.

3.4.4.3 Anchorage Seismic Capacity

An anchorage seismic capacity was calculated for all equipment [3.7] and is shown in the Anchorage field in Table 3.2, except:

- In-line equipment –(e.g. valves, temperature elements, and dampers).
- Equipment whose anchorage capacity is obviously high (e.g., a small circuit breaker panel anchored to a reinforced concrete wall with four expansion anchors).

The anchorage calculations followed GIP procedures (Section II.4.4 and Appendix C of Reference 14) with the following exceptions:

The SME floor response spectra were used. These are what the GIP calls “realistic, median-centered”, but the 1.25 factor of conservatism specified in GIP Table 4-3 is not required and was not applied.

The GIP allows the use of 1.5x the ground response spectrum as the floor response spectrum under certain conditions. This option was not used in these calculations; only the SME floor response spectra were used (the unfactored ground response spectra was used as the floor response spectrum for the basement of the reactor building).

The GIP requires that reduction factors be applied to anchor bolt capacities under certain conditions. All of these reduction factors were applied, where needed, except for the essential relay reduction factor for concrete expansion anchors.

The GIP requirements for bolt tightness checks are not required for IPEEE equipment and were not applied.

3.4.4.4 Block Wall Seismic Capacity

During the walkdown of each item of equipment, a note was made of any block walls that were judged to be potential interaction hazards. Block wall seismic capacities were calculated (see Section 3.4.3). Each item of equipment was then assigned a block wall seismic capacity equal to the minimum capacity of all block walls adjacent to that item of equipment. That capacity is listed in the Block Wall field in Table 3.1.

3.4.4.5 Equipment that Did Not Screen

All equipment - whether on the A-46 SSEL or not - was evaluated per the GIP [3.14]. All equipment that met GIP requirements was considered to have been "screened", and was assigned an *equipment* seismic capacity based on Table 2-4 of NP-6041 [3.5]. Note that the *equipment* seismic capacity does not consider anchorage or block walls.

Equipment that did not "screen" includes the following:

Outliers: The A-46 and IPEEE outliers are identified in Table 3.2. An "x" appears in both the A-46 SSEL field and the outlier field for A-46 outliers. The IPEEE outliers are identified by an "x" in the outlier field only. The SME anchorage capacity of A-46 outliers is provided in Table 3.3.

Atmospheric storage tanks: Table 2-4 of NP-6041 [3.5] requires that all atmospheric storage tanks be evaluated. These are discussed in Table 3.3.

3.4.5 Other Equipment

3.4.5.1 NSSS Primary Coolant System

The NSSS primary coolant system was assigned a seismic capacity of 0.5g PGA. Per Table 2-4 of NP-6041 [3.5], this equipment can be assigned a seismic capacity equal to the second earthquake level with no evaluation except for piping with intergranular stress corrosion cracking (IGSCC).

Programs that mitigate the effects of IGSCC in NSSS piping welds are in effect at Dresden. These include ongoing inspections, contingency weld overlay repair planning, and the use of hydrogen water chemistry . These programs are sufficient to address the IGSCC issue for the Structural Margins Assessment.

3.4.5.2 NSSS Supports

The NSSS supports were assigned a seismic capacity of 0.3g PGA. Per Table 2-4 of NP-6041 [3.5], this equipment can be assigned a seismic capacity equal to the first earthquake level with no evaluation if the supports are designed for combined loadings of SSE and pipe break. Per Section 3.9.3.1.1 of the updated FSAR [3.3], the reactor pressure vessel and its supports as well as component supports have been analyzed for seismic loads combined with pipe rupture loads.

3.4.5.3 Reactor Internals

Reference 3.13 removed the evaluation of reactor internals from the scope of the seismic IPEEE for focused scope plants.

3.4.5.4 Control Rod Drive Housing and Mechanisms

The control rod drive (CRD) housing and mechanisms were assigned a seismic capacity of 0.3g PGA. Per Table 2-4 of NP-6041 [3.5], this equipment can be assigned a seismic capacity equal to the first earthquake level if the control rod drive housing is laterally supported.

The typical longer control rod drive housing cylinders project about 13' below the bottom of the reactor vessel. At their bottom, the cylinders are supported in a steel gridwork that is suspended from steel beams with threaded rods. No documentation was found substantiating that the gridwork is laterally restrained to the inside wall of the reactor vessel pedestal. Therefore, it was decided to evaluate the CRD housing for lateral bending due to the cantilever bending deformation of the CRD housing. Stresses due the RLE loads were relatively small (well less than 10 ksi per Reference 3.11) and are, therefore, screened out at the 0.3g PGA level.

3.4.6 Distribution Systems

3.4.6.1 Category I Piping

Category I piping was assigned a seismic capacity of 0.5g PGA. Per Table 2-4 and Appendix A of NP-6041 [3.5], piping systems in nuclear power plants have capacities greater than 0.5g PGA, but certain details need to be investigated by a walkdown.

A piping walkdown was performed. The walkdown criteria followed Section 5 and Appendix A of EPRI NP-6041 [3.5]. Specifically, the walkdown looked for:

- threaded or mechanically coupled (Victaulic type) connections
- cast iron bodies
- inflexible branch lines
- long unsupported spans
- insufficient "rattle space" and close proximity of valve operators to interferences
- "unzipping" of threaded supports

- shock isolators
- sufficient flexibility of piping across structural joints (between buildings)

Safety related piping throughout the plant was "walked-by". Adequate flexibility was found at building interfaces. No issues as listed above were identified for safety related piping. Some drain lines were observed to have Victaulic couplings, but these lines are not over safety-related equipment, nor are they normally full of water.

In the seismic margins assessment (SMA), the success path piping systems are as follows: Control Rod Drive System (CRD), Automatic Depressurization System (ADS), High Pressure Coolant Injection (HPCI), and Low Pressure Coolant Injection System/Containment Cooling Service Water (LPCI/CCSW). LPCI piping was chosen as the system for a detailed walkdown in accordance with NP-6041 requirements. The A loop in the SE corner room and the B loop into the Torus compartment were walked down from end to end. Both systems are extremely well supported with obvious seismic supports. No anomalies were found.

Some drain lines were observed to have Victualic couplings, but these lines are normally not over safety-related equipment, nor are they normally full of water.

In conclusion, all seismically designed piping is screened at a 0.5g PGA.

3.4.6.2 HVAC Ducting and Dampers

HVAC ducting and dampers was assigned a seismic capacity of 0.3g PGA. Per Table 2-4 and Appendix A of NP-6041 [3.5], HVAC ducting can be assigned a 0.3g PGA seismic capacity pending a walk down.

The ductwork throughout the safety-related areas of the plant was walked down and found to be adequately supported by either threaded rod trapeze supports anchored to embedded strut or light metal straps anchored by 1/4" diameter concrete expansion anchors. Both support systems are ductile, and given the light weight of ductwork, anchorage failure was judged not credible.

Note that the major concern raised in Appendix A of NP-6041 - HVAC equipment such as fans and coolers mounted on vibration isolators - was addressed above under *Electrical and Mechanical Equipment*.

3.4.6.3 Cable Trays and Electrical Conduit

The cable and conduit raceway review performed at the Dresden Station Units 2 and 3, follows the Generic Implementation Procedure (GIP) developed by the Seismic Qualification Utility Group (SQUG) [14]. If the raceway system meets the GIP caveats and limited analytical review (LAR) evaluations, it is screened for the RLE at a 0.3g PGA.

The raceway review was performed as specified in GIP Section 8. Raceway systems were walked-down, checked against the Inclusion Rules and Other Seismic Performance Concerns as specified in Section 8.2 of the GIP, and examined for seismic spatial interactions with adjacent equipment and structures. Twelve(12) representative, worst-case raceway supports were selected and as-built. These supports then received a Limited Analytical Review per GIP Section 8.3 of the GIP. Seven outliers were identified and documented. The HCLPF capacities for the seven outlier raceways systems were analytically screened out by GIP LAR resolution analysis with the exception of the occurrence of short rod hangers interspersed among longer rod hangers within a raceway system. This type of design is found in the Reactor, Turbine and Service buildings. A HCLPF based on the use of the Conservative Deterministic Failure Margin (CDFM) method yields a 0.15g PGA capacity with regard to the RLE based on a worst case representation.

3.4.7 Other seismic issues

3.4.7.1 Seismically Induced Flooding

Seismically induced flooding was evaluated by first assembling a list of potential flooding sources in the areas of the plant containing IPEEE equipment, then performing a walk down to assess whether the sources are both significant hazards and seismically vulnerable.

A walkdown was conducted the week of October 2, 1995 by Mr. B. M. Lory of ComEd and Mr. W. Djordjevic of S&A to address the seismic vulnerability of potential internal flooding sources. Initially, an inventory of flooding sources was compiled. This included non-seismically designed piping (seismically designed piping is screened out with a High Confidence Low Probability of Failure (HCLPF) value of at least 0.5g peak ground acceleration (PGA) and large tanks greater than 1000 gallons. In particular, non-critical - thus, non-seismically designed - piping such as fire protection, non-critical main steam and non-critical service water piping were studied. All areas of the Reactor, Turbine, and Cribhouse buildings were walked down.

One concern was identified with regard to fluid retaining tanks and should be evaluated against the findings of the internal flooding study:

The MG Set Area Drain Tank 2/3-4911 at Col. G-44 (Turbine Building, El. 534') is saddle mounted, but not welded or positively anchored to its saddles. The tank capacity is approximately 1000 gallons. If the tank slides off the saddles, the drain line valve could be broken and thus cause an oil release. It is located approximately 15' away from MCC 39-2. This is an interaction concern that is being tracked in Table 3.3 of this report.

The Unit 2 Isolation Condenser is not guided (supported) transversely except for center support. Conversely, the Unit 3 condenser is supported transversely. These heat exchangers were evaluated in the Dresden SEP and Sargent & Lundy calculation 8900-15-EO-S and are seismically adequate.

All other tanks seen are positively anchored.

The fire protection piping was walked down in general to identify any potential for seismic degradation. In the vicinity of the IPEEE equipment, no credible incidences were recorded. The fire protection piping is generally gravity, rod hung welded steel piping. An extreme minority of piping segments are connected by threaded components and mechanical (Victualic) couplings. The fire protection piping headers run throughout the plant; and, although they are not seismically designed, they are gravity rod-hung, welded steel piping that frame through concrete walls (which act as lateral guides) and are not seen as plausible failure sources at moderate earthquake levels. One possible hazard was identified in the Crib House adjacent to the service water pumps where mechanically coupled (Victualic) piping is routed adjacent to some of the service water pumps. Large, laterally unsupported spans may develop large bending moments which may result in water leakage. This area and potential event were addressed in Section 4.4.4 of the Dresden IPE [3.17] and found not to have a high probability for potential core damage.

Spray-down hazards on equipment associated with fire piping (or other piping systems) were included as part of the specific IPEEE equipment walkdown and none were noted at Dresden.

Other internal flooding hazards were investigated throughout the plant site. The cubicle coolers in each of the corner rooms were assessed since they were found to be rod-hung and flexible, thus likely to translate during a seismic event. DCPs 9900168 and 9900278 will be installed during the D3R16 and D2R17 refueling outages to laterally restrain the coolers, thus there will be minimal potential for a rupture in the attached service water cooling water piping due to translation of the coolers.

3.4.7.2 Seismic / Fire Interaction Walkdown

A walkdown was conducted the week of October 2, 1995 by Mr. B. M. Lory of ComEd and Mr. W. Djordjevic of S&A to address the seismic vulnerability of potential fire sources as well as seismic-fire interactions. Three topics were studied as part of the seismic-fire interaction issue: (1) seismically induced fires, (2) inadvertent actuation of fire suppression systems, and (3) seismic degradation of fire suppression systems.

Observations and potential issues resulting from the walkdown are listed below. The Dresden Fire Risk Scoping Study response (see Section 4.9.3.1) addresses each of the potential issues given below.

<u>Seismically induced fires</u> were evaluated by first assembling a list of significant combustion sources (significant quantities with ignition points below about 500(F), then performing a walk down to assess whether the sources are both significant hazards and seismically vulnerable. All potential fire sources were walked down in the Reactor, Turbine, and Cribhouse buildings. Combustible sources such as fuel oil tanks, waste gas tanks, hydrogen gas bottles, flammable liquid storage cabinets, and hydrogen piping were assessed.

The following potential hazards and observations were noted:

POTENTIAL ISSUE: Hydrogen Seal Oil Control Panel and Hydrogen Monitors
The Hydrogen seal oil control panel on TB, El. 538', Unit 3 was unanchored. The Unit 2 seal oil panel was anchored, but the anchorage welds were of questionable quality. Hydrogen lines are routed through these cabinets, so the potential for hydrogen gas release in this area existed. The shelf-mounted Hydrogen Monitors located near each seal oil panel were also not

positively anchored. These concerns have been resolved by DCPs 9900205 (Unit 2) and 9900204 (Unit 3) that positively anchor the panels and seismically restrain the monitors.

OBSERVATION: Flammables Storage Cabinets
All such cabinets are unanchored and have doors which are secured (latched) closed by three point latches. None of the cabinets were observed to contain anything other than oil, grease or lubricants and some contained no flammables at all. The cabinets are 3' tall, but based upon their aspect ratio they are not considered hazards since they will slide, but not tip over. The two locations noted are:

Turbine Bldg., El 538', adjacent to Hydrogen Seal Oil Control Panel
Reactor Bldg., El 545

OBSERVATION: Oil Cooled Switchgear
The Switchgear associated step-down transformers contain oil (oil cooled transformers) and are not anchored. Given their aspect ratio, they may tip and, as a result, could potentially release oil. The transformer area is diked (curb) and will contain the oil release; thus, this issue is resolved. The observed switchgear are:

Switchgear 35 & 36
Switchgear 25 & 26
Switchgear 27 & 37

OBSERVATION: PCB Holding Tanks
PCB holding tanks behind Switchgear 23-1 and 34-1 were suggested as release hazards (for fire) due to site glass at bottom which could break. It was determined that they serve only as temporary storage for PCB liquid when collecting PCB fluid that is associated with 480V SWGR 28, 29, 38 and 39. They are otherwise empty during plant operation; thus, this issue is resolved.

OBSERVATION: Hydrogen Piping
The hydrogen piping that is routed throughout portions of the Turbine buildings is not seismically designed; however, it is primarily socket welded piping with lateral and longitudinal bracing; however, some portions of the piping near valve manifolds and other equipment are joined by flanged and threaded connections. No obvious weak areas, or areas of discontinuity were identified. As such, it is screened out with a HCLPF of 0.3g PGA.

OBSERVATION: Hydrogen Tanks
The hydrogen tank farm is located outside. The tanks themselves are free to translate longitudinally and could credibly break the tubing connected to each tank. This does not pose a significant risk as the tank farm is located outside where the hydrogen gas is free to disperse. Moreover, the tank farm is sufficiently removed from the power block buildings and does not pose a hazard.

OBSERVATION: Waste Oil Tanks
The waste oil room and the waste oil tanks (1/2-5101 & -5102) are rugged and are screened out at HCLPF level of 0.3g PGA.

Inadvertent actuation of fire suppression was studied by means of a walkdown. During the walkdown, care was taken to observe potential for spray-down or release of fire suppression media due to seismic interaction. No such instances were observed at Dresden. In addition, fire control equipment (panels and cabinets) were walked down to ensure they were properly anchored and not subject to potential seismic interactions. The following observations were made:

OBSERVATION: Relays Controlling Fire Suppression System
This item is addressed in Section 4.9.3.1.2. Refer to this section for further discussion.

Seismic degradation of fire suppression systems was reviewed by walking down fire piping and looking for poor structural design features or potential interactions with SMA success path equipment. This was routinely performed for each success path equipment item during the walkdown phase. No such potential interactions were noted except for the fire protection piping in the vicinity of panels D02 and D03-2203-0073A&B. The fire piping header in front of (and overhead) the panels has a somewhat long laterally unsupported span and is made up of mechanical couplings (Victualic manufacture). The fire piping has been seismically evaluated and determined to be adequate (ref. DOC ID# 6012466).

3.5 *Analysis of Containment Performance*

A specific walkdown for containment integrity was conducted on August 23, 1995. The purpose of the containment integrity walkdown is to identify any vulnerabilities associated with early containment failure due to a postulated seismic event. This includes the integrity of the containment itself, isolation systems such as valves, mechanical and electrical penetrations, bypass systems and plant-unique containment systems such as igniters or active seals.

Virtually all power-actuated valves were reviewed either as part of the USI A-46 or IPEEE program efforts. In addition, all other isolation valves along with their associated solenoid valves were at least walked by and no concerns were found. Typical configurations were assessed from both the inside and outside of the drywell. No piping supports were observed to provide a "hard point" so all systems have sufficient flexibility to withstand differential displacement between the reactor building and the drywell containment.

The personnel and equipment hatches were walked down. The personnel air lock and equipment access hatch are rugged with no credible seismic vulnerabilities. No "active" isolation systems are utilized.

The main steam and other mechanical penetrations are welded to the steel containment. Some can accommodate thermal movement and there are also those which experience relatively little thermal stress. No concerns were noted.

Electrical penetration areas are also welded assemblies that are leak tight and exhibit no credible seismic vulnerabilities.

Instrument line penetrations also are welded configurations with numerous small diameter lines welded to header plates on the penetrations. As with previous penetrations, no plausible seismic vulnerabilities leading to early containment failure could be reasonably postulated.

In summary, from the viewpoint of seismic hazard no early containment failure features could be identified by virtue of the drywell (containment) walkdown.

3.6 IPEEE Relay Evaluation

Introduction

The Guidelines for the IPEEE relay evaluation, contained in NUREG-1407 [3.5], designate Dresden as a "focused scope" plant. A focused scope plant that is conducting an A-46 review is required to only conduct a bad actor search of the relays controlling IPEEE equipment. The IPEEE bad actor search was conducted for equipment and relays exclusive to the IPEEE program. Equipment items common to both the IPEEE and A-46 programs were evaluated under the A-46 relay evaluation [3.16].

Technical Approach

An IPEEE Only Relay List was developed by examining the control circuits for approximately 130 exclusively IPEEE, electrically controlled equipment (Appendix E1). The list is documented on Relay Screening and Evaluation Tabulation Sheets (Appendix E2) The term "exclusively IPEEE" denotes that this safe shutdown equipment was either not on the USI A-46 SSEL, and therefore was identified solely to satisfy IPEEE seismic requirements, or the equipment was on the USI A-46 SSEL, but required a separate relay review for it's IPEEE function(s), which was not bounded by the USI A-46 review. A detailed circuit analysis was performed on the IPEEE equipment to determine the IPEEE Only Essential relays (Appendix E2) in order to screen for bad actor (low ruggedness) devices.

Finally, the cabinets housing the IPEEE-only relays (Appendix E3) were inspected for seismic adequacy by a seismic review team as part of the USI A-46 resolution following the guidance provided in the GIP [3.14]. The results of this review are included in the USI-A46 submittal report [3.7].

IPEEE Relay Evaluation Results

More than 1500 (approximately 780 Unit 2 and common systems and approximately 790 Unit 3 systems) contacts were identified during this study. Four bad actor (low ruggedness) relays per unit were found and were replaced by Modification M12-2(3)-94-002. Based on a review of the modification, this modification has been designed in accordance with current seismic design criteria. The remaining low ruggedness relays that were identified during the USI A-46 review will be resolved by replacement, or by additional evaluation. Appendix F1 provides a listing of these relays.

An initial screening of SQUG/IPEEE relays associated with the Isolation Condenser System, which was added subsequent to the USI A-46 submittal, had been completed at the time of the original submittal of this report. The initial screening was for the purpose of identifying essential contacts. Invulnerable and chatter-acceptable contacts were screened out. Completion of the evaluation of the remaining contacts is pending relay walkdowns. Therefore, the list of these relays is considered an open item and is included in this report as Appendix F2.

3.7 *USI A-45, GI-131 and Other Seismic Safety Issues*

GI-131 Flux Mapping Cart
This generic issue deals with mobile flux mapping carts designed by Westinghouse Corporation and is not applicable to Dresden Station, a boiling water reactor.

Charleston Earthquake Issue (GL 88-20)
The NRC states in response to industry question 7.13 on page D-13 of NUREG 1407 [3.1]: "The issue of the 1886 Charleston earthquake has been resolved. The issue of eight outlier plants identified through the Eastern U.S. Seismicity program has been subsumed in the IPEEE and no specific reporting is required to close this issue."

USI A-45 Shutdown Decay Heat Removal Requirements
This effort has been subsumed in the USI A-46 program as these components make up part of the SSEL list of equipment (and, thus, the SMA Success Path) and have been reviewed in detail.

3.8 Resolution of Open Items

The December 1997 IPEEE Submittal to the NRC identified open items that still required assessment in order to resolve seismic concerns and to determine their HCLPFs. Subsequently, all open items have been assessed with the exception of relays that must be walked down as plant status permits. The results are provided below:

Equipment ID	Description	Capacity (PGA)
Relays	Relays	Inaccessible relays need to be walked down
D02-0902-0040-P05, D02-0902-0008-P05, 2-0902-18, D03-0903-0040-P05, 3-0903-18	Control Panels	>0.3g
D02-2202-0028-P05, D02-2202-0076-P05, D03-2203-0076-P05	Instrument Panels	>0.3g
D02-1301-0003-V20, D03-1301-0003-V20	Condensate return gate valves	0.3g
D02-1340-0002-LI, D03-1340-0002-LI	Level Indicators	0.3g

The following equipment has been deleted from the open items list, as they were evaluated at the time of the December 1997 submittal and included as outliers in Table 3.3 of that report:

- MCCs D02-7820-02---M05 and D02-7820-03---M-5.
- AOVs D02-8501-0005A---V05 and D02-8501-0005B---V05
- Cable Tray and Conduit Raceway System LAR009
- Unit 2 Turbine Building RACE10

3.9 References

3.1 USNRC, NUREG-1407 "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities", June 1991.

3.2 USNRC, NUREG/CR-0098 "Development of Criteria for Seismic Review of Selected Nuclear Power Plants, May 1978.

3.3 Commonwealth Edison Company, Dresden Final Safety Analysis, Updated.

3.4 "Seismic Design Criteria for Dresden Station - Units 2 and 3 Reactor-Turbine Building", by Sargent & Lundy, Project No. 7355-00, DC-SE-002-DR, Rev. 2, December 5, 1988.

3.5 Jack R. Benjamin & Associates, et. al., "A Methodology for Assessment of Nuclear Power Plant Seismic Margin (Rev 1)", EPRI NP-6041-SL, August 1991.

3.6 USNRC, "Damping Values for Seismic Design of Nuclear Power Plants", USNRC Regulatory Guide 1.61, October, 1973.

3.7 Stevenson & Associates, "Resolution of USI A-46 Seismic Evaluation Report Dresden Nuclear Power Station, Units 2 and 3", June, 1996

3.8 Dresden Nuclear Power Station Units 2 and 3 Design Report for Suppression Chamber, Vent System, Vacuum Breaker Header, and Internal Structures", Revision 1, Pacific Nuclear Systems, Inc. January, 1994.

3.9 NDIT No. SEC-DR-96-003 dated 2/23/95, Bechtel Power Calculations for USNRC IE Bulletin No. 80-11 for Dresden Masonry Walls 40,41,47, 50-54, 82-84, 93, 100, 112-115, 120-121.

3.10 Stevenson & Associates, "HCLPF for Dresden Block Walls ", Calculation 93C2806.04-C-004, Rev. 0, November 1996

3.11 Stevenson & Associates, "HCLPF Analysis of Selected Components", Calculation 93C2806.04-C-005, Rev. 0, November 1996

3.12 Bechtel Corporation, Masonry Wall Calculations C9K-108B, 1981.

3.13 USNRC, "NRC Generic Letter 88-20, Supplement 5", September 8, 1995.

3.14 SQUG, "Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment", Revision 2, Corrected, 2/14/92.

3.15 American Concrete Institute, 1995 ACI Manual of Concrete Practice Part 5, Building Code Requirements for Masonry Structures (ACI 530-92 / ASCE 5-92 / TMS 402-92).

3.16 Relay Evaluation Report for the Resolution of Generic Letter 87-02, prepared by Sargent & Lundy Engineers, dated June, 1996.

3.17 Dresden Nuclear Generating Station Units 2 and 3 Individual Plant Examination Submittal Report, Revision 1, June 1996

3.18 Sargent & Lundy, "In-Structure Seismic Response Spectra for SMA Commonwealth Edison Company Dresden Nuclear Power Station Units 2 & 3, Dresden Nuclear Power Station, Units 2 & 3" , May, 1995

3.19 USNRC, Generic Letter (GL) 87-02, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46," February 19, 1987.

3.20 USNRC, Generic Letter (GL) 87-02, Supplement 1, "Supplemental Safety Evaluation Report No. 2 (SSER No. 2) on SQUG Generic Implementation Procedure, Revision 2, as corrected on February 14, 1992 (GIP-2)," May 22, 1992.

Table 3.1 - Masonry Block Wall Seismic Capacities

ID	Equipment Adjacent	Seismic Capacity HCLPF (PGA)
D2-549-32F-47	D02-9802-2ANEGB05	0.5g
D2-549-32F-47	D02-9802-2APOSB05	0.5g
D2-549-32F-47	D02-9802-2BNEGB05	0.5g
D2-549-32F-47	D02-9802-2BPOSB05	0.5g
D2-549-32F-47	D00-83250-0---B05	0.5g
D2-549-32F-47	D02-83250-2---B05	0.5g
D3-517-49J-93	D03-2203-0008	>0.3g
D2-534-33H-22	D00-5741-0048AV72	0.38g
D2-534-33H-22	D00-5741-0048BV72	0.38g
D2-517-31F-82	D02-0902-0028	0.3g
D2-517-32G-84	D03-0903-0028	0.3g
D2-517-32G-84	D03-0903-0041	0.3g
D2-517-31G-105	D03-2203-0070A	>0.3g
D2-517-31G-105	D03-2203-0070B	>0.3g
VENTILATION ROOM	D00-9400-0104BF05	Screened Out > 0.3g (externally reinforced)
VENTILATION ROOM	D00-5741-0056-D05	Screened Out > 0.3g (externally reinforced)
VENTILATION ROOM	D00-5741-0058-D05	Screened Out > 0.3g (externally reinforced)
VENTILATION ROOM	D00-5741-0059BD05	Screened Out > 0.3g (externally reinforced)
VENTILATION ROOM	D00-9400-0100-F05	Screened Out > 0.3g (externally reinforced)
VENTILATION ROOM	D00-9400-0101-F10	Screened Out > 0.3g (externally reinforced)
D2-549-32G-50	D02-83250-----M05	0.40g
D2-545-40N-41	D02-67231-----S35	0.31g
D2-545-40N-41	D02-67241-----S35	0.31g
D3-545-48N-40	D03-67331-----S35	0.31g
D3-545-48N-40	D03-67341-----S35	0.31g
#51 to #54	D02-9802-A----B04	0.40g
#51 to #54	D02-9802-B----B04	0.40g
#100	D00-5202------T05	0.26g
#114 and #115	D03-5202------T05	0.38g and 0.32g, respectively
#112 and #113	D02-5202------T05	0.47g and 0.46g, respectively

Table 3.2 - Success Path Equipment List Seismic Margin Capacities

I D	Description	SMA Seismic Capacity (PGA) Equm't	Anchor-age	Block Wall	Mini-mum	A-46 SSEL	Outlier
D02-5746-A----H15	LPCI/ LPCI Emergency Room Air Cooler	.3g	>0.3G	NA	.3g	x	x
D03-5746-A----H15	LPCI/ LPCI Emergency Room Air Cooler	.3g	>0.3G	NA	.3g	x	x
D02-5746-B----H15	LPCI/ LPCI Emergency Room Air Cooler	.3g	>0.3G	NA	.3g	x	x
D03-5746-B----H15	LPCI/ LPCI Emergency Room Air Cooler	.3g	>0.3G	NA	.3g	x	x
D02-0902-0004	CONTROL PANELS/ Control Panel 902-4	.3g	0.22g	NA	0.22g	x	x
D02-0902-0015	CONTROL PANELS/ Control Panel 902-15	.3g	0.22g	NA	0.22g	x	x
D02-0902-0017	CONTROL PANELS/ Control Panel 902-17	.3g	0.22g	NA	0.22g	x	x
D03-0903-0003	CONTROL PANELS/ Control Panel 903-3	.3g	0.61g	NA	.3g	X	X
D03-0903-0015	CONTROL PANELS/ Control Panel 903-15	.3g	0.22g	NA	0.22g	x	X
D03-0903-0017	CONTROL PANELS/ Control Panel 903-17	.3g	0.22g	NA	0.22g	X	x
D02-0902-0019	CONTROL PANELS/ Control Panel 902-19	.3g	0.22g	NA	0.22g	X	X
D02-0902-0036	CONTROL PANELS/ Control Panel 902-36	.3g	0.22g	NA	0.22g	X	X
D03-0903-0019	CONTROL PANELS/ Control Panel 903-19	.3g	0.22g	NA	0.22g	X	X
D03-0903-0036	CONTROL PANELS/ Control Panel 903-36	.3g	0.22g	NA	0.22g	X	X
D02-0902-0003	CONTROL PANELS/ Control Panel 902-3	.3g	0.23g	NA	0.23g	X	X
D03-0903-0004	CONTROL PANELS/ Control Panel 903-4	.3g	0.22g	NA	0.22g	x	X
D02-83125-1---P06	125V DC/ TB Res Bus #2, Feed to TB Res Bus #2B-1	.3g	0.29g	NA	0.29g	x	X
D02-8302B-1---P06	125V DC/ TB Res Bus #2B-1 (ROB-TB Res Bus #2B)	.3g	0.29g	NA	0.29g	x	X
D02-83250-2---B05	250V DC/ Battery Charger #2	.3g	.3g	0.5g	.3g	x	X
D00-83250-0---B05	250V DC/ Battery Charger #2/3	.3g	.3g	0.5g	.3g	x	X
D03-8300--3---B05	125V DC/ Battery Charger #3	.3g	.3g	NA	.3g	x	X
D02-8302A-----M05	250V DC/ MCC Bus #2A (ROB-RB MCC #2)	.3g	0.17g	NA	0.17g	x	X
D02-8302B-----M05	250V DC/ MCC Bus #2B (ROB-RB MCC #2)	.3g	0.17g	0.40g	0.17g	x	X
D03-8303A-----M05	250V DC/ MCC Bus #3A (ROB-RB MCC #3)	.3g	0.17g	NA	0.17g	x	X
D03-8303B-----M05	250V DC/ MCC Bus #3B (ROB-RB MCC #3)	.3g	0.17g	NA	0.17g	x	x
D02-83125-----P06	125V DC/ RB 125V DC Distribution Panel #2	.3g	0.17g	NA	0.17g	x	x
D03-83125-----P06	125V DC/ RB 125V DC Distribution Panel #3	.3g	0.17g	NA	0.17g	x	X
D02-2252-0010	CONTROL PANEL/ DG Metering and Relay Cabinet	.3g	0.17g	NA	0.17g	x	X
D00-3303-A----T05	CONDENSATE/ Contaminated Condensate Storage Tank	NA	0.2g	NA	0.2g	x	X
D00-3303-B----T05	CONDENSATE/ Contaminated Condensate Storage Tank	NA	0.2g	NA	0.2g	x	X
D03-83250-3---B05	250V DC/ Battery Charger #3	.3g	.3g	NA	.3g	x	X
D03-0903-0028	CONTROL PANELS/ Control Panel 903-28	.3g	0.23g	0.3g	0.23g	x	
D02-0902-0028	CONTROL PANELS/ Control Panel 902-28	.3g	0.23g	0.3g	0.23g	x	X
D00-5202------T05	DIESEL GENERATOR/ Diesel Fuel Oil Storage Day Tank	NA	0.61g	0.26g	0.26g	x	x
D02-8302A1----P06	125V DC/ TB Main Bus #2A-1 (ROB-Main Bus #2A)	.3g	0.27g	NA	0.27g	x	
D02-83125-2---P06	125V DC/ TB Battery Bus #2, Feed to Main Bus #2A-1	.3g	0.27g	NA	0.27g	x	
D02-9802-B----P06	24/48V DC/ Distribution Panel #2B	.3g	0.27g	NA	0.27g	x	
D02-9802-A----P06	24/48V DC/ Distribution Panel #2A	.3g	0.27g	NA	0.27g	x	x
D02-7329------S35	480V AC/ Switchgear 29	.3g	0.27g	NA	0.27g	x	x
D02-7328------S35	480V AC/ Switchgear 28	.3g	0.27g	NA	0.27g	x	x
D02-8300--2---B05	125V DC/ Battery Charger #2	.3g	0.27g	NA	0.27g	x	
D02-8300--2A--B05	125V DC/ Battery Charger #2A	.3g	0.27g	NA	0.27g	x	
D03-8300--3A--B05	125V DC/ Battery Charger #3A	.3g	0.28g	NA	0.28g	x	
D02-2252-0021	CONTROL PANEL/ DG Excitation Cabinet	.3g	0.28g	NA	0.28g	x	x
D02-83250-----M05	250V DC/ TB MCC #2	.3g	0.29g	NA	0.29g	x	x
D02-7826-4----M05	480V AC/ MCC 26-4	.3g	0.29g	NA	0.29g	x	
D02-0305------H20	CRD/ Hydraulic Control Unit	.3g	0.36g	NA	.3g	x	x
D03-0305------H20	CRD/ Hydraulic Control Unit	.3g	0.36g	NA	.3g	x	x
D03-8303B1----P06	125V DC/ TB Res Bus #3B-1 (ROB-TB Res Bus #3B)	.3g	0.46g	NA	.3g	x	x
D02-7829-4----M05	480V AC/ MCC 29-4	.3g	0.32g	NA	.3g	x	x
D03-8303A1-1--P06	125V DC/ TB Main Bus #3A, Feed to Res Bus #2	.3g	0.46g	NA	.3g	x	x
D03-83125-3---P06	125V DC/ TB Battery Bus #3, Feed to Main Bus #3A	.3g	0.52g	NA	.3g	x	x
D03-83250-----M05	250V DC/ TB MCC #3	.3g	0.52g	NA	.3g	x	

I D	Description	SMA Seismic Capacity (PGA)				A-46 SSEL	Outlier
		Equm't	Anchor-age	Block Wall	Mini-mum		
D02-7828-1----M05	480V AC/ MCC 28-1	.3g	0.32g	NA	.3g	x	
D02-7828-7----M05	480V AC/ MCC 28-7	.3g	0.32g	NA	.3g	x	x
D03-7839-1----M05	480V AC/ MCC 39-1	.3g	0.32g	NA	.3g	x	
D03-7838-1----M05	480V AC/ MCC 38-1	.3g	0.32g	NA	.3g	x	
D03-7838-4----M05	480V AC/ MCC 38-4	.3g	0.32g	NA	.3g	x	
D03-7838-7----M05	480V AC/ MCC 38-7	.3g	0.32g	NA	.3g	x	
D03-8303A1-2--P06	125V DC/ TB Main Bus #3A-1 (ROB-Main Bus #3A)	.3g	0.46g	NA	.3g	x	x
D03-7839-2----M05	480V AC/ MCC 39-2	.3g	0.44g	NA	.3g	x	x
D02-7829-2----M05	480V AC/ MCC 29-2	.3g	0.85g	NA	.3g	x	x
D02-7828-2----M05	480V AC/ MCC 28-2	.3g	0.49g	NA	.3g	x	
D02-7828-3----M05	480V AC/ MCC 28-3	.3g	0.49g	NA	.3g	x	x
D03-7838-2----M05	480V AC/ MCC 38-2	.3g	0.52g	NA	.3g	x	x
D03-7838-3----M05	480V AC/ MCC 38-3	.3g	0.49g	NA	.3g	x	x
D03-9802-A----P06	24/48V DC/ Distribution Panel #3A	.3g	0.46g	NA	.3g	x	x
D03-9802-B----P06	24/48V DC/ Distribution Panel #3B	.3g	0.46g	NA	.3g	x	x
D02-7829-8----M05	480V AC/ MCC 29-8	.3g	0.44g	NA	.3g	x	x
D02-7829-1----M05	480V AC/ MCC 29-1	.3g	0.32g	NA	.3g	x	
D03-7839-7----M05	480V AC/ MCC 39-7	.3g	0.32g	NA	.3g	x	
D02-7829-7----M05	480V AC/ MCC 29-7	.3g	0.32g	NA	.3g	x	
D03-7339------S35	480V AC/ 480V Switchgear 39	.3g	>0.3g	NA	.3g	x	x
D03-7338------S35	480V AC/ Switchgear 38	.3g	>0.3g	NA	.3g	x	x
D03-67341-----S35	4160V AC/ Switchgear 34-1	.3g	>0.3g	0.31g	.3g	x	x
D02-67241-----S35	4160V AC/ Switchgear 24-1	.3g	>0.3g	0.31g	.3g	x	x
D02-67231-----S35	4160V AC/ Switchgear 23-1	.3g	>0.3g	0.31g	.3g	x	x
D03-67331-----S35	4160V AC/ Switchgear 33-1	.3g	>0.3g	0.31g	.3g	x	x
D02-6723------S35	4160V AC/ Switchgear 23	.3g	>0.3g	NA	.3g	x	x
D02-6724------S35	4160V AC/ Switchgear 24	.3g	>0.3g	NA	.3g	x	x
D03-6733------S35	4160V AC/ Switchgear 33	.3g	>0.3g	NA	.3g	x	
D03-6734------S35	4160V AC/ Switchgear 34	.3g	>0.3g	NA	.3g	x	x
D00-6740------S35	4160V AC/ Switchgear 40	.3g	0.7g	NA	.3g	x	
D03-7239------T10	480V AC/ 480V Transformer 39, Feed to Switchgear 39	.3g	>0.3g	NA	.3g	x	
D02-7229------T10	480V AC/ Transformer 29, Feed to Switchgear 29	.3g	>0.3g	NA	.3g	x	x
D03-7138------T10	480V AC/ Transformer 38, Feed to Switchgear 38	.3g	>0.3g	NA	.3g	x	
D02-7128------T10	480V AC/ Transformer 28, Feed to Switchgear 28	.3g	>0.3g	NA	.3g	x	
2/3-4002-A	Refuse Pumps	.3g	>.3g	NA	.3g	x	
2/3-4002-B	Refuse Pumps	.3g	>.3g	NA	.3g	x	
D02-2302------P30	HPCI/ HPCI Pump	.3g	5.02g	NA	.3g	x	
D02-2301------T20	HPCI/ HPCI Turbine	.3g	1.54g	NA	.3g	x	
D02-2301-0057-P30	HPCI/ HPCI Turbine Cooling Water Pump	.3g	1.54g	NA	.3g	x	
D03-2302------P30	HPCI/ HPCI Pump	.3g	5.02g	NA	.3g	x	
D03-2301------T20	HPCI/ HPCI Turbine	.3g	1.54g	NA	.3g	x	
D03-2301-0057-P30	HPCI/ HPCI Turbine Cooling Water Pump	.3g	1.54g	NA	.3g	x	
D03-3903------P30	SERVICE WATER/ Diesel Generator Cooling Water Pump	.3g	2.34g	NA	.3g	x	
D02-3903------P30	SERVICE WATER/ Diesel Generator Cooling Water Pump	.3g	2.34g	NA	.3g	x	
D02-5203------P30	DIESEL GENERATOR/ Fuel Oil Transfer Pump	.3g	>0.3g	NA	.3g	x	x
D00-5203------P30	DIESEL GENERATOR/ Fuel Oil Transfer Pump	.3g	>0.3g	NA	.3g	x	
D03-5203------P30	DIESEL GENERATOR/ Fuel Oil Transfer Pump	.3g	>0.3g	NA	.3g	x	x
D02-1501-0044CP30	"CCSW/ CCSW Pump ""C"""	.3g	6.57g	NA	.3g	x	
D02-1501-0044DP30	"CCSW/ CCSW Pump ""D"""	.3g	6.89g	NA	.3g	x	
D02-1501-0044AP30	"CCSW/ CCSW Pump ""A"""	.3g	8.58g	NA	.3g	x	
D02-1501-0044BP30	"CCSW/ CCSW Pump ""B"""	.3g	8.58g	NA	.3g	x	
D03-1501-0044CP30	"CCSW/ CCSW Pump ""C"""	.3g	8.58g	NA	.3g	x	x
D03-1501-0044DP30	"CCSW/ CCSW Pump ""D"""	.3g	8.58g	NA	.3g	x	x
D03-1501-0044AP30	"CCSW/ CCSW Pump ""A"""	.3g	8.58g	NA	.3g	x	x

I D	Description	SMA Seismic Capacity (PGA) Equm't	 Anchor-age	 Block Wall	 Mini-mum	A-46 SSEL	Outlier
D03-1501-0044BP30	"CCSW/ CCSW Pump ""B"""	.3g	8.58g	NA	.3g	x	x
D00-3903------P30	SERVICE WATER/ Diesel Generator Cooling Water Pump	.3g	2.34g	NA	.3g	x	
D02-2301-CONDPP	HPCI/ Condenser Hotwell Condensate Pump	.3g	1.54g	NA	.3g	x	
D03-2301-CONDPP	HPCI/ Condenser Hotwell Condensate Pump	.3g	1.54g	NA	.3g	x	
D03-1502-B----P30	"LPCI/ LPCI Injection Pump ""B"""	.3g	1.85g	NA	.3g	x	
D02-1502-D----P30	"LPCI/ LPCI Injection Pump ""D"""	.3g	1.17g	NA	.3g	x	x
D02-1502-C----P30	"LPCI/ LPCI Injection Pump ""C"""	.3g	1.17g	NA	.3g	x	x
D03-1502-A----P30	"LPCI/ LPCI Injection Pump ""A"""	.3g	2.12g	NA	.3g	x	
D02-1502-B----P30	"LPCI/ LPCI Injection Pump ""B"""	.3g	2.70g	NA	.3g	x	x
D02-1502-A----P30	"LPCI/ LPCI Injection Pump ""A"""	.3g	2.70g	NA	.3g	x	x
D03-1502-C----P30	"LPCI/ LPCI Injection Pump ""C"""	.3g	1.63g	NA	.3g	x	
D03-1502-D----P30	"LPCI/ LPCI Injection Pump ""D"""	.3g	0.54g	NA	.3g	x	
D02-1301-0017-V05	Steam Line Globe Valves	.3g	NA	NA	.3g	x	
D02-1301-0020-V05	Steam Line Globe Valves	.3g	NA	NA	.3g	x	
D03-1301-0017-V05	Steam Line Globe Valves	.3g	NA	NA	.3g	x	
D03-1301-0020-V05	Steam Line Globe Valves	.3g	NA	NA	.3g	x	
D02-1301-0017-V27	Solenoid Valve	.3g	NA	NA	.3g	x	
D02-1301-0020-V27	Solenoid Valve	.3g	NA	NA	.3g	x	
D03-1301-0017-V27	Solenoid Valve	.3g	NA	NA	.3g	x	
D03-1301-0020-V27	Solenoid Valve	.3g	NA	NA	.3g	x	
D02-0203-0004AV26	ADS/ Reactor Overpressure Relief Valve	.3g	NA	NA	.3g	x	
D03-0203-0004AV26	ADS/ Reactor Overpressure Relief Valve	.3g	NA	NA	.3g	x	
D02-0203-0004BV26	ADS/ Reactor Overpressure Relief Valve	.3g	NA	NA	.3g	x	
D03-0203-0004BV26	ADS/ Reactor Overpressure Relief Valve	.3g	NA	NA	.3g	x	
D02-0203-0004CV26	ADS/ Reactor Overpressure Relief Valve	.3g	NA	NA	.3g	x	
D02-0203-0004EV26	ADS/ Reactor Overpressure Relief Valve	.3g	NA	NA	.3g	x	
D02-0203-0004FV26	ADS/ Reactor Overpressure Relief Valve	.3g	NA	NA	.3g	x	
D02-0203-0004GV26	ADS/ Reactor Overpressure Relief Valve	.3g	NA	NA	.3g	x	
D02-0203-0004HV26	ADS/ Reactor Overpressure Relief Valve	.3g	NA	NA	.3g	x	
D03-0203-0004CV26	ADS/ Reactor Overpressure Relief Valve	.3g	NA	NA	.3g	x	
D03-0203-0004DV26	ADS/ Reactor Overpressure Relief Valve	.3g	NA	NA	.3g	x	
D03-0203-0004EV26	ADS/ Reactor Overpressure Relief Valve	.3g	NA	NA	.3g	x	
D03-0203-0004FV26	ADS/ Reactor Overpressure Relief Valve	.3g	NA	NA	.3g	x	
D03-0203-0004GV26	ADS/ Reactor Overpressure Relief Valve	.3g	NA	NA	.3g	x	
D03-0203-0004HV26	ADS/ Reactor Overpressure Relief Valve	.3g	NA	NA	.3g	x	
D02-0203-0003AV26	ADS/ Target Rock Valve	.3g	NA	NA	.3g	x	
D03-0203-0003AV26	ADS/ Target Rock Valve	.3g	NA	NA	.3g	x	
D02-0203-0001AV05	MAIN STEAM/ Isolation Valve	.3g	NA	NA	.3g	x	
D02-0203-0001BV05	MAIN STEAM/ Isolation Valve	.3g	NA	NA	.3g	x	
D02-0203-0001CV05	MAIN STEAM/ Isolation Valve	.3g	NA	NA	.3g	x	
D02-0203-0001DV05	MAIN STEAM/ Isolation Valve	.3g	NA	NA	.3g	x	
D03-0203-0001AV05	MAIN STEAM/ Isolation Valve	.3g	NA	NA	.3g	x	
D03-0203-0001BV05	MAIN STEAM/ Isolation Valve	.3g	NA	NA	.3g		
D03-0203-0001CV05	MAIN STEAM/ Isolation Valve	.3g	NA	NA	.3g		
D03-0203-0001DV05	MAIN STEAM/ Isolation Valve	.3g	NA	NA	.3g		
D02-0203-0004DV26	ADS/ Reactor Overpressure Relief Valve	.3g	NA	NA	.3g	x	
D02-0203-0002AV05	MAIN STEAM/ Isolation Valve	.3g	NA	NA	.3g		

I D	Description	SMA Seismic Capacity (PGA) Equm't	Anchor-age	Block Wall	Mini-mum	A-46 SSEL	Outlier
D02-0302-0161AV05	CRD/ West Bank Scram Discharge Volume Vent Valve	.3g	NA	NA	.3g	x	
D02-0302-0160AV05	CRD/ West Bank Scram Discharge Volume Vent Valve	.3g	NA	NA	.3g	x	
D03-0302-0161AV05	CRD/ East Bank Scram Discharge Volume Vent Valve	.3g	NA	NA	.3g	x	
D03-0302-0160AV05	CRD/ East Bank Scram Discharge Volume Vent Valve	.3g	NA	NA	.3g	x	
D02-0302-0161BV05	CRD/ East Bank Scram Discharge Volume Vent Valve	.3g	NA	NA	.3g	x	
D02-0302-0160BV05	CRD/ East Bank Scram Discharge Volume Vent Valve	.3g	NA	NA	.3g	x	
D03-0302-0161BV05	CRD/ West Bank Scram Discharge Volume Vent Valve	.3g	NA	NA	.3g	x	
D03-0302-0160BV05	CRD/ West Bank Scram Discharge Volume Vent Valve	.3g	NA	NA	.3g	x	
D02-0302-0157AV05	CRD/ West Bank Scram Discharge Volume Drain Valve	.3g	NA	NA	.3g	x	
D02-0302-0158AV05	CRD/ West Bank Scram Discharge Volume Drain Valve	.3g	NA	NA	.3g	x	
D02-0302-0157BV05	CRD/ East Bank Scram Discharge Volume Drain Valve	.3g	NA	NA	.3g	x	
D02-0302-0158BV05	CRD/ East Bank Scram Discharge Volume Drain Valve	.3g	NA	NA	.3g	x	
D03-0302-0157AV05	CRD/ East Bank Scram Discharge Volume Drain Valve	.3g	NA	NA	.3g	x	x
D03-0302-0158AV05	CRD/ East Bank Scram Discharge Volume Drain Valve	.3g	NA	NA	.3g	x	
D03-0302-0157BV05	CRD/ West Bank Scram Discharge Volume Drain Valve	.3g	NA	NA	.3g	x	
D03-0302-0158BV05	CRD/ West Bank Scram Discharge Volume Drain Valve	.3g	NA	NA	.3g	x	
D00-5741-0048BV72	CONTROL ROOM VENTILATION/ Service Water Supply Valve	.3g	NA	0.38g	.3g	x	
D00-5741-0048AV72	CONTROL ROOM VENTILATION/ CCSW Cooling Supply Valve	.3g	NA	0.38g	.3g	x	
D02-0203-0002BV05	MAIN STEAM/ Isolation Valve	.3g	NA	NA	.3g		
D02-0203-0002CV05	MAIN STEAM/ Isolation Valve	.3g	NA	NA	.3g		
D02-0203-0002DV05	MAIN STEAM/ Isolation Valve	.3g	NA	NA	.3g		
D03-0203-0002AV05	MAIN STEAM/ Isolation Valve	.3g	NA	NA	.3g		
D03-0203-0002BV05	MAIN STEAM/ Isolation Valve	.3g	NA	NA	.3g		x
D03-0203-0002CV05	MAIN STEAM/ Isolation Valve	.3g	NA	NA	.3g		
D03-0203-0002DV05	MAIN STEAM/ Isolation Valve	.3g	NA	NA	.3g		
D02-0220-0044-V05	REACTOR RECIRCULATION/ Recirc Loop Sample Line Valve	.3g	NA	NA	.3g		
D02-0220-0045-V05	REACTOR RECIRCULATION/ Recirc Loop Sample Line Valve	.3g	NA	NA	.3g		
D03-0220-0044-V05	REACTOR RECIRCULATION/ Recirc Loop Sample Line Valve	.3g	NA	NA	.3g		
D03-0220-0045-V05	REACTOR RECIRCULATION/ Recirc Loop Sample Line Valve	.3g	NA	NA	.3g		
D03-2001-0005-V05	RB EQUIPMENT DRAIN/ Drywell Equipment Drain Line Valve	.3g	NA	NA	.3g		
D03-2001-0006-V05	RB EQUIPMENT DRAIN/ Drywell Equipment Drain Line Valve	.3g	NA	NA	.3g		
D03-2001-0105-V05	RB EQUIPMENT DRAIN/ Drywell Floor Drain Line Valve	.3g	NA	NA	.3g		
D03-2001-0106-V05	RB EQUIPMENT DRAIN/ Drywell Floor Drain Line Valve	.3g	NA	NA	.3g		
D02-2001-0005-V05	RB EQUIPMENT DRAIN/ Drywell Equipment Drain Line Valve	.3g	NA	NA	.3g		

I D	Description	SMA Seismic Capacity (PGA) Equm't	Anchor-age	Block Wall	Mini-mum	A-46 SSEL	Outlier
D02-2001-0006-V05	RB EQUIPMENT DRAIN/ Drywell Equipment Drain Line Valve	.3g	NA	NA	.3g		
D02-2001-0105-V05	RB EQUIPMENT DRAIN/ Drywell Floor Drain Line Valve	.3g	NA	NA	.3g		
D02-2001-0106-V05	RB EQUIPMENT DRAIN/ Drywell Floor Drain Line Valve	.3g	NA	NA	.3g		
D03-1601-0023-V05	PRESSURE SUPPRESSION/ Drywell Ventilation Line Valve	.3g	NA	NA	.3g		
D03-1601-0024-V05	PRESSURE SUPPRESSION/ Drywell and Torus Vent Valve	.3g	NA	NA	.3g		
D02-1601-0062-V05	PRESSURE SUPPRESSION/ Drywell Ventilation Line Valve	.3g	NA	NA	.3g		
D02-1601-0063-V05	PRESSURE SUPPRESSION/ Containment to SBGT Line Valve	.3g	NA	NA	.3g		
D02-1601-0023-V05	PRESSURE SUPPRESSION/ Drywell Ventilation Line Valve	.3g	NA	NA	.3g		
D02-1601-0024-V05	PRESSURE SUPPRESSION/ Drywell and Torus Vent Valve	.3g	NA	NA	.3g		
D03-1601-0062-V05	PRESSURE SUPPRESSION/ Drywell Ventilation Line Valve	.3g	NA	NA	.3g		
D03-1601-0063-V05	PRESSURE SUPPRESSION/ Containment to SBGT Line Valve	.3g	NA	NA	.3g		
D03-1601-0060-V05	PRESSURE SUPPRESSION/ Torus Ventilation Line Valve	.3g	NA	NA	.3g		
D03-1601-0061-V05	PRESSURE SUPPRESSION/ Torus Ventilation Line Valve	.3g	NA	NA	.3g		
D02-1601-0060-V05	PRESSURE SUPPRESSION/ Torus Ventilation Line Valve	.3g	NA	NA	.3g		
D02-1601-0061-V05	PRESSURE SUPPRESSION/ Torus Ventilation Line Valve	.3g	NA	NA	.3g		
D03-1601-0021-V05	PRESSURE SUPPRESSION/ Drywell Purge Line Valve	.3g	NA	NA	.3g		
D03-1601-0022-V05	PRESSURE SUPPRESSION/ Drywell/Torus Purge Line Valve	.3g	NA	NA	.3g		
D03-1601-0055-V05	PRESSURE SUPPRESSION/ Drywell Purge Inerting Line Valve	.3g	NA	NA	.3g		
D03-1601-0056-V05	PRESSURE SUPPRESSION/ Torus Purge and Inert Line Valve	.3g	NA	NA	.3g		
D03-1601-0058-V05	PRESSURE SUPPRESSION/ Torus Nitrogen Makeup Line Valve	.3g	NA	NA	.3g		
D02-1601-0021-V05	PRESSURE SUPPRESSION/ Drywell Purge Line Valve	.3g	NA	NA	.3g		
D02-1601-0022-V05	PRESSURE SUPPRESSION/ Drywell/Torus Purge Line Valve	.3g	NA	NA	.3g		
D02-1601-0055-V05	PRESSURE SUPPRESSION/ Drywell Purge Inerting Line Valve	.3g	NA	NA	.3g		
D02-1601-0056-V05	PRESSURE SUPPRESSION/ Torus Purge and Inert Line Valve	.3g	NA	NA	.3g		
D02-1601-0059-V05	PRESSURE SUPPRESSION/ Drywell Nitr. Makeup Line Valve	.3g	NA	NA	.3g		
D03-1601-0059-V05	PRESSURE SUPPRESSION/ Drywell Nitr. Makeup Line Valve	.3g	NA	NA	.3g		
D03-8501-0001AV05	CONTAINMENT SAMPLING/ Torus Oxygen Sampling Line Valve	.3g	NA	NA	.3g		
D03-8501-0001BV05	CONTAINMENT SAMPLING/ Torus Oxygen Sampling Line Valve	.3g	NA	NA	.3g		
D02-8501-0003AV05	CONTAINMENT SAMPLING/ Drywell Oxy. Sampling Line Valve	.3g	NA	NA	.3g		
D03-8501-0003BV05	CONTAINMENT SAMPLING/ Drywell Oxy. Sampling Line Valve	.3g	NA	NA	.3g		
D02-8501-0005AV05	CONTAINMENT SAMPLING/ Drywell Oxy. Sampling Line Valve	.3g	NA	NA	.3g		

I D	Description	SMA Seismic Capacity (PGA) Equm't	Anchor-age	Block Wall	Mini-mum	A-46 SSEL	Outlier
D03-8501-0005BV05	CONTAINMENT SAMPLING/ Drywell Oxy. Sampling Line Valve	.3g	NA	NA	.3g		
D02-9205-A----V05	CONTAINMENT SAMPLING/ Drywell Oxy. Sampling Line Valve	.3g	NA	NA	.3g		x
D03-9205-A----V05	CONTAINMENT SAMPLING/ Drywell Oxy. Sampling Line Valve	.3g	NA	NA	.3g		
D03-9206-A----V05	CONTAINMENT SAMPLING/ Drywell Oxy. Sampling Line Valve	.3g	NA	NA	.3g		x
D02-9206-B----V05	CONTAINMENT SAMPLING/ Drywell Oxy. Sampling Line Valve	.3g	NA	NA	.3g		x
D03-9208-A----V05	CONTAINMENT SAMPLING/ Drywell Sampling Line Valve	.3g	NA	NA	.3g		
D03-9207-B----V05	CONTAINMENT SAMPLING/ Drywell Sampling Line Valve	.3g	NA	NA	.3g		
D02-9207-A----V05	CONTAINMENT SAMPLING/ Drywell Sampling Line Valve	.3g	NA	NA	.3g		
D02-9208-B----V05	CONTAINMENT SAMPLING/ Drywell Sampling Line Valve	.3g	NA	NA	.3g		
D02-8501-0001AV05	CONTAINMENT SAMPLING/ Torus Oxygen Sampling Line Valve	.3g	NA	NA	.3g		
D02-8501-0001BV05	CONTAINMENT SAMPLING/ Torus Oxygen Sampling Line Valve	.3g	NA	NA	.3g		
D03-8501-0003AV05	CONTAINMENT SAMPLING/ Drywell Oxy. Sampling Line Valve	.3g	NA	NA	.3g		
D02-8501-0003BV05	CONTAINMENT SAMPLING/ Drywell Oxy. Sampling Line Valve	.3g	NA	NA	.3g		
D03-8501-0005AV05	CONTAINMENT SAMPLING/ Drywell Oxy. Sampling Line Valve	.3g	NA	NA	.3g		
D02-8501-0005BV05	CONTAINMENT SAMPLING/ Drywell Oxy. Sampling Line Valve	.3g	NA	NA	.3g		
D02-9205-B----V05	CONTAINMENT SAMPLING/ Drywell Oxy. Sampling Line Valve	.3g	NA	NA	.3g		x
D02-9206-A----V05	CONTAINMENT SAMPLING/ Drywell Oxy. Sampling Line Valve	.3g	NA	NA	.3g		x
D03-9206-B----V05	CONTAINMENT SAMPLING/ Drywell Oxy. Sampling Line Valve	.3g	NA	NA	.3g		x
D03-9207-A----V05	CONTAINMENT SAMPLING/ Drywell Sampling Line Valve	.3g	NA	NA	.3g		
D02-9207-B----V05	CONTAINMENT SAMPLING/ Drywell Sampling Line Valve	.3g	NA	NA	.3g		
D02-9208-A----V05	CONTAINMENT SAMPLING/ Drywell Sampling Line Valve	.3g	NA	NA	.3g		
D03-9208-B----V05	CONTAINMENT SAMPLING/ Drywell Sampling Line Valve	.3g	NA	NA	.3g		
D03-4720------V05	INSTRUMENT AIR/ Drywell Pneumatic Supply Valve	.3g	NA	NA	.3g		
D03-4721------V05	INSTRUMENT AIR/ Drywell Pneumatic Supply Valve	.3g	NA	NA	.3g		
D02-4720------V05	INSTRUMENT AIR/ Drywell Pneumatic Supply Valve	.3g	NA	NA	.3g		
D02-4721------V05	INSTRUMENT AIR/ Drywell Pneumatic Supply Valve	.3g	NA	NA	.3g		
D02-1601-0058-V05	PRESSURE SUPPRESSION/ Torus Nitrogen Makeup Line Valve	.3g	NA	NA	.3g		
D03-9205-B----V05	CONTAINMENT SAMPLING/ Drywell Oxy. Sampling Line Valve	.3g	NA	NA	.3g		
D02-2301-0036-V20	HPCI/ Suppression Pool Suction Line Valve	.3g	NA	NA	.3g	x	
D02-2301-0035-V20	HPCI/ Suppression Pool Suction Line Valve	.3g	NA	NA	.3g	x	
D02-2301-0006-V20	HPCI/ Condensate Tank Supply to HPCI Pump Valve	.3g	NA	NA	.3g	x	
D03-2301-0004-V20	HPCI/ Turbine Steam Line Valve	.3g	NA	NA	.3g		
D02-2301-0005-V20	HPCI/ Turbine Steam Line Valve	.3g	NA	NA	.3g	x	x
D03-2301-0003-V20	HPCI/ Turbine Steam Line Valve	.3g	NA	NA	.3g	x	
D02-2301-0014-V20	HPCI/ HPCI Pump Test Line Valve	.3g	NA	NA	.3g	x	
D02-2301-0008-V20	HPCI/ HPCI Pump Injection Line Valve	.3g	NA	NA	.3g	x	
D03-2301-0006-V20	HPCI/ Condensate Tank Supply to HPCI Pump Valve	.3g	NA	NA	.3g	x	
D03-2301-0036-V20	HPCI/ Suppression Pool Suction Line Valve	.3g	NA	NA	.3g	x	

I D	Description	SMA Seismic Capacity (PGA) Equm't	Anchor-age	Block Wall	Mini-mum	A-46 SSEL	Outlier
D03-2301-0035-V20	HPCI/ Suppression Pool Suction Line Valve	.3g	NA	NA	.3g	x	
D03-2301-0008-V20	HPCI/ HPCI Pump Injection Line Valve	.3g	NA	NA	.3g	x	x
D03-2301-0014-V20	HPCI/ HPCI Pump Test Line Valve	.3g	NA	NA	.3g	x	
D02-2301-0004-V20	HPCI/ Turbine Steam Line Valve	.3g	NA	NA	.3g		
D03-2301-0005-V20	HPCI/ Turbine Steam Line Valve	.3g	NA	NA	.3g	x	x
D02-2301-0003-V20	HPCI/ Turbine Steam Line Valve	.3g	NA	NA	.3g	x	
D03-1501-0005BV20	"LPCI/ Suppression Pool Suction Line ""B"" Valve"	.3g	NA	NA	.3g		
D03-1501-0005AV20	"LPCI/ Suppression Pool Suction Line ""A"" Valve"	.3g	NA	NA	.3g		
D02-1501-0005CV20	"LPCI/ Suppression Pool Suction Line ""C"" Valve"	.3g	NA	NA	.3g		
D02-1501-0005AV20	"LPCI/ Suppression Pool Suction Line ""A"" Valve"	.3g	NA	NA	.3g		
D02-1501-0011BV20	LPCI/ LPCI Heat Exchanger Bypass Line Valve	.3g	NA	NA	.3g	x	
D03-1501-0011BV20	LPCI/ LPCI Heat Exchanger Bypass Line Valve	.3g	NA	NA	.3g	x	
D02-1501-0032BV20	LPCI/ LPCI Header Crosstie Line Valve	.3g	NA	NA	.3g	x	x
D03-1501-0032AV20	LPCI/ LPCI Header Crosstie Line Valve	.3g	NA	NA	.3g	x	
D02-1501-0038BV20	LPCI/ Suppression Chamber Spray Line Valve	.3g	NA	NA	.3g	x	
D03-1501-0038BV20	LPCI/ Suppression Chamber Spray Line Valve	.3g	NA	NA	.3g	x	
D03-1501-0020BV20	LPCI/ Suppression Chamber Spray Line Valve	.3g	NA	NA	.3g	x	
D02-1501-0020BV20	LPCI/ Suppression Chamber Spray Line Valve	.3g	NA	NA	.3g	x	
D03-1501-0022AV20	LPCI/ LPCI Injection Line Valve	.3g	NA	NA	.3g	x	
D03-1501-0013BV20	LPCI/ LPCI Minimum Flow Bypass Line Valve	.3g	NA	NA	.3g	x	
D02-0203-0003BV26	ADS/ Electromatic Relief Valve	.3g	NA	NA	.3g	x	
D02-0203-0003EV26	ADS/ Electromatic Relief Valve	.3g	NA	NA	.3g	x	
D02-0203-0003CV26	ADS/ Electromatic Relief Valve	.3g	NA	NA	.3g	x	
D02-0203-0003DV26	ADS/ Electromatic Relief Valve	.3g	NA	NA	.3g	x	
D03-0203-0003BV26	ADS/ Electromatic Relief Valve	.3g	NA	NA	.3g	x	
D03-0203-0003EV26	ADS/ Electromatic Relief Valve	.3g	NA	NA	.3g	x	
D03-0203-0003CV26	ADS/ Electromatic Relief Valve	.3g	NA	NA	.3g	x	
D03-0203-0003DV26	ADS/ Electromatic Relief Valve	.3g	NA	NA	.3g	x	
D02-1501-0005BV20	"LPCI/ Suppression Pool Suction Line ""B"" Valve"	.3g	NA	NA	.3g		
D02-1501-0005DV20	"LPCI/ Suppression Pool Suction Line ""D"" Valve"	.3g	NA	NA	.3g		
D02-1501-0011AV20	LPCI/ LPCI Heat Exchanger Bypass Line Valve	.3g	NA	NA	.3g	x	
D02-1501-0032AV20	LPCI/ LPCI Header Crosstie Line Valve	.3g	NA	NA	.3g	x	
D02-1501-0038AV20	LPCI/ Suppression Chamber Spray Line Valve	.3g	NA	NA	.3g	x	
D02-1501-0022AV20	LPCI/ LPCI Injection Line Valve	.3g	NA	NA	.3g	x	
D02-1501-0013BV20	LPCI/ LPCI Minimum Flow Bypass Line Valve	.3g	NA	NA	.3g	x	
D03-1501-0011AV20	LPCI/ LPCI Heat Exchanger Bypass Line Valve	.3g	NA	NA	.3g	x	
D03-1501-0038AV20	LPCI/ Suppression Chamber Spray Line Valve	.3g	NA	NA	.3g	x	
D02-1501-0020AV20	LPCI/ Suppression Chamber Spray Line Valve	.3g	NA	NA	.3g	x	
D02-1501-0013AV20	LPCI/ LPCI Minimum Flow Bypass Line Valve	.3g	NA	NA	.3g	x	
D03-1501-0005CV20	"LPCI/ Suppression Pool Suction Line ""C"" Valve"	.3g	NA	NA	.3g		
D03-1501-0005DV20	"LPCI/ Suppression Pool Suction Line ""D"" Valve"	.3g	NA	NA	.3g		
D03-1501-0032BV20	LPCI/ LPCI Header Crosstie Line Valve	.3g	NA	NA	.3g	x	x
D03-1501-0020AV20	LPCI/ Suppression Chamber Spray Line Valve	.3g	NA	NA	.3g	x	
D03-1501-0013AV20	LPCI/ LPCI Minimum Flow Bypass Line Valve	.3g	NA	NA	.3g	x	
D02-1501-0003BV20	CCSW/ Heat Exchanger Outlet Service Water Line Valve	.3g	NA	NA	.3g	x	
D02-1501-0003AV20	CCSW/ Heat Exchanger Outlet Service Water Line Valve	.3g	NA	NA	.3g	x	
D03-1501-0003AV20	CCSW/ Heat Exchanger Outlet Service Water Line Valve	.3g	NA	NA	.3g	x	
D03-1501-0003BV20	CCSW/ Heat Exchanger Outlet Service Water Line Valve	.3g	NA	NA	.3g	x	

I D	Description	SMA Seismic Capacity (PGA) Equm't	Anchor-age	Block Wall	Mini-mum	A-46 SSEL	Outlier
D02-0202-0005AV20	REACTOR RECIRCULATION/ Recirc Pump A Discharge Valve	.3g	NA	NA	.3g	x	
D03-0202-0005AV20	REACTOR RECIRCULATION/ Recirc Pump A Discharge Valve	.3g	NA	NA	.3g	x	
D02-0202-0009AV20	REACTOR RECIRCULATION/ Recirc Crosstie Bypass Valve	.3g	NA	NA	.3g	x	
D02-0302-0020AV27	CRD/ Scram Dump Solenoid Valve	.3g	NA	NA	.3g	x	
D02-0302-0020BV27	CRD/ Scram Dump Solenoid Valve	.3g	NA	NA	.3g	x	
D03-0302-0020AV25	CRD/ Scram Dump Solenoid Valve	.3g	NA	NA	.3g	x	
D03-0302-0020BV27	CRD/ Scram Dump Solenoid Valve	.3g	NA	NA	.3g	x	
D02-0302-0019AV27	CRD/ Backup Scram Solenoid Valve	.3g	NA	NA	.3g	x	x
D03-0302-0019AV27	CRD/ Backup Scram Solenoid Valve	.3g	NA	NA	.3g	x	
D03-0302-0019BV27	CRD/ Backup Scram Solenoid Valve	.3g	NA	NA	.3g	x	
D00-5790--EP2-V27	DIESEL GENERATOR/ Vent. Fan Dampers Solenoid Valve	.3g	NA	NA	.3g	x	
D02-5790--EP2-V27	DIESEL GENERATOR/ Room Vent. Fan Dampers Solenoid Valve	.3g	NA	NA	.3g	x	
D02-5790--EP3-V27	DIESEL GENERATOR/ Normal Vent. Damper Solenoid Valve	.3g	NA	NA	.3g	x	
D03-5790--EP2-V27	DIESEL GENERATOR/ Room Vent. Fan Dampers Solenoid Valve	.3g	NA	NA	.3g	x	
D03-5790--EP3-V27	DIESEL GENERATOR/ Normal Vent. Damper Solenoid Valve	.3g	NA	NA	.3g	x	
D03-1301-0001-V20	ISOLATION CONDENSER/ Steam Line Isolation Valve	.3g	NA	NA	.3g		
D03-1301-0002-V20	ISOLATION CONDENSER/ Steam Line Isolation Valve	.3g	NA	NA	.3g		
D03-1301-0004-V20	ISOLATION CONDENSER/ Steam Return Line Isolation Valve	.3g	NA	NA	.3g		
D02-1301-0001-V20	ISOLATION CONDENSER/ Steam Line Isolation Valve	.3g	NA	NA	.3g		
D02-1301-0002-V20	ISOLATION CONDENSER/ Steam Line Isolation Valve	.3g	NA	NA	.3g		
D02-1301-0004-V20	ISOLATION CONDENSER/ Steam Return Line Isolation Valve	.3g	NA	NA	.3g		
D02-1601-0057-V20	PRESSURE SUPPRESSION/ Drywell/Torus Nitr. Makeup Valve	.3g	NA	NA	.3g		x
D03-1201-0001-V20	REACTOR WATER CLEAN UP/ Aux Pump Suction Line Valve	.3g	NA	NA	.3g		
D03-1201-0002-V20	REACTOR WATER CLEAN UP/ Pump Suction Bypass Line Valve	.3g	NA	NA	.3g		
D02-1201-0001-V20	REACTOR WATER CLEAN UP/ Aux Pump Suction Line Valve	.3g	NA	NA	.3g		
D02-1201-0002-V20	REACTOR WATER CLEAN UP/ Pump Suction Bypass Line Valve	.3g	NA	NA	.3g		
D03-1402-0003AV20	CORE SPRAY/ Pump Injection Line Valve	.3g	NA	NA	.3g		
D03-1402-0003BV20	CORE SPRAY/ Pump Injection Line Valve	.3g	NA	NA	.3g		
D03-1402-0024AV20	CORE SPRAY/ Pump Discharge Injection Line Valve	.3g	NA	NA	.3g		
D02-1402-0024BV20	CORE SPRAY/ Pump Discharge Injection Line Valve	.3g	NA	NA	.3g		
D02-1402-0003AV20	CORE SPRAY/ Pump Injection Line Valve	.3g	NA	NA	.3g		
D02-1402-0003BV20	CORE SPRAY/ Pump Injection Line Valve	.3g	NA	NA	.3g		
D02-1402-0024AV20	CORE SPRAY/ Pump Discharge Injection Line Valve	.3g	NA	NA	.3g		
D03-1402-0024BV20	CORE SPRAY/ Pump Discharge Injection Line Valve	.3g	NA	NA	.3g		
D02-0302-0019BV27	CRD/ Backup Scram Solenoid Valve	.3g	NA	NA	.3g	x	
D03-1601-0057-V20	PRESSURE SUPPRESSION/ Drywell/Torus Nitr. Makeup Valve	.3g	NA	NA	.3g		x
D02-0220-0044-V27	REACTOR RECIRCULATION/ Recirc Loop Line Solenoid Valve	.3g	NA	NA	.3g		
D02-0220-0045-V27	REACTOR RECIRCULATION/ Recirc Loop Line Solenoid Valve	.3g	NA	NA	.3g		
D02-2001-0005-V27	RB EQUIPMENT DRAIN/ Drywell Equipment Solenoid Valve	.3g	NA	NA	.3g		

I D	Description	SMA Seismic Capacity (PGA) Equm't	Anchor-age	Block Wall	Mini-mum	A-46 SSEL	Outlier
D02-2001-0006-V27	RB EQUIPMENT DRAIN/ Drywell Equipment Solenoid Valve	.3g	NA	NA	.3g		
D02-2001-0105-V27	RB EQUIPMENT DRAIN/ Drywell Floor Drain Solenoid Valve	.3g	NA	NA	.3g		
D02-2001-0106-V27	RB EQUIPMENT DRAIN/ Drywell Floor Drain Solenoid Valve	.3g	NA	NA	.3g		
D02-1601-0023-V27	PRESSURE SUPPRESSION/ Drywell Vent. Solenoid Valve	.3g	NA	NA	.3g		x
D02-1601-0024-V27	PRESSURE SUPPRESSION/ Drywell/Torus Solenoid Valve	.3g	NA	NA	.3g		
D02-1601-0062-V27	PRESSURE SUPPRESSION/ Drywell Vent. Solenoid Valve	.3g	NA	NA	.3g		
D02-1601-0063-V27	PRESSURE SUPPRESSION/ Containment to SBGT Solen. Valve	.3g	NA	NA	.3g		
D02-1601-0060-V27	PRESSURE SUPPRESSION/ Torus Vent. Solenoid Valve	.3g	NA	NA	.3g		
D02-1601-0061-V27	PRESSURE SUPPRESSION/ Torus Vent. Solenoid Valve	.3g	NA	NA	.3g		
D02-1601-0021-V27	PRESSURE SUPPRESSION/ Drywell Purge Line Solenoid Valve	.3g	NA	NA	.3g		
D02-1601-0022-V27	PRESSURE SUPPRESSION/ Drywell/Torus Purge Solen. Valve	.3g	NA	NA	.3g		
D02-1601-0055-V27	PRESSURE SUPPRESSION/ Drywell Purge Inert. Solen. Valve	.3g	NA	NA	.3g		
D02-1601-0056-V27	PRESSURE SUPPRESSION/ Torus Purge/Inert Solenoid Valve	.3g	NA	NA	.3g		
D02-1601-0058-V27	PRESSURE SUPPRESSION/ Torus Nitrogen Make. Solen. Valve	.3g	NA	NA	.3g		
D02-1601-0059-V27	PRESSURE SUPPRESSION/ Drywell Nitr. Makeup Solen. Valve	.3g	NA	NA	.3g		
D02-8501-0001AV27	CONTAINMENT SAMPLING/ Torus Oxygen Sampl. Solen. Valve	.3g	NA	NA	.3g		
D02-8501-0001BV27	CONTAINMENT SAMPLING/ Torus Oxygen Sampl. Solen. Valve	.3g	NA	NA	.3g		
D02-8501-0003AV27	CONTAINMENT SAMPLING/ Drywell Oxy. Sampl. Solen. Valve	.3g	NA	NA	.3g		
D02-8501-0003BV27	CONTAINMENT SAMPLING/ Drywell Oxy. Sampl. Solen. Valve	.3g	NA	NA	.3g		
D02-8501-0005AV27	CONTAINMENT SAMPLING/ Drywell Oxy. Sampl. Solen. Valve	.3g	NA	NA	.3g		x
D02-8501-0005BV27	CONTAINMENT SAMPLING/ Drywell Oxy. Sampl. Solen. Valve	.3g	NA	NA	.3g		x
D02-9205-A----V27	CONTAINMENT SAMPLING/ Drywell Oxy. Sampl. Solen. Valve	.3g	NA	NA	.3g		
D02-9205-B----V27	CONTAINMENT SAMPLING/ Drywell Oxy. Sampl. Solen. Valve	.3g	NA	NA	.3g		
D02-9206-B----V27	CONTAINMENT SAMPLING/ Drywell Oxy. Sampl. Solen. Valve	.3g	NA	NA	.3g		
D02-9206-A----V27	CONTAINMENT SAMPLING/ Drywell Oxy. Sampl. Solen. Valve	.3g	NA	NA	.3g		
D02-9207-A----V27	CONTAINMENT SAMPLING/ Drywell Sampling Solenoid Valve	.3g	NA	NA	.3g		
D02-9207-B----V27	CONTAINMENT SAMPLING/ Drywell Sampling Solenoid Valve	.3g	NA	NA	.3g		
D02-9208-A----V27	CONTAINMENT SAMPLING/ Drywell Sampling Solenoid Valve	.3g	NA	NA	.3g		
D02-9208-B----V27	CONTAINMENT SAMPLING/ Drywell Sampling Solenoid Valve	.3g	NA	NA	.3g		
D03-0220-0044-V27	REACTOR RECIRCULATION/ Recirc Loop Line Solenoid Valve	.3g	NA	NA	.3g		
D03-0220-0045-V27	REACTOR RECIRCULATION/ Recirc Loop Line Solenoid Valve	.3g	NA	NA	.3g		

I D	Description	SMA Seismic Capacity (PGA) Equm't	Anchor-age	Block Wall	Mini-mum	A-46 SSEL	Outlier
D03-2001-0005-V27	RB EQUIPMENT DRAIN/ Drywell Equipment Solenoid Valve	.3g	NA	NA	.3g		
D03-2001-0006-V27	RB EQUIPMENT DRAIN/ Drywell Equipment Solenoid Valve	.3g	NA	NA	.3g		
D03-2001-0105-V27	RB EQUIPMENT DRAIN/ Drywell Floor Drain Solenoid Valve	.3g	NA	NA	.3g		
D03-2001-0106-V27	RB EQUIPMENT DRAIN/ Drywell Floor Drain Solenoid Valve	.3g	NA	NA	.3g		
D03-1601-0023-V27	PRESSURE SUPPRESSION/ Drywell Vent. Solenoid Valve	.3g	NA	NA	.3g		
D03-1601-0024-V27	PRESSURE SUPPRESSION/ Drywell/Torus Solenoid Valve	.3g	NA	NA	.3g		
D03-1601-0062-V27	PRESSURE SUPPRESSION/ Drywell Vent. Solenoid Valve	.3g	NA	NA	.3g		
D03-1601-0063-V27	PRESSURE SUPPRESSION/ Containment to SBGT Solen. Valve	.3g	NA	NA	.3g		
D03-1601-0060-V27	PRESSURE SUPPRESSION/ Torus Vent. Solenoid Valve	.3g	NA	NA	.3g		
D03-1601-0061-V27	PRESSURE SUPPRESSION/ Torus Vent. Solenoid Valve	.3g	NA	NA	.3g		
D03-1601-0021-V27	PRESSURE SUPPRESSION/ Drywell Purge Line Solenoid Valve	.3g	NA	NA	.3g		
D03-1601-0022-V27	PRESSURE SUPPRESSION/ Drywell/Torus Purge Solen. Valve	.3g	NA	NA	.3g		
D03-1601-0055-V27	PRESSURE SUPPRESSION/ Drywell Purge Inert. Solen. Valve	.3g	NA	NA	.3g		
D03-1601-0056-V27	PRESSURE SUPPRESSION/ Torus Purge/Inert Solenoid Valve	.3g	NA	NA	.3g		
D03-1601-0058-V27	PRESSURE SUPPRESSION/ Torus Nitrogen Make. Solen. Valve	.3g	NA	NA	.3g		
D03-1601-0059-V27	PRESSURE SUPPRESSION/ Drywell Nitr. Makeup Solen. Valve	.3g	NA	NA	.3g		
D03-8501-0001AV27	CONTAINMENT SAMPLING/ Torus Oxygen Sampl. Solen. Valve	.3g	NA	NA	.3g		
D03-8501-0001BV27	CONTAINMENT SAMPLING/ Torus Oxygen Sampl. Solen. Valve	.3g	NA	NA	.3g		
D03-8501-0003AV27	CONTAINMENT SAMPLING/ Drywell Oxy. Sampl. Solen. Valve	.3g	NA	NA	.3g		
D03-8501-0003BV27	CONTAINMENT SAMPLING/ Drywell Oxy. Sampl. Solen. Valve	.3g	NA	NA	.3g		
D03-8501-0005AV27	CONTAINMENT SAMPLING/ Drywell Oxy. Sampl. Solen. Valve	.3g	NA	NA	.3g		
D03-8501-0005BV27	CONTAINMENT SAMPLING/ Drywell Oxy. Sampl. Solen. Valve	.3g	NA	NA	.3g		
D03-9205-A----V27	CONTAINMENT SAMPLING/ Drywell Oxy. Sampl. Solen. Valve	.3g	NA	NA	.3g		
D03-9205-B----V27	CONTAINMENT SAMPLING/ Drywell Oxy. Sampl. Solen. Valve	.3g	NA	NA	.3g		
D03-9206-A----V27	CONTAINMENT SAMPLING/ Drywell Oxy. Sampl. Solen. Valve	.3g	NA	NA	.3g		
D03-9206-B----V27	CONTAINMENT SAMPLING/ Drywell Oxy. Sampl. Solen. Valve	.3g	NA	NA	.3g		
D03-9207-A----V27	CONTAINMENT SAMPLING/ Drywell Sampling Solenoid Valve	.3g	NA	NA	.3g		
D03-9207-B----V27	CONTAINMENT SAMPLING/ Drywell Sampling Solenoid Valve	.3g	NA	NA	.3g		
D03-9208-A----V27	CONTAINMENT SAMPLING/ Drywell Sampling Solenoid Valve	.3g	NA	NA	.3g		
D03-9208-B----V27	CONTAINMENT SAMPLING/ Drywell Sampling Solenoid Valve	.3g	NA	NA	.3g		
D02-0203-001A1V27	MAIN STEAM/ Isolation DC Solenoid Valve	.3g	NA	NA	.3g		
D02-0203-001A2V27	MAIN STEAM/ Isolation AC Solenoid Valve	.3g	NA	NA	.3g		
D02-0203-001B1V27	MAIN STEAM/ Isolation DC Solenoid Valve	.3g	NA	NA	.3g		

I D	Description	SMA Seismic Capacity (PGA) Equm't	Anchor-age	Block Wall	Mini-mum	A-46 SSEL	Outlier
D02-0203-001B2V27	MAIN STEAM/ Isolation AC Solenoid Valve	.3g	NA	NA	.3g		
D02-0203-001C1V27	MAIN STEAM/ Isolation DC Solenoid Valve	.3g	NA	NA	.3g		
D02-0203-001C2V27	MAIN STEAM/ Isolation AC Solenoid Valve	.3g	NA	NA	.3g		
D02-0203-001D1V27	MAIN STEAM/ Isolation DC Solenoid Valve	.3g	NA	NA	.3g		
D02-0203-001D2V27	MAIN STEAM/ Isolation AC Solenoid Valve	.3g	NA	NA	.3g		
D02-0203-002A1V27	MAIN STEAM/ Isolation DC Solenoid Valve	.3g	NA	NA	.3g		
D02-0203-002A2V27	MAIN STEAM/ Isolation AC Solenoid Valve	.3g	NA	NA	.3g		
D02-0203-002B1V27	MAIN STEAM/ Isolation DC Solenoid Valve	.3g	NA	NA	.3g		
D02-0203-002B2V27	MAIN STEAM/ Isolation AC Solenoid Valve	.3g	NA	NA	.3g		
D02-0203-002D1V27	MAIN STEAM/ Isolation DC Solenoid Valve	.3g	NA	NA	.3g		
D02-0203-002C2V27	MAIN STEAM/ Isolation AC Solenoid Valve	.3g	NA	NA	.3g		
D02-0203-002C1V27	MAIN STEAM/ Isolation DC Solenoid Valve	.3g	NA	NA	.3g		
D02-0203-002D2V27	MAIN STEAM/ Isolation AC Solenoid Valve	.3g	NA	NA	.3g		
D03-0203-001A1V27	MAIN STEAM/ Isolation DC Solenoid Valve	.3g	NA	NA	.3g		
D03-0203-001A2V27	MAIN STEAM/ Isolation AC Solenoid Valve	.3g	NA	NA	.3g		
D03-0203-001B1V27	MAIN STEAM/ Isolation DC Solenoid Valve	.3g	NA	NA	.3g		
D03-0203-001B2V27	MAIN STEAM/ Isolation AC Solenoid Valve	.3g	NA	NA	.3g		
D03-0203-001C1V27	MAIN STEAM/ Isolation DC Solenoid Valve	.3g	NA	NA	.3g		
D03-0203-001C2V27	MAIN STEAM/ Isolation AC Solenoid Valve	.3g	NA	NA	.3g		
D03-0203-001D1V27	MAIN STEAM/ Isolation DC Solenoid Valve	.3g	NA	NA	.3g		
D03-0203-001D2V27	MAIN STEAM/ Isolation AC Solenoid Valve	.3g	NA	NA	.3g		
D03-0203-002B1V27	MAIN STEAM/ Isolation DC Solenoid Valve	.3g	NA	NA	.3g		
D03-0203-002B2V27	MAIN STEAM/ Isolation AC Solenoid Valve	.3g	NA	NA	.3g		
D03-0203-002A1V27	MAIN STEAM/ Isolation DC Solenoid Valve	.3g	NA	NA	.3g		
D03-0203-002A2V27	MAIN STEAM/ Isolation AC Solenoid Valve	.3g	NA	NA	.3g		
D03-0203-002C1V27	MAIN STEAM/ Isolation DC Solenoid Valve	.3g	NA	NA	.3g		
D03-0203-002C2V27	MAIN STEAM/ Isolation AC Solenoid Valve	.3g	NA	NA	.3g		
D03-0203-002D1V27	MAIN STEAM/ Isolation DC Solenoid Valve	.3g	NA	NA	.3g		
D03-0203-002D2V27	MAIN STEAM/ Isolation AC Solenoid Valve	.3g	NA	NA	.3g		
D00-5790-0003BV10	DIESEL GENERATOR/ Vent. Fan Outlet Damper Solen. Oper.	.3g	NA	NA	.3g	x	
D00-5790-0003AV10	DIESEL GENERATOR/ Vent. Fan Inlet Damper Solen. Oper.	.3g	NA	NA	.3g	x	
D02-4741-0011-V27	INSTRUMENT AIR/ Drywell Pneumatic Supply Solenoid Valve	.3g	NA	NA	.3g		
D02-4741-0012-V27	INSTRUMENT AIR/ Drywell Pneumatic Supply Solenoid Valve	.3g	NA	NA	.3g		
D03-4741-0011-V27	INSTRUMENT AIR/ Drywell Pneumatic Supply Solenoid Valve	.3g	NA	NA	.3g		
D03-4741-0012-V27	INSTRUMENT AIR/ Drywell Pneumatic Supply Solenoid Valve	.3g	NA	NA	.3g		
D02-5790-0003-V10	DIESEL GENERATOR/ Vent. Fan Damper Solenoid Operator	.3g	NA	NA	.3g	x	
D03-5790-0003-V10	DIESEL GENERATOR/ Vent. Fan Damper Solenoid Operator	.3g	NA	NA	.3g	x	

I D	Description	SMA Seismic Capacity (PGA) Equm't	Anchor-age	Block Wall	Mini-mum	A-46 SSEL	Outlier
D02-0302-0157AV27	CRD/ West Bank Scram Discharge Volume Drain Sol. Valve	.3g	NA	NA	.3g	x	
D02-0302-0158AV27	CRD/ West Bank Scram Discharge Volume Drain Sol. Valve	.3g	NA	NA	.3g	x	
D02-0302-0157BV27	CRD/ East Bank Scram Discharge Volume Drain Sol. Valve	.3g	NA	NA	.3g	x	
D02-0302-0158BV27	CRD/ East Bank Scram Discharge Volume Drain Sol. Valve	.3g	NA	NA	.3g	x	
D03-0302-0157AV27	CRD/ East Bank Scram Discharge Volume Drain Sol. Valve	.3g	NA	NA	.3g	x	
D03-0302-0158AV27	CRD/ East Bank Scram Discharge Volume Drain Sol. Valve	.3g	NA	NA	.3g	x	
D03-0302-0157BV27	CRD/ West Bank Scram Discharge Volume Drain Sol. Valve	.3g	NA	NA	.3g	x	
D03-0302-0158BV27	CRD/ West Bank Scram Discharge Volume Drain Sol. Valve	.3g	NA	NA	.3g	x	
D02-0302-0161AV27	CRD/ West Bank Scram Discharge Volume Vent Sol. Valve	.3g	NA	NA	.3g	x	
D02-0302-0160AV27	CRD/ West Bank Scram Discharge Volume Vent Sol. Valve	.3g	NA	NA	.3g	x	
D02-0302-0161BV27	CRD/ East Bank Scram Discharge Volume Vent Sol. Valve	.3g	NA	NA	.3g	x	
D02-0302-0160BV27	CRD/ East Bank Scram Discharge Volume Vent Sol. Valve	.3g	NA	NA	.3g	x	
D03-0302-0161AV27	CRD/ East Bank Scram Discharge Volume Vent Sol. Valve	.3g	NA	NA	.3g	x	
D03-0302-0160AV27	CRD/ East Bank Scram Discharge Volume Vent Sol. Valve	.3g	NA	NA	.3g	x	
D03-0302-0161BV27	CRD/ West Bank Scram Discharge Volume Vent Sol. Valve	.3g	NA	NA	.3g	x	
D03-0302-0160BV27	CRD/ West Bank Scram Discharge Volume Vent Sol. Valve	.3g	NA	NA	.3g	x	
D00-5741-0054-V27	"CONTROL ROOM VENTILATION/ Train ""A"" Iso. Damper Solen."	.3g	NA	NA	.3g	x	
D02-5790------F10	DIESEL GENERATOR/ Room Ventilation Fan	.3g	>0.3g	NA	.3g	x	
D03-5790------F10	DIESEL GENERATOR/ Room Ventilation Fan	.3g	>0.3g	NA	.3g	x	
D00-9400-0104AF05	CONTROL ROOM VENTILATION/ AFU Booster Fan	.3g	>0.3g	NA	.3g	x	
D00-9400-0104BF05	CONTROL ROOM VENTILATION/ AFU Booster Fan	.3g	>0.3g	>0.3g	.3g	x	
D00-5790------F10	DIESEL GENERATOR/ Room Ventiliation Fan	.3g	>0.3g	NA	.3g	x	
D03-2320-GSCE-F05	HPCI/ Gland Seal Condenser Exhaust Fan	.3g	>0.3g	NA	.3g	x	
D02-2320-GSCE-F05	HPCI/ Gland Seal Condenser Exhaust Fan	.3g	>0.3g	NA	.3g	x	
D03-LOC	HPCI/ Lube Oil Cooler	.3g	>0.3g	NA	.3g	x	
D02-5747------H15	HPCI/ HPCI Emergency Air Cooler	.3g	>0.3g	NA	.3g	x	
D02-LOC	HPCI/ Lube Oil Cooler	.3g	>0.3g	NA	.3g	x	
D03-5747------H15	HPCI/ HPCI Emergency Air Cooler	.3g	>0.3g	NA	.3g	x	
D00-5772-0100-D05	DIESEL GENERATOR/ Ventiliation Fan Inlet Damper	.3g	>0.3g	NA	.3g	x	
D02-5700-0030CH15	"CCSW/ CCSW Pump Cooler ""C"""	.3g	0.44g	NA	.3g	x	x
D02-5700-0030DH15	"CCSW/ CCSW Pump Cooler ""D"""	.3g	0.44g	NA	.3g	x	x
D02-5700-0030AH15	"CCSW/ CCSW Pump Cooler ""A"""	.3g	0.44g	NA	.3g	x	x
D02-5700-0030BH15	"CCSW/ CCSW Pump Cooler ""B"""	.3g	0.44g	NA	.3g	x	x
D03-5700-0030CH15	"CCSW/ CCSW Pump Cooler ""C"""	.3g	0.44g	NA	.3g	x	x
D03-5700-0030DH15	"CCSW/ CCSW Pump Cooler ""D"""	.3g	0.44g	NA	.3g	x	x
D00-5741-0058-D05	CONTROL ROOM VENTILATION/ AFU Inlet Damper	.3g	>0.3g	>0.3g	.3g	x	

I D	Description	SMA Seismic Capacity (PGA) Equm't	Anchor-age	Block Wall	Mini-mum	A-46 SSEL	Outlier
D00-5741-0055-D05	CONTROL ROOM VENTILATION/ AFU Booster Fan Outlet Damper	.3g	>0.3g	NA	.3g	x	
D00-5741-0056-D05	CONTROL ROOM VENTILATION/ AFU Booster Fan Outlet Damper	.3g	>0.3g	>0.3g	.3g	x	
D00-5741-0059BD05	CONTROL ROOM VENTILATION/ AHU Inlet Damper	.3g	>0.3g	>0.3g	.3g	x	
D00-5741-0059AD05	CONTROL ROOM VENTILATION/ AHU Outlet Damper	.3g	>0.3g	NA	.3g	x	
D00-9400-0100-F05	CONTROL ROOM VENTILATION/ Air Handling Unit	.3g	>0.3g	>0.3g	.3g	x	
D00-5741-0054CD05	"CONTROL ROOM VENTILATION/ Train ""A"" Isolation Damper"	.3g	>0.3g	NA	.3g	x	
D00-5741-0054BD05	"CONTROL ROOM VENTILATION/ Train ""A"" Isolation Damper"	.3g	>0.3g	NA	.3g	x	
D00-5741-0057-D05	CONTROL ROOM VENTILATION/ AFU Recirculation Damper	.3g	>0.3g	NA	.3g	x	
D00-5772-0101-D05	DIESEL GENERATOR/ Ventiliation Fan Outlet Damper	.3g	>0.3g	NA	.3g	x	
D02-5772-0100-D05	DIESEL GENERATOR/ Ventiliation Fan Inlet Damper	.3g	>0.3g	NA	.3g	x	
D02-5772-0101-D05	DIESEL GENERATOR/ Ventiliation Fan Outlet Damper	.3g	>0.3g	NA	.3g	x	
D02-5772-0102-D05	DIESEL GENERATOR/ Normal Ventiliation Duct Damper	.3g	>0.3g	NA	.3g	x	
D03-5772-0100-D05	DIESEL GENERATOR/ Ventiliation Fan Inlet Damper	.3g	>0.3g	NA	.3g	x	
D03-5772-0101-D05	DIESEL GENERATOR/ Ventiliation Fan Outlet Damper	.3g	>0.3g	NA	.3g	x	
D03-5772-0102-D05	DIESEL GENERATOR/ Normal Ventiliation Duct Damper	.3g	>0.3g	NA	.3g	x	x
D03-5700-0030AH15	"CCSW/ CCSW Pump Cooler ""A"""	.3g	0.44g	NA	.3g	x	x
D03-5700-0030BH15	"CCSW/ CCSW Pump Cooler ""B"""	.3g	0.44g	NA	.3g	x	x
D00-5741-0054DD05	"CONTROL ROOM VENTILATION/ Train ""A"" Isolation Damper"	.3g	>0.3g	NA	.3g	x	
D00-5741-0054AD05	"CONTROL ROOM VENTILATION/ Train ""A"" Isolation Damper"	.3g	>0.3g	NA	.3g	x	
D00-9400-0101-F10	CONTROL ROOM VENTILATION/ Air Filtration Unit Heater	.3g	0.59g	>0.3g	.3g	x	x
D00-9400-0102-R15	CONTROL ROOM VENTILATION/ Refrigeration Condensing Unit	.3g	1.54g	NA	.3g	x	x
D02-4600-B----T05	"DIESEL GENERATOR/ Primary Gas Air Receiver Unit ""A1"""	.3g	1.1g	NA	.3g	x	
D02-4600-C----T05	"DIESEL GENERATOR/ Primary Gas Air Receiver Unit ""A2"""	.3g	1.1g	NA	.3g	x	
D02-4600-G----T05	"DIESEL GENERATOR/ Primary Gas Air Receiver Unit ""B1"""	.3g	1.1g	NA	.3g	x	
D02-4600-H----T05	"DIESEL GENERATOR/ Primary Gas Air Receiver Unit ""B2"""	.3g	1.1g	NA	.3g	x	
D03-4600-B----T05	"DIESEL GENERATOR/ Primary Gas Air Receiver Unit ""A1"""	.3g	1.1g	NA	.3g	x	
D03-4600-C----T05	"DIESEL GENERATOR/ Primary Gas Air Receiver Unit ""A2"""	.3g	1.1g	NA	.3g	x	
D03-4600-H----T05	"DIESEL GENERATOR/ Primary Gas Air Receiver Unit ""B2"""	.3g	1.1g	NA	.3g	x	
D03-4600-G----T05	"DIESEL GENERATOR/ Primary Gas Air Receiver Unit ""B1"""	.3g	1.1g	NA	.3g	x	
D00-4600-B----T05	"DIESEL GENERATOR/ Primary Gas Air Receiver Unit ""A1"""	.3g	1.1g	NA	.3g	x	
D00-4600-C----T05	"DIESEL GENERATOR/ Primary Gas Air Receiver Unit ""A2"""	.3g	0.9g	NA	.3g	x	
D00-4600-G----T05	"DIESEL GENERATOR/ Primary Gas Air Receiver Unit ""B1"""	.3g	0.9g	NA	.3g	x	
D00-4600-H----T05	"DIESEL GENERATOR/ Primary Gas Air Receiver Unit ""B2"""	.3g	0.9g	NA	.3g	x	

I D	Description	SMA Seismic Capacity (PGA) Equm't	Anchor-age	Block Wall	Mini-mum	A-46 SSEL	Outlier
D02-83250-----B04	250V DC/ Battery #2	.3g	0.34g	NA	.3g	x	x
D03-8300--BC--B04	125V DC/ Battery #3, Feed to TB Battery Bus #3	.3g	0.31g	NA	.3g	x	x
D03-83250-----B04	250V DC/ Battery #3, Feed to TB MCC #3	.3g	0.36g	NA	.3g	x	x
D02-8300--BC--B04	125V DC/ Battery #2, Feed to TB Battery Bus #2	.3g	0.34g	NA	.3g	x	x
D02-9802-B----B04	24/48V DC/ Battery #2B	.3g	0.77g	0.40g	.3g	x	x
D02-9802-A----B04	24/48V DC/ Battery #2A	.3g	0.77g	0.40g	.3g	x	x
D03-9802-A----B04	24/48V DC/ Battery #3A	.3g	0.62g	NA	.3g	x	x
D03-9802-B----B04	24/48V DC/ Battery #3B	.3g	0.62g	NA	.3g	x	x
D02-9802-2APOSB05	24/48V DC/ Battery Charger #2A (+)	.3g	>0.3g	0.5g	.3g	x	
D02-9802-2BPOSB05	24/48V DC/ Battery Charger #2B (+)	.3g	>0.3g	0.5g	.3g	x	
D03-9802-3APOSB05	24/48V DC/ Battery Charger #3A (+)	.3g	>0.3g	NA	.3g	x	
D03-9802-3BPOSB05	24/48V DC/ Battery Charger #3B (+)	.3g	>0.3g	NA	.3g	x	x
D03-9802-3ANEGB05	24/48V DC/ Battery Charger #3A (-)	.3g	>0.3g	NA	.3g	x	
D03-9802-3BNEGB05	24/48V DC/ Battery Charger #3B (-)	.3g	>0.3g	NA	.3g	x	x
D02-9802-2ANEGB05	24/48V DC/ Battery Charger #2A (-)	.3g	>0.3g	0.5g	.3g	x	
D02-9802-2BNEGB05	24/48V DC/ Battery Charger #2B (-)	.3g	>0.3g	0.5g	.3g	x	
D02-6601------G05	DIESEL GENERATOR/ Diesel Engine Driven Generator	.3g	1.24g	NA	.3g	x	
D03-6601------G05	DIESEL GENERATOR/ Diesel Engine Driven Generator	.3g	1.24g	NA	.3g	x	
D00-6601------G05	DIESEL GENERATOR/ Diesel Engine Driven Generator	.3g	0.75g	NA	.3g	x	
D02-0903-0050-P06	Distribution Panel (902-50)	.3g	.3g	NA	.3g	x	
D03-0903-0050-P06	Distribution Panel (903-50)	.3g	.3g	NA	.3g	x	
D02-1341----LT	Level Transmitter	.3g	.3g	NA	.3g	x	
D03-1341----LT	Level Transmitter	.3g	.3g	NA	.3g	x	
D00-2350-A-----LS	HPCI/ Storage Tank Level Switch	.3g	>0.3g	NA	.3g	x	
D00-2350-C-----LS	HPCI/ Storage Tank Level Switch	.3g	>0.3g	NA	.3g	x	
D03-2203-0007	INSTRUMENT RACKS/ Instrument Rack 2203-7	.3g	>0.3g	NA	.3g	x	
D03-2203-0008	INSTRUMENT RACKS/ Instrument Rack 2203-8	.3g	>0.3g	NA	.3g	x	
D03-2203-0006	INSTRUMENT RACKS/ Instrument Rack 2203-6	.3g	>0.3g	NA	.3g	x	x
D03-2203-0005	INSTRUMENT RACKS/ Instrument Rack 2203-5	.3g	>0.3g	NA	.3g	x	
D02-2202-0005	INSTRUMENT RACKS/ Instrument Rack 2202-5	.3g	>0.3g	NA	.3g	x	
D02-2202-0006	INSTRUMENT RACKS/ Instrument Rack 2202-6	.3g	>0.3g	NA	.3g	x	x
D02-2202-0007	INSTRUMENT RACKS/ Instrument Rack 2202-7	.3g	>0.3g	NA	.3g	x	
D02-2202-0008	INSTRUMENT RACKS/ Instrument Rack 2202-8	.3g	>0.3g	NA	.3g	x	x
D02-2202-0036	INSTRUMENT RACKS/ Instrument Rack 2202-36	.3g	>0.3g	NA	.3g	x	
D02-2202-0070A	INSTRUMENT RACKS/ Instrument Rack 2202-70A	.3g	1.1g	NA	.3g	x	x
D02-2202-0073A	INSTRUMENT RACKS/ Instrument Rack 2202-73A	.3g	0.46g	NA	.3g	x	x
D02-2202-0073B	INSTRUMENT RACKS/ Instrument Rack 2202-73B	.3g	0.46g	NA	.3g	x	x
D03-2203-0029	INSTRUMENT RACKS/ Instrument Rack 2203-29	.3g	1.1g	NA	.3g	x	
D03-2203-0032	INSTRUMENT RACKS/ Instrument Rack 2203-32	.3g	>0.3g	NA	.3g	x	
D03-2203-0036	INSTRUMENT RACKS/ Instrument Rack 2203-36	.3g	1.1g	NA	.3g	x	
D03-2203-0073A	INSTRUMENT RACKS/ Instrument Rack 2203-73A	.3g	0.46g	NA	.3g	x	x
D03-2203-0073B	INSTRUMENT RACKS/ Instrument Rack 2203-73B	.3g	0.46g	NA	.3g	x	x
D02-2202-0070B	INSTRUMENT RACKS/ Instrument Rack 2202-70B	.3g	1.1g	NA	.3g	x	x
D03-2203-0070A	INSTRUMENT RACKS/ Instrument Rack 2203-70A	.3g	1.1g	NA	.3g	x	x
D03-2203-0070B	INSTRUMENT RACKS/ Instrument Rack 2203-70B	.3g	1.1g	NA	.3g	x	x
D00-2350-B-----LS	HPCI/ Storage Tank Level Switch	.3g	>0.3g	NA	.3g	x	
D00-2350-D-----LS	HPCI/ Storage Tank Level Switch	.3g	>0.3g	NA	.3g	x	
D02-2351-A-----LS	HPCI/ Torus Water Level Switch	.3g	>0.3g	NA	.3g	x	
D02-2351-B-----LS	HPCI/ Torus Water Level Switch	.3g	>0.3g	NA	.3g	x	
D03-2351-A-----LS	HPCI/ Torus Water Level Switch	.3g	>0.3g	NA	.3g	x	

		SMA Seismic Capacity (PGA)					
I D	Description	Equm't	Anchor-age	Block Wall	Mini-mum	A-46 SSEL	Outlier
D03-2351-B-----LS	HPCI/ Torus Water Level Switch	.3g	>0.3g	NA	.3g	x	
D02-2370-A-----TS	HPCI/ Steam Leak Detection Temperature Switch	.3g	>0.3g	NA	.3g	x	
D02-2370-B-----TS	HPCI/ Steam Leak Detection Temperature Switch	.3g	>0.3g	NA	.3g	x	
D02-2370-C-----TS	HPCI/ Steam Leak Detection Temperature Switch	.3g	>0.3g	NA	.3g	x	
D02-2370-D-----TS	HPCI/ Steam Leak Detection Temperature Switch	.3g	>0.3g	NA	.3g	x	
D02-2371-A-----TS	HPCI/ Steam Leak Detection Temperature Switch	.3g	>0.3g	NA	.3g	x	
D02-2371-B-----TS	HPCI/ Steam Leak Detection Temperature Switch	.3g	>0.3g	NA	.3g	x	
D02-2371-C-----TS	HPCI/ Steam Leak Detection Temperature Switch	.3g	>0.3g	NA	.3g	x	
D02-2371-D-----TS	HPCI/ Steam Leak Detection Temperature Switch	.3g	>0.3g	NA	.3g	x	
D02-2372-A-----TS	HPCI/ Steam Leak Detection Temperature Switch	.3g	>0.3g	NA	.3g	x	
D02-2372-B-----TS	HPCI/ Steam Leak Detection Temperature Switch	.3g	>0.3g	NA	.3g	x	
D02-2372-C-----TS	HPCI/ Steam Leak Detection Temperature Switch	.3g	>0.3g	NA	.3g	x	
D02-2372-D-----TS	HPCI/ Steam Leak Detection Temperature Switch	.3g	>0.3g	NA	.3g	x	
D02-2373-A-----TS	HPCI/ Steam Leak Detection Temperature Switch	.3g	>0.3g	NA	.3g	x	
D02-2373-B-----TS	HPCI/ Steam Leak Detection Temperature Switch	.3g	>0.3g	NA	.3g	x	
D02-2373-C-----TS	HPCI/ Steam Leak Detection Temperature Switch	.3g	>0.3g	NA	.3g	x	
D02-2373-D-----TS	HPCI/ Steam Leak Detection Temperature Switch	.3g	>0.3g	NA	.3g	x	
D03-2370-A-----TS	HPCI/ Steam Leak Detection Temperature Switch	.3g	>0.3g	NA	.3g	x	
D03-2370-B-----TS	HPCI/ Steam Leak Detection Temperature Switch	.3g	>0.3g	NA	.3g	x	
D03-2370-D-----TS	HPCI/ Steam Leak Detection Temperature Switch	.3g	>0.3g	NA	.3g	x	
D03-2370-C-----TS	HPCI/ Steam Leak Detection Temperature Switch	.3g	>0.3g	NA	.3g	x	
D03-2371-A-----TS	HPCI/ Steam Leak Detection Temperature Switch	.3g	>0.3g	NA	.3g	x	
D03-2371-B-----TS	HPCI/ Steam Leak Detection Temperature Switch	.3g	>0.3g	NA	.3g	x	
D03-2371-C-----TS	HPCI/ Steam Leak Detection Temperature Switch	.3g	>0.3g	NA	.3g	x	
D03-2371-D-----TS	HPCI/ Steam Leak Detection Temperature Switch	.3g	>0.3g	NA	.3g	x	
D03-2372-A-----TS	HPCI/ Steam Leak Detection Temperature Switch	.3g	>0.3g	NA	.3g	x	
D03-2372-B-----TS	HPCI/ Steam Leak Detection Temperature Switch	.3g	>0.3g	NA	.3g	x	
D03-2372-C-----TS	HPCI/ Steam Leak Detection Temperature Switch	.3g	>0.3g	NA	.3g	x	
D03-2372-D-----TS	HPCI/ Steam Leak Detection Temperature Switch	.3g	>0.3g	NA	.3g	x	
D03-2373-A-----TS	HPCI/ Steam Leak Detection Temperature Switch	.3g	>0.3g	NA	.3g	x	
D03-2373-B-----TS	HPCI/ Steam Leak Detection Temperature Switch	.3g	>0.3g	NA	.3g	x	
D03-2373-D-----TS	HPCI/ Steam Leak Detection Temperature Switch	.3g	>0.3g	NA	.3g	x	
D03-2373-C-----TS	HPCI/ Steam Leak Detection Temperature Switch	.3g	>0.3g	NA	.3g	x	
D02-2252-0084	INSTRUMENT RACKS/ Instrument Rack 2252-84	.3g	>0.3g	NA	.3g	x	x
D03-2253-0083	INSTRUMENT RACKS/ Instrument Rack 2253-83	.3g	>0.3g	NA	.3g	x	
D03-2253-0084	INSTRUMENT RACKS/ Instrument Rack 2253-84	.3g	>0.3g	NA	.3g	x	
D03-LCS1-------LS	HPCI/ Gland Seal Condenser Drain Pump Level Switch	.3g	>0.3g	NA	.3g	x	
D03-LCS2-------LS	HPCI/ Gland Seal Condenser Drain Pump Level Switch	.3g	>0.3g	NA	.3g	x	
D03-2301-PS4---PS	HPCI/ Emergency Bearing Oil Pump Pressure Switch	.3g	>0.3g	NA	.3g	x	
D02-LCS1-------LS	HPCI/ Gland Seal Condenser Drain Pump Level Switch	.3g	>0.3g	NA	.3g	x	
D02-LCS2-------LS	HPCI/ Gland Seal Condenser Drain Pump Level Switch	.3g	>0.3g	NA	.3g	x	
D02-2301-PS4---PS	HPCI/ Emergency Bearing Oil Pump Pressure Switch	.3g	>0.3g	NA	.3g	x	
D02-2202-0029	INSTRUMENT RACKS/ Instrument Rack 2202-29	.3g	>0.3g	NA	.3g	x	
D03-2203-0075	INSTRUMENT RACKS/ Instrument Rack 2203-75	.3g	>0.3g	NA	.3g	x	
D02-2202-0075	INSTRUMENT RACKS/ Instrument Rack 2202-75	.3g	>0.3g	NA	.3g	x	
D02-2252-0083	INSTRUMENT RACKS/ Instrument Rack 2252-83	.3g	0.52g	NA	.3g	x	
D02-0902-0032	CONTROL PANELS/ Control Panel 902-32	.3g	0.5g	NA	.3g	x	
D02-0902-0033	CONTROL PANELS/ Control Panel 902-33	.3g	0.5g	NA	.3g	x	x
D02-0902-0039	CONTROL PANELS/ Control Panel 902-39	.3g	>0.3g	NA	.3g	x	x
D02-0902-0046	CONTROL PANELS/ Control Panel 902-46	.3g	>0.3g	NA	.3g	x	x
D02-0902-0047	CONTROL PANELS/ Control Panel 902-47	.3g	>0.3g	NA	.3g	x	x
D03-0903-0033	CONTROL PANELS/ Control Panel 903-33	.3g	0.49g	NA	.3g	x	x
D03-0903-0039	CONTROL PANELS/ Control Panel 903-39	.3g	0.59g	NA	.3g	x	
D03-0903-0046	CONTROL PANELS/ Control Panel 903-46	.3g	0.73g	NA	.3g	x	x
D03-0903-0047	CONTROL PANELS/ Control Panel 903-47	.3g	0.73g	NA	.3g	x	x
D03-0903-0032	CONTROL PANELS/ Control Panel 903-32	.3g	0.39g	NA	.3g	x	x
D02-7828-1-1--P06	DISTRIBUTION PANELS/ Distribution Panel 28-1-1	.3g	0.48g	NA	.3g	x	

I D	Description	SMA Seismic Capacity (PGA) Equm't	Anchor-age	Block Wall	Mini-mum	A-46 SSEL	Outlier
D03-7838-1-1--P06	DISTRIBUTION PANELS/ Distribution Panel 38-1-1	.3g	0.48g	NA	.3g	x	x
D00-2223-0109	CONTROL PANEL/ DG Cooling Pump Transfer Switch Status	.3g	>0.3g	NA	.3g	x	
D00-2223-0041	CONTROL PANEL/ DG Excitation Cabinet	.3g	>0.3g	NA	.3g	x	x
D03-2253-0010	CONTROL PANEL/ DG Metering and Relay Cabinet	.3g	1.54g	NA	.3g	x	
D03-2253-0021	CONTROL PANEL/ DG Excitation Cabinet	.3g	0.47g	NA	.3g	x	
D00-2223-0033	CONTROL PANEL/ DG Relaying and Metering Cabinet	.3g	>0.3g	NA	.3g	x	x
D00-9400-0102	CONTROL PANEL/ RCU Control Panel	.3g	1.7g	NA	.3g	x	
D00-9400-0103	CONTROL PANEL/ Control Cabinet 9400-103	.3g	>0.3g	NA	.3g	x	
D00-9400-0105	CONTROL PANEL/ Control Cabinet 9400-105	.3g	1.3g	NA	.3g	x	
D00-ACP	CONTROL PANEL/ Unit 2/3 Auxiliary Control Panel	.3g	>0.3g	NA	.3g	x	
D03-0923-0005	CONTROL PANELS/ Control Panel 923-5	.3g	0.41g	NA	.3g	x	
D02-0902-0061	CONTROL PANELS/ Control Panel 902-61	.3g	>0.3g	NA	.3g	x	
D02-0902-0062	CONTROL PANELS/ Control Panel 902-62	.3g	>0.3g	NA	.3g	x	
D03-0903-0061	CONTROL PANELS/ Control Panel 903-61	.3g	>0.3g	NA	.3g	x	
D03-0903-0062	CONTROL PANELS/ Control Panel 903-62	.3g	>0.3g	NA	.3g	x	
D02-0902-0041	CONTROL PANELS/ Control Panel 902-41	.3g	>0.3g	NA	.3g	x	
D03-0903-0041	CONTROL PANELS/ Control Panel 903-41	.3g	0.33g	0.3g	.3g	x	x
D00-NGC	CONTROL PANEL/ Unit 2/3 Neutral Grounding Cabinet	.3g	>0.3g	NA	.3g	x	x
D00-2223-0053	CONTROL PANEL/ Diesel Generator Fire Protection Panel	.3g	>0.3g	NA	.3g	x	
D02-1342-A----TE	Isolation Condenser Temperature Element	.3g	NA	Na	.3g	x	
D02-1342-B----TE	Isolation Condenser Temperature Element	.3g	NA	Na	.3g	x	
D03-1342-A----TE	Isolation Condenser Temperature Element	.3g	NA	Na	.3g	x	
D03-1342-B----TE	Isolation Condenser Temperature Element	.3g	NA	Na	.3g	x	
D02-1340-1--TR	Chart Recorder	.3g	NA	NA	.3g	x	
D02-1340-1--TR	Chart Recorder	.3g	NA	NA	.3g	x	
D02-1302----H15	Isolation Condenser Heat Exchanger	NA	*.3g*	NA	.3g	x	
D03-1302----H15	Isolation Condenser Heat Exchanger	NA	*.3g*	NA	.3g	x	
D02-2320-GSC--H15	HPCI/ Gland Seal Condenser	NA	>0.3g	NA	.3g	x	x
D03-2320-GSC--H15	HPCI/ Gland Seal Condenser	NA	>0.3g	NA	.3g	x	
D02-0220-0082AA10	MAIN STEAM/ Isolation Valve Accumulator	NA	>0.3g	NA	.3g	x	
D02-0220-0082BA10	MAIN STEAM/ Isolation Valve Accumulator	NA	>0.3g	NA	.3g		
D02-0220-0082CA10	MAIN STEAM/ Isolation Valve Accumulator	NA	>0.3g	NA	.3g		
D02-0220-0082DA10	MAIN STEAM/ Isolation Valve Accumulator	NA	>0.3g	NA	.3g	x	
D03-0220-0082AA10	MAIN STEAM/ Isolation Valve Accumulator	NA	>0.3g	NA	.3g	x	
D03-0220-0082BA10	MAIN STEAM/ Isolation Valve Accumulator	NA	>0.3g	NA	.3g	x	
D03-0220-0082CA10	MAIN STEAM/ Isolation Valve Accumulator	NA	>0.3g	NA	.3g	x	
D03-0220-0082DA10	MAIN STEAM/ Isolation Valve Accumulator	NA	>0.3g	NA	.3g	x	
D00-5201------T05	DIESEL GENERATOR/ Diesel Fuel Oil Storage Tank	NA	>0.3g	NA	.3g	x	x
D03-5201------T05	DIESEL GENERATOR/ Diesel Fuel Oil Storage Tank	NA	>0.3g	NA	.3g	x	x
D02-5201------T05	DIESEL GENERATOR/ Diesel Fuel Oil Storage Tank	NA	>0.3g	NA	.3g	x	x
D03-4798-A----A10	INSTRUMENT AIR/ Target Rock Accumulator	NA	>0.3g	NA	.3g	x	
D02-4798-A----A10	INSTRUMENT AIR/ Target Rock Accumulator	NA	>0.3g	NA	.3g	x	x
D02-0318-A----T05	CRD/ West Bank Scram Discharge Volume Tank	NA	>0.3g	NA	.3g	x	
D02-0318-B----T05	CRD/ East Bank Scram Discharge Volume Tank	NA	>0.3g	NA	.3g	x	
D03-0409-A----T05	CRD/ East Bank Scram Discharge Volume Tank	NA	>0.3g	NA	.3g	x	
D03-0409-B----T05	CRD/ West Bank Scram Discharge Volume Tank	NA	>0.3g	NA	.3g	x	
D02-0220-0083AA10	MAIN STEAM/ Isolation Valve Accumulator	NA	>0.3g	NA	.3g		
D02-0220-0083BA10	MAIN STEAM/ Isolation Valve Accumulator	NA	>0.3g	NA	.3g		
D02-0220-0083CA10	MAIN STEAM/ Isolation Valve Accumulator	NA	>0.3g	NA	.3g		
D02-0220-0083DA10	MAIN STEAM/ Isolation Valve Accumulator	NA	>0.3g	NA	.3g		
D03-0220-0083AA10	MAIN STEAM/ Isolation Valve Accumulator	NA	>0.3g	NA	.3g		

I D	Description	SMA Seismic Capacity (PGA) Equm't	Anchor-age	Block Wall	Mini-mum	A-46 SSEL	Outlier
D03-0220-0083BA10	MAIN STEAM/ Isolation Valve Accumulator	NA	>0.3g	NA	.3g		
D03-0220-0083CA10	MAIN STEAM/ Isolation Valve Accumulator	NA	>0.3g	NA	.3g		
D03-0220-0083DA10	MAIN STEAM/ Isolation Valve Accumulator	NA	>0.3g	NA	.3g		
D03-5202------T05	DIESEL GENERATOR/ Diesel Fuel Oil Storage Day Tank	NA	0.61g	0.32g	0.32g	x	x
D02-5202------T05	DIESEL GENERATOR/ Diesel Fuel Oil Storage Day Tank	NA	0.61g	0.46g	0.46g	x	x
D03-1503-B----H15	LPCI/ LPCI Heat Exchanger	NA	0.47g	NA	0.47g	x	
D02-1503-A----H15	LPCI/ LPCI Heat Exchanger	NA	0.47g	NA	0.47g	x	x
D02-1503-B----H15	LPCI/ LPCI Heat Exchanger	NA	0.47g	NA	0.47g	x	x
D03-1503-A----H15	LPCI/ LPCI Heat Exchanger	NA	0.47g	NA	0.47g	x	x

Table 3.3 SME Anchorage Capacity of A-46 Outliers

ID	EQUIPMENT	OUTLIER FINDING	RESOLUTION (Referenced resolution numbers are detailed in Section 7.1.1)	ANCHORAGE HCLPF (PGA)
1	Hydraulic Control Unit D02-0305------H20 D03-0305------H20	1) No seismic capacity based on earthquake experience or generic seismic testing ruggedness data is available for Class 0 equipment. 2) Nearby gas bottles are restrained by only a single chain.	1) No actions are required. Upon further review, Hydraulic Control Unit consists of accumulator and valves that are in the earthquake experience and generic seismic testing ruggedness data. Rack load path and anchorage are separately analyzed and acceptable. CRD piping and scram header adequately seismically supported. RESOLUTION #11 2) No actions are required. The worst-case gas bottle with a single chain has been verified for adequacy based on seismic testing (ref. NDIT No. SEC-DR-00-001). RESOLUTION #1	>0.3g

ID	EQUIPMENT	OUTLIER FINDING	RESOLUTION (Referenced resolution numbers are detailed in Section 7.1.1)	ANCHORAGE HCLPF (PGA)
2	Motor Control Centers D02-7820-03- - M05 D02-7820-02- - M05	MCCs are not anchored to the equipment pad.	Outlier has been deleted from the Program. The MCCs power the Refuse Pumps. The Refuse Pumps are not being credited for Isolation Condenser makeup for the loss of dam failure scenario, therefore the MCCs are not required for safe shutdown or included in the SPEL. RESOLUTION #9	N/A
3	Bus and Panel D02-8302A-----M05	1) Locker/storage cabinet located next to bus on rear side. 2) Seismic demand exceeds capacity.	1) The locker/cabinet has been relocated. RESOLUTION #5 2) Refined floor spectra have been developed. Evaluation is in process. RESOLUTION #11	0.27g (revised from 0.17g)
4	Motor Control Center D03-7839-2----M05	1) Adjacent MCC 35-1 is inadequately anchored (only tack welded in rear). 2) Tank 2/3-4911 is not positively anchored to its saddle. Drain valve may potentially break and cause an oil release.	1) Anchorage improvements are pending DCP 9900205. RESOLUTION #9 2) Perform an evaluation or positively anchor the tank. RESOLUTION #7	>0.3g
5	Bus and Panel D02-83125-----P06	Seismic demand exceeds capacity.	Refined floor spectra have been developed. Evaluation is in process. RESOLUTION #11	0.27g (revised from 0.17g)

ID	EQUIPMENT	OUTLIER FINDING	RESOLUTION (Referenced resolution numbers are detailed in Section 7.1.1)	ANCHORAGE HCLPF (PGA)
6	Bus and Panel D03-8303A-----M05 D03-8303B-----M05 D03-83125-----P06 D02-8302B-----M05	1) Seismic demand exceeds capacity. 2) Embedded angle pullout capacity is insufficient to hold down the cabinet during the event of SSE.	1&2) Refined floor spectra have been developed. Evaluation is in process. RESOLUTION #11	.202g (revised from 0.17g)
7	Switchgear D02-67241----S35	Spare Breakers are too close	The spare breakers have been relocated. RESOLUTION #5	>0.3g
8	Motor Control Center D02-7829-8----M05	Overhead unanchored emergency light # 251C (battery powered).	The light has been positively anchored. RESOLUTION #3	>0.3g
9	Panel D02-8302B-1---P06 D02-83125-1---P06	A 6' long (tall) cable tray raceway system hanger which is rod hung and laterally unbraced may sufficiently displace to impact the Panel.	Panels do not contain essential relays, therefore no actions are required to preclude impact potential. RESOLUTION #6	0.29g
10	Panel D03-8303A1-2-P06	1) Lights overhead have open S-hooks. 2) D.G. stack rises adjacent to panel and is a potential interaction hazard.	1) The S-hooks have been closed. RESOLUTION #2 2) No actions are required. The seismic structural integrity is acceptable based on evaluation (ref. SEWS, Rev. 1). RESOLUTION #6	>0.3g

ID	EQUIPMENT	OUTLIER FINDING	RESOLUTION (Referenced resolution numbers are detailed in Section 7.1.1)	ANCHORAGE HCLPF (PGA)
11	Motor Control Center D02-7829-2----M05	1) Nearby gas bottles secured by only 1 chain. 2) Lights overhead have open S-hooks. 3) Unrestrained flammable storage cabinet adjacent to MCC.	1) No actions are required. The worst-case gas bottle with a single chain has been verified for adequacy based on seismic testing (ref. NDIT No. SEC-DR-00-001).RESOLUTION #1 2) The S-hooks have been closed. RESOLUTION #2 3) The cabinet has been relocated. RESOLUTION #5	>0.3g
12	Panel D03-8303B1----P06 D03-9802-A----P06 D03-9802-B----P06	1) Lights overhead have open S-hooks. 2) Emergency light #329 (battery powered) not secured.	1) The S-hooks have been closed. RESOLUTION #2 2) The light has been positively secured. RESOLUTION #3	>0.3g
13	Panel D03-8303A1-1--P06 D03-83125-3---P06	Lights overhead have open S-hooks.	The S-hooks have been closed. RESOLUTION #2	>0.3g

ID	EQUIPMENT	OUTLIER FINDING	RESOLUTION (Referenced resolution numbers are detailed in Section 7.1.1)	ANCHORAGE HCLPF (PGA)
14	Motor Control Center and Panel D02-7828-3----M05 D02-7829-4----M05 D03-7838-2----M05 D03-7838-3----M05 D02-2202-0073A-P05 D02-2202-0073B-P05 D03-2203-0006-P06 D02-2252-0084-P05	Lights overhead have open S-hooks.	S-hooks have been closed. RESOLUTION #2	>0.3g
15	Motor Control Center and Panel D02-7828-7----M05 D03-7838-1-1--P06 D02-9205-A----V05 D02-9205-B----V05 D02-9206-A----V05 D02-9206-B----V05 D03-9206-A----V05 D03-9206-B----V05	Nearby gas bottles only restrained by a single chain	The worst-case gas bottle with a single chain has been verified for adequacy based on seismic testing (ref. NDIT No. SEC-DR-00-001), except the gas bottle near D03-7838-1-1 which requires relocation of its restraining chain, pending a maintenance work request. RESOLUTION #1	>0.3g

ID	EQUIPMENT	OUTLIER FINDING	RESOLUTION (Referenced resolution numbers are detailed in Section 7.1.1)	ANCHORAGE HCLPF (PGA)
16	Motor Control Center and Panel D02-83250-----M05	1) Nearby gas bottles only restrained by a single chain. 2) Anchorage capacity less than demand	1) No actions are required. The worst-case gas bottle with a single chain has been verified for adequacy based on seismic testing (ref. NDIT No. SEC-DR-00-001). RESOLUTION #1 2) No actions are required. Refer to Section 1.4.1 of this report for further information. RESOLUTION #11	0.29g
17	Motor Control Center and Panel D02-9802-A----P06	1) Nearby gas bottles only restrained by a single chain. 2) Anchorage capacity less than demand	1) No actions are required. The worst-case gas bottle with a single chain has been verified for adequacy based on seismic testing (ref. NDIT No. SEC-DR-00-001).RESOLUTION #1 2) Re-evaluation based on refined floor spectra. RESOLUTION #11	0.27g

ID	EQUIPMENT	OUTLIER FINDING	RESOLUTION (Referenced resolution numbers are detailed in Section 7.1.1)	ANCHORAGE HCLPF (PGA)
18	Panel D02-0902-0015-P06 D02-0902-0017-P06 D02-0902-0019-P06 D02-0902-0036-P06D03-0903-0015-P06 D03-0903-0017-P06 D03-0903-0019-P06 D03-0903-0036-P06 D02-0902-0003-P06 D02-0902-0004-P06 D03-0903-0003-P06 D03-0903-0004-P06	The internal cable tray raceway system hanger is rod hung and transversely unrestrained. It can displace sufficiently to impact the Panel which has only 1" clearance from the end of the cross-member. Cabinets do contain essential relays.	DCPs 9900185 and 9900186 have been issued to restrain cross-members by "tying" them to front and back panels. This will preclude impact hazard and increase anchorage capacity seismic margin. RESOLUTION #6	>0.3g (revised from 0.22g) New capacity pending installation of DCPs 9900185 and 9900186.
19	Panel D03-0903-0046-P06	1) Located adjacent to a conduit pull-box and it is an impact hazard. 2) Adjacent tool box adjudged a hazard.	1) Neoprene has been installed between the pull-box and cabinet in accordance with WR 99005643 to reduce significance of impact. RESOLUTION #6 2) The tool box has been relocated. RESOLUTION #5	>0.3g
20	Panel and Transformer D02-2202-0008-P06 D02-7229------T10 D02-0902-0041-P06	Adjacent emergency lights #293, #299, #220, and #313 (battery powered) are missing straps or are unrestrained.	The lights have been positively restrained. RESOLUTION #3	>0.3g

ID	EQUIPMENT	OUTLIER FINDING	RESOLUTION (Referenced resolution numbers are detailed in Section 7.1.1)	ANCHORAGE HCLPF (PGA)
21	Battery Charger D03-9802-3BNEGB05 D03-9802-3BPOSB05	Overhead emergency light #329 (battery powered) is shelf mounted and unrestrained.	The lights have been positively restrained. RESOLUTION #3	>0.3g
22	Switchgear D02-7328------S35 D02-7329------S35	1) Overhead trolley hoist is an impact hazard and needs to be parked. 2) Emergency light #229 (battery powered) not restrained. 3) Anchorage capacity less than demand	1) A device has been designed to prevent the hoist from rolling freely. Installation is pending work package preparation. RESOLUTION #4 2) The light has been positively restrained. RESOLUTION #3 3) No action required. Refer to Section 1.4.1 of this report for further information. RESOLUTION #11	0.27g
23	Switchgear D03-7338------S35	Overhead trolley hoist, open S-hook on overhead florescent light and racking crank hanging between breaker and transformer are all impact hazards.	1) No actions are required since it has been verified that there are no essential relays within the switchgear. RESOLUTION #4 2) The S-Hooks have been closed. RESOLUTION #2 3) The racking crank has been relocated in accordance with E12-3-95-220. RESOLUTION #5	>0.3g

ID	EQUIPMENT	OUTLIER FINDING	RESOLUTION (Referenced resolution numbers are detailed in Section 7.1.1)	ANCHORAGE HCLPF (PGA)
24	Switchgear D03-7339------S35	Overhead trolley hoist, open S-hook on overhead florescent light, racking crank hanging between breaker and transformer and adjacent unanchored 4KV gear box locker are all impact hazards.	1) No actions are required since it has been verified that there are no essential relays within the switchgear. RESOLUTION #4 2) The S-Hooks have been closed. RESOLUTION #2 3) The racking crank has been relocated in accordance with E12-3-95-221. RESOLUTION #5 4) The locker has been relocated in accordance with E12-3-95-220. RESOLUTION #5	>0.3g
25	Switchgear D03-6734------S35	Safety equipment locker adjacent to cubicle, spare breakers marginally chained to wall, and additional unanchored spare cubicles 5' away from line-up are interaction hazards.	Equipment has been relocated. RESOLUTION #5	>0.3g

ID	EQUIPMENT	OUTLIER FINDING	RESOLUTION (Referenced resolution numbers are detailed in Section 7.1.1)	ANCHORAGE HCLPF (PGA)
26	Panel D03-2203-0070A-P05 D03-2203-0070B-P05 D02-0902-0033-P06 D02-2202-0070A-P05 D02-2202-0070B-P05 D02-0902-0039-P06 D02-0902-0047-P06 D03-0903-0047-P06 D03-0903-0032-P06	1) Panel is not bolted to adjacent panel 2) Panel D03-0903-0052 is missing a bolt and is adjacent to Safe Shutdown Panels D03-2203-70A & B	1) Panels D02-2202-70A and D03-2203-70A are bolted to adjacent Panels D02-2202-70B and D03-2203-70B, respectively. Panel D03-2203-70B is also bolted to the adjacent panel 903-27 in accordance with ECN No. 12-00854E. U2 DCP 9900187 and U3 DCP 9900188 have been issued to bolt the remaining panels to adjacent panels. RESOLUTION #8 2) No actions are required. Appendix I of the USI A-46 Report identified a missing bolt on Panel D03-0903-52, however a subsequent walkdown has verified that all bolts are tightened in place. RESOLUTION #14	>0.3g
27	Panel D02-0902-0046-P06	Panel not bolted to adjacent "low boy" console.	No actions are required. The gap between Panel D02-0902-46 and the "low boy" console has been judged as large enough to preclude seismic interaction based on a subsequent walkdown. RESOLUTION #8	>0.3g

ID	EQUIPMENT	OUTLIER FINDING	RESOLUTION (Referenced resolution numbers are detailed in Section 7.1.1)	ANCHORAGE HCLPF (PGA)
28	Panel D02-0902-0028-P06	1) One anchor projected out and loose. 2) There is 1/4" gap between the panel and the surface of the concrete and panel contains essential relays. 3) Anchorage capacity less than demand.	1) The loose anchor has been fixed. RESOLUTION #14 2) No actions are required. A subsequent walkdown found the panel to be true level and any gap between the concrete and panel to be insufficient to create an interaction. RESOLUTION #6 3) No action required. Refer to Section 1.4.1 of this report for further information. RESOLUTION #11	0.23g
29	Panel D03-0903-0033	1) Panel is not bolted to adjacent panel. 2) There is 1/8" gap between the panel and the surface of the concrete and panel contains essential relays.	1) DCP 9900188 has been issued to bolt panels together to preclude impact potential. RESOLUTION #8 2) No actions are required. A subsequent walkdown found the panel to be true level and any gap between the concrete and panel to be insufficient to create an interaction. RESOLUTION #6	>0.3g

ID	EQUIPMENT	OUTLIER FINDING	RESOLUTION (Referenced resolution numbers are detailed in Section 7.1.1)	ANCHORAGE HCLPF (PGA)
30	Cooler D03-5746-A----H15 D03-5746-B----H15 D02-5746-A----H15 D02-5746-B----H15	The attached piping may not have enough flexibility.	U3 DCP 9900168 has been issued – scheduled for installation during D3R16 – to laterally restrain coolers to preclude piping rupture potential. Similar improvements in Unit 2 are pending DCP 9900278. RESOLUTION #7	>0.3g
31	Battery Charger D03-8300--3---B05 D03-83250-3	1) D03-8300-3 anchorage is inadequate (2 clips). 2) Improve anchorage for D03-83250-3.	1 & 2) Anchorage has been improved in accordance with E12-3-96-224 and E12-3-96-225. RESOLUTION #9	>0.3g
32	Battery Charger D00-83250-0---B05 D02-83250-2---B05	Unanchored.	Anchorage has been provided in accordance with DCPs E12-0-96-213 and E12-2-96-236. RESOLUTION #9	>0.3g
33	Vertical Pump D02-1502-A----P30 D02-1502-B----P30 D02-1502-C----P30 D02-1502-D----P30	Nozzle loads should be considered in the ANCHOR evaluation but information on nozzle loads are incomplete.	No actions are required. The anchorage has been verified to be adequate based on an evaluation considering nozzle loads (ref. SEWS, Rev. 1) RESOLUTION #9	>0.3g

ID	EQUIPMENT	OUTLIER FINDING	RESOLUTION (Referenced resolution numbers are detailed in Section 7.1.1)	ANCHORAGE HCLPF (PGA)
34	Horizontal Pump D03-1501-0044AP30 D03-1501-0044BP30 D03-1501-0044CP30 D03-1501-0044DP30	Piping vertically supported but has very little lateral restraint and therefore piping loads should be considered in evaluating the anchorage but information on nozzle loads is incomplete.	No actions are required. The anchorage has been verified to be adequate based on an evaluation considering nozzle loads (ref. SEWS, Rev. 1) RESOLUTION #9	>0.3g
35	Valve D02-2301-0005-V20 D03-2301-0005-V20 D03-2301-0008-V20	Did not meet GIP offset rules.	Calculation CE091.0208 shows all valves have greater than 0.3g capacity. RESOLUTION #11	>0.3g
36	Valve D02-8501-0005AV05 D02-8501-0005BV05	AOVs are supported off 1/2" diameter tubing.	The 1/2" tubing has been qualified based on subsequent evaluation (ref. DOC ID# 5995877). RESOLUTION #11	0.3g
37	Valve D02-0302-0019AV27	The condulet coming from the valve was open, exposing the wires.	A condulet cover has been installed (ref. AR 940038586). RESOLUTION #14	>0.3g
38	Horizontal Pump D02-5203------P30 D03-5203------P30	Unrestrained emergency light #209 (battery powered) is overhead.	The light has been positively secured. RESOLUTION #3	>0.3g

ID	EQUIPMENT	OUTLIER FINDING	RESOLUTION (Referenced resolution numbers are detailed in Section 7.1.1)	ANCHORAGE HCLPF (PGA)
39	Switchgear D02-6723------S35 D02-6724------S35	1) Spare breakers are too close. 2) Open S-hooks on light fixtures is overhead. 3) Panel 2-6724 is not bolted to adjacent panels.	1) The spare breakers have been relocated. RESOLUTION #5 2) The S-hooks have been closed. RESOLUTION #2 3) No actions are required. A subsequent walkdown verified that the panels are bolted together. RESOLUTION #8	>0.3g
40	Switchgear D02-67231-----S35 D03-67341-----S35	1) A safety equipment locker immediately adjacent to switchgear is an interaction hazard. 2) PCB storage tank behind switchgear is rod hanger restrained at mid-height. It is a flooding hazard because of sight glass at bottom of tank and essential relays inside the switchgear.	1) The locker has been relocated. RESOLUTION #5 2) Sight glass valves have been closed to prevent loss of oil if the glass breaks. RESOLUTION #6	>0.3g

ID	EQUIPMENT	OUTLIER FINDING	RESOLUTION (Referenced resolution numbers are detailed in Section 7.1.1)	ANCHORAGE HCLPF (PGA)
41	Panel D03-2203-0073A D03-2203-0073B	1) Overhead heater has only support rod and should have two. Thus, it is an impact hazard. 2) Overhead lighting has one open S-hook and is an impact hazard. 3) Overhead fire line is judged not to be seismically qualified by SRT. It is a flooding hazard because of essential relays inside panel.	1) The heater support has been repaired in accordance with WR 960044408. RESOLUTION #14 2) The S-hook has been closed. RESOLUTION #2 3) No actions are required. Fire piping has been seismically evaluated to be adequate (ref. DOC ID# 6012466). RESOLUTION #7	>0.3g
42	Valve D03-0302-0157AV05	Suction line of pump (4-1/2" in diameter) touches the valve yoke.	Pending DCP 9900346 for restraining the line to prevent excessive force on the valve. RESOLUTION #6	>0.3g
43	Valve D03-1501-0032BV20 D02-1601-0023--V27 D03-0203-0002BV05	1) Potential impact with nearby piping, grating or steel. 2) Valve 2-1601-23 is missing a bolt.	1) No actions are required. Valve 3-1501-32B has been evaluated for its yoke displacement and found to be within the available clearance with the grating (ref. DOC ID# 5996462). Valves 2-1601-23 and 3-0203-2B are pending evaluation. RESOLUTION #6 2) Bolt has been installed in accordance with Work Request 950096788-01. RESOLUTION #14	>0.3g

ID	EQUIPMENT	OUTLIER FINDING	RESOLUTION (Referenced resolution numbers are detailed in Section 7.1.1)	ANCHORAGE HCLPF (PGA)
44	Panel D02-2252-0010-P06	Base channel is secured to floor by 4 friction clips (one in each corner) Neighboring Neutral Grounding Cabinet is secured by friction clips.	DCP 9900322 has been issued to implement anchorage improvements. RESOLUTION #10	>0.3g (revised from 0.17g) New capacity pending installation of DCP 9900322.
45	Panel D00-NGC	Panel is right against a column support on one side and an adjacent panel on the other.	DCP 9900321 has been issued to secure the panel to the column and bolt it to the adjacent panel. RESOLUTION #6	>0.3g
46	Panel D00-2223-0033-P06 D00-2223-0041-P06	Panel is too close to two adjacent panels that it is not bolted to.	DCP 9900321 has been issued to connect D00-2223-33 to the adjacent NGC, and to trim and bend the D00-2223-41 drip cover to prevent interaction. RESOLUTION #8	>0.3g
47	Damper D03-5772-0102-D05	1) Damper hung in poorly supported ductwork (already sagging). 2) Overhead light has open S-hooks.	1) The duct support rod has been replaced in accordance with WR 960043286. RESOLUTION #14 2) The S-hooks have been closed. RESOLUTION #2	>0.3g

ID	EQUIPMENT	OUTLIER FINDING	RESOLUTION (Referenced resolution numbers are detailed in Section 7.1.1)	ANCHORAGE HCLPF (PGA)
48	Battery Rack D02-9802-A----B04 D02-9802-B----B04 D03-9802-A----B04 D03-9802-B----B04	1) There are spaces between the batteries and along the front and back of the batteries. 2) The battery cells are more than 10 years old.	New batteries have been installed in accordance with DCPs M12-2-95-003 and M12-3-95-003. RESOLUTION #12 RESOLUTION #13	>0.3g
49	Battery Rack D02-8300--BC--B04 D02-83250-----B04 D03-8300--BC--B04 D03-83250-----B04	The Styrofoam, on the front and back, is not full-height and could easily slip out during a seismic event.	An engineering letter to implement repairs has been issued (ref. DOC ID# 6043969). RESOLUTION #12	>0.3g
50	CCSW Pump Cooler D02-5700-0030AH15 D02-5700-0030BH15 D02-5700-0030CH15 D02-5700-0030DH15 D03-5700-0030AH15 D03-5700-030BH157 D03-5700-0030CH15 D03-5700-0030DH15	Bolt type is not covered by the GIP - Cinch Anchor.	No actions are required. The anchorage capacity exceeds the design basis seismic demand loads based on "Lead Expansion Anchor Load Capacity in Reactor Building at the Savannah River Site", Westinghouse Savannah River Company, RTR-2661, Aug. 15, 1989 (Refs. 22 & 23). RESOLUTION #10	>0.3g
51	Switchgear D03-67331------S35	Spare breakers which are unanchored are potential interaction hazards.	The spare breakers have been relocated RESOLUTION #5	>0.3g

ID	EQUIPMENT	OUTLIER FINDING	RESOLUTION (Referenced resolution numbers are detailed in Section 7.1.1)	ANCHORAGE HCLPF (PGA)
52	Vertical Tank D00-3303-A----T05, D00-3303-B----T05	Tank is supported by ring foundation.	No actions are required. The ring foundation has been qualified based on evaluation (ref. Stevensen & Associates Calculation 98Q4006-C-002). RESOLUTION #9	0.20g
53	D02-NGC	Base channel is secured to floor by 4 friction clips (one in each corner)	DCP 9900322 has been issued to implement anchorage improvements. RESOLUTION #10	>0.3g (revised from 0.28g) New capacity pending installation of DCP 9900322.
54	Pressure Indicators 2/3-4041-1A & 1B	Indicators are supported by piping and not attached to wall which could result in seismic interaction.	Outlier has been deleted from the Program. The pressure indicators are for the Refuse Pumps. The Refuse Pumps are not being credited for CCSW makeup for the loss of dam failure scenario, therefore the pressure indicators are not required for safe shutdown or included in the SPEL. RESOLUTION #6	N/A

ID	EQUIPMENT	OUTLIER FINDING	RESOLUTION (Referenced resolution numbers are detailed in Section 7.1.1)	ANCHORAGE HCLPF (PGA)
55	Iso Cond Heat Exchangers D02-1302---H15, D03-1302---H15	Isolation condenser has no anchor bolts at outboard piers. Load taken by center pier.	No actions are required. The existing configuration has been verified to be adequate per S&L calculation 8900-15-EO-S. RESOLUTION # 14	>0.3g
56	Heat Exchanger D02-1503-A----H15 D03-1503-A----H15 D02-1503-B----H15 D03-1503-B----H15	Support steel requires evaluation.	Support steel is adequate based on anchorage improvements in accordance with DCPs E12-2-95-242, E12-2-95-243, E12-3-95-258, and E12-3-95-259. RESOLUTION #9	>0.3g
57	Accumulator D02-4798-A----A10	The accumulator has one bolt missing.	No actions are required. Upon further review, the accumulator is designed to be installed with one U-bolt. A subsequent walkdown verified that the U-bolt is in place (ref. DOC ID# 6034842). RESOLUTION #14	>0.3g
58	Control Panel / DG Exciter D02-2252-0021-P06	Neighboring Neutral Grounding Cabinet is secured by friction clips.	DCP 9900322 has been issued to implement anchorage improvements. RESOLUTION #10	>0.3g
59	D02-2202-0006-P06	Overhead light is only secured by one chain.	The light has been secured. RESOLUTION #14	>0.3g

ID	EQUIPMENT	OUTLIER FINDING	RESOLUTION (Referenced resolution numbers are detailed in Section 7.1.1)	ANCHORAGE HCLPF (PGA)
60	Cable Tray & Conduit Raceway Systems LAR001	An enveloping support in the Unit 2 Battery Room area was chosen for limited analytical review. The support is a 4-tier, rod hung trapeze supporting 4 trays. The support has two different types of ceiling anchorages: embedded strut and weldment to building steel. Loads exceed the allowables for the rod fatigue check.	This outlier is resolved by outlier analysis. A limit analysis per Section 8.4.8 of Reference 14 was performed and the hanger passes. Dresden rod fatigue data obtained from actual cyclic testing of the Dresden field threaded rods including the weldment anchorage plate from SEP Project 8050 (Ref. 22) was used to evaluate the rods. Based on the "Generic Rod Acceptability Curves", it was shown that the Dresden rods will sustain the SSE demand loads. RESOLUTION #15	>0.3g

ID	EQUIPMENT	OUTLIER FINDING	RESOLUTION (Referenced resolution numbers are detailed in Section 7.1.1)	ANCHORAGE HCLPF (PGA)
61	Cable Tray & Conduit Raceway Systems LAR004	An enveloping support in the Auxiliary Electrical Equipment Room was chosen for limited analytical review. The support is a four tier rod hung trapeze supporting 4 trays. The support has two different types of ceiling anchorages: embedded strut and weldment to building steel. The loads for the embedded strut version exceed the allowables for the vertical capacity check. The loads for the welded attachment to building steel version exceed, by a small amount, the allowables for the rod fatigue check and are considered acceptable.	This outlier is resolved by outlier analysis. A limit analysis per Section 8.4.8 of Reference 14 was performed and the hanger passes. RESOLUTION #15	>0.3g
62	Cable Tray & Conduit Raceway Systems LAR005	The main "spine of the cables connecting the Reactor building with the Control room. A large two-bay system carrying 12-44" trays in one bay (one side) and 4-44" trays on the other side in a floor-to-ceiling system. The bolting connections at the ceiling connection (anchorage) do not pass the vertical capacity check.	This outlier is resolved by outlier analysis. A limit analysis per Section 8.4.8 of Reference 14 was performed and the hanger passes. RESOLUTION #15	>0.3g

ID	EQUIPMENT	OUTLIER FINDING	RESOLUTION (Referenced resolution numbers are detailed in Section 7.1.1)	ANCHORAGE HCLPF (PGA)
63	Cable Tray & Conduit Raceway Systems LAR006	An enveloping support in the Unit 3 Turbine Building, El. 538, was chosen for limited analytical review. The support is a two tier, rod hung trapeze supporting 2 trays. The support anchorage is weldment to building steel. The support loads exceed the allowables for the rod fatigue check.	This outlier is resolved by outlier analysis. A redundancy and consequence analysis per Section 8.4.8 of Reference 14 was performed and the hanger passes. Dresden rod fatigue data obtained from actual cyclic testing of the Dresden field threaded rods including the weldment anchorage plate from SEP Project 8050 (Ref. 22) was used to evaluate the rods. Based on the "Generic Rod Acceptability Curves", it was shown that the Dresden rods will sustain the SSE demand loads. RESOLUTION #15	>0.3g
64	Cable Tray & Conduit Raceway Systems LAR007	An enveloping support in the Unit 2 Turbine Building, El. 517, was chosen for limited analytical review. The support is a four tier, rod hung trapeze supporting 4 trays. The support anchorage is an embedded strut. The support loads exceed the allowables for the rod fatigue check.	No actions are required. The worst-case hangers have been reassessed for actual tray spans and loading. The actual spans and loading were found to be smaller than those used in the original assessment. Calculation DRE99-0029 shows that the worst-case hangers are acceptable without modifications. RESOLUTION #15	>0.3g (revised from 0.15g)

ID	EQUIPMENT	OUTLIER FINDING	RESOLUTION (Referenced resolution numbers are detailed in Section 7.1.1)	ANCHORAGE HCLPF (PGA)
65	Cable Tray & Conduit Raceway Systems LAR008	An enveloping support in the Reactor Building, El. 517, was chosen for limited analytical review. The support is a two bay, three tier/bay, rod hung trapeze supporting 1 tray/tier. The support has two different types of ceiling anchorages: embedded strut and weldment to building steel. The loads for the embedded strut version exceed the allowables for the vertical capacity check. The loads for the welded attachment to building steel version exceed the allowables for the rod fatigue check.	This outlier is resolved by outlier analysis. A limit analysis per Section 8.4.8 of Reference 14 was performed and the hanger passes. Dresden rod fatigue data obtained from actual cyclic testing of the Dresden field threaded rods including the weldment anchorage plate from SEP Project 8050 (Ref. 22) was used to evaluate the rods. Based on the "Generic Rod Acceptability Curves", it was shown that the Dresden rods will sustain the SSE demand loads.	>0.3g
66	Cable Tray & Conduit Raceway Systems LAR009	The support is a 1/2" field threaded rod trapeze supporting 16"dia. bus duct, 24" to "top tier". The limiting seismic capacity based on rod fatigue is 98 lbs/ft. The actual weight (amount of cable inside the conduit) is currently unknown.	No actions are required. The actual weight of the bus duct is less than 20lbs/ft, compared to the 98 lbs/ft capacity (ref. NDIT No. SEC-DR-98-046). RESOLUTION #15	>0.3g
67	Unit 2 Turbine Building (including D/G Room), El. 517 RACE010	The hangers at the entrance to the turbine building, east of the truck bay, are rod hung on one side and bolted to a strut embedded in a block wall on the other side.	The wall capacity is no longer required since the cable tray is not supported off the wall per Doc ID# 6065125. The cable tray has been verified to be adequate per the resolution to ID 64. RESOLUTION #15	>0.3g

ID	EQUIPMENT	OUTLIER FINDING	RESOLUTION (Referenced resolution numbers are detailed in Section 7.1.1)	ANCHORAGE HCLPF (PGA)
68	Diesel Fuel Oil Storage Day Tank .1 D00-5202----T05	Sight glass interaction issue.	No action required. Refer to Section 1.4.1 of this report for further information. RESOLUTION #6	0.26g
69	Drywell/Torus Nitrogen Makeup Valves D02-1601-0057---V20 D03-1601-0057---V20	3g analysis shows yoke overstress.	Evaluate actual demand accelerations. RESOLUTION #11	0.3g
70	Hydrogen Seal Oil Panel and Hydrogen Monitors 2-5341-15 3-5341-15 2252-7 2253-7	Inadequate anchorage	Anchorage improvements and seismic restraint of monitors is pending DCPs 9900205 (Unit 2) and 9900204 (Unit 3).	>0.3g
71	Procedure DHP 0220-02	Ensure adequate procedure control.	Revise procedure DHP 0220-02.	N/A

4.0 INTERNAL FIRE EVALUATION

4.1 METHODOLOGY

In June of 1991, the NRC issued Supplement 4 to Generic Letter 88-20 (Ref. 4-1), asking all licensees holding operating licenses and construction permits for nuclear power reactor facilities to perform an Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities. The five external events requested to be assessed included internal fires. Both the Generic Letter and NUREG-1407 (Ref. 4-2) state these objectives for the IPEEE:

1. To develop an appreciation of severe accident behavior;
2. To understand the most likely severe accident sequences that could occur under full-power operations;
3. To gain a qualitative understanding of the overall likelihood of core damage and radioactive release; and
4. If necessary to reduce the overall likelihood of core damage and radioactive release by modifying hardware and procedures that would help prevent or mitigate severe accidents.

Although neither the Generic Letter nor NUREG-1407 prescribes a specific methodology for assessment of internal fires, they do require that these steps be included:

- Identification of critical areas of vulnerability;
- Calculation of the frequency of fire initiation in each area;
- Analysis of the likelihood of critical safety functions disabled by a fire;
- Assessment of fire-induced accident sequences leading to core damage;
- Evaluation of the containment mitigating function under fire; and
- Assessment of issues identified in the Fire Risk Scoping Study (FRSS) (Ref. 4-3).

In recent years, Fire-Induced Vulnerability Evaluation (FIVE) (Ref. 4-4) and Fire PRA (Ref. 4-5) methodologies have been developed by the Electric Power Research Institute (EPRI) for evaluation of fire risk at nuclear power plants. While FIVE is a screening method to identify critical fire scenarios, the Fire PRA method provides the robust tools needed for more realistic assessment of core damage risk associated with these scenarios. Both methods draw from earlier methodologies such as those described in NUREG/CR-2815 (Ref. 4-6) and NUREG/CR-4840 (Ref. 4-7), and benefit from the insights gained from over 1000 reactor-years of operating experience and various fire tests conducted by Sandia National Laboratories (SNL), National Institute of

Standards and Technology (NIST), and others. Both these methodologies explicitly contain the steps requested and are capable of achieving the goals set in the Generic Letter for the Fire IPEEE.

ComEd elected to use FIVE (Ref. 4-4), supplemented by the EPRI Fire PRA methodology, for assessment of internal fire risk at the Dresden Nuclear Power Station because of its ability to best achieve the objectives of the IPEEE. FIVE provides effective methods for screening out fire compartments that are not risk-significant. The methods in the Fire PRA Implementation Guide (Ref. 4-5) allow for development of a plant fire risk model that integrates all aspects of the plant fire protection design and practice, and at the same time makes use of the available data on fire history and tests. The detailed and integrated nature of the method helps to better understand the most likely severe accident sequences and identify the most effective solutions to prevent or mitigate severe accidents resulting from fires.

ComEd previously submitted the results of a fire risk analysis for Dresden (Ref. 4-19) in 1997. ComEd has since upgraded this fire risk analysis by developing and applying additional plant specific information related to spatial location, failure modes, and failure consequences of critical cables and circuits, modifying the analysis to realistically treat the use of plant operating procedures, and incorporating the upgraded plant PRA model. The upgraded analysis eliminated numerous sources of conservatism and produced a much more accurate and usable fire risk assessment. The conservatism in the original Fire IPEEE is discussed further in Section 4.9.1.

The upgraded Dresden fire risk analysis used EPRI's FIVE methodology (Ref. 4-4) for initial screening, determination of ignition source frequencies, fire compartment boundary requirements, and plant walkdowns. EPRI's Fire PRA Implementation Guide (Ref. 4-5) was used to provide enhancements for the development of individual fire-induced scenarios and the multi-compartment analysis. The sequence of major analyses developed during the project and their contents are as follows:

<u>Identification of Fire Compartments</u>

Plant fire areas and compartments were defined based on guidance described in FIVE. Compartments containing safe shutdown (SSD) circuits and equipment, as well as other equipment important to plant safety (such as offsite power) were selected for further evaluation. Qualitative screening was limited to fire areas that did not contain Appendix R credited post fire safe shutdown features and were not expected to cause a plant trip or require a shutdown, and fire compartments which satisfied the FIVE screening criteria. The fire analysis upgrade effort did not alter the fire compartment definitions used in the original analysis.

Develop Compartment Fire Frequencies

Fire compartment ignition frequencies were determined for each fire compartment that was not qualitatively screened. Compartment frequencies were determined by counting each individual fire ignition source located in each compartment and applying the process described in FIVE Phase II, Step 1, to determine a frequency for each source. The sum of the source frequencies in each compartment represented the compartment fire frequency. The fire analysis upgrade effort did not alter the fire compartment ignition frequencies calculated for the original analysis.

Preliminary Screening

The purpose of this task was to eliminate from further consideration fire compartments that are not risk significant. The original Fire IPEEE used a Core Damage Frequency (CDF) screening criteria of 10^{-6}/yr. The upgraded fire risk analysis applied a CDF screening criteria of 10^{-7}/yr. to be consistent with the EPRI Fire PRA Implementation Guide (Ref. 4-5). The upgraded fire risk analysis screening basically assumed that any fire in the compartment will damage all PRA targets. The process used the aforementioned compartment fire frequencies and the Conditional Core Damage Probability (CCDP) determined from the plant Fire PRA Model. Adjustment of the ignition frequency by eliminating ignition sources, using severity factors, or credit for automatic or manual fire suppression was not applied for any of the screened cases. CDF was determined by multiplying compartment fire frequency and CCDP.

Detailed Fire Modeling / Analysis of Single Fire Compartments

This analysis evaluated the fire compartments which remained unscreened after the preliminary screening. The approach used for each fire compartment CDF was: 1) evaluate each individual fire source that can damage credited fire PRA targets; 2) define fire scenarios taking into account fire protection features such as detection and suppression; 3) determine a CCDP for the specific PRA targets damaged; and 4) calculate a scenario specific CDF. The sum of the scenario CDFs represents the final compartment CDF. The upgraded fire analysis did not screen any fire compartments in which fire modeling was performed regardless of the calculated CDF contribution.

Analysis of Multi-compartment Fires

This analysis was performed to evaluate the potential for core damage in the event that the fire barriers credited in the single compartment analyses were unable to prevent fire propagation (and equipment damage) in adjacent compartments. The analysis evaluated the risk significance of postulated fire scenarios that represent challenges to fire barrier integrity. The methodology used in this analysis was based on the Fire PRA Implementation Guide (Ref. 4-5). The upgraded fire risk analysis evaluated all compartment boundaries including those which are maintained as rated fire barriers as well as those which are not.

Status of Appendix R Modifications

The Dresden Station Fire Protection Report (Ref. 4-8) provides the basis for plant modifications needed to implement the Appendix R rule. As of the date of the original submittal no unresolved Appendix R modifications have been identified.

The remainder of Section 4 contains the documentation for the assessment of internal fire risk at Dresden. Table 4.1 provides the key for cross-listing the information provided in this assessment with the information requested in the Generic Letter 88-20, Supplement 4, Pages 23 and 24 (Ref. 4-1).

Table 4-1
Key for Information Requested by Generic Letter in the Fire IPEEE Submittal

Requested Information	Section of the Fire IPEEE
1. A description of the methodology and key assumptions used in performing the fire IPEEE and a discussion of the status of Appendix R modifications.	Section 4.0 contains a summary description of the methodology. Details of the individual steps, assumptions, and bases are documented in Sections 4.4, 4.5 & 4.6. Status of Appendix R modifications is discussed in Section 4.1.
2. A summary of walkdown findings and a concise description of the walkdown team and the procedure used.	Sections 4.2 and 4.5.2.1.5.
3. A discussion of the criteria used to identify critical fire areas and a list of critical areas, including (a) single areas in which equipment failure represents a serious erosion of safety margin, (b) same as (a), but for double or multiple areas sharing common barriers, penetration seals, HVAC ducting, etc.	Section 4.3.1 identifies single areas and Section 4.7.3 contains evaluation of the double or multiple areas.
4. A discussion of the criteria used for fire size and duration and the treatment of cross-zone fire spread and associated major assumptions.	The criteria for fire size and duration is discussed in Section 4.4. Treatment of cross-zone fire spread and assumptions are documented in Section 4.7.3.

Table 4-1
Key for Information Requested by Generic Letter in the Fire IPEEE Submittal

Requested Information	Section of the Fire IPEEE
5. A discussion of the fire initiation database, including the plant-specific database used. Describe the handling method, including major assumptions, the role of expert judgment, and identification and evaluation of sources of data uncertainties. A discussion of each case where the plant-specific data used is less conservative than the database used in approved fire vulnerability methodologies.	Discussion of the fire initiation database, including plant-specific data, is provided in Section 4.3.2. Discussion of plant-specific design and operation data is included in Section 4.5.3, cable location in Section 4.5.2, detection and suppression in Section 4.6. In no case is less conservative plant-specific data used.
6. A discussion of the treatment of fire growth and spread, the spread of hot gases and smoke, and the analysis of detection and suppression and their associated assumptions, including treatment of suppression-induced damage to equipment.	Sections 4.4 and 4.6 provide a discussion of the method for analysis of single-area fire growth, detection and suppression. Treatment of multi-compartment and Control Room fires growth, detection and suppression are discussed in Sections 4.7.3 and 4.7.4, respectively. Assessment of suppression-induced damage to equipment is addressed under FRSS issues and documented in Section 4.10.2.
7. A discussion of fire damage Modeling, including definition of the fire-induced failures related to fire barriers and control systems and fire-induced damage to the cabinets. A discussion of how human intervention is treated and how fire-induced and non-fire-induced failures are combined. Identify recovery actions and types of fire mitigating actions taken credit for in these sequences.	Section 4.5. Fire barrier failures are discussed in Section 4.7.3 as part of the multi-compartment analysis.
8. Discuss the treatment of detection and suppression including fire fighting procedures, fire brigade training, and adequacy of existing fire brigade equipment, and treatment of access routes versus existing barriers.	Discussion of fire detection and suppression is provided in Section 4.7.2, 4.7.3 and 4.7.4 for single area, multiple areas, and Control Room, respectively. Fire Brigade Training and equipment is addressed under FRSS issues in Section 4.10.2.
9. All functional/systemic event trees associated with fire-initiated sequences.	Discussion of the fire-initiated sequences is provided in Section 4.5.4.

Table 4-1
Key for Information Requested by Generic Letter in the Fire IPEEE Submittal

Requested Information	Section of the Fire IPEEE
10. A description of dominant functional/systemic sequences leading to core damage, along with their frequencies and percentage contribution to overall fire core-damage frequencies. Sequence selection criteria are provided in GL 88-20 and NUREG-1335. The description of the sequences should include a discussion of specific assumptions and human recovery actions.	Discussion of the dominant fire-induced sequences is provided in Section 4.7.2. The results of the fire-induced sequences are summarized in Section 4.7.5.
11. The estimated core-damage frequency, the timing of the associated core damage, a list of analytical assumptions including their bases, the sources of uncertainties.	Section 4
12. Any fire-induced containment failures identified as being different than those identified in the internal events analysis and other containment performance insights.	Section 4.8
13. Documentation with regard to fire risk scoping study issues addressed by the submittal, the bases and assumptions used to address these issues, and a discussion of the findings and conclusions. Evaluation results and potential improvements associated with the decay heat removal function should be specifically highlighted.	FRSS issues are addressed in Section 4.10. Evaluation results and potential improvements associated with decay heat removal are addressed in Section 4.11.
14. When an existing PRA is used to address the fire IPEEE, the licensee should describe sensitivity studies related to the use of the initial hazard supplemental plant walkdown results and subsequent evaluations. The licensee should examine the aforementioned list to fill in those items missed in the existing fire PRA.	Not applicable.

4.2 REVIEW OF PLANT INFORMATION AND WALKDOWNS

An extensive volume of plant information was used to perform the Dresden Fire IPEEE Analysis. This information was obtained from plant documentation. When necessary, this information was supplemented by additional investigation and walkdowns.

The following is a discussion of the sources of information used in this analysis:

The Dresden Fire Hazards Analysis (FHA) in the Dresden Fire Protection Report (FPR) (Ref. 4-8) was used to obtain:

- Plant layout for defining the fire areas and compartments;
- Barrier information for calculating failure probability; and

- Detection and suppression data for fire modeling.

The Appendix R Safe Shutdown Analysis (SSA) in the Dresden FPR (Ref. 4-8) was used to determine:

- Systems and components used for Appendix R safe shutdown;
- Location and function of the Appendix R safe shutdown cables and circuits; and
- Post-fire manual actions.

The Sargent & Lundy Interactive Cable Engineering (SLICE) Cable Database (Ref. 4-9) in conjunction with plant electrical drawings was used as a source for cable location. In addition SLICE was also used in conjunction with plant electrical drawings to identify and locate offsite power cables.

The PRA model resulting from the 1999 Dresden PRA Model Upgrade was used to develop the probabilistic model to quantify fire-induced CCDPs.

Transient Combustible Control (Ref. 4-11) and Housekeeping (Ref. 4-12). Procedures were reviewed to assist in the process of selecting transient fire scenarios.

Plant Drawings were used in nearly every task to obtain and/or confirm data including; plant layout, enclosure data, location of fire hazards and protective systems, location of the cables and circuits, etc.

Plant Safe Shutdown Procedures (DSSPs) were used to identify and model post-fire manual actions.

Throughout the project, walkdowns were conducted by the IPEEE Team to obtain and/or confirm data. The composition of the walkdown teams varied depending on the information to be collected or confirmed.

Detailed preparation preceded each walkdown followed by documentation of the data obtained. The following is a list of the walkdowns performed and their purpose:

- Walkdowns were conducted to identify fire ignition sources.
- A walkdown of the Control Room was conducted to determine (a) ignition source loading and separation, (b) location of the detectors, and (c) control room ventilation and smoke removal capabilities.
- Walkdowns were conducted for all unscreened compartments to determine the locations of the fixed ignition sources with respect to

potential targets, the locations of detection and suppression systems with respect to the source and target, and the placements of combustibles near the fire barriers. This information was used in the preliminary screening of fire compartments (Section 4.7.1) and for detailed fire modeling (Section 4.7.2).

- Additional walkdowns of unscreened compartments were completed to evaluate transient fires.

4.3 FIRE HAZARD ANALYSIS

This section presents the evaluation of potential fire hazards at Dresden. It begins with defining the fire areas and compartments. It then characterizes the fire hazard by determining the types of fires and their frequency in each of the compartments.

4.3.1 Fire Area and Compartment Designations

4.3.1.1 Methodology

Fire area and compartment designations were developed based on the guidance provided in the FIVE methodology (Ref. 4-4). Each fire area, as defined in FIVE, Definition 2.2, was found to be sufficiently consistent with the Appendix R fire area definitions to support qualitative screening. A fire area was screened if it did not contain any Appendix R safe shutdown features and a postulated fire did not result or cause a plant trip initiator. The unscreened fire areas were then examined to subdivide them into fire compartments. Qualitative screening of fire compartments required that the compartment boundaries satisfy the FIVE criteria. Fire compartment screening also required that the compartment not contain Appendix R safe shutdown features.

The original fire analysis was based on 83 fire zones, which was then subdivided into 163 fire compartments. During the course of the re-analysis, it was determined that this level of resolution did not have any notable impact on the analysis results as compared to a case that was based on the existing Appendix R Fire Zone definitions. In order to simplify the process of integrating the fire compartment definition with the available cable data, the analysis upgrade effort elected to rely on the Appendix R fire area and zone definitions and the basis for the fire risk analysis compartments. Subdivision into smaller compartments was not performed. The fire PRA upgrade effort was based on the division of the Dresden plant into 19 fire areas which are then subdivided into 83 fire compartments. The following provides a description and fire area designation of these 83 fire compartments.

Table 4-2
Upgraded Fire PRA Analysis Fire Compartments

Fire Compartment ID	Fire Area ID	Fire Compartment Description
1.1.1.1	RB3-II	U3 TORUS BASEMENT
1.1.1.2	RB3-II	U3 RX BLDG GROUND FLOOR
1.1.1.3	RB3-II	U3 SECOND FLOOR RX BLDG
1.1.1.4	RB3-II	U3 RX BLDG SWITCHGEAR AREA
1.1.1.5.A	RB3-I	U3 ISCO FLOOR
1.1.1.5.B	RB3-I	U3 ISCO PIPE CHASE
1.1.1.5.C	RB3-I	U3 ISCO PIPE CHASE
1.1.1.5.D	RB3-II	U3 SKIMMER SURGE TANK ROOM
1.1.1.6	RB2-II/RB3-II	U3 REFUEL FLOOR
1.1.2.1	RB2-II	U2 TORUS BASEMENT
1.1.2.2	RB2-II	U2 RX BLDG GROUND FLOOR
1.1.2.3	RB2-II	U2 SECOND FLOOR RX BLDG
1.1.2.4	RB2-II	U2 RX BLDG SWITCHGEAR AREA
1.1.2.5.A	RB2-I	U2 ISCO FLOOR
1.1.2.5.B	RB2-I	U2 ISCO PIPE CHASE
1.1.2.5.C	RB2-I	U2 ISCO PIPE CHASE
1.1.2.5.D	RB2-II	U2 SKIMMER SURGE TANK ROOM
1.1.2.6	RB2-II/RB3-II	U2 REFUEL FLOOR
1.2.1	U3-PC	U3 DRYWELL
1.2.2	U2-PC	U2 DRYWELL
1.3.1	RB3-II	U3 SHUTDOWN COOLING PUMP ROOM
1.3.2	RB2-I	U2 SHUTDOWN COOLING PUMP ROOM
1.4.1	RB3-I	U3 TIP ROOM
2.0	TB-V	CONTROL ROOM
6.1	TB-III	U3 BATTERY CHARGER ROOM
6.2	TB-V	AUX ELEC EQUIP ROOM
7.0.A.1	TB-I	U2 BATTERY ROOM
7.0.A.2	TB-I	125 VDC BATT. RM.
7.0.A.3	TB-I	250 VDC BATT. RM.
7.0.B	TB-III	U3 STATION BATT. ROOM
8.1	TB-I	CLEAN/DIRTY OIL ROOM

Table 4-2
Upgraded Fire PRA Analysis Fire Compartments

Fire Compartment ID	Fire Area ID	Fire Compartment Description
8.2.1.A	TB-I	U2 COND. PP AREA
8.2.1.B	TB-III	U3 COND. PP AREA
8.2.2.A	TB-I	U2 CRD PUMP ROOM
8.2.2.B	TB-III	U3 CRD PUMP ROOM
8.2.4	TB-III	U3 CABLE TUNNEL
8.2.5.A	TB-I	U2 NORTH TRACKWAY/SWGR AREA
8.2.5.B	TB-I	U2 LP HEATER BAY
8.2.5.C	TB-II	2/3 TB CORRIDOR
8.2.5.D	TB-III	U3 LP HEATER BAY
8.2.5.E	TB-III	U3 WEST CORRIDOR AND TRACKWAY
8.2.6.A	TB-I	CONTROL ROOM BKUP VENTILATION
8.2.6.B	TB-I	U2 MEZZANINE
8.2.6.C	TB-II	U2/3 SBGT & TBCCW HX
8.2.6.D	TB-III	U3 MEZZANINE FLOOR
8.2.6.E	TB-III	U3 MEZZANINE FLOOR
8.2.7	TB-I	VENT ROOM OVER NE SWGR
8.2.8.A	TB-IV	U2/3 TURBINE OPERATING FLOOR
8.2.8.B	TB-IV	VENT FLOOR
8.2.8.C	TB-IV	VENT FLOOR
8.2.8.D	TB-IV	VENT FLOOR
9.0.A	TB-I	U2 D/G
9.0.B	TB-III	U3 D/G
9.0.C	RB-2/3	U2/3 DG
11.1.1	RB3-II	U3 LPCI PP ROOM DIV II
11.1.2	RB3-II	U3 LPCI PP ROOM DIV I
11.1.3	RB-2/3	U3 HPCI
11.2.1	RB2-II	U2 LPCI PP ROOM DIV II
11.2.2	RB2-II	U2 LPCI PP ROOM DIV I
11.2.3	RB-2/3	U2 HPCI
11.3	CRIB	CRIBHOUSE UPPER
14.1	RW	RADWASTE BLDG.

Table 4-2
Upgraded Fire PRA Analysis Fire Compartments

Fire Compartment ID	Fire Area ID	Fire Compartment Description
14.2.A	TB-IV	U2 STEAM JET AIR EJECTOR ROOM
14.2.B	TB-IV	U2 OFF GAS RECOMBINER
14.2.C	TB-IV	U2 OFF GAS CONDENSER
14.3.A	TB-IV	U3 STEAM JET AIR EJECTOR ROOM
14.3.B	TB-IV	U3 OFF GAS RECOMBINER
14.3.C	TB-IV	U3 OFF GAS CONDENSER
14.4	OG	OFF GAS FILTER BUILDING
14.5	RW	RADWASTE BLDG.
14.6	RW	MAX RECYCLE
18.1.1	XFMR	U3 MPT
18.1.2	XFMR	U2 MPT
18.2.1	XFMR	U3 UAT
18.2.2	XFMR	U2 UAT
18.3.1	XFMR	U3 RAT
18.3.2	XFMR	U2 RAT
18.4	OUTDOOR	AUX BOILER HOUSE
18.6	OUTDOOR	SBO BATT RM
18.7.1	MISC	ISCO PP HOUSE NORTH
18.7.2	MISC	ISCO PP HOUSE SOUTH
OUTDOOR	OUTDOOR	H2 STORAGE FACILITY
OUTDOOR	OUTDOOR	UNIT 1

4.3.1.2 Results

A total of 19 fire areas were used to describe the Dresden Units 2 and 3 plant site. The two fire areas corresponding to the drywell areas were screened as were the fire compartments representing the Auxiliary Boiler House, the Hydrogen Storage Facility, and Unit 1. The bases for this screening are as follows:

1. The Units 2 and 3 Primary Containment fire areas (IPEEE Fire Zone /Compartment Nos. 1.2.2 and 1.2.1) are a special case as described in detail in the Dresden FPR (Ref. 4-8). The Primary Containment for each unit is the drywell and suppression chamber (Torus) which are normally inerted with Nitrogen when the Units are at power operation.

2. The outdoor Hydrogen Storage Facilities, the Auxiliary Boiler House, and the Unit 1 areas were screened because they do not contain safe shutdown equipment and no plant trip initiator would result from a fire in the zone.

The remaining 78 fire compartments were subjected to additional analysis to determine the potential ignition sources in each compartment and to develop a fire ignition frequency for each compartment.

4.3.2 Fire Hazard Characterization

4.3.2.1 Methodology

Fire compartment ignition frequency walkdowns were done for unscreened fire compartments based on guidance provided in FIVE, (Ref. 4-4), and the Fire PRA Implementation Guide (Ref. 4-5). The process consisted of reviewing the FHA to identify likely ignition sources within each fire compartment. If accessible, a walkdown for each fire compartment was completed. If a fire compartment was not accessible, an alternate method was used to determine the ignition sources within the room (i.e., interviews with personnel knowledgeable of plant configuration). The quantity of cables within each fire compartment was allocated using plant information.

4.3.2.1.1 Ignition Frequency Walkdowns

Ignition sources were identified in each compartment primarily via plant walkdowns. However, some compartments were inaccessible to the walkdown team during plant operation. These were handled as follows:

1. If an inaccessible compartment had a mirror-image counterpart which was accessible in the opposite unit, ignition sources were counted in the accessible compartment and assumed to be nearly identical in both compartments; or,
2. In some cases, where compartments were inaccessible for radiological reasons, they were viewed via videotapes, or photographs, and/or plant drawings were used to count the components; or,
3. In some cases, ignition source counts in inaccessible compartments were obtained by interviewing plant personnel familiar with specific areas of the plant.

4.3.2.1.2 Ignition Frequency Calculations

After all of the accessible fire compartments were walked down, a fire ignition frequency was calculated for each fire compartment per the methodology given in FIVE (Ref. 4-4, Section 6.3.1). The methodology assigned fire frequencies to each compartment in proportion to the number and types of plant equipment and components located there. The fire ignition frequency calculations took into account both fixed and transient ignition sources.

The FIVE method for calculating ignition frequencies calls for development of weighting factors to adjust the generic frequencies provided in FIVE for application to particular locations and ignition sources in the subject plant. Location weighting factors (W_L) were calculated in accordance with the guidance provided in Reference Table 1.1 of Attachment 10.3 in FIVE, with some modifications as explained below. Location weighting factors used in the Dresden fire ignition frequency calculations are shown in Table 4.3. Ignition source weighting factors (W_S) were calculated in accordance with the guidance provided in Reference Table 1.2 of Attachment 10.3 in FIVE, with some modifications as explained below.

The FIVE method for calculating W_L and W_S for switchgear rooms assumes that 4 kV switchgear are segregated in separate switchgear rooms and that the electrical cabinets are distributed uniformly among all the switchgear rooms. However, at Dresden, high voltage switchgear are distributed among several zones in the Turbine, Reactor, and Diesel Generator Buildings, (namely 1.1.2.3, 1.1.1.3, 1.1.2.4, 1.1.1.4, 8.2.5.A, 8.2.5.E, 8.2.6.A, 8.2.6.C, 8.2.6.E, 9.0.C). Moreover, the number of switchgear cubicles are not uniform among the Dresden switchgear areas. Therefore, an adjustment of the method was needed to obtain more accurate fire ignition frequencies for the areas where switchgear are located at Dresden. W_L was set equal to 2 to account for the switchgear equipment for two units at Dresden. Then W_S was used to apportion the generic frequency for switchgear fires to each switchgear area according to the number of switchgear cubicles present. W_S was calculated by dividing the number of switchgear cubicles in a switchgear area by the total number of switchgear cubicles in all of the switchgear areas for Dresden Unit 2 and 3.

The FIVE method for calculating W_L and W_S for battery rooms assumes all batteries in the plant are segregated in separate battery rooms, and that batteries are uniformly distributed among the rooms. Batteries are found in some locations of the Dresden plant outside of battery rooms (namely fire zones 8.2.8.A and 8.2.6.E). In these two locations, batteries were conservatively assumed to represent a frequency contribution for battery fires over and above that of the normal complement of batteries in a typical two unit plant. W_S was used to assign a frequency contribution to the additional batteries proportional to the number of battery cells present. For example, there are two full banks of batteries in 8.2.8.A (the Units 2 and 3 Turbine Operating Floor). W_S was set equal to 2.0 for battery fires in 8.2.8.A to account for a frequency contribution approximately equivalent to two battery rooms.

The FIVE method for calculating W_L and W_S for intake structures assumes the equipment in the intake structures is evenly distributed among the intake structures. Dresden has two intake structures, one serving Unit 1 and one serving Units 2 and 3. Except for diesel fire pumps in the Unit 1 Intake Structure, all of the ignition sources which could impact operation of Units 2 and 3 is concentrated in the Unit 2/3 intake structure. Therefore, W_L was assigned a value of 2.0 for the Unit 2/3 Intake Structure, and, conservatively, a value of 1 for the Unit 1 Intake Structure. The FIVE method does not provide a weighting factor method for W_S for pumps in the Intake Structure. (The method calls for weighting factor method A, that is, no ignition source weighting factor method is used.) However, the Dresden Unit 2/3 Intake Structure is divided into two fire zones, one zone containing more pumps than the other. Therefore, for the Dresden ignition frequency calculation, W_S was used to apportion the frequency for pumps between the two zones, where W_S was calculated as the number of pumps in the zone divided by the number of pumps in the Intake Structure.

Transient ignition source weighting factors were assigned according to the activities permitted or likely to occur in the compartment, per the guidance in FIVE (Ref. 4-4), Attachment 10.3. Transient fires due to maintenance preheating activities, extension cords, heaters, hot pipes and welding were considered credible and were assumed to be equally likely throughout the plant. Neither candles nor cigarettes are allowed in the plant. Therefore, they are not considered credible transient ignition sources at Dresden.

Table 4.3
Location Weighting Factors

Location	W_L	Discussion
Reactor Building	1	All zones in the Unit 2 and 3 Reactor Buildings except those designated otherwise below.
Diesel Generator Rooms	1	9.0.A, 9.0.B, and 9.0.C
Switchgear Rooms	2	1.1.1.3, 1.1.2.3, 1.1.2.4, 1.1.1.4, 8.2.5.A, 8.2.5.E, 8.2.6.A, 8.2.6.C, 8.2.6.E, 9.0.C
Battery Rooms	0.5	7.0.A.2, 7.0.A.3, 7.0.B, 18.6
Control Room	2	2.0
Cable Spreading Room	N/A	(There is no cable spreading room at Dresden Station)
Unit 1 Intake Structure	1	U1 Cribhouse
Unit 2/3 Intake Structure	2	11.3.1 and 11.3.2
Turbine Buildings	1	All Turbine Building zones except those designated otherwise above or below; 6.1, 7.0.A.1; 14.2.A, B and C; 14.3.A, B, and C; 14.4
Radwaste Areas	2	14.1, 14.5, 14.6
Switchyards	.67	18.1.1, 18.1.2, 18.2.1, 18.2.2, 18.3.1, 18.3.2, U1 Switchyard
Plant-Wide Components	2	All plant zones

The fire frequencies associated with open junction boxes and auto-ignition of exposed unqualified cable in trays were assigned to each zone in proportion to the length of cable routed through each fire zone, as documented in the Fire Hazards Analysis portion of the Dresden Fire Protection Report (Ref. 4-8). The total length of cable was apportioned equally among all the compartments in a given zone.

4.3.2.2 Results

Fire ignition frequencies were developed for the Dresden IPEEE fire compartments remaining after FIVE Phase I (Ref. 4-4) qualitative screening.

All reported ComEd fires were reviewed and dispositioned according to the EPRI Fire Events Database (FEDB) (Ref. 4-13), which is the basis for the generic fire frequencies reported in FIVE. This was done to identify any major differences in the ComEd data which would affect the final fire compartment frequency calculations. There are two fire ignition sources observed in the ComEd data that did not appear in the EPRI database: (1) refuel hoists, and (2) isophase bus ducts. A separate location-specific fire frequency term was added to the applicable locations as follows:

Refuel hoists: 4 events/1264 reactor years = 3.2E-3/reactor-year
Isophase bus ducts: 3 events/1264 reactor years = 2.4E-3/reactor-year

4.4 FIRE GROWTH AND PROPAGATION

4.4.1 Detailed Fire Modeling

The unscreened fire compartments were initially evaluated based on an assumed exposure fire wherein all cables and equipment within the boundaries defined by the compartment were assumed to be damaged. In many cases, the resultant calculated CDF contribution satisfied the screening criteria of 1.0E-7/yr. No further analysis was performed for screened fire compartments other than the multi-compartment analysis as described in Section 4.7.3. Unscreened fire compartment were analyzed further to develop a more realistic treatment of credible fire events. These analysis refinements included performing fire modeling analyses using the guidance provided in the EPRI FIVE Methodology (Ref. 4-4) and the EPRI Fire PRA Implementation Guide (Ref. 4-5). The modeling effort was based on extensive plant walkdowns, and review of controlled drawings and related fire protection documents. Ignition sources were examined in the field to determine if they were capable of propagating a fire. Fire scenario geometry reflecting the locations of the ignition sources, PRA targets, and intervening combustibles was determined in the field. Data determined in the field also included identification of fire protection features such as suppression, detection, and other features which serve to protect PRA targets.

4.4.1.1 Development of Fire Scenarios

A fire scenario is a physical description of the ignition and development of a fire and the resulting potential consequences. Characterizing a fire scenario involves defining the amount of material involved in the fire, the intensity of burning, and the timing and physical placement of the fire with respect to PRA targets (equipment and circuits).

Fire scenarios are defined based on the data collected in the field. A scenario includes the following:

- Location of the ignition source and its burning characteristics;
- Location and geometry of intervening combustible materials (close enough to be ignited by the ignition source), and their burning characteristics;
- Location of the PRA targets;
- Type of raceway containing the PRA targets (i.e., solid bottom cable tray, conduit);
- Geometry of the compartment; and
- Fire protection features in close proximity to the source and targets.

Fire modeling of scenarios initially assumed that suppression systems were not functioning. This approach provided an assessment of the potential impact on PRA targets from a fire in the event it was not suppressed. Suppression systems were subsequently credited, where appropriate and justified, in calculating the potential for damage to PRA targets as addressed in Section 4.7.2.

4.4.1.2 Calculating Fire Exposure

Fire exposure temperatures were calculated at the PRA targets based on fire modeling correlations described in FIVE (Ref. 4-4), and supplemented by fire modeling data presented in the Fire PRA Implementation Guide (Ref. 4-5). The correlations define pre-flashover exposure temperatures in the following regions of fire influence:

- Targets directly above the source, i.e., those targets exposed to convective heating from the plume;
- Targets located to the side of the fire source and close to the ceiling, i.e., those targets exposed to the ceiling jet;
- Targets located in close proximity to the fire source, i.e., those targets exposed to radiative heating from the exposure fire; and
- Targets outside the plume and below the ceiling jet, i.e., those targets exposed to the hot gas layer (HGL).

The input variables in the correlations include distances, ignition source, and intervening combustible burning characteristics. The key inputs to the EPRI FIVE fire modeling worksheets are described below:

- Target Damage Threshold: 425°F - the plant cables consist of IEEE 383 and non-IEEE 383 qualified cables. Since the damage threshold from the FIVE Methodology (Ref. 4-4) for non-IEEE 383 cables is lower than that of IEEE 383 cables, the lower value was selected.
- Heat Loss Factor: 0.70 – the recommended value from FIVE is applied. In the case of the multi-compartment analysis, a Heat Loss Factor of 0.85 is used. This is discussed further in Section 4.7.3.
- Critical Radiant Flux: 0.5 Btu/s/ft^2 and 1.0 Btu/s/ft^2 - for non-qualified and all other components, respectively.
- Target Thermal Response Parameter: 16 - PE/PVC Electrical Cables - limiting.
- Rated Actuation Temperature of Detector/Suppression: selected based on the installation specifics.
- Time Constant of Detection Device: selected based on the installation specific.
- Ambient Temperature: 90°F - this assumed value was used for the maximum expected ambient temperature during normal plant operations. For purposes of suppression modeling, the minimum ambient temperature was assumed to be 75°F. The time for suppression actuation is dependent on the time to raise the ambient temperature to the actuation temperature of the detector. The lower the ambient temperature the longer the time for actuation.

4.4.2 Equipment and Material Burning Characteristics

The burning characteristics of equipment and materials involved in the fire scenarios are characterized by a heat release rate (HRR) and heat content (Q_{tot}). FIVE (Ref. 4-4) and the Fire PRA Implementation Guide (Ref. 4-5, Appendix E) were the primary sources for this information. The following paragraphs address the equipment/ material burning characteristics for the fire types which have been specifically selected to represent Dresden (English Units - BTUs, feet).

4.4.2.1 Oil Spill Fires

Oil spill fires were treated using the guidance provided in FIVE (Ref 4-4) and the EPRI Fire PRA Implementation Guide (Ref. 4-5). 18% of the postulated oil spill fires were treated as "large" oil spills. The remaining 82% were treated as "small" oil spills. In the case of air compressors, the guidance provided in the EPRI Fire PRA Implementation Guide (Ref. 4-5) recommends 2% of oil spills should be treated as "large". However, the Dresden fire analysis did not credit this lower severity factor. The selection of these severity factors was consistent with the information in the EPRI Fire Events Database (Ref. 4-13).

A postulated "large" oil spill fire was evaluated based on the volume of oil contained within an individual cavity of a component, such as a pump or motor, being evaluated as an ignition source. This did not necessarily involve the total volume of oil for the entire component. A small oil spill was assumed to involve the larger of 2 quarts or 10% of the large oil spill. In cases where the largest oil spill associated with a component was of the same order of magnitude as a small fire as defined here, such as an electric motor containing ½ to 1 gallon of oil, the small fire scenario considered failure of the individual component itself with no fire propagation.

The parameters for all postulated oil spills are provided below. These parameters are based on heavy fuel oil for heat characteristics and DTE 797 for spill characteristics. These bases were selected in order to provide bounding results for the fire modeling analyses.

1) Mobile DTE 797:
   a) Unconfined spill thickness: 0.013 inch
   b) Spill Specific Area: 120 ft^2/gal

2) Fuel Oil, Heavy:
   a) Net Heat of Combustion: 17,111 Btu/lbm
   b) Ideal Unit Mass Loss Rate: 0.007 lbm/s-ft2
   c) Estimated Combustion Efficiency: 0.9
   d) Approximate Unit Heat Release Rate: 110 Btu/s-ft2
   e) Density: 60 lbm/ft3

4.4.2.2 Electrical Fires

Electrical fires were evaluated using criteria and methods consistent with FIVE (Ref. 4-4) and the EPRI Fire PRA Implementation Guide (Ref. 4-5). Five types of electrical fires were evaluated. These types were low voltage cabinets and panels, Motor Control Centers (MCCs), low voltage Buses, medium voltage Switchgears, and Transformers.

Low Voltage Cabinets and Panels: This classification addressed miscellaneous electrical enclosures containing low voltage circuits (less than 600 Volts). These enclosures are not susceptible to explosive type events. Enclosures in this category include control panels and cabinets, termination cabinets, and enclosed power distribution panels not addressed by the other classifications. Fire propagation is not considered to be credible if the panel is substantially sealed. If fire propagation is considered credible a 400 BTU/s heat rate and 15 minute fire duration were used.

Motor Control Centers: This classification addressed both AC and DC MCCs. Fires in MCCs typically involve individual cubicle fires. However, the treatment of a postulated MCC fire assumed the functional failure of the entire MCC (loss of power). Walkdown inspections of the MCCs which were located below raceways containing circuits associated with credited safe shutdown functions found that the cable entries were sealed. Cables which entered the top of the MCC were provided with individual fittings. While these fittings were not considered to be a rated fire barrier, the lack of a concentrated mass of combustibles through a common penetration greatly reduced the likelihood of fire propagation. As such, propagation of a postulated fire beyond the boundaries of the MCC is not considered likely. However, in order to address potential uncertainty, 10% of postulated fires were assumed to propagate vertically beyond the boundaries of the MCC and potentially challenge targets directly above the MCC.

<u>Low Voltage Buses</u>: This classification addressed 480 VAC and DC buses. Fires in these enclosures typically involve individual cubicle fires. However, the treatment of a postulated bus fire assumed the functional failure of the entire bus (loss of power). Walkdown inspections of the bus enclosures found that the cable entries are sealed. As such, propagation of a postulated fire beyond the boundaries of the enclosure is not considered likely. However, in order to address potential uncertainty, 10% of postulated fires were assumed to propagate vertically beyond the boundaries of the enclosure and potentially challenge targets directly above.

<u>Medium Voltage Switchgears</u>: This classification addressed 4.16kV switchgear. Fires in switchgears typically involve individual cubicle fires. However, the treatment of a postulated switchgear fire assumed the functional failure of the entire switchgear. Walkdown inspections of the switchgears found that the cable entries are sealed. As such, propagation of a postulated fire beyond the boundaries of the switchgear is not considered likely. However, in order to address potential uncertainty, 20% of postulated fires were assumed to propagate beyond the boundaries of the switchgear. This included horizontally through the switchgear front due to an explosive event. Horizontal propagation through the back of the switchgear was not considered to be a credible consequence of a fire event.

<u>Transformers</u>: This classification addresses 4kV, 480VAC and off-site transformers. The 4kV/480V transformers are filled with Pyranol which was previously evaluated and determined to be not combustible. As such, the fire hazard which is presented is not significant when compared to a combustible oil filled transformer. The upgraded fire analysis took credit for this noncombustible oil and did not assume any oil based fire involving the transformers filled with Pyranol. A postulated transformer fire assumed the functional failure of the entire transformer. Oil based fires were considered for the off-site transformers without consideration of severity factors.

<u>Self-Initiated Cable Fires</u>: A review of the Dresden Appendix R related documentation and discussions with design engineering staff determined that the majority of the cabling in the plant is not IEEE 383 qualified. The upgraded fire analysis evaluated the implication of this by explicitly evaluating cable tray fires in the Unit 3 cable tunnel, and those raceways containing circuits associated with the Automatic Depressurization System (ADS). Recognizing the potential risk significance of spurious ADS actuation

scenarios, a series of self-initiated cable tray fire scenarios was developed to investigate the potential risk significance of fire induced spurious ADS actuation. The scenarios involving self-initiated cable fires resulting in failure in all circuits in the raceway were analyzed with a severity factor of 0.20. This factor was developed based on a review of incidents in the EPRI Fire Events Database (Ref. 4-13).

4.4.2.3 Transient Fires

Transient fires can be postulated to occur. However, such fires require both an ignition source and combustibles. The fire modeling analyses presented in this report explicitly evaluated in-situ ignition sources and combustibles. Transient ignition sources were not explicitly treated. Transient ignition sources such as that associated with hot work are controlled by plant procedures which include a requirement that a fire watch be posted. The likelihood that a significant fire resulting from a transient ignition source that causes damage to critical targets due to failure of pre-work controls and the fire watch is judged to be low enough to be excluded from explicit treatment. Other potential transient ignition sources such as those considered in the development of the compartment ignition frequencies were also screened. This is based on the specific fire modeling analyses and walkdown inspection that were performed. The fire modeling analyses evaluated the consequences of postulated fires involving in-situ sources. The treatment of these sources included the explicit consideration of non-severe fire events. The combination of the scenarios for the severe and non-severe fire events bounds the potential impact of the screened transient ignition sources.

Another potential source of a transient related fire involves combustible material storage. However, such a fire requires the presence of an ignition source of sufficient intensity to cause ignition and continued combustion. The bounding treatment of fires in this report is considered to be sufficient to address this issue given the lack of significant transient combustible storage adjacent to ignition sources observed during plant walkdowns. In addition, it was observed that containers intended for normal occupancy refuse were metal and fitted with FM approved self-extinguishing covers. As such, this transient combustible source was screened.

4.4.3 Fire Growth and Propagation Characteristics

The modeling generally assumed that fires reached their peak heat release rate immediately upon ignition. This minimized the time for automatic and manual suppression to react and therefore produced bounding results.

Propagation of cable fires was approximated as follows:

- Vertical runs of cable, if ignited, were assumed to propagate up to the room boundary or until the cable changed direction and traversed horizontally.
- Fires in horizontal tray stacks (ladderback trays) were assumed to propagate upward from tray to tray as described in Appendix I of the Fire PRA Implementation Guide (Ref. 4-5).

- Barriers are credited with limiting damage to PRA targets as described in Appendix J of the Fire PRA Implementation Guide (Ref. 4-5).

The potential for a given fire generating a hot gas layer within its respective fire compartment was not postulated for those spaces that have substantial ventilation openings. Only those openings that provide an ascending vertical vent path were considered. Available ventilation pathways as well as potential fire induced boundary failures were also evaluated in the multi-compartment analysis as discussed in Section 4.7.3. In order to ensure that the potential for localized heating outside the fire plume and ceiling jet region was considered for substantially ventilated compartments, a minimum allowable margin of 50 °F between the calculated target temperature and the damage threshold was used in lieu of the hot gas layer analysis developed in the FIVE worksheets.

4.5 EVALUATION OF COMPONENT FRAGILITIES AND FAILURE MODES

The potential damage to a component that could result due to a postulated fire was determined in several steps. First, the scope of plant systems, associated equipment, and functions credited for post fire plant trip response was determined using the plant fire PRA model. A review of plant design drawings and cable databases, including Appendix R data sources was then performed to identify those cables and circuits whose failure given a postulated fire event would result in the equipment failure(s) treated in the PRA model and their location in the plant. This data was then linked to the fire compartment designators and the fire PRA model. The analysis assumed that fire induced cable or circuit failure resulted in loss of associated equipment functionality unless supplemental evaluations are performed to confirm and justify no loss of equipment function.

4.5.1 Equipment Damage Criteria

Threshold values for damage and ignition of PRA targets and intervening combustible materials are specified in the Fire Risk Analysis Implementation Guide (Ref. 4-5) as follows:

- Damage temperature for unqualified cables: 425 °F
- Ignition temperature for unqualified cables: 425 °F
- Critical Heat Flux for unqualified cable 0.5 $Btu/s/ft.^2$
- Thermal response of IEEE unqualified cable 16 $(Btu/sec\text{-}ft.^2)^{1/2}$
- Damage temperature of solid state electronics: 150 °F

The timing of target damage followed the guidelines that are based on the Fire PRA Implementation Guide (Ref. 4-5). The time to target damage given the expose to temperatures at or above the damage threshold was determined using the transient fire modeling worksheets developed as part of the EPRI FIVE Methodology (Ref. 4-4).

4.5.2 Cable/Circuit Location

This section of the report describes the development of the cable/circuit and equipment location database for the upgrade fire risk analysis. The database was used to identify the spatial distribution of plant system features credited for post fire plant shutdown that could be exposed to fire induced damage.

4.5.2.1 Scope

The scope of this report is limited to those plant systems credited for performing post fire plant shutdown functions as described in Section 4.5.4.2. These systems can be grouped into three categories - inventory control systems, decay heat removal systems, and support systems. The cables and circuits necessary for these systems to function were identified using a variety of data sources.

4.5.2.1.1 Methodology

The cable/circuit and equipment data was developed by using the information available from a variety of sources and documents located at Dresden Station. The information was evaluated for accuracy and verified with as many sources as possible. How each data source was used is discussed in the following sections.

4.5.2.1.2 Safe Shutdown Equipment List

The Safe Shutdown Analysis (SSA) and other related Appendix R documents were used to obtain critical spatial location information for certain plant system equipment. This information included cable and circuit information. The cable and circuit information associated with the Appendix R Program are contained in a computerized database called SLICE. The SLICE data files were used to identify the cables associated with the Appendix R equipment, their location in terms of fire compartments, and their individual routing throughout the plant. The scope of plant systems whose cable information was obtained directly from the Appendix R related information sources were:

- High Pressure Coolant Injection (HPCI)
- Isolation Condenser (IC)
- Automatic Depressurization System (ADS)
- Low Pressure Coolant Injection – Injection Mode
- Low Pressure Coolant Injection - Suppression Pool Cooling mode
- Shutdown Cooling (SDC)
- Support Systems – i.e., AC/DC power, RHRSW, etc.

4.5.2.1.3 Electrical Elementary/Schematic Diagrams

The scope of plant systems credited in the fire PRA model included selected non-Appendix R systems. The cable and circuit information for these additional systems was identified by examining the electrical diagrams. The fire PRA model was reviewed to identify the non-Appendix R plant system components and functions. The appropriate Dresden design drawings were then reviewed to identify the circuits and cables whose fire induced failure may result in the failure of the component. This cable mapping process included those cables whose failure individually would result in component failure. It also included those cables whose failure must occur in combination with other circuit failures in order to cause component failure. It was recognized that this mapping process would result in cases where the data would indicate component failure when an insufficient set of cables were concurrently impacted. Supplemental reviews were performed on a fire compartment and scenario basis following initial fire PRA model quantification to identify and modify the treatment of these occurrences on an as-needed basis.

The scope of plant systems whose cable information was obtained by specific review of drawings were Feedwater (FW), Condensate (Cond), Control Rod Drive (CRD), Core Spray (CS), the non-Appendix R credited train of LPCI, Torus/Drywell Vent (TDV), and their associated support systems. The scope of support systems included the offsite power supply circuit.

4.5.2.1.4 SLICE Engineering Tool

The SLICE Engineering Tool (Ref. 4-9) consists of a computer database and computer program that provides information about the electrical cables installed at Dresden. The database contains cable information such as associated SSD equipment, cable routing, fire zones, etc.

The SLICE database files include detailed routing information for each of the cables considered in this analysis. This routing information was used along with plant design documents, and walkdown inspections to determine the location of the cables. The location information was developed on two different levels of detail. The initial data set associated fire compartment designators to each of the cables. In this manner initial quantifications on a compartment basis could be performed. This data was then refined on an as-needed basis to develop the specific raceway routing points within a fire compartment that contain the various cables of concern to support fire modeling analyses.

4.5.2.1.5 Plant Walkdowns

The fire protection engineers completed site walkdowns as part of the fire ignition frequency task within the original IPEEE Project. The walkdowns were primarily used to identify fire ignition frequencies for the project but the walkdowns did provide additional information on equipment locations.

Additional plant walkdowns were performed to support the fire modeling effort and the multi-compartment analysis. These walkdowns focused on the arrangements of fire ignition sources, fuels, combustibles, targets, and the features and characteristics of the compartment boundaries. Many of the Dresden fire compartments in the Turbine Building were treated as ventilated spaces. While this provides the means for heat and smoke removal, it also creates the potential for postulated fires to impact targets in an adjacent compartment. The walkdowns examined these ventilation pathways to ensure that any targets or interactions with the adjacent compartment were properly addressed.

4.5.2.2 Development of the FRANC Cable Data Files

The cable data developed as described in the prior section was used to develop one of the Fire Risk Analysis Code (FRANC) data files (Ref. 4-20). For each piece of equipment credited in the fire PRA model, a data link was made to relate the set of cables whose fire induced failure would directly impact the equipment functionality. For each of these cables, another link to the fire compartments through which it is routed was also provided. The process of assigning fire compartment designators was structured to ensure that terminal end locations of the cable were properly reflected. For example, the routing information for a bottom entry cable into a motor control center may not properly show the fire compartment which contains the actual motor control center. The methodology addressed this potential. The resultant file contained necessary data relationships to provide a list of equipment potentially disabled for a fire in any given fire compartment.

4.5.2.3 FRANC Cable Data File Implementation

The FRANC cable data file was linked to another data file which contained the fire PRA model basic events. The FRANC program, in concert with the Fully Optimized Risk & Reliability Quantification Engine (FORTE) program (Ref. 4-21), performs quantifications of the fire PRA model by either setting the linked basic events to 'true' or 1.0, based on user inputs. Events set to 'true' typically represent equipment rendered unavailable by the fire, while events set to 1.0 represent equipment or operator actions that are not credited in the quantification.

4.5.3 Cable/Circuit Fire Induced Failure Modes

The upgraded fire analysis considered three potential fire induced cable failure modes - open circuit, short circuit, and hot shorts. A key feature of the analysis is that spurious equipment actuation is considered for all three failure modes.

Open circuit - postulated open circuit conditions would result in the interruption of power or control signals. In this case, components would align themselves in a de-energized state. Valves which require power to maintain a desired position would be assumed to change state. Relays which are normally energized would de-energize. The consequences of this relay action may include spurious actuation of mechanical system components. In no case were fire induced failures credited to assist components in achieving the desired state for this analysis. The open circuit failure mode was treated using a scenario specific conditional failure probability of 1.0.

Short circuit - postulated short circuit conditions were defined as those fire induced failures wherein the conductors of an individual cable become 'connected' together in any combination. The failure modes which were considered included shorting of all conductors in power circuits, and the selected shorting of conductors within individual control cables to cause spurious equipment actuation. For example, a control cable between a motor control center and the control room was treated using failure modes which included the shorting of conductors to generate a spurious valve open or close signal. As in the prior case, fire induced cable failures were not credited to assist components achieve the desired state for this analysis. This failure mode was treated using a scenario specific conditional failure probability of 1.0.

Hot short - this cable failure mode is a special case of the more general short circuit failure mode. This case involves an instance wherein the energized conductor(s) of a given cable become connected to the de-energized conductor(s) of another cable causing undesired spurious actuation of equipment associated with the second cable. This failure mode is very unlikely since it also requires that these 'shorted' conductors not include certain other conductors such as neutral or ground, and be connected long enough to cause the affected component to change state. As such, the application of a non-unity conditional failure probability is appropriate. However, the upgraded fire risk analysis does not credit the use of a non-unity conditional failure probability.

4.5.4 Fire PRA Model

4.5.4.1 Purpose

This section describes the method used to determine fire-induced conditional core damage probability (CCDP) for each of the fire compartments that were not qualitatively screened. The CCDPs were used in conjunction with the fire compartment fire ignition frequencies to determine the calculated core damage frequency (CDF) contributions.

The key differences in the fire PRA model developed for this upgraded analysis and that applied in the original Fire IPEEE are the treatment of operator actions and the treatment of postulated fire initiated events. The original analysis results were based primarily on the implementation of the Appendix R based post-fire safe shutdown procedures which address a bounding exposure fire event. The original analysis did not fully credit the Emergency Operating Procedures (EOPs). The upgraded analysis relied on implementation of the EOPs. As a result, all but one of the postulated fire scenarios in this upgraded analysis were quantified based only on the provisions of the EOPs. The Appendix R post-fire safe shutdown procedures are credited only for the bounding control room fire event. The original Fire IPEEE also considered only a limited set of fire initiated events. The upgraded fire analysis expanded the scope of initiating events to more realistically analyze the potential challenges.

4.5.4.2 Description of the Fire PRA Model

The Dresden fire PRA model was developed from the plant internal events PRA model. The development began with the selection of the potential fire initiated events. It was determined that the internal events PRA model structure was adequate to address all postulated fire induced initiating events except for an event involving spurious actuation of all ADS valves. This was treated by creating a new initiating event, %TP, for multiple spurious ADS valve openings. The event tree sequences for the %TP initiator were developed based on the structure for a large LOCA event from the plant internal events PRA model.

The set of fire initiated events that were addressed in the fire PRA model are listed below.

- %TT - Turbine Trip
- %TC - Loss of Main Condenser
- %TIA - Loss of Instrument Air
- %TI - Single Spurious ADS Valve Opening
- %TP - Multiple Spurious ADS Valve Opening
- %LOOP - Single Unit Loss of Offsite Power
- %DLOOP - Dual Unit Loss of Offsite Power

The determination of the appropriate initiating event given a postulated fire was determined as described in Section 4.5.4.5.

The systems in operation and use at Dresden were considered in the development of the fire PRA model. The front-line systems modeled and included were:

- Isolation Condenser (IC) System
- Feedwater and Condensate (FW) System
- Control Rod Hydraulic (CRD) System
- High Pressure Coolant Injection (HPCI) System
- Automatic Depressurization System (ADS)
- Low Pressure Cooling Injection (LPCI) System – Injection and Suppression Pool Cooling Modes
- Core Spray (CS) System
- Shutdown Cooling (SDC) System
- Containment Vents including the Torus and Drywell vent paths

The scope of front-line systems was supplemented with the necessary set of support systems. These support systems included the offsite power supply connections as well as the Emergency and Station Blackout Diesel Generators. The performance of these plant systems was modeled by fault trees to depict combinations of hardware faults, human errors, test and maintenance unavailabilities, and other events that can lead to a failure.

4.5.4.3 Cable and Equipment Spatial Location Information

The system functions and associated equipment treated in the Dresden fire PRA model were linked to spatial location information developed as described in Section 4.5.2. This data relationship allowed the analysis to identify those fire PRA model functions that were adversely impacted by a postulated fire event. The analysis was structured such that individual basic events in the fire PRA model were set to 'true' in the quantification for individual fire scenarios.

The analysis process considered basic events to be 'true' if the associated equipment or attendant cable(s) was within the critical spacing determined by the fire modeling analyses. If fire modeling was not performed for a given fire compartment, all basic events associated with that compartment were set to 'true'. While many of the model basic events were treated in this fashion, some mechanical equipment, such as piping, heat exchangers, tanks, valves without external automatic operators, were not considered to be susceptible to fire damage. Component failures associated with equipment in this category were left at their random failure probabilities.

4.5.4.4 Operator Actions

The fire PRA model incorporated all of the operator actions included in the plant internal events PRA model. A review of these operator actions was performed to identify those actions which occur within the control room and those which occur outside the control room. The incorporation of these operator actions effectively structured the model to be consistent with the plant normal, abnormal, and emergency operating procedure provisions. The additional recoveries that are potentially available via the Appendix R post-fire safe shutdown procedures were not explicitly incorporated into the model structure and were credited only for the bounding control room fire scenario.

All of the operator actions incorporated in the fire PRA model were reviewed to determine the timeline associated with the action and binned into six groups. Three bins were defined for actions which occur in the control room with three additional bins for actions outside the control room. One bin was defined for actions that must be completed within 30 minutes, another for the 30 to 60 minute time interval, and a final bin for actions with greater than 1 hour available to complete the action.

4.5.4.4.1 *Operator Actions in the Control Room*

Operator actions which are performed in the main control room were not considered to be adversely impacted by postulated fire events outside the control room. In the event of a postulated fire in the control room, operator actions may be affected. This was treated in the analysis assuming that any operator action with a required response time of 30 minutes or less fails given a postulated control room fire. The fire PRA model contained 9 operator actions which occur in the control room with a required response time of less than 30 minutes.

2HISY--------H--	Operator Controls HPCI and Prevents Overfill
2LI—CNTSINIAH--	Operator Initiate Containment Sprays (non-ATWS)
2LIOPLIA-INJ-H--	Operator Injects Through A Loop Given B Loop Failure
2LIOPLIB-INJ-H--	Operator Fails to Inject Through B Loop Given A Loop Failure
2MSOP-5699--AH--	Operator Fails to Control Pressure with Bypass Valve
OFWC2	Operator Fails to Transfer to Alt. Pwr – "C" RFP (SLOCA)
OP-ACT-2-3906D--	Operator Fails to Open SW to FP Cross-Tie (MOV 2-3906) from Control Room
OP-ACT-IC-MU	Operator Fails to Initiate IC Shell Side Makeup
OP-ACT-IC-MU2	Operator Fails to Initiate IC Shell Side Makeup Using Alternate Supply

4.5.4.4.2 *Operator Actions Outside the Control Room*

The fire PRA model included events representing human actions which may be taken outside the control room to mitigate the effects of system failures. These included actions such as local equipment actuation or manual alignment of backup systems. In the event of a fire, smoke and heat may delay or prevent the operator from performing the intended action. Therefore, an initial screening quantification was performed with all ex-control room human actions assigned an HEP of 1.0. This provides a bounding result in that this assumes there is no possibility of success regardless of the location of the fire and required action. The results of the initial screening quantification determined that two operator actions outside of the control room were risk significant.

- Operator fails to switch to alternate battery charger
- Operator fails to link to alternate 125 VDC battery bank

These two actions were evaluated on a fire scenario basis to determine whether it was appropriate to credit the operator action with a non-unity failure probability. All other operator actions outside of the control room were maintained in the analysis with a HEP of 1.0. The extensive use of a HEP of 1.0 for potential operator actions outside the control room is conservative but does not have a significant impact on the overall analysis results. This is because these events do not appear in the dominant cutsets for the analysis.

The HEP value calculated for the internal events analysis for the two operator actions outside the control room was considered to be applicable to the fire analysis without modification for selected fire compartments. This was based on the following conditions being satisfied. Fire compartments that did not satisfy items 2 and 3 below were analyzed based on these two operator actions having their HEP set to 1.0.

1. At least 30 minutes was available for completion of the action,
2. the action was not taken in the fire compartment as the fire location, and
3. the fire was not in the pathway from the control room to the location of the required action.

4.5.4.5 Fire Initiated Events

The postulated fire initiated event was determined based on an assessment of the fire induced failures. This was done by identifying fire affected circuits and equipment. The fire PRA model was structured to address the following initiating events.

- %TT - Turbine Trip
- %TC - Loss of Main Condenser
- %TIA - Loss of Instrument Air
- %TI - Single Spurious ADS Valve Opening
- %TP - Multiple Spurious ADS Valve Opening
- %LOOP - Single Unit Loss of Offsite Power
- %DLOOP - Dual Unit Loss of Offsite Power

The default initiating event for all CCDP/CDF quantifications was %TC – Loss of Main Condenser. However, many of the fire compartments required a different initiating event. For example, the majority of the Turbine Building was evaluated using the %TIA initiator because of the soldered copper piping used for the Instrument Air system. A postulated fire could significantly degrade or cause the failure of soldered joints. Such a failure could result in failure of the entire air system. The %TIA initiator also causes loss of the main condenser and feedwater systems, as well as the Torus/Drywell Vent. The majority of the Reactor Building was evaluated using the %TC initiator. The %TI, %TP, %LOOP, and %DLOOP initiators were selected based on the routing of cables whose fire induced failure would cause these events. The %TT initiator was selected when it was determined that the fire would cause a plant trip, but the impact of the fire was such that the main condenser would remain available.

A specific study was performed to assess the risk significance of the %TP initiator. This initiator would be caused by fire induced spurious actuation of multiple ADS valves. The control circuit design for the Dresden ADS system was reviewed to identify those circuits whose fire induced failure could cause spurious valve operation. The specific routing of the cables was obtained from the SLICE database and explicitly integrated into the upgraded fire analysis.

4.6 FIRE DETECTION AND SUPPRESSION

4.6.1 Automatic System Performance

A listing of automatic detection and suppression systems in the compartments analyzed is provided in Table 4-4. Incorporation of the effects of the detection and suppression systems in the analysis was based on data provided in FIVE (Ref. 4-4) and the Fire PRA Implementation Guide (Ref. 4-5) and are summarized in Tables 4.5 and 4.6. Automatic suppression was evaluated in those instances where there was a chance for the suppression system to actuate and extinguish a fire before damage would occur to a PRA target, for example, where a pump fire might damage a PRA target in the overhead. Sprinkler and spray systems are also credited for cooling hot gases and mitigating damaging HGL scenarios. CO_2 systems were not expected to cool hot gases, and were not credited in mitigating HGL damage.

Table 4-4
Compartment Fire Detection and Suppression

Fire Compartment	Fire Compartment Description	Suppression	Detection
1.1.1.1	U3 TORUS BASEMENT	none	partial
1.1.1.2	U3 RX BLDG GROUND FLOOR	partial	full
1.1.1.3	U3 SECOND FLOOR RX BLDG	partial	partial
1.1.1.4	U3 RX BLDG SWITCHGEAR AREA	none	full
1.1.1.5.A	U3 ISCO FLOOR	none	full
1.1.1.5.B	U3 ISCO PIPE CHASE	none	none
1.1.1.5.C	U3 ISCO PIPE CHASE	none	none
1.1.1.5.D	U3 SKIMMER SURGE TANK ROOM	none	partial
1.1.1.6	U3 REFUEL FLOOR	none	none
1.1.2.1	U2 TORUS BASEMENT	none	partial
1.1.2.2	U2 RX BLDG GROUND FLOOR	partial	full
1.1.2.3	U2 SECOND FLOOR RX BLDG	partial	partial
1.1.2.4	U2 RX BLDG SWITCHGEAR AREA	partial	full
1.1.2.5.A	U2 ISCO FLOOR	none	full
1.1.2.5.B	U2 ISCO PIPE CHASE	none	none
1.1.2.5.C	U2 ISCO PIPE CHASE	none	none

Table 4-4
Compartment Fire Detection and Suppression

Fire Compartment	Fire Compartment Description	Suppression	Detection
1.1.2.5.D	U2 SKIMMER SURGE TANK ROOM	none	partial
1.1.2.6	U2 REFUEL FLOOR	none	none
1.2.1	U3 DRYWELL	screened	screened
1.2.2	U2 DRYWELL	screened	screened
1.3.1	U3 SHUTDOWN COOLING PUMP ROOM	none	full
1.3.2	U2 SHUTDOWN COOLING PUMP ROOM	none	full
1.4.1	U3 TIP ROOM	none	full
2.0	CONTROL ROOM	none	full
6.1	U3 BATTERY CHARGER ROOM	none	full
6.2	AUX ELEC EQUIP ROOM	full	none
7.0.A.1	U2 BATTERY ROOM	none	full
7.0.A.2	125 VDC BATT. RM.	none	full
7.0.A.3	250 VDC BATT. RM.	none	full
7.0.B	U3 STATION BATT. ROOM	none	full
8.1	CLEAN/DIRTY OIL ROOM	full	none
8.2.1.A	U2 COND. PP AREA	full	none
8.2.1.B	U3 COND. PP AREA	full	none
8.2.2.A	U2 CRD PUMP ROOM	full	none
8.2.2.B	U3 CRD PUMP ROOM	full	none
8.2.4	U3 CABLE TUNNEL	full	full
8.2.5.A	U2 NORTH TRACKWAY/SWGR AREA	partial	partial
8.2.5.B	U2 LP HEATER BAY	full	none
8.2.5.C	2/3 TB CORRIDOR	partial	partial
8.2.5.D	U3 LP HEATER BAY	full	none
8.2.5.E	U3 WEST CORRIDOR AND TRACKWAY	partial	partial
8.2.6.A	CONTROL ROOM BKUP VENTILATION	partial	partial
8.2.6.B	U2 MEZZANINE	full	none
8.2.6.C	U2/3 SBGT & TBCCW HX	full	none
8.2.6.D	U3 MEZZANINE FLOOR	full	none

Table 4-4
Compartment Fire Detection and Suppression

Fire Compartment	Fire Compartment Description	Suppression	Detection
8.2.6.E	U3 MEZZANINE FLOOR	partial	partial
8.2.7	VENT ROOM OVER NE SWGR	none	full
8.2.8.A	U2/3 TURBINE OPERATING FLOOR	partial	none
8.2.8.B	VENT FLOOR	partial	none
8.2.8.C	VENT FLOOR	none	none
8.2.8.D	VENT FLOOR	none	none
9.0.A	U2 D/G	full	none
9.0.B	U3 D/G	full	none
9.0.C	U2/3 DG	full	none
11.1.1	U3 LPCI PP ROOM DIV II	none	full
11.1.2	U3 LPCI PP ROOM DIV I	none	full
11.1.3	U3 HPCI	full	full
11.2.1	U2 LPCI PP ROOM DIV II	none	full
11.2.2	U2 LPCI PP ROOM DIV I	none	full
11.2.3	U2 HPCI	full	full
11.3	CRIBHOUSE UPPER	full	partial
14.1	RADWASTE BLDG.	none	none
14.2.A	U2 STEAM JET AIR EJECTOR ROOM	none	none
14.2.B	U2 OFF GAS RECOMBINER	none	none
14.2.C	U2 OFF GAS CONDENSER	none	none
14.3.A	U3 STEAM JET AIR EJECTOR ROOM	none	none
14.3.B	U3 OFF GAS RECOMBINER	none	none
14.3.C	U3 OFF GAS CONDENSER	none	none
14.4	OFF GAS FILTER BUILDING	none	none
14.5	RADWASTE BLDG.	none	none
14.6	MAX RECYCLE	none	none
18.1.1	U3 MPT	full	none
18.1.2	U2 MPT	full	none
18.2.1	U3 UAT	full	none
18.2.2	U2 UAT	full	none
18.3.1	U3 RAT	full	none

Table 4-4
Compartment Fire Detection and Suppression

Fire Compartment	Fire Compartment Description	Suppression	Detection
18.3.2	U2 RAT	full	none
18.4	AUX BOILER HOUSE	screened	screened
18.6	SBO BATT RM	full	full
18.7.1	ISCO PP HOUSE NORTH	full	full
18.7.2	ISCO PP HOUSE SOUTH	full	Full
OUTDOOR	H2 STORAGE FACILITY	screened	screened
OUTDOOR	UNIT 1	screened	screened

Legend:
full = full compartment detection or suppression coverage
partial = partial compartment detection or suppression coverage
screened = fire compartment was screened in the qualitative Phase 1 portion of the analysis

Table 4-5
Suppression System Response Parameters

Detector Time Constants (Ref. 4-4)	
• Wet Pipe Solder Type	60 – 120 seconds
• Wet Pipe Bulb Type	120 – 240 seconds

Automatic Suppression System Unavailability

Suppression system reliabilities provided in the Fire PRA Implementation Guide (Ref. 4-5) were used.

Table 4-6
Compartment Fire Detection and Suppression

Suppression Type	System Unreliability
Wet Pipe Sprinkler	2.0E-02
Preaction Sprinkler	5.0E-02
Deluge Sprinkler	5.0E-02

4.6.2 Manual Fire Suppression

Manual suppression was not credited as being effective in preventing damage to critical targets. This is because of the time delay between detection of the fire and the time required for the fire brigade to respond. The main control room was an exception to this case because it is continuously manned. Another case would involve the fire watch that would be posted during planned hot work activities. In these cases, the effectiveness of the manual fire suppression activities were expected to be very high.

4.7 ANALYSIS OF PLANT SYSTEMS, SEQUENCES, AND PLANT RESPONSES

FIVE (Ref. 4-4) was used for initial screening, determination of ignition source frequencies, fire compartment interactions analysis, plant walkdowns, and fire modeling. The EPRI Fire PRA Implementation Guide (Ref. 4-5) was used to supplement the guidance for detailed analysis of unscreened areas. A preliminary screening process was used to eliminate from further consideration those compartments that have negligible fire risk potential. Subsequent detailed analyses evaluated the unscreened compartments in more detail.

The overall analysis approach involved three phases. The first phase involved application of the qualitative FIVE Phase 1 screening criteria. This was followed by an initial quantitative screening which assumed all cables, circuits, and equipment within the boundaries of a fire compartment were disabled due to a postulated fire. Fire compartments with a calculated CDF contribution of less than 1.0E-7/yr were screened. The remaining fire compartments were analyzed using a graded approach that involved a combination of supplemental circuit function reviews and fire modeling. The CDF contributions from all scenarios associated with unscreened fire compartments were retained and included in the total plant CDF reported herein regardless of the calculated contribution associated with an individual scenario.

The use of a 1.0E-7/yr. screening criteria is an order of magnitude lower than that used in the original Fire IPEEE analysis for Dresden. The selection of the lower screening criteria for the upgraded fire risk analysis was chosen to be consistent with the analysis for the Quad Cities Station.

The subsections that follow provide a brief discussion of the results obtained for the screening analysis and the subsequent detailed analyses for individual fire compartments, the Control Room, and multi-compartment fire scenarios.

4.7.1 Preliminary Screening Analysis

Fire areas and compartments were screened from further consideration based on guidance provided in FIVE (Ref. 4-4). Fire areas that did not contain Appendix R safe shutdown equipment or cables, AND would not cause or necessitate a reactor trip in the event of fire were screened from further evaluation. Unscreened fire areas where then evaluated using the FIVE Methodology (Ref. 4-4) and subdivided into fire compartments. The fire compartments were defined using the Appendix R fire zone definitions in order to maintain consistency with available plant documentation. Fire compartments were screened if the boundaries satisfied at least one of the FIVE boundary criteria AND did not contain Appendix R safe shutdown features. All fire areas and compartments were also evaluated separately in the Multi-Compartment Analysis (Section 4.7.3) to assess the risk significance of potential fire induced boundary failure.

4.7.1.1 Qualitative Screening of Fire Compartments

The qualitative screening process began with the division of the plant into the 19 fire areas defined in the Appendix R Program. These nineteen fire areas were then evaluated using the FIVE screening criteria. This criteria requires that the area be bounded by rated wall, floor, and ceiling, and any postulated exposure fire within a fire area not result in a safety challenge (plant trip or loss of critical system functions). Two of the 19 fire areas were screened on this basis as shown in Table 4-7.

Table 4-7
Fire Area Screening Results

Fire Area	Fire Area Description	Screened
Crib House	Cribhouse	N
MISC	Isolation Condenser Pump House	N
OG BLDG	Offgas Filter Building	N
OUTDOOR	Areas Outside the Plant and Other Fire Areas	N
RB-2/3	Unit 2/3 HPCI and EDG Areas	N
RB2-I	Unit 2 Reactor Building – Isolation Condenser	N
RB2-II	Unit 2 Reactor Building	N
RB2-II/RB3-II	Unit 2/3 Refueling Floor	N
RB3-I	Unit 3 Reactor Building – Isolation Condenser	N
RB3-II	Unit 3 Reactor Building	N
RW	Radwaste Building	N
TB-I	Unit 2 Turbine Building	N
TB-II	Unit 2/3 Turbine Building Common Areas	N
TB-III	Unit 3 Turbine Building	N
TB-IV	Unit 2/3 Turbine Building, Turbine Deck	N
TB-V	Main Control and Auxiliary Electric Equipment Rooms	N
U2 PC	Unit 2 Primary Containment	Y (1)
U3 PC	Unit 3 Primary Containment	Y (1)

Table 4-7
Fire Area Screening Results

Fire Area	Fire Area Description	Screened
XMFR	Station Main and Auxiliary Transformer Areas	N

Note 1: Primary Containment is enclosed by complete three-hour barriers. Ignition sources were not counted in this compartment based on the fact that primary containment is inerted, thus, a fire cannot occur in this compartment and all methods of shutdown are available.

The remaining 17 fire areas were subdivided into fire compartments using the Appendix R fire zone definitions. The Appendix R fire zone definitions were used because the existing data structure for cable location information was based on these definitions. It was recognized that use of these definitions required careful tracking to ensure that all compartments were properly considered for potential fire related consequences, including propagation to adjacent compartments.

This second level qualitative screening required that a fire compartment be bounded by walls, floors, and ceilings that satisfy the FIVE boundary criteria and contain no Appendix R safe shutdown equipment or cables. This criteria is consistent with the EPRI FIVE Methodology as described in Step 6 of the Phase 1 screening process. Three fire compartments were screened in this fashion and are listed in Table 4-9. The 2 fire areas which were screened as discussed above consisted of 2 fire compartments resulting in a total of 5 out of 83 fire compartments being screened on a qualitative basis. The remaining 78 fire compartments required quantitative analysis as discussed in Section 4.7.1.2.

Table 4-9
Fire Compartments Qualitatively Screened

Fire Compartment	Compartment Description
1.2.1	U3 DRYWELL
1.2.2	U2 DRYWELL
18.4	AUX BOILER HOUSE
OUTDOOR	H2 STORAGE FACILITY
OUTDOOR	UNIT 1

4.7.1.2 Quantitative Screening of Fire Compartments

The original Dresden Fire IPEEE applied a two step quantitative screening process that included the consideration of fire severity factors and credit for fire suppression. The upgraded fire analysis simplified the process by applying a single bounding scenario for each fire compartment and did not apply severity factors nor credit for automatic fire suppression system operation.

The fire ignition frequencies were developed based on walkdowns of the fire compartments. During the walkdown, fire ignition sources were identified and counted. The information was then assimilated into overall plant fire ignition source frequencies and fire compartment ignition source frequencies. This information then determined the fire ignition frequency for each fire compartment as described in Section 4.3.2.

The CCDPs were developed using equipment and cable information about each fire compartment. All equipment and cables contained within a given fire compartment were considered damaged and a CCDP for the fire compartment was determined. The core damage frequency (CDF) was then calculated using the following equation:

$$\mathbf{CDF}_{\text{compartment}} = \mathbf{CCDP}_{\text{compartment}} * \mathbf{FF}_{\text{compartment}}$$

where:

- **CDF** is the core damage frequency for the compartment.
- **FF** is the fire ignition frequency for the compartment. It includes all the ignition sources identified in the fire ignition frequency development task.
- **CCDP** is the conditional core damage probability. In this equation it includes the guaranteed failure of all equipment identified within the fire compartment.

Fire compartments with a calculated CDF contribution of less than 1E-07/yr. were screened and no longer considered in the evaluations. No other refinements were applied as part of the screening process. Compartments that were screened can be expected to have a realistic CDF contribution significantly lower than the 1E-07/yr. screening criteria because of the bounding assumption in the overall screening methodology.

The quantification for Dresden Unit 2 found that 62 of the 78 fire compartments that were not qualitatively screened could be quantitatively screened on this basis of a bounding calculated CDF contribution of less than 1.0E-7/yr. The 62 screened fire compartments are listed in Table 4-10. The table provides the screening CDF for each of these fire compartments. Unscreened fire compartments (those with an initially calculated CDF contribution greater than 1.0E-7/yr.) do not have a reported CDF. An entry of 'Ns' is provided instead. These remaining 16 fire compartments for Unit 2 were evaluated further using fire modeling tools as discussed in Section 4.7.2.

The quantification for Dresden Unit 3 found that 57 of the 78 fire compartments that were not qualitatively screened could be quantitatively screened on the basis of a bounding calculated CDF contribution of less than 1.0E-7/yr. These 57 screened fire compartments are also listed in Table 4-10. These remaining 21 fire compartments for Unit 3 were evaluated further using fire modeling tools as discussed in Section 4.7.2.

Table 4-10
Quantitative Screening Results

Fire Compartment	Fire Compartment Description	Unit 2 CDF	Unit 3 CDF
1.1.1.1	U3 TORUS BASEMENT	6.21E-12	Ns
1.1.1.2	U3 RX BLDG GROUND FLOOR	4.72E-09	Ns
1.1.1.3	U3 SECOND FLOOR RX BLDG	6.10E-09	Ns
1.1.1.4	U3 RX BLDG SWITCHGEAR AREA	2.71E-09	Ns
1.1.1.5.A	U3 ISCO FLOOR	6.27E-12	3.35E-10
1.1.1.5.B	U3 ISCO PIPE CHASE	3.86E-10	3.86E-10
1.1.1.5.C	U3 ISCO PIPE CHASE	3.86E-10	2.16E-12
1.1.1.5.D	U3 SKIMMER SURGE TANK ROOM	2.88E-09	1.61E-11
1.1.1.6	U3 REFUEL FLOOR	1.77E-09	1.77E-09
1.1.2.1	U2 TORUS BASEMENT	Ns	6.24E-12
1.1.2.2	U2 RX BLDG GROUND FLOOR	Ns	6.24E-10
1.1.2.3	U2 SECOND FLOOR RX BLDG	Ns	7.05E-10
1.1.2.4	U2 RX BLDG SWITCHGEAR AREA	Ns	1.34E-10
1.1.2.5.A	U2 ISCO FLOOR	1.00E-10	1.52E-09
1.1.2.5.B	U2 ISCO PIPE CHASE	2.16E-12	3.87E-10
1.1.2.5.C	U2 ISCO PIPE CHASE	2.57E-11	3.87E-10
1.1.2.5.D	U2 SKIMMER SURGE TANK ROOM	1.29E-11	2.32E-09
1.1.2.6	U2 REFUEL FLOOR	1.77E-09	1.77E-09
1.3.1	U3 SHUTDOWN COOLING PUMP ROOM	7.20E-12	4.13E-10
1.3.2	U2 SHUTDOWN COOLING PUMP ROOM	4.47E-10	7.09E-12
1.4.1	U3 TIP ROOM	3.86E-10	Ns
2.0	CONTROL ROOM	Ns	Ns
6.1	U3 BATTERY CHARGER ROOM	Ns	Ns
6.2	AUX ELEC EQUIP ROOM	Ns	Ns
7.0.A.1	U2 BATTERY ROOM	Ns	Ns

Table 4-10
Quantitative Screening Results

Fire Compartment	Fire Compartment Description	Unit 2 CDF	Unit 3 CDF
7.0.A.2	125 VDC BATT. RM.	5.00E-08	5.69E-09
7.0.A.3	250 VDC BATT. RM.	1.81E-09	7.49E-09
7.0.B	U3 STATION BATT. ROOM	1.23E-08	1.88E-09
8.1	CLEAN/DIRTY OIL ROOM	3.06E-09	3.06E-09
8.2.1.A	U2 COND. PP AREA	2.43E-08	2.17E-09
8.2.1.B	U3 COND. PP AREA	4.56E-11	Ns
8.2.2.A	U2 CRD PUMP ROOM	8.72E-10	4.17E-08
8.2.2.B	U3 CRD PUMP ROOM	7.44E-09	6.01E-08
8.2.4	U3 CABLE TUNNEL	1.21E-09	Ns
8.2.5.A	U2 NORTH TRACKWAY/SWGR AREA	Ns	Ns
8.2.5.B	U2 LP HEATER BAY	3.96E-12	8.57E-09
8.2.5.C	2/3 TB CORRIDOR	Ns	Ns
8.2.5.D	U3 LP HEATER BAY	1.60E-09	1.28E-08
8.2.5.E	U3 WEST CORRIDOR AND TRACKWAY	Ns	Ns
8.2.6.A	CONTROL ROOM BKUP VENTILATION	Ns	Ns
8.2.6.B	U2 MEZZANINE	Ns	9.27E-08
8.2.6.C	U2/3 SBGT & TBCCW HX	Ns	Ns
8.2.6.D	U3 MEZZANINE FLOOR	2.71E-09	Ns
8.2.6.E	U3 MEZZANINE FLOOR	Ns	Ns
8.2.7	VENT ROOM OVER NE SWGR	1.27E-09	Ns
8.2.8.A	U2/3 TURBINE OPERATING FLOOR	2.23E-10	1.13E-08
8.2.8.B	VENT FLOOR	5.99E-10	7.24E-09
8.2.8.C	VENT FLOOR	9.84E-10	1.19E-08
8.2.8.D	VENT FLOOR	2.97E-09	3.59E-08
9.0.A	U2 D/G	1.01E-10	1.01E-10
9.0.B	U3 D/G	1.72E-10	Ns
9.0.C	U2/3 DG	5.44E-09	5.32E-09
11.1.1	U3 LPCI PP ROOM DIV II	2.60E-09	1.03E-08
11.1.2	U3 LPCI PP ROOM DIV I	7.71E-10	1.02E-09

Table 4-10
Quantitative Screening Results

Fire Compartment	Fire Compartment Description	Unit 2 CDF	Unit 3 CDF
11.1.3	U3 HPCI	1.53E-09	4.35E-10
11.2.1	U2 LPCI PP ROOM DIV II	5.54E-08	1.52E-11
11.2.2	U2 LPCI PP ROOM DIV I	9.53E-08	1.45E-11
11.2.3	U2 HPCI	4.63E-10	4.63E-10
11.3	CRIBHOUSE UPPER	Ns	Ns
14.1	RADWASTE BLDG.	1.18E-10	1.12E-08
14.2.A	U2 STEAM JET AIR EJECTOR ROOM	5.31E-10	6.42E-09
14.2.B	U2 OFF GAS RECOMBINER	5.29E-08	5.29E-08
14.2.C	U2 OFF GAS CONDENSER	3.86E-10	4.66E-09
14.3.A	U3 STEAM JET AIR EJECTOR ROOM	1.01E-09	1.23E-08
14.3.B	U3 OFF GAS RECOMBINER	5.29E-08	5.29E-08
14.3.C	U3 OFF GAS CONDENSER	3.86E-10	4.66E-09
14.4	OFF GAS FILTER BUILDING	2.93E-09	2.93E-09
14.5	RADWASTE BLDG.	1.25E-08	1.25E-08
14.6	MAX RECYCLE	1.20E-08	1.20E-08
18.1.1	U3 MPT	3.31E-09	3.31E-09
18.1.2	U2 MPT	3.31E-09	3.31E-09
18.2.1	U3 UAT	3.31E-09	3.31E-09
18.2.2	U2 UAT	3.31E-09	3.31E-09
18.3.1	U3 RAT	6.60E-09	6.60E-09
18.3.2	U2 RAT	7.83E-08	2.12E-09
18.6	SBO BATT RM	7.60E-12	7.60E-12
18.7.1	ISCO PP HOUSE NORTH	4.87E-11	4.87E-11
18.7.2	ISCO PP HOUSE SOUTH	4.01E-11	4.01E-11
	Number of Screened Compartments Number of Unscreened Compartments	62 16	57 21

4.7.1.3 Summary of Screening Results

The evaluation of the 83 fire compartments determined that many of them could be screened using qualitative and quantitative screening criteria. The results are summarized in Table 4-11. The application of the screening criteria and the associated bounding assumptions are such that potentially risk significant core damage sequences would not be inadvertently excluded from detailed analysis. The unscreened fire compartments were then analyzed in greater detail using fire modeling tools and cable function evaluations to develop discrete fire scenarios. This is discussed in Section 4.7.2. The fire compartments retained for more detailed analysis are listed in Table 4-12 and are identified by an entry of 'No' for the applicable Unit.

Table 4-11
Summary of Screening Results

	Unit 2	Unit 3
Number of fire compartments	83	83
Number of compartments associated with qualitatively screened fire areas(Table 4-8)	2	2
Number of additional compartments qualitatively screened (Table 4-9)	3	3
Number of compartments quantitatively screened (Table 4-10)	62	57
Number of compartments remaining unscreened (Table 4-11)	16	21

Table 4-12
Unscreened Fire Compartments

Fire Compartment	Compartment Description	Unit 2 Screened	Unit 3 Screened
1.1.1.1	U3 TORUS BASEMENT	Yes	No
1.1.1.2	U3 RX BLDG GROUND FLOOR	Yes	No
1.1.1.3	U3 SECOND FLOOR RX BLDG	Yes	No
1.1.1.4	U3 RX BLDG SWITCHGEAR AREA	Yes	No
1.1.2.1	U2 TORUS BASEMENT	No	Yes
1.1.2.2	U2 RX BLDG GROUND FLOOR	No	Yes
1.1.2.3	U2 SECOND FLOOR RX BLDG	No	Yes
1.1.2.4	U2 RX BLDG SWITCHGEAR AREA	No	Yes
1.4.1	U3 TIP ROOM	Yes	No
2.0	CONTROL ROOM	No	No

Table 4-12
Unscreened Fire Compartments

Fire Compartment	Compartment Description	Unit 2 Screened	Unit 3 Screened
6.1	U3 BATTERY CHARGER ROOM	No	No
6.2	AUX ELEC EQUIP ROOM	No	No
7.0.A.1	U2 BATTERY ROOM	No	No
8.2.1.B	U3 COND. PP AREA	Yes	No
8.2.4	U3 CABLE TUNNEL	Yes	No
8.2.5.A	U2 NORTH TRACKWAY/SWGR AREA	No	No
8.2.5.C	2/3 TB CORRIDOR	No	No
8.2.5.E	U3 WEST CORRIDOR AND TRACKWAY	No	No
8.2.6.A	CONTROL ROOM BKUP VENTILATION	No	No
8.2.6.B	U2 MEZZANINE	No	Yes
8.2.6.C	U2/3 SBGT & TBCCW HX	No	No
8.2.6.D	U3 MEZZANINE FLOOR	Yes	No
8.2.6.E	U3 MEZZANINE FLOOR	No	No
8.2.7	VENT ROOM OVER NE SWGR	Yes	No
9.0.B	U3 D/G	Yes	No
11.3	CRIBHOUSE UPPER	No	No

4.7.2 Analysis of Single Compartment Fires

This section summarizes the analysis refinements that were performed for the unscreened fire compartments. These refinements involved a combination of fire modeling analyses and focused circuit function reviews. The results of these refinements are summarized in the following sections. The discussions are presented by unit in the order as they appear in Table 4-12. The Unit 2 fire compartment refinements discussions are followed by the Unit 3 discussions. The results for all unscreened fire compartments for Unit 2 are summarized in Table 4-14. The Unit 3 summary is provided in Table 4-16.

4.7.2.1 Unit 2 Analysis - Fire Compartment 1.1.2.1

The initial quantification for this fire compartment resulted in a calculated CDF contribution slightly greater than the screening criteria of 1.0E-7/yr. Since the bounding value was only slightly greater than the screening criteria, no further refinements were performed.

4.7.2.2 Unit 2 Analysis - Fire Compartment 1.1.2.2

The initial quantification for this fire compartment resulted in a calculated CDF contribution greater than the screening criteria of 1.0E-7/yr. This fire compartment is the ground floor of the Reactor Building. The Reactor Building, in general, contains relatively few significant ignition sources or heavy concentration of combustible materials. The ground floor of the Reactor Building shares a common ventilation path with the upper elevations through the >400 ft^2 open equipment hatch. This ventilation path was not credited in the fire modeling analyses.

A review of the configuration, equipment, cables, and fire ignition sources in this compartment concluded that the development of multiple explicit fire scenarios to bound the analysis was appropriate. These scenarios considered fires originating at the Drywell/Suppression Pool Pump Back Air Compressor, or the MCCs located throughout the compartment. The analysis of this fire compartment also included a specific investigation of the risk significance of fire induced spurious actuation of the ADS valves.

4.7.2.3 Unit 2 Analysis - Fire Compartment 1.1.2.3

The initial quantification for this fire compartment resulted in a calculated CDF contribution greater than the screening criteria of 1.0E-7/yr. This fire compartment is the second floor of the Reactor Building. The Reactor Building, in general, contains relatively few significant ignition sources or heavy concentration of combustible materials. The ground floor of the Reactor Building shares a common ventilation path with the upper elevations through the >400 ft^2 open equipment hatch. This ventilation path was not credited in the fire modeling analyses.

A review of the configuration, equipment, cables, and fire ignition sources in this compartment concluded that the development of multiple explicit fire scenarios to bound the analysis was appropriate. The walkdown inspection of this fire compartment concluded that each of the 4kV switchgear buses in the compartment required fire modeling. The analysis for the CDF contribution due to the switchgear fires was based on loss of the switchgear along with failure of the circuits in the cable tray directly above. The scenarios also considered the potential for an 'explosive' type failure of a switchgear breaker wherein the damage could propagate horizontally. Separate scenarios were also generated to address the RBCCW pump/motor as well as specific evaluation of the ADS circuits.

4.7.2.4 Unit 2 Analysis - Fire Compartment 1.1.2.4

The initial quantification for this fire compartment resulted in a calculated CDF contribution greater than the screening criteria of 1.0E-7/yr. This fire compartment is the switchgear floor of the Reactor Building. The Reactor Building, in general, contains relatively few significant ignition sources or heavy concentration of combustible materials. The ground floor of the Reactor Building shares a common ventilation path with the upper elevations through the >400 ft^2 open equipment hatch. This ventilation path was not credited in the fire modeling analyses.

A review of the configuration, equipment, cables, and fire ignition sources in this compartment concluded that the development of multiple explicit fire scenarios to bound the analysis was appropriate. The walkdown inspection of this fire compartment concluded that each of the 480 Vac switchgear buses in the compartment required fire modeling as well as the DC MCCs.

4.7.2.5 Unit 2 Analysis - Fire Compartment 2.0

This fire compartment is the main control room. The discussion and results of the main control room analysis are presented in Section 4.7.4.

4.7.2.6 Unit 2 Analysis - Fire Compartment 6.1

This fire compartment is the Unit 3 Battery Charger room. The walkdown inspection of this compartment determined that the cabinets located in this compartment are substantially sealed. These cabinets consist of battery chargers and distribution panels. The analysis considered individual fire scenarios for each cabinet. Each scenario assumed the functional failure of the equipment and the failure of circuits in the cables trays located above them.

4.7.2.7 Unit 2 Analysis - Fire Compartment 6.2

The examination of the Dresden Unit 2/3 Auxiliary Electric Equipment Room (AEER) was performed to develop the credible fire scenarios for the upgraded fire analysis. The overall methodology involved identifying each discrete cabinet, determining the plant system functions associated with the devices and features within the cabinet based on the cables terminated therein, and assessing the likelihood for fire propagation between cabinets, and to overhead raceways based on the structural features of the cabinets. The configuration of the room and the low heat rate of the cabinet fire did not result in a hot gas layer damage potential within the room.

Several cabinets in this compartment are not considered to be sealed and are either open or have ventilation louvers that could create a propagation path to overhead cable trays. The analysis of these cabinets assumed that any postulated fire propagates to the tray directly above.

4.7.2.7.1 Cabinet Ignition Frequency

The total fire ignition frequency for this fire compartment is 2.0E-2. The fire ignition frequency for an individual cabinet or group of cabinets is determined by taking the fraction of cabinet of interest and multiplying it to the total frequency of 2.0E-2. The auxiliary relay room was treated on the basis of 147 cabinets. The number of sections associated with each cabinet was not intended to represent the number of compartments within the cabinet. Instead, the number of sections was used as a means to evaluate the relative size of each of the cabinets with each cabinet section corresponding to a width of approximately 30" - 36". The ignition frequency of an individual cabinet is 1.36E-4/yr.

4.7.2.7.2 Individual Cabinet Details

The information which was developed for each of the control cabinets is summarized in Table 4-13. In certain cases, the cabinet configuration is such that fire propagation beyond the panel boundaries to the overhead cable trays is credible. In these cases, the consequences of this propagation is determined by identifying the affected cable trays, referring to the SLICE database to obtain the associated cables, and relating these cables to PSA model basic events.

Table 4-13
Auxiliary Relay Room Cabinets Fire Scenario Summary

Cabinet ID	Sections	Scenario	Propagation	Fire Consequences
Computer Peripherals	1	B	No	No impact on credited systems – use %TT
Data Acquisition Cabinet #1	1	B	No	No impact on credited systems - use %TT
Data Acquisition Cabinet #2	1	B	No	No impact on credited systems - use %TT
EPA 2A-1	1	B	No	No impact on credited systems - use %TT
EPA 2A-2	1	B	No	No impact on credited systems - use %TT
EPA 2AB-1	1	B	No	No impact on credited systems - use %TT
EPA 2AB-2	1	B	No	No impact on credited systems - use %TT
EPA 2B-1	1	B	No	No impact on credited systems - use %TT
EPA 2B-2	1	B	No	No impact on credited systems - use %TT
EPA 3A-1	1	B	No	No impact on credited systems - use %TT
EPA 3A-2	1	B	No	No impact on credited systems - use %TT
EPA 3AB-1	1	B	No	No impact on credited systems - use %TT
EPA 3AB-2	1	B	No	No impact on credited systems - use %TT
EPA 3B-1	1	B	No	No impact on credited systems - use %TT
EPA 3B-2	1	B	No	No impact on credited systems - use %TT
Old EHC Panel	1	B	No	No impact on credited systems - use %TT
RLC #13	1	B	No	No impact on credited systems - use %TT
2TB-71	2	M	No	Loss of various specific system functions based on cables in cable trays.

Table 4-13
Auxiliary Relay Room Cabinets Fire Scenario Summary

Cabinet ID	Sections	Scenario	Propagation	Fire Consequences
2202-70A	1	B	No	No impact on credited systems - use %TT - Loss of ATWS/RPT functions
2202-70B	1	B	No	No impact on credited systems - use %TT - Loss of ATWS/RPT functions
2203-70A	1	B	No	No impact on credited systems - use %TT - Loss of ATWS/RPT functions
2203-70B	1	B	No	No impact on credited systems - use %TT - Loss of ATWS/RPT functions
2253-91	1	B	No	No impact on credited systems – use %TT
2253-92	1	B	No	No impact on credited systems – use %TT
2-8001-A	1	B	No	No impact on credited systems - use %TT - Unit 2 RPS MG Set
2-8001-B	1	B	No	No impact on credited systems - use %TT - Unit 2 RPS MG Set
3-8001-A	1	B	No	No impact on credited systems - use %TT - Unit 3 RPS MG Set
3-8001-B	1	B	No	No impact on credited systems - use %TT - Unit 3 RPS MG Set
902-27	3	N	Yes	No impact on credited systems - potential propagation to trays 257B and 258B.
902-28	4	B	No	No impact on credited systems – use %TT
902-29	4	D	No	Loss of Unit 2 Offsite Power - TR21 and 22
902-31A	6	E	Yes	Loss of Unit 2 Main Condenser - also potential propagation to trays 225 and 249.
902-31B	-	-	-	"
902-31C	-	-	-	"
902-31D	-	-	-	"
902-31E	-	-	-	"
902-31F	-	-	-	"
902-32	3	F	No	Loss of Unit 2 ECCS Div I CS and RHR systems and ADS
902-33	3	G	No	Loss of Unit 2 ECCS Div II CS, RHR, and HPCI systems

Table 4-13
Auxiliary Relay Room Cabinets Fire Scenario Summary

Cabinet ID	Sections	Scenario	Propagation	Fire Consequences
902-34	7	O	Yes	No impact on credited systems – potential propagation trays 245B, 246B, and 247.
902-38	1	B	No	No impact on credited systems – use %TT
902-39	2	H	No	Loss of Unit 2 HPCI
902-40	2	C	No	Loss of Unit 2 Main Condenser due to MSIV closure
902-41	2	C	No	Loss of Unit 2 Main Condenser due to MSIV closure
902-46	1	I	No	Loss of Unit 2 LPCI Line Break Detection - ESS I
902-47	1	J	No	Loss of Unit 2 LPCI Line Break Detection – ESS II
902-49	1	K	No	Loss of Unit 2 Essential Services Bus
902-50	1	L	No	Loss of Unit 2 Instrument Bus
902-51	1	P	No	Loss of Unit 2 Main Condenser
902-52A	1	B	No	Loss of Unit 2 RPS Bus 2B
902-52B	1	B	No	Loss of Unit 2 RPS Bus 2A
902-61	2	C	No	Loss of Unit 2 Main Condenser due to MSIV closure
902-63	1	K	No	Loss of Essential Services Bus Auto Transfer Switch
902-63A	2	K	No	Loss of Unit 2 Essential Services Bus Inverter - conservatively treated based on loss of the Unit 2 ESS Bus
902-63B	2	K	No	Loss of Unit 2 Essential Services Bus Static Switch - conservatively treated based on loss of the Unit 2 ESS Bus
902-63C	1	K	No	Loss of Unit 2 Essential Services Bus Line Regulator - conservatively treated based on loss of the Unit 2 ESS Bus
902-64A	1	B	No	No impact on credited systems – use %TT
902-64B	1	B	No	No impact on credited systems – use %TT
902-68	1	B	No	No impact on credited systems – use %TT
902-69	1	B	No	No impact on credited systems – use %TT

Table 4-13
Auxiliary Relay Room Cabinets Fire Scenario Summary

Cabinet ID	Sections	Scenario	Propagation	Fire Consequences
902-70	1	B	No	No impact on credited systems – use %TT
902-71	1	B	No	No impact on credited systems – use %TT
902-74	1	D	No	Loss of Unit 2 UAT/RAT Supplemental Protection Cabinet – conservatively treat as Unit 2 LOOP
903-27	3	Q	Yes	No impact on credited systems – potential propagation to trays 650B and 671B.
903-28	4	B	No	No impact on credited systems – use %TT
903-29	3	D	No	Loss of Unit 3 Offsite Power – TR31 and 32 – conservatively treat as Unit 2 LOOP
903-31A	6	R	Yes	Loss of Unit 3 Main Condenser – also potential propagation tray 655B.
903-31B	-	-	-	"
903-31C	-	-	-	"
903-31D	-	-	-	"
903-31E	-	-	-	"
903-31F	-	-	-	"
903-32	3	B	No	Loss of Unit 3 ECCS Div I CS and RHR systems and ADS
903-33	3	B	No	Loss of Unit 3 ECCS Div II CS, RHR, and HPCI systems
903-34	7	S	Yes	No impact on credited systems - potential propagation to tray 229.
903-38	1	B	No	No impact on credited systems - use %TT
903-39	2	B	No	Loss of Unit 3 HPCI
903-40	2	B	No	Loss of Unit 3 Main Condenser due to MSIV closure
903-41	2	B	No	Loss of Unit 3 Main Condenser due to MSIV closure
903-46	1	B	No	Loss of Unit 3 LPCI Line Break Detection – ESS I
903-47	1	B	No	Loss of Unit 3 LPCI Line Break Detection - ESS II
903-49	1	K	No	Loss of Unit 3 Essential Services Bus
903-50	1	L	No	Loss of Unit 3 Instrument Bus

Table 4-13
Auxiliary Relay Room Cabinets Fire Scenario Summary

Cabinet ID	Sections	Scenario	Propagation	Fire Consequences
903-51	1	B	No	Loss of Unit 3 Main Condenser
903-52A	1	B	No	Loss of Unit 3 RPS Bus 3B
903-52B	1	B	No	Loss of Unit 3 RPS Bus 3A
903-60	1	B	No	No impact on credited systems – use %TT
903-62	2	B	No	Loss of Unit 3 Main Condenser due to MSIV closure
903-63	1	K	No	Loss of Essential Services Bus Auto Transfer Switch
903-63A	2	K	No	Loss of Unit 3 Essential Services Bus Inverter – conservatively treated based on loss of the ESS Bus
903-63B	2	K	No	Loss of Unit 3 Essential Services Bus Static Switch – conservatively treated based on loss of the ESS Bus
903-63C	1	K	No	Loss of Unit 3 Essential Services Bus Line Regulator – conservatively treated based on loss of the ESS Bus
903-64A	1	B	No	No impact on credited systems - use %TT
903-64B	1	B	No	No impact on credited systems - use %TT
903-68	1	B	No	No impact on credited systems – use %TT
903-69	1	B	No	No impact on credited systems – use %TT
903-70	1	B	No	No impact on credited systems – use %TT
903-71	1	B	No	No impact on credited systems – use %TT
903-74	1	D	No	Loss of Unit 2 UAT/RAT Supplemental Protection Cabinet – conservatively treat as Unit 2 LOOP
TOTAL	147			

In addition to the scenarios described in Table 4-13, additional cases were considered because of specific issues related to potential fire induced spurious actuation of ADS valves. These additional cases considered self-initiated cable tray fires.

4.7.2.8 Unit 2 Analysis - Fire Compartment 7.0.A.1

The initial quantification for this fire compartment resulted in a calculated CDF contribution greater than the screening criteria of 1.0E-7/yr. This fire compartment is adjacent to the Unit 2 Battery Room. A review of the configuration, equipment and cables in this fire compartment concluded that the development of multiple explicit fire scenarios to bound the analysis was appropriate. These scenarios considered postulated fires originating in the electrical cabinets, MCCs, and battery chargers.

4.7.2.9 Unit 2 Analysis – Fire Compartment 8.2.5.A

The initial quantification for this fire compartment resulted in a calculated CDF contribution greater than the screening criteria of 1.0E-7. This fire compartment is the ground floor of the Unit 2 Turbine Building. This area consists of the non-Essential 480 Vac switchgear, Turbine Building Trackway, and Reactor Feedwater pump areas. A review of the configuration, equipment and cables in this fire compartment concluded that the development of multiple explicit fire scenarios to bound the analysis was appropriate. These scenarios considered postulated fires originating in the MCCs, air compressors, and pumps. A self-initiated cable fire was also considered to address the potential for fire induced spurious actuation of ADS valves. The evaluation of this area credited the automatic fire suppression system installed in the Reactor Feedwater Pump area.

4.7.2.10 Unit 2 Analysis - Fire Compartment 8.2.5.C

The initial quantification for this fire compartment resulted in a calculated CDF contribution slightly greater than the screening criteria of 1.0E-7/yr. This fire compartment is the ground floor Turbine Building common area between Units 2 and 3. A review of the configuration, equipment and cables in this fire compartment concluded that the development of multiple explicit fire scenarios to bound the analysis was appropriate. These scenarios considered postulated fires originating in the MCCs, air compressors, and pumps. The evaluation of this area credited the automatic fire suppression system installed over the air compressor.

4.7.2.11 Unit 2 Analysis - Fire Compartment 8.2.5.E

The initial quantification for this fire compartment resulted in a calculated CDF contribution greater than the screening criteria of 1.0E-7/yr. This fire compartment is the ground floor of the Unit 3 Turbine Building. This area is similar to and is the Unit 3 equivalent of fire compartment 8.2.5.A. A review of the configuration, equipment and cables in this fire compartment concluded that the development of multiple explicit fire scenarios to bound the analysis was appropriate. These scenarios generally divided the area into two sub-compartment based on the location of the trackway.

4.7.2.12 Unit 2 Analysis - Fire Compartment 8.2.6.A

The initial quantification for this fire compartment resulted in a calculated CDF contribution greater than the screening criteria of 1.0E-7/yr. This fire compartment is the mezzanine level of the Unit 2 Turbine Building and includes both the switchgear and general areas. A review of the configuration, equipment and cables in this fire compartment concluded that the development of multiple explicit fire scenarios to bound the analysis was appropriate. These scenarios considered postulated fire originating in the switchgears, MCCs, and electrical panels. The postulated switchgear fires included the consideration of an 'explosive' type event wherein horizontal propagation is possible. This is was considered to ensure a bounding assessment of the area. In addition, a self-initiated cable tray fire was included to explicitly treat the potential for a fire induced spurious actuation of the ADS valves.

4.7.2.13 Unit 2 Analysis - Fire Compartment 8.2.6.B

The initial quantification for this fire compartment resulted in a calculated CDF contribution greater than the screening criteria of 1.0E-7/yr. This fire compartment is the mezzanine level of the Unit 2 Turbine Building. A review of the configuration, equipment and cables in this fire compartment concluded that there are no significant ignition sources of concern. However, the area does contain circuits whose fire induced failure could cause a spurious actuation of multiple ADS valves. Therefore, fire scenarios were developed to evaluate postulated self-initiated cable tray fires.

4.7.2.14 Unit 2 Analysis - Fire Compartment 8.2.6.C

The initial quantification for this fire compartment resulted in a calculated CDF contribution slightly greater than the screening criteria of 1.0E-7/yr. This fire compartment is the mezzanine level of the Turbine Building common area between Units 2 and 3. A review of the configuration, equipment and cables in this fire compartment concluded that the development of multiple explicit fire scenarios to bound the analysis was appropriate. These scenarios considered postulated fires originating in the MCCs, air compressors, and pumps.

4.7.2.15 Unit 2 Analysis – Fire Compartment 8.2.6.E

The initial quantification for this fire compartment resulted in a calculated CDF contribution greater than the screening criteria of 1.0E-7/yr. This fire compartment is the mezzanine level of the Unit 3 Turbine Building. This area is similar to and is the Unit 3 equivalent of fire compartment 8.2.6.A. A review of the configuration, equipment and cables in this fire compartment concluded that the development of multiple explicit fire scenarios to bound the analysis was appropriate. These scenarios generally divided the area into two sub-compartment based on the location of the trackway.

4.7.2.16 Unit 2 Analysis – Fire Compartment 11.3

The initial quantification for this fire compartment resulted in a calculated CDF contribution slightly greater than the screening criteria of 1.0E-7/yr. Since the bounding value was only slightly greater than the screening criteria, no further refinements were performed.

Table 4-14
Unit 2 Unscreened Fire Compartment Analysis Details

Fire Compartment	Scen	Scenario Description	Fire Compartment Description	Ignition Frequency	NSP	SF	CCDP	CDF
1.1.2.1	A	Screening Scenario	U2 TORUS BASEMENT	1.83E-03	1	1	6.03E-05	1.10E-07
1.1.2.2	B	MCC 28-1	U2 RX BLDG GROUND FLOOR	1.99E-03	1	1	5.41E-08	1.08E-10
1.1.2.2	C	MCCs 28-7/29-5/29-6/39-5/39-6	U2 RX BLDG GROUND FLOOR	1.99E-03	1	1	5.04E-09	1.00E-11
1.1.2.2	D	MCC 29-4	U2 RX BLDG GROUND FLOOR	1.99E-03	1	1	2.01E-07	3.99E-10
1.1.2.2	E	MCC 29-1	U2 RX BLDG GROUND FLOOR	1.99E-03	1	1	5.04E-09	1.00E-11
1.1.2.2	F	MCC 29-7	U2 RX BLDG GROUND FLOOR	1.99E-03	1	1	5.04E-09	1.00E-11
1.1.2.2	G	Pump Back Air Compressor Large Fire	U2 RX BLDG GROUND FLOOR	1.80E-03	1	0.18	6.17E-05	2.00E-08
1.1.2.2	H	Pump Back Air Compressor Small Fire	U2 RX BLDG GROUND FLOOR	1.80E-03	1	0.82	4.79E-09	7.07E-12
1.1.2.2	I	Self-Initiated Cable Fire - %TP	U2 RX BLDG GROUND FLOOR	5.35E-05	1	1	4.84E-04	2.59E-08
1.1.2.2	J	Self-Initiated Cable Fire - %TI	U2 RX BLDG GROUND FLOOR	5.35E-05	0.1	1	8.25E-05	4.42E-10
1.1.2.3	B	RBCCW Pump Large Fire	U2 SECOND FLOOR RX BLDG	3.36E-03	1	0.18	1.36E-05	8.25E-09
1.1.2.3	C	RBCCW Pump Small Fire	U2 SECOND FLOOR RX BLDG	3.36E-03	1	0.82	5.04E-09	1.39E-11

Table 4-14
Unit 2 Unscreened Fire Compartment Analysis Details

Fire Compartment	Scen	Scenario Description	Fire Compartment Description	Ignition Frequency	NSP	SF	CCDP	CDF
1.1.2.3	D	RWCU Pumps	U2 SECOND FLOOR RX BLDG	2.24E-03	1	1	6.42E-04	1.44E-06
1.1.2.3	F	Bus 23-1 Large Fire	U2 SECOND FLOOR RX BLDG	1.85E-03	1	1	4.43E-05	8.20E-08
1.1.2.3	G	Bus 24-1	U2 SECOND FLOOR RX BLDG	1.85E-03	1	1	7.93E-07	1.47E-09
1.1.2.3	H	Self-Initiated Cable Fire - %TP	U2 SECOND FLOOR RX BLDG	7.48E-05	1	1	4.51E-04	3.38E-08
1.1.2.3	I	Self-Initiated Cable Fire - %TI	U2 SECOND FLOOR RX BLDG	7.48E-05	1	1	2.81E-04	2.10E-08
1.1.2.3	J	Bus 23-1 Small Fire	U2 SECOND FLOOR RX BLDG	1.85E-03	1	0.8	4.43E-05	6.56E-08
1.1.2.4	B	Bus 28	U2 RX BLDG SWITCHGEAR AREA	1.93E-03	1	1	2.00E-08	3.86E-11
1.1.2.4	C	Bus 29	U2 RX BLDG SWITCHGEAR AREA	1.93E-03	1	1	1.96E-07	3.78E-10
1.1.2.4	D	250VDC MCC 2A	U2 RX BLDG SWITCHGEAR AREA	5.12E-03	1	1	6.95E-07	3.56E-09
1.1.2.4	E	250VDC MCC 2B/125VDC Panel 2	U2 RX BLDG SWITCHGEAR AREA	5.12E-03	1	1	1.89E-06	9.69E-09
11.3	A	Screening Scenario	CRIBHOUSE UPPER	2.50E-02	1	1	9.81E-06	2.45E-07
2.0	B	Control Panel 902-3 Subsection 1 - ECCS Div II	CONTROL ROOM	2.47E-04	1	1	6.30E-05	1.56E-08

Table 4-14
Unit 2 Unscreened Fire Compartment Analysis Details

Fire Compartment	Scen	Scenario Description	Fire Compartment Description	Ignition Frequency	NSP	SF	CCDP	CDF
2.0	C	Control Panel 902-3 Subsection 2 - HPCI	CONTROL ROOM	1.24E-04	1	1	3.27E-07	4.05E-11
2.0	D	Control Panel 902-3 Subsection 3 - ECCS Div I	CONTROL ROOM	1.24E-04	1	1	1.04E-04	1.29E-08
2.0	E	Control Panel 902-3 Subsection 3 - ECCS Div I	CONTROL ROOM	1.24E-04	1	1	3.10E-04	3.84E-08
2.0	F	Control Panel 902-4 Subsection 1 - SDC and CRD	CONTROL ROOM	4.95E-04	1	1	1.04E-07	5.13E-11
2.0	G	Control Panels 902-5, 6 - CRD, FW, Cond, Condenser	CONTROL ROOM	1.61E-03	1	1	6.27E-06	1.01E-08
2.0	H	Control Panel 902-8 Subsection 1 - LOOP + Div II	CONTROL ROOM	7.42E-04	1	1	6.72E-04	4.98E-07
2.0	I	Control Panel 902-8 Subsection 2 - Div I	CONTROL ROOM	1.24E-04	1	1	1.45E-06	1.79E-10
2.0	J	Control Panel 923-1 - IA,SA,SW,RBCCW,TBCCW,IC MU	CONTROL ROOM	3.71E-04	1	1	2.65E-05	9.83E-09
2.0	K	Control Panel 923-2 - Loss of 345kV Swyd Conn	CONTROL ROOM	4.95E-04	1	1	2.62E-07	1.30E-10
2.0	L	Control Panel 901-P18 - Loss of 138kV Swyd Conn	CONTROL ROOM	4.95E-04	1	1	7.61E-06	3.77E-09
2.0	M	Control Panel 902-18 - FW	CONTROL ROOM	2.47E-04	1	1	6.27E-06	1.55E-09
2.0	N	Severe Fire w/evacuation	CONTROL ROOM	1.93E-02	0.0034	0.1	1.00E+00	6.56E-06

Table 4-14
Unit 2 Unscreened Fire Compartment Analysis Details

Fire Compartment	Scen	Scenario Description	Fire Compartment Description	Ignition Frequency	NSP	SF	CCDP	CDF
6.1	B	250VDC MCC #3	U3 BATTERY CHARGER ROOM	4.21E-04	1	1	6.95E-07	2.93E-10
6.1	C	125VDC Dist'n Panel Reserve Bus #3	U3 BATTERY CHARGER ROOM	4.21E-04	1	1	2.71E-06	1.14E-09
6.1	D	125VDC Battery Charger 3	U3 BATTERY CHARGER ROOM	1.12E-03	1	1	2.71E-06	3.04E-09
6.1	E	250VDC Battery Charger 3	U3 BATTERY CHARGER ROOM	1.12E-03	1	1	3.41E-09	3.82E-12
6.2	AA	Trays 223T & 225T - %TP	AUX ELEC EQUIP ROOM	1.42E-06	1	1	7.56E-02	1.07E-07
6.2	B	No Loss of System Function - %TT	AUX ELEC EQUIP ROOM	9.66E-03	1	1	3.41E-09	3.29E-11
6.2	BB	Tray 226M - %TP	AUX ELEC EQUIP ROOM	1.42E-06	1	1	3.75E-02	5.33E-08
6.2	C	Loss of Main Condenser - %TC	AUX ELEC EQUIP ROOM	8.16E-04	1	1	6.11E-07	4.99E-10
6.2	CC	Trays 226T & 227T - %TP	AUX ELEC EQUIP ROOM	1.42E-06	1	1	7.61E-03	1.08E-08
6.2	D	Loss of Offsite Power - %LOOP	AUX ELEC EQUIP ROOM	1.22E-03	1	1	3.17E-06	3.87E-09
6.2	DD	Tray 241M - %TP	AUX ELEC EQUIP ROOM	1.42E-06	1	1	4.64E-03	6.58E-09
6.2	E	Propagation to Trays 225B & 249B - %TC	AUX ELEC EQUIP ROOM	8.16E-04	1	1	1.55E-04	1.26E-07
6.2	EE	Tray 241T - %TP	AUX ELEC EQUIP ROOM	1.42E-06	1	1	5.72E-03	8.13E-09
6.2	F	ECCS Div I & Spurious ADS - %TP	AUX ELEC EQUIP ROOM	4.08E-04	1	1	2.66E-04	1.09E-07

Table 4-14
Unit 2 Unscreened Fire Compartment Analysis Details

Fire Compartment	Scen	Scenario Description	Fire Compartment Description	Ignition Frequency	NSP	SF	CCDP	CDF
6.2	FF	Tray 242T - %TP	AUX ELEC EQUIP ROOM	1.42E-06	1	1	5.88E-03	8.35E-09
6.2	G	ECCS Div II - %TT	AUX ELEC EQUIP ROOM	4.08E-04	1	1	3.41E-09	1.39E-12
6.2	GG	Trays 243T & 244T - %TP	AUX ELEC EQUIP ROOM	1.42E-06	1	1	8.02E-03	1.14E-08
6.2	H	HPCI - %TT	AUX ELEC EQUIP ROOM	2.72E-04	1	1	1.72E-07	4.68E-11
6.2	HH	Trays 254M & 306M - %TP	AUX ELEC EQUIP ROOM	1.42E-06	1	1	2.84E-02	4.04E-08
6.2	I	LPCI Div I - %TT	AUX ELEC EQUIP ROOM	1.36E-04	1	1	3.41E-09	4.64E-13
6.2	II	Tray 257M - %TP	AUX ELEC EQUIP ROOM	1.42E-06	1	1	1.91E-04	2.72E-10
6.2	J	LPCI Div II - %TT	AUX ELEC EQUIP ROOM	1.36E-04	1	1	3.41E-09	4.64E-13
6.2	JJ	Tray 257T - %TP	AUX ELEC EQUIP ROOM	1.42E-06	1	1	5.06E-03	7.18E-09
6.2	K	Essential Services Bus - %TC	AUX ELEC EQUIP ROOM	1.90E-03	1	1	6.05E-07	1.15E-09
6.2	KK	Trays 242M, 258M, & 260M - %TP	AUX ELEC EQUIP ROOM	1.42E-06	1	1	1.91E-04	2.71E-10
6.2	L	Instrument Bus - %TT	AUX ELEC EQUIP ROOM	2.72E-04	1	1	6.05E-07	1.65E-10
6.2	LL	Tray 258T - %TP	AUX ELEC EQUIP ROOM	1.42E-06	1	1	5.88E-03	8.35E-09
6.2	M	Panel 2TB-71 - %TC	AUX ELEC EQUIP ROOM	2.72E-04	1	1	7.39E-07	2.01E-10
6.2	MM	Tray 260T - %TP	AUX ELEC EQUIP ROOM	1.42E-06	1	1	5.88E-03	8.35E-09
6.2	N	Propagation to Trays 257B & 258B - %TC	AUX ELEC EQUIP ROOM	4.08E-04	1	1	2.63E-06	1.07E-09
6.2	O	Propagation to Trays 245B, 246B & 247B - %TC	AUX ELEC EQUIP ROOM	9.52E-04	1	1	5.04E-09	4.80E-12
6.2	P	Panel 902-51 - %TC	AUX ELEC EQUIP ROOM	1.36E-04	1	1	6.11E-07	8.31E-11

Table 4-14
Unit 2 Unscreened Fire Compartment Analysis Details

Fire Compartment	Scen	Scenario Description	Fire Compartment Description	Ignition Frequency	NSP	SF	CCDP	CDF
6.2	Q	Propagation to Trays 650B & 671B - %TC	AUX ELEC EQUIP ROOM	4.08E-04	1	1	3.41E-09	1.39E-12
6.2	R	Propagation to Tray 655B - %TC	AUX ELEC EQUIP ROOM	8.16E-04	1	1	1.10E-08	8.95E-12
6.2	S	Propagation to Tray 229 - %TC	AUX ELEC EQUIP ROOM	9.52E-04	1	1	1.27E-06	1.21E-09
6.2	V	Bounding Fire - %LOOP	AUX ELEC EQUIP ROOM	2.21E-04	1	1	9.96E-05	2.20E-08
7.0.A.1	B	250VDC MCC #2	U2 BATTERY ROOM	3.28E-04	1	1	3.41E-09	1.12E-12
7.0.A.1	C	125VDC Dist'n Panel Main Bus	U2 BATTERY ROOM	3.28E-04	1	1	2.14E-05	7.02E-09
7.0.A.1	D	125VDC Dist'n Panel Reserve Bus	U2 BATTERY ROOM	3.28E-04	1	1	7.37E-06	2.42E-09
7.0.A.1	E	125VDC Battery Chargers 2 & 2A	U2 BATTERY ROOM	1.78E-03	1	1	2.14E-05	3.81E-08
7.0.A.1	F	250VDC Battery Chargers 2 & 2/3	U2 BATTERY ROOM	1.78E-03	1	1	3.41E-09	6.07E-12
7.0.A.1	G	Self-Initiated Cable Fire - Non-Severe - %TP	U2 BATTERY ROOM	5.35E-05	0.033	0.8	1.07E-03	1.51E-09
7.0.A.1	H	Self-Initiated Cable Fire - %TP	U2 BATTERY ROOM	5.35E-05	0.033	0.2	2.79E-02	9.85E-09
8.2.5.A	B	2-4706A IA Compressor Fire	U2 NORTH TRACKWAY/SWGR AREA	5.39E-04	1	1	2.97E-05	1.60E-08

Table 4-14
Unit 2 Unscreened Fire Compartment Analysis Details

Fire Compartment	Scen	Scenario Description	Fire Compartment Description	Ignition Frequency	NSP	SF	CCDP	CDF
8.2.5.A	C	2-4601 SA Comp Fire - Large w/o Supp	U2 NORTH TRACKWAY/SWGR AREA	5.39E-04	0.02	0.18	2.98E-02	5.78E-08
8.2.5.A	D	2-4601 SA Comp Fire - Large w/Supp	U2 NORTH TRACKWAY/SWGR AREA	5.39E-04	0.98	0.18	1.69E-05	1.61E-09
8.2.5.A	E	2-4601 SA Compressor Fire - Small	U2 NORTH TRACKWAY/SWGR AREA	5.39E-04	1	0.82	7.45E-06	3.29E-09
8.2.5.A	F	480 V Bus/Xfmr 25 - Large	U2 NORTH TRACKWAY/SWGR AREA	2.07E-03	1	0.2	7.51E-04	3.11E-07
8.2.5.A	G	480 V Bus/Xfmr 26 - Large	U2 NORTH TRACKWAY/SWGR AREA	2.07E-03	1	0.2	2.25E-04	9.33E-08
8.2.5.A	H	480 V MCC 25-2	U2 NORTH TRACKWAY/SWGR AREA	5.24E-04	1	1	9.84E-06	5.15E-09
8.2.5.A	I	480 V MCC 26-8	U2 NORTH TRACKWAY/SWGR AREA	5.24E-04	1	1	8.71E-06	4.56E-09
8.2.5.A	J	480 V MCC 29-2	U2 NORTH TRACKWAY/SWGR AREA	5.24E-04	1	1	1.75E-07	9.16E-11
8.2.5.A	K	480 V Bus/Xfmr 25 - Small	U2 NORTH TRACKWAY/SWGR AREA	2.07E-03	1	0.8	7.38E-06	1.22E-08
8.2.5.A	L	480 V Bus/Xfmr 26 - Small	U2 NORTH TRACKWAY/SWGR AREA	2.07E-03	1	0.8	7.44E-06	1.23E-08
8.2.5.A	M	RFP Area - Large Fire w/o Suppression	U2 NORTH TRACKWAY/SWGR AREA	4.42E-03	0.02	0.18	9.56E-03	1.52E-07
8.2.5.A	N	RFP C - Large Fire w/Suppression	U2 NORTH TRACKWAY/SWGR AREA	1.47E-03	0.98	0.18	9.56E-03	2.48E-06

Table 4-14
Unit 2 Unscreened Fire Compartment Analysis Details

Fire Compartment	Scen	Scenario Description	Fire Compartment Description	Ignition Frequency	NSP	SF	CCDP	CDF
8.2.5.A	O	RFP C - Small Fire w/o Suppression	U2 NORTH TRACKWAY/SWGR AREA	1.47E-03	0.02	0.82	9.56E-03	2.31E-07
8.2.5.A	P	Misc Cond Pumps - MCC 26-1 - Large Fire	U2 NORTH TRACKWAY/SWGR AREA	1.81E-03	1	0.2	4.65E-03	1.68E-06
8.2.5.A	Q	Misc Cond Pumps - MCC 26-1 - Small Fire	U2 NORTH TRACKWAY/SWGR AREA	1.81E-03	1	0.8	6.80E-06	9.84E-09
8.2.5.A	R	Self-Initiated Cable Fire - Severe - %TP	U2 NORTH TRACKWAY/SWGR AREA	4.01E-05	1	0.2	2.96E-02	2.37E-07
8.2.5.A	S	Self-Initiated Cable Fire - Non-Severe - %TP	U2 NORTH TRACKWAY/SWGR AREA	4.01E-05	1	0.8	9.49E-04	3.04E-08
8.2.5.A	T	RFP Area - Small Fire w/Suppression	U2 NORTH TRACKWAY/SWGR AREA	4.42E-03	0.98	0.82	5.90E-06	2.09E-08
8.2.5.A	U	RFP A/B - Large w/Supp & Small w/o Supp	U2 NORTH TRACKWAY/SWGR AREA	2.95E-03	0.98	1	9.09E-06	2.63E-08
8.2.5.C	B	Instrument Air Compressor 2B Large Fire w/o Supp	2/3 TB CORRIDOR	5.63E-04	0.02	0.18	7.31E-02	1.48E-07
8.2.5.C	C	Resin Air Compressors	2/3 TB CORRIDOR	1.13E-03	1	1	7.65E-06	8.64E-09
8.2.5.C	D	MCC 27-1	2/3 TB CORRIDOR	4.09E-04	1	1	7.38E-06	3.02E-09
8.2.5.C	E	MCC 37-1	2/3 TB CORRIDOR	4.09E-04	1	1	7.38E-06	3.02E-09
8.2.5.C	F	Cond Demin Panel 2252-11	2/3 TB CORRIDOR	4.09E-04	1	1	7.38E-06	3.02E-09
8.2.5.C	G	Cond Demin Panel 2253-11	2/3 TB CORRIDOR	4.09E-04	1	1	7.38E-06	3.02E-09
8.2.5.C	H	EHC Pumps	2/3 TB CORRIDOR	7.80E-04	1	1	7.66E-06	5.97E-09

Table 4-14
Unit 2 Unscreened Fire Compartment Analysis Details

Fire Compartment	Scen	Scenario Description	Fire Compartment Description	Ignition Frequency	NSP	SF	CCDP	CDF
8.2.5.C	I	Instrument Air Compressor 2B Small Fire	2/3 TB CORRIDOR	5.63E-04	1	0.82	1.38E-04	6.39E-08
8.2.5.C	J	Instrument Air Compressor 2B Large Fire w/Supp	2/3 TB CORRIDOR	5.63E-04	0.98	0.18	1.38E-04	1.38E-08
8.2.5.E	B	RFP Area - Large Fire w/o Suppression	U3 WEST CORRIDOR AND TRACKWAY	5.40E-03	0.02	0.18	1.02E-03	1.97E-08
8.2.5.E	C	RFP Area - Large Fire w/ Suppression	U3 WEST CORRIDOR AND TRACKWAY	5.40E-03	0.98	0.18	1.40E-05	1.33E-08
8.2.5.E	D	RFP Area - Small Fire	U3 WEST CORRIDOR AND TRACKWAY	5.40E-03	1	0.82	1.40E-05	6.20E-08
8.2.5.E	E	Switchgear Area Fire	U3 WEST CORRIDOR AND TRACKWAY	1.30E-02	1	1	1.72E-06	2.24E-08
8.2.6.A	B	4 kV Bus 21	CONTROL ROOM BKUP VENTILATION	6.59E-04	1	1	5.90E-06	3.88E-09
8.2.6.A	C	4 kV Bus 22	CONTROL ROOM BKUP VENTILATION	6.59E-04	1	1	5.90E-06	3.88E-09
8.2.6.A	D	4 kV Bus 23 - Large Fire	CONTROL ROOM BKUP VENTILATION	1.46E-03	1	0.2	8.73E-05	2.55E-08
8.2.6.A	E	4 kV Bus 24 - Large Fire	CONTROL ROOM BKUP VENTILATION	1.64E-03	1	0.2	1.64E-03	5.38E-07
8.2.6.A	F	Exciter Panel	CONTROL ROOM BKUP VENTILATION	2.77E-04	1	1	7.65E-08	2.12E-11
8.2.6.A	G	480 MCC 28-2/28-3	CONTROL ROOM BKUP VENTILATION	9.79E-04	1	1	6.94E-06	6.80E-09

Table 4-14
Unit 2 Unscreened Fire Compartment Analysis Details

Fire Compartment	Scen	Scenario Description	Fire Compartment Description	Ignition Frequency	NSP	SF	CCDP	CDF
8.2.6.A	H	4 kV Bus 23 - Small Fire	CONTROL ROOM BKUP VENTILATION	1.46E-03	1	0.8	2.52E-07	2.95E-10
8.2.6.A	I	4 kV Bus 24 - Small Fire	CONTROL ROOM BKUP VENTILATION	1.64E-03	1	0.8	4.73E-06	6.20E-09
8.2.6.A	J	Self-Initiated Cable Fire - %TP	CONTROL ROOM BKUP VENTILATION	5.36E-05	1	1	3.17E-05	1.70E-09
8.2.6.B	B	Tray 3BA - %TP	U2 MEZZANINE	5.36E-05	1	1	1.25E-02	6.72E-07
8.2.6.B	C	Tray 3BA - %TC	U2 MEZZANINE	5.36E-05	1	1	3.17E-05	1.70E-09
8.2.6.C	B	MCC 25-1	U2/3 SBGT & TBCCW HX	3.80E-04	1	1	8.16E-06	3.10E-09
8.2.6.C	C	MCCs 35-1 and 39-2	U2/3 SBGT & TBCCW HX	7.60E-04	1	1	1.10E-08	8.34E-12
8.2.6.C	D	480V Swgr 27	U2/3 SBGT & TBCCW HX	1.78E-03	1	1	7.65E-08	1.36E-10
8.2.6.C	E	480V Swgr 37	U2/3 SBGT & TBCCW HX	1.78E-03	1	1	3.41E-09	6.07E-12
8.2.6.C	F	2/3 Sparging Air Compressors	U2/3 SBGT & TBCCW HX	1.01E-03	1	1	1.10E-08	1.11E-11
8.2.6.C	G	Unit 2 TBCCW Pumps	U2/3 SBGT & TBCCW HX	2.74E-04	1	1	7.40E-06	2.03E-09
8.2.6.C	H	Unit 3 TBCCW Pumps	U2/3 SBGT & TBCCW HX	2.74E-04	1	1	1.10E-08	3.01E-12
8.2.6.E	B	DC Panel Fire	U3 MEZZANINE FLOOR	1.16E-03	1	1	3.38E-05	3.92E-08
8.2.6.E	C	MCC 38-2/38-3 Fire	U3 MEZZANINE FLOOR	1.16E-03	1	1	6.93E-08	8.04E-11
8.2.6.E	D	Bounding Switchgear Fire	U3 MEZZANINE FLOOR	3.42E-03	1	1	6.93E-08	2.37E-10
					TOTAL UNIT 2 CDF			**1.69E-5**

4.7.2.17 Unit 3 Analysis - Fire Compartment 1.1.1.1

The initial quantification for this fire compartment resulted in a calculated CDF contribution slightly greater than the screening criteria of 1.0E-7/yr. This fire compartment is the Unit 3 Torus Compartment. A review of the configuration, equipment, cables, and fire ignition sources in this compartment concluded that the development of multiple explicit fire scenarios to bound the analysis was appropriate. These scenarios considered postulated fires impacting subsets of the target population based on their spatial location.

4.7.2.18 Unit 3 Analysis - Fire Compartment 1.1.1.2

The initial quantification for this fire compartment resulted in a calculated CDF contribution greater than the screening criteria of 1.0E-7/yr. This fire compartment is the ground floor of the Reactor Building. The Reactor Building, in general, contains relatively few significant ignition sources or heavy concentration of combustible materials. The ground floor of the Reactor Building shares a common ventilation path with the upper elevations through the >400 ft^2 open equipment hatch. This ventilation path was not credited in the fire modeling analyses.

A review of the configuration, equipment, cables, and fire ignition sources in this compartment concluded that the development of multiple explicit fire scenarios to bound the analysis was appropriate. These scenarios considered fires originating at the Drywell/Suppression Pool Pump Back Air Compressor, or the MCCs located throughout the compartment. The analysis of this fire compartment also included a specific investigation of the risk significance of fire induced spurious actuation of the ADS valves.

4.7.2.19 Unit 3 Analysis - Fire Compartment 1.1.1.3

The initial quantification for this fire compartment resulted in a calculated CDF contribution greater than the screening criteria of 1.0E-7/yr. This fire compartment is the second floor of the Reactor Building. The Reactor Building, in general, contains relatively few significant ignition sources or heavy concentration of combustible materials. The ground floor of the Reactor Building shares a common ventilation path with the upper elevations through the >400 ft^2 open equipment hatch. This ventilation path was not credited in the fire modeling analyses.

A review of the configuration, equipment, cables, and fire ignition sources in this compartment concluded that the development of multiple explicit fire scenarios to bound the analysis was appropriate. The walkdown inspection of this fire compartment concluded that each of the 4kV switchgear buses in the compartment required fire modeling. The analysis for the CDF contribution due to the switchgear fires was based on loss of the switchgear along with failure of the circuits in the cable tray directly above. The scenarios also considered to potential for an 'explosive' type event wherein the damage could propagate horizontally. Separate scenarios were also generated to address the RBCCW pump/motor as well as specific evaluation of the ADS circuits.

4.7.2.20 Unit 3 Analysis - Fire Compartment 1.1.1.4

The initial quantification for this fire compartment resulted in a calculated CDF contribution greater than the screening criteria of 1.0E-7/yr. This fire compartment is the switchgear floor of the Reactor Building. The Reactor Building, in general, contains relatively few significant ignition sources or heavy concentration of combustible materials. The ground floor of the Reactor Building shares a common ventilation path with the upper elevations through the >400 ft^2 open equipment hatch. This ventilation path was not credited in the fire modeling analyses.

A review of the configuration, equipment, cables, and fire ignition sources in this compartment concluded that the development of multiple explicit fire scenarios to bound the analysis was appropriate. The walkdown inspection of this fire compartment concluded that each of the 480 Vac switchgear buses in the compartment required fire modeling as well as the DC MCCs.

4.7.2.21 Unit 3 Analysis - Fire Compartment 1.4.1

The initial quantification for this fire compartment resulted in a calculated CDF contribution greater than the screening criteria of 1.0E-7/yr. This fire compartment is the TIP Room. Since the bounding value was only slightly greater than the screening criteria, no further refinements were performed. The bounding assessment included consideration of the ADS circuits whose postulated failure could cause the spurious actuation of the ADS valves.

4.7.2.22 Unit 3 Analysis - Fire Compartment 2.0

This fire compartment is the main control room. The discussion and results of the main control room analysis are presented in Section 4.7.4.

4.7.2.23 Unit 3 Analysis - Fire Compartment 6.1

This fire compartment is the Unit 3 Battery Charger room. The walkdown inspection of this compartment determined that the cabinets located in this compartment are substantially sealed. These cabinets consist of battery chargers and distribution panels. The analysis considered individual fire scenarios for each cabinet. Each scenario assumed the functional failure of the equipment and the failure of circuits in the cables trays located above them.

4.7.2.24 Unit 3 Analysis - Fire Compartment 6.2

The examination of the Dresden Unit 2/3 Auxiliary Electric Equipment Room (AEER) was performed to develop the credible fire scenarios for the upgraded fire analysis. The overall methodology that was applied for the Unit 3 analysis was identical to that for the Unit 2 analysis. Refer to Section 4.7.2.7 for a discussion of the analysis methodology.

The information which was developed for each of the control cabinets is summarized in Table 4-15. In certain cases, the cabinet configuration is such that fire propagation beyond the panel boundaries to the overhead cable trays is credible. In these cases, the consequences of this propagation is determined by identifying the affected cable trays, referring to the SLICE database to obtain the associated cables, and relating these cables to PSA model basic events.

Table 4-15
Auxiliary Relay Room Cabinets Fire Scenario Summary

Cabinet ID	Sections	Scenario	Propagation	Fire Consequences
Computer Peripherals	1	B	No	No impact on credited systems – use %TT
Data Acquisition Cabinet #1	1	B	No	No impact on credited systems - use %TT
Data Acquisition Cabinet #2	1	B	No	No impact on credited systems - use %TT
EPA 2A-1	1	B	No	No impact on credited systems - use %TT
EPA 2A-2	1	B	No	No impact on credited systems - use %TT
EPA 2AB-1	1	B	No	No impact on credited systems - use %TT
EPA 2AB-2	1	B	No	No impact on credited systems - use %TT
EPA 2B-1	1	B	No	No impact on credited systems - use %TT
EPA 2B-2	1	B	No	No impact on credited systems - use %TT
EPA 3A-1	1	B	No	No impact on credited systems - use %TT
EPA 3A-2	1	B	No	No impact on credited systems - use %TT
EPA 3AB-1	1	B	No	No impact on credited systems - use %TT
EPA 3AB-2	1	B	No	No impact on credited systems - use %TT
EPA 3B-1	1	B	No	No impact on credited systems - use %TT
EPA 3B-2	1	B	No	No impact on credited systems - use %TT
Old EHC Panel	1	B	No	No impact on credited systems - use %TT

Table 4-15
Auxiliary Relay Room Cabinets Fire Scenario Summary

Cabinet ID	Sections	Scenario	Propagation	Fire Consequences
RLC #13	1	B	No	No impact on credited systems - use %TT
2TB-71	2	M	No	Loss of various specific system functions based on cables in cable trays.
2202-70A	1	B	No	No impact on credited systems - use %TT – Loss of ATWS/RPT functions
2202-70B	1	B	No	No impact on credited systems - use %TT – Loss of ATWS/RPT functions
2203-70A	1	B	No	No impact on credited systems - use %TT – Loss of ATWS/RPT functions
2203-70B	1	B	No	No impact on credited systems - use %TT – Loss of ATWS/RPT functions
2253-91	1	B	No	No impact on credited systems – use %TT
2253-92	1	B	No	No impact on credited systems – use %TT
2-8001-A	1	B	No	No impact on credited systems - use %TT – Unit 2 RPS MG Set
2-8001-B	1	B	No	No impact on credited systems - use %TT – Unit 2 RPS MG Set
3-8001-A	1	B	No	No impact on credited systems - use %TT – Unit 3 RPS MG Set
3-8001-B	1	B	No	No impact on credited systems - use %TT – Unit 3 RPS MG Set
902-27	3	N	Yes	No impact on credited systems - potential propagation to trays 257B and 258B
902-28	4	B	No	No impact on credited systems – use %TT
902-29	4	D	No	Loss of Unit 2 Offsite Power - TR21 and 22 – conservatively treat as Unit 3 LOOP
902-31A	6	E	Yes	Loss of Unit 2 Main Condenser - also potential propagation to trays 225 and 249
902-31B	-	-	-	"

Table 4-15
Auxiliary Relay Room Cabinets Fire Scenario Summary

Cabinet ID	Sections	Scenario	Propagation	Fire Consequences
902-31C	-	-	-	"
902-31D	-	-	-	"
902-31E	-	-	-	"
902-31F	-	-	-	"
902-32	3	B	No	Loss of Unit 2 ECCS Div I CS and RHR systems and ADS
902-33	3	B	No	Loss of Unit 2 ECCS Div II CS, RHR, and HPCI systems
902-34	7	O	Yes	No impact on credited systems - potential propagation trays 245B, 246B, and 247
902-38	1	B	No	No impact on credited systems - use %TT
902-39	2	B	No	Loss of Unit 2 HPCI
902-40	2	B	No	Loss of Unit 2 Main Condenser due to MSIV closure
902-41	2	B	No	Loss of Unit 2 Main Condenser due to MSIV closure
902-46	1	B	No	Loss of Unit 2 LPCI Line Break Detection - ESS I
902-47	1	B	No	Loss of Unit 2 LPCI Line Break Detection – ESS II
902-49	1	K	No	Loss of Unit 2 Essential Services Bus
902-50	1	L	No	Loss of Unit 2 Instrument Bus
902-51	1	B	No	Loss of Unit 2 Main Condenser
902-52A	1	B	No	Loss of Unit 2 RPS Bus 2B
902-52B	1	B	No	Loss of Unit 2 RPS Bus 2A
902-61	2	B	No	Loss of Unit 2 Main Condenser due to MSIV closure

Table 4-15
Auxiliary Relay Room Cabinets Fire Scenario Summary

Cabinet ID	Sections	Scenario	Propagation	Fire Consequences
902-63	1	K	No	Loss of Essential Services Bus Auto Transfer Switch
902-63A	2	K	No	Loss of Unit 2 Essential Services Bus Inverter - conservatively treated based on loss of the Unit 3 ESS Bus
902-63B	2	K	No	Loss of Unit 2 Essential Services Bus Static Switch - conservatively treated based on loss of the Unit 3 ESS Bus
902-63C	1	K	No	Loss of Unit 2 Essential Services Bus Line Regulator - conservatively treated based on loss of the Unit 3 ESS Bus
902-64A	1	B	No	No impact on credited systems - use %TT
902-64B	1	B	No	No impact on credited systems - use %TT
902-68	1	B	No	No impact on credited systems – use %TT
902-69	1	B	No	No impact on credited systems – use %TT
902-70	1	B	No	No impact on credited systems – use %TT
902-71	1	B	No	No impact on credited systems – use %TT
902-74	1	D	No	Loss Of Unit 2 UAT/RAT Supplemental Protection Cabinet – Conservatively Treat As Unit 3 LOOP
903-27	3	Q	Yes	No impact on credited systems - potential propagation to trays 650B and 671B
903-28	4	B	No	No impact on credited systems - use %TT
903-29	3	D	No	Loss of Unit 3 Offsite Power - TR31 and 32
903-31A	6	R	Yes	Loss of Unit 3 Main Condenser – also potential propagation tray 655B
903-31B	-	-	-	"
903-31C	-	-	-	"

Table 4-15
Auxiliary Relay Room Cabinets Fire Scenario Summary

Cabinet ID	Sections	Scenario	Propagation	Fire Consequences
903-31D	-	-	-	"
903-31E	-	-	-	"
903-31F	-	-	-	"
903-32	3	F	No	Loss of Unit 3 ECCS Div I CS and RHR systems and ADS
903-33	3	G	No	Loss of Unit 3 ECCS Div II CS, RHR, and HPCI systems
903-34	7	S	Yes	No impact on credited systems - potential propagation to tray 229
903-38	1	B	No	No impact on credited systems - use %TT
903-39	2	H	No	Loss of Unit 3 HPCI
903-40	2	C	No	Loss of Unit 3 Main Condenser due to MSIV closure
903-41	2	C	No	Loss of Unit 3 Main Condenser due to MSIV closure
903-46	1	I	No	Loss of Unit 3 LPCI Line Break Detection – ESS I
903-47	1	J	No	Loss of Unit 3 LPCI Line Break Detection - ESS II
903-49	1	K	No	Loss of Unit 3 Essential Services Bus
903-50	1	L	No	Loss of Unit 3 Instrument Bus
903-51	1	P	No	Loss of Unit 3 Main Condenser
903-52A	1	B	No	Loss of Unit 3 RPS Bus 3B
903-52B	1	B	No	Loss of Unit 3 RPS Bus 3A
903-60	1	B	No	No impact on credited systems – use %TT
903-62	2	C	No	Loss of Unit 3 Main Condenser due to MSIV closure

Table 4-15
Auxiliary Relay Room Cabinets Fire Scenario Summary

Cabinet ID	Sections	Scenario	Propagation	Fire Consequences
903-63	1	K	No	Loss of Essential Services Bus Auto Transfer Switch
903-63A	2	K	No	Loss of Unit 3 Essential Services Bus Inverter – conservatively treated based on loss of the ESS Bus
903-63B	2	K	No	Loss of Unit 3 Essential Services Bus Static Switch – conservatively treated based on loss of the ESS Bus
903-63C	1	K	No	Loss of Unit 3 Essential Services Bus Line Regulator – conservatively treated based on loss of the ESS Bus
903-64A	1	B	No	No impact on credited systems - use %TT
903-64B	1	B	No	No impact on credited systems - use %TT
903-68	1	B	No	No impact on credited systems – use %TT
903-69	1	B	No	No impact on credited systems – use %TT
903-70	1	B	No	No impact on credited systems – use %TT
903-71	1	B	No	No impact on credited systems – use %TT
903-74	1	D	No	Loss of Unit 2 UAT/RAT Supplemental Protection Cabinet – conservatively treat as Unit 2 LOOP
TOTAL	147			

In addition to the scenarios described in Table 4-15, additional cases were considered because of specific issues related to potential fire induced spurious actuation of ADS valves. These additional cases considered self-initiated cable tray fires.

4.7.2.25 Unit 2 Analysis - Fire Compartment 7.0.A.1

The initial quantification for this fire compartment resulted in a calculated CDF contribution greater than the screening criteria of 1.0E-7/yr. This fire compartment is adjacent to the Unit 2 Battery Room. A review of the configuration, equipment and cables in this fire compartment concluded that the development of multiple explicit fire scenarios to bound the analysis was appropriate. These scenarios considered postulated fires originating in the electrical cabinets, MCCs, and battery chargers. A self-initiated cable fire was also considered to address the potential for fire induced spurious actuation of ADS valves.

4.7.2.26 Unit 3 Analysis - Fire Compartment 8.2.1.B

The initial quantification for this fire compartment resulted in a calculated CDF contribution greater than the screening criteria of 1.0E-7/yr. This fire compartment is the Unit 3 Condensate Pump Area. Since the bounding value was only slightly greater than the screening criteria, no further refinements were performed.

4.7.2.27 Unit 3 Analysis - Fire Compartment 8.2.4

This fire compartment is the Unit 3 Cable Tunnel which runs below the ground floor of the Turbine Building. The tunnel contains stacks of cable trays and conduits. No significant ignition sources exist in the tunnel. The tunnel was not significantly ventilated as floor, ceiling and wall penetrations were sealed and access doors were closed.

The cable trays in the cable tunnel are solid bottom trays with extensive fire suppression throughout. Even though no significant ignition sources were present, individual cable tray fire scenarios were developed by assuming a self-induced fire in a tray. No propagation was considered due to the extensive suppression system. A severe fire scenario was developed considering ignition of multiple cable trays and suppression system unavailability.

4.7.2.28 Unit 3 Analysis - Fire Compartment 8.2.5.A

The initial quantification for this fire compartment resulted in a calculated CDF contribution greater than the screening criteria of 1.0E-7/yr. This fire compartment is the ground floor of the Unit 2 Turbine Building. This area is similar to and is the Unit 2 equivalent of fire compartment 8.2.5.E. A review of the configuration, equipment and cables in this fire compartment concluded that the development of multiple explicit fire scenarios to bound the analysis was appropriate. These scenarios generally divided the area into two sub-compartment based on the location of the trackway and credited the automatic fire suppression system installed in the Reactor Feedwater Pump area. The evaluation of this compartment also considered the potential for fire induced failure of critical ADS circuits causing a spurious actuation of ADS valves.

4.7.2.29 Unit 3 Analysis - Fire Compartment 8.2.5.C

The initial quantification for this fire compartment resulted in a calculated CDF contribution slightly greater than the screening criteria of 1.0E-7/yr. This fire compartment is the ground floor Turbine Building common area between Units 2 and 3. A review of the configuration, equipment and cables in this fire compartment concluded that the development of multiple explicit fire scenarios to bound the analysis was appropriate. These scenarios considered postulated fires originating in the MCCs, air compressors, and pumps. The evaluation of this compartment also considered the potential for fire induced failure of critical ADS circuits causing a spurious actuation of ADS valves.

4.7.2.30 Unit 3 Analysis – Fire Compartment 8.2.5.E

The initial quantification for this fire compartment resulted in a calculated CDF contribution greater than the screening criteria of 1.0E-7. This fire compartment is the ground floor of the Unit 3 Turbine Building. This area consists of the non-Essential 480 Vac switchgear, Turbine Building Trackway, and Reactor Feedwater pump areas. A review of the configuration, equipment and cables in this fire compartment concluded that the development of multiple explicit fire scenarios to bound the analysis was appropriate. These scenarios considered postulated fires originating in the MCCs, air compressors, and pumps. The evaluation of this area credited the automatic fire suppression system installed in the Reactor Feedwater Pump area.

4.7.2.31 Unit 3 Analysis – Fire Compartment 8.2.6.A

The initial quantification for this fire compartment resulted in a calculated CDF contribution greater than the screening criteria of 1.0E-7/yr. This fire compartment is the mezzanine level of the Unit 2 Turbine Building. This area is similar to and is the Unit 2 equivalent of fire compartment 8.2.6.E. A review of the configuration, equipment and cables in this fire compartment concluded that a single bounding scenario, in addition to an explicit scenario to address the ADS circuits, was appropriate.

4.7.2.32 Unit 3 Analysis - Fire Compartment 8.2.6.C

The initial quantification for this fire compartment resulted in a calculated CDF contribution slightly greater than the screening criteria of 1.0E-7/yr. This fire compartment is the mezzanine level of the Turbine Building common area between Units 2 and 3. A review of the configuration, equipment and cables in this fire compartment concluded that the development of multiple explicit fire scenarios to bound the analysis was appropriate. These scenarios considered postulated fires originating in the MCCs, air compressors, and pumps. The evaluation of this compartment also considered the potential for fire induced failure of critical ADS circuits causing a spurious actuation of ADS valves.

4.7.2.33 Unit 3 Analysis - Fire Compartment 8.2.6.D

The initial quantification for this fire compartment resulted in a calculated CDF contribution greater than the screening criteria of 1.0E-7/yr. This fire compartment is the mezzanine level of the Unit 3 Turbine Building. A review of the configuration, equipment and cables in this fire compartment concluded that there are no significant ignition sources of concern. However, the area does contain circuits whose fire induced failure could cause a spurious actuation of multiple ADS valves. Therefore, fire scenarios were developed to evaluate postulated self-initiated cable tray fires.

4.7.2.34 Unit 3 Analysis - Fire Compartment 8.2.6.E

The initial quantification for this fire compartment resulted in a calculated CDF contribution greater than the screening criteria of 1.0E-7/yr. This fire compartment is the mezzanine level of the Unit 3 Turbine Building and includes both the switchgear and general areas. A review of the configuration, equipment and cables in this fire compartment concluded that the development of multiple explicit fire scenarios to bound the analysis was appropriate. These scenarios considered postulated fire originating in the switchgears, MCCs, and electrical panels. The postulated switchgear fires included the consideration of an 'explosive' type event wherein horizontal propagation is possible. This is was considered to ensure a bounding assessment of the area.

4.7.2.35 Unit 3 Analysis – Fire Compartment 8.2.7

The initial quantification for this fire compartment resulted in a calculated CDF contribution slightly greater than the screening criteria of 1.0E-7/yr. Since the bounding value was only slightly greater than the screening criteria, no further refinements were performed.

4.7.2.36 Unit 3 Analysis – Fire Compartment 9.0.B

The initial quantification for this fire compartment resulted in a calculated CDF contribution slightly greater than the screening criteria of 1.0E-7/yr. Since the bounding value was only slightly greater than the screening criteria, no further refinements were performed.

4.7.2.37 Unit 3 Analysis – Fire Compartment 11.3

The initial quantification for this fire compartment resulted in a calculated CDF contribution slightly greater than the screening criteria of 1.0E-7/yr. Since the bounding value was only slightly greater than the screening criteria, no further refinements were performed.

Table 4-16
Unit 3 Unscreened Fire Compartment Analysis Details

Fire Compartment	Scen	Scenario Description	Fire Compartment Description	Ignition Frequency	NSP	SF	CCDP	CDF
1.1.1.1	B	Trays 844 & 1029-1034	U3 TORUS BASEMENT	9.10E-04	1	1	6.46E-06	5.88E-09
1.1.1.1	C	Trays 845 & 1031-1034	U3 TORUS BASEMENT	9.10E-04	1	1	1.86E-06	1.69E-09
1.1.1.2	B	MCC 38-1	U3 RX BLDG GROUND FLOOR	1.81E-03	1	1	1.13E-05	2.05E-08
1.1.1.2	C	MCC 38-4	U3 RX BLDG GROUND FLOOR	1.81E-03	1	1	8.93E-06	1.62E-08
1.1.1.2	D	MCC 38-7	U3 RX BLDG GROUND FLOOR	1.81E-03	1	1	3.41E-09	6.17E-12
1.1.1.2	E	MCC 39-1	U3 RX BLDG GROUND FLOOR	1.81E-03	1	1	1.92E-07	3.47E-10
1.1.1.2	F	MCC 39-7	U3 RX BLDG GROUND FLOOR	1.81E-03	1	1	8.90E-07	1.61E-09
1.1.1.2	G	Pump Back Air Compressor Lg Fire	U3 RX BLDG GROUND FLOOR	1.44E-03	1	0.18	1.56E-03	4.04E-07
1.1.1.2	H	Pump Back Air Compressor Sm Fire	U3 RX BLDG GROUND FLOOR	1.44E-03	1	0.82	5.04E-09	5.95E-12
1.1.1.2	I	Self-Initiated Cable Fire - %TI	U3 RX BLDG GROUND FLOOR	3.03E-05	1	1	9.00E-03	2.73E-07
1.1.1.3	B	RBCCW Pumps	U3 SECOND FLOOR RX BLDG	2.18E-03	1	1	3.41E-09	7.43E-12
1.1.1.3	C	RWCU Pumps	U3 SECOND FLOOR RX BLDG	2.18E-03	1	1	3.48E-05	7.58E-08
1.1.1.3	D	Bus 33-1	U3 SECOND FLOOR RX BLDG	1.94E-03	1	1	4.46E-05	8.65E-08

Table 4-16
Unit 3 Unscreened Fire Compartment Analysis Details

Fire Compartment	Scen	Scenario Description	Fire Compartment Description	Ignition Frequency	NSP	SF	CCDP	CDF
1.1.1.3	E	Bus 34-1	U3 SECOND FLOOR RX BLDG	1.94E-03	1	1	1.38E-04	2.67E-07
1.1.1.3	F	Self-Initiated Cable Fire - %TI	U3 SECOND FLOOR RX BLDG	7.08E-05	1	1	1.24E-02	8.78E-07
1.1.1.3	G	Self-Initiated Cable Fire - Trays 959-973 - %TP	U3 SECOND FLOOR RX BLDG	7.08E-05	1	1	2.71E-02	1.92E-06
1.1.1.3	H	Self-Initiated Cable Fire - Trays 974-987 - %TP	U3 SECOND FLOOR RX BLDG	7.08E-05	1	1	4.41E-03	3.12E-07
1.1.1.4	B	Switchgear 38	U3 RX BLDG SWITCHGEAR AREA	3.15E-03	1	1	1.27E-08	3.99E-11
1.1.1.4	C	Switchgear 39	U3 RX BLDG SWITCHGEAR AREA	3.15E-03	1	1	1.86E-07	5.87E-10
1.1.1.4	D	250VDC MCC 3A	U3 RX BLDG SWITCHGEAR AREA	7.82E-03	1	1	6.95E-07	5.44E-09
1.1.1.4	E	250VDC MCC 3B/125VDC Panel #3	U3 RX BLDG SWITCHGEAR AREA	7.82E-03	1	1	1.85E-06	1.45E-08
1.4.1	B	Self-Initiated Cable Fire - %TP	U3 TIP ROOM	6.32E-04	1	1	1.73E-04	1.10E-07
1.4.1	C	Self-Initiated Cable Fire - %TC	U3 TIP ROOM	6.32E-04	1	1	3.41E-09	2.16E-12
11.3	A	Screening Scenario	CRIBHOUSE UPPER	2.50E-02	1	1	9.53E-06	2.38E-07
2.0	B	Control Panel 903-3 Subsection 1 - ECCS Div II	CONTROL ROOM	2.47E-04	1	1	6.30E-05	1.56E-08

Table 4-16
Unit 3 Unscreened Fire Compartment Analysis Details

Fire Compartment	Scen	Scenario Description	Fire Compartment Description	Ignition Frequency	NSP	SF	CCDP	CDF
2.0	C	Control Panel 903-3 Subsection 2 - HPCI	CONTROL ROOM	1.24E-04	1	1	3.27E-07	4.05E-11
2.0	D	Control Panel 903-3 Subsection 3 - ECCS Div I	CONTROL ROOM	1.24E-04	1	1	1.04E-04	1.28E-08
2.0	E	Control Panel 903-3 Subsection 3 - ECCS Div I	CONTROL ROOM	1.24E-04	1	1	3.10E-04	3.84E-08
2.0	F	Control Panel 903-4 Subsection 1 - SDC and (R1)	CONTROL ROOM	4.95E-04	1	1	1.04E-07	5.13E-11
2.0	G	Control Panel 903-5,6 - CRD, FW, Cond, Condenser	CONTROL ROOM	1.61E-03	1	1	6.27E-06	1.01E-08
2.0	H	Control Panel 903-8 Subsection I -- LOOP + Div II	CONTROL ROOM	7.42E-04	1	1	6.28E-04	4.66E-07
2.0	I	Control Panel 903-8 Subsection 2 - Div I	CONTROL ROOM	1.24E-04	1	1	1.44E-06	1.79E-10
2.0	J	Control Panel 923-1 IA, SA, SW, RBCCW, TBCCW, ICMU	CONTROL ROOM	3.71E-04	1	1	2.65E-05	9.83E-09
2.0	K	Control Panel 923-2 Loss of 345kV Swyd Conn	CONTROL ROOM	4.95E-04	1	1	7.10E-09	3.52E-12
2.0	L	Control Panel 901-P18 Loss of 138kV Swyd Conn	CONTROL ROOM	4.95E-04	1	1	2.56E-07	1.27E-10
2.0	M	Control Panel 902-18 FW	CONTROL ROOM	2.47E-04	1	1	6.27E-06	1.55E-09
2.0	N	Severe Fire w/evacuation	CONTROL ROOM	1.93E-02	0.0034	0.1	1.00E+00	6.56E-06
6.1	B	250VDC MCC #3	U3 BATTERY CHARGER ROOM	4.21E-04	1	1	3.41E-09	1.44E-12

Table 4-16
Unit 3 Unscreened Fire Compartment Analysis Details

Fire Compartment	Scen	Scenario Description	Fire Compartment Description	Ignition Frequency	NSP	SF	CCDP	CDF
6.1	C	125VDC Dist'n Panel Reserve Bus #3	U3 BATTERY CHARGER ROOM	4.21E-04	1	1	5.67E-07	2.39E-10
6.1	D	125VDC Battery Charger 3	U3 BATTERY CHARGER ROOM	1.12E-03	1	1	7.39E-07	8.27E-10
6.1	E	250VDC Battery Charger 3	U3 BATTERY CHARGER ROOM	1.12E-03	1	1	3.41E-09	3.82E-12
6.2	AA	Self-Initiated Cable Fire - Trays 671B/672B - %TP	AUX ELEC EQUIP ROOM	3.07E-06	1	1	1.90E-03	5.83E-09
6.2	B	No Loss of System Function - %TT	AUX ELEC EQUIP ROOM	9.66E-03	1	1	3.41E-09	3.29E-11
6.2	BB	Self-Initiated Cable Fire - Trays 655B/677B - %TP	AUX ELEC EQUIP ROOM	3.07E-06	1	1	1.57E-03	4.81E-09
6.2	C	Loss of Main Condenser - %TC	AUX ELEC EQUIP ROOM	8.16E-04	1	1	6.11E-07	4.99E-10
6.2	D	Loss of Offsite Power - %LOOP	AUX ELEC EQUIP ROOM	1.22E-03	1	1	3.17E-06	3.87E-09
6.2	E	Propagation to Trays 225 & 249 - %TC	AUX ELEC EQUIP ROOM	8.16E-04	1	1	3.78E-05	3.09E-08
6.2	F	ECCS Div I & Spurious ADS - %TP	AUX ELEC EQUIP ROOM	4.08E-04	1	1	2.66E-04	1.09E-07
6.2	G	ECCS Div II - %TT	AUX ELEC EQUIP ROOM	4.08E-04	1	1	3.41E-09	1.39E-12
6.2	H	HPCI - %TT	AUX ELEC EQUIP ROOM	2.72E-04	1	1	1.72E-07	4.68E-11
6.2	I	LPCI Div I - %TT	AUX ELEC EQUIP ROOM	1.36E-04	1	1	1.33E-06	1.81E-10

Table 4-16
Unit 3 Unscreened Fire Compartment Analysis Details

Fire Compartment	Scen	Scenario Description	Fire Compartment Description	Ignition Frequency	NSP	SF	CCDP	CDF
6.2	J	LPCI Div II - %TT	AUX ELEC EQUIP ROOM	1.36E-04	1	1	1.33E-06	1.81E-10
6.2	K	Essential Services Bus - %TC	AUX ELEC EQUIP ROOM	1.90E-03	1	1	6.05E-07	1.15E-09
6.2	L	Instrument Bus - %TT	AUX ELEC EQUIP ROOM	2.72E-04	1	1	6.05E-07	1.65E-10
6.2	M	Panel 2TB-71 - %TC	AUX ELEC EQUIP ROOM	2.72E-04	1	1	3.41E-09	9.28E-13
6.2	N	Propagation to Trays 257B & 258B - %TC	AUX ELEC EQUIP ROOM	4.08E-04	1	1	2.38E-06	9.70E-10
6.2	O	Propagation to Trays 245B, 246B & 247 - %TC	AUX ELEC EQUIP ROOM	9.52E-04	1	1	7.65E-08	7.28E-11
6.2	P	Panel 902-51 - %TC	AUX ELEC EQUIP ROOM	1.36E-04	1	1	6.11E-07	8.31E-11
6.2	Q	Propagation to Trays 650B & 671B - %TC	AUX ELEC EQUIP ROOM	4.08E-04	1	1	2.96E-06	1.21E-09
6.2	R	Propagation to Tray 655B - %TC	AUX ELEC EQUIP ROOM	8.16E-04	1	1	1.49E-04	1.22E-07
6.2	S	Propagation to Tray 229 - %TC	AUX ELEC EQUIP ROOM	9.52E-04	1	1	1.24E-04	1.18E-07
6.2	T	Propagation to Trays 650B & 671B - %TP	AUX ELEC EQUIP ROOM	4.08E-04	1	1	1.60E-03	6.51E-07
6.2	U	Propagation to Tray 655B - %TP	AUX ELEC EQUIP ROOM	8.16E-04	1	1	1.57E-03	1.28E-06
6.2	V	Bounding Fire - %LOOP	AUX ELEC EQUIP ROOM	2.21E-04	1	1	9.96E-05	2.20E-08
6.2	W	Self-Initiated Cable Fire - Tray 651B - %TP	AUX ELEC EQUIP ROOM	3.07E-06	1	1	1.59E-03	4.87E-09

Table 4-16
Unit 3 Unscreened Fire Compartment Analysis Details

Fire Compartment	Scen	Scenario Description	Fire Compartment Description	Ignition Frequency	NSP	SF	CCDP	CDF
6.2	X	Self-Initiated Cable Fire - Tray 653B - %TP	AUX ELEC EQUIP ROOM	3.07E-06	1	1	1.99E-04	6.10E-10
6.2	Y	Self-Initiated Cable Fire - Tray 659B/M1/T - %TP	AUX ELEC EQUIP ROOM	3.07E-06	1	1	2.85E-02	8.76E-08
6.2	Z	Self-Initiated Cable Fire - Tray 660M1/T - %TP	AUX ELEC EQUIP ROOM	3.07E-06	1	1	2.85E-02	8.76E-08
7.0.A.1	B	250VDC MCC #2	U2 BATTERY ROOM	3.28E-04	1	1	7.40E-07	2.43E-10
7.0.A.1	C	125VDC Dist'n Panel Main Bus	U2 BATTERY ROOM	3.28E-04	1	1	7.41E-06	2.43E-09
7.0.A.1	D	125VDC Dist'n Panel Reserve Bus	U2 BATTERY ROOM	3.28E-04	1	1	3.41E-09	1.12E-12
7.0.A.1	E	125VDC Battery Chargers 2 & 2A	U2 BATTERY ROOM	1.78E-03	1	1	7.41E-06	1.32E-08
7.0.A.1	F	250VDC Battery Chargers 2 & 2/3	U2 BATTERY ROOM	1.78E-03	1	1	3.41E-09	6.07E-12
8.2.1.B	A	Screening Scenario	U3 COND. PP AREA	4.16E-03	1	1	1.16E-04	4.85E-07
8.2.4	AA	Tray CT29M5 - %TC	U3 CABLE TUNNEL	2.35E-05	1	0.8	8.73E-06	1.64E-10
8.2.4	B	Tray CT7T - %TI	U3 CABLE TUNNEL	3.36E-05	1	0.8	3.77E-06	1.01E-10
8.2.4	BB	Tray CT29B - %TC	U3 CABLE TUNNEL	2.35E-05	1	0.8	3.41E-09	6.41E-14
8.2.4	C	Tray CT7M1 - %TP	U3 CABLE TUNNEL	3.36E-05	1	0.8	8.23E-04	2.21E-08
8.2.4	CC	Bounding Fire - Tray Stack CT7 - %TP	U3 CABLE TUNNEL	2.35E-04	0.02	0.2	1.00E+00	9.40E-07

Table 4-16
Unit 3 Unscreened Fire Compartment Analysis Details

Fire Compartment	Scen	Scenario Description	Fire Compartment Description	Ignition Frequency	NSP	SF	CCDP	CDF
8.2.4	D	Tray CT7M2 - %TP	U3 CABLE TUNNEL	3.36E-05	1	0.8	2.87E-02	7.71E-07
8.2.4	DD	Bounding Fire - Tray Stack CT7 - %TC	U3 CABLE TUNNEL	2.35E-04	0.02	0.2	2.58E-02	2.42E-08
8.2.4	E	Tray CT7M3 - %TC	U3 CABLE TUNNEL	3.36E-05	1	0.8	2.42E-05	6.52E-10
8.2.4	EE	Bounding Fire - Tray Stack CT8 - %TP	U3 CABLE TUNNEL	2.35E-04	0.02	0.2	2.34E-01	2.20E-07
8.2.4	F	Tray CT7M4 - %TC	U3 CABLE TUNNEL	3.36E-05	1	0.8	5.63E-05	1.51E-09
8.2.4	FF	Bounding Fire - Tray Stack CT8 - %TC	U3 CABLE TUNNEL	2.35E-04	0.02	0.2	3.11E-03	2.92E-09
8.2.4	G	Tray CT7M5 - %TC	U3 CABLE TUNNEL	3.36E-05	1	0.8	1.37E-04	3.68E-09
8.2.4	GG	Bounding Fire - Tray Stack CT17 - %TI	U3 CABLE TUNNEL	2.35E-04	0.02	0.2	1.26E-02	1.18E-08
8.2.4	H	Tray CT7B - %TC	U3 CABLE TUNNEL	3.36E-05	1	0.8	3.41E-09	9.17E-14
8.2.4	HH	Bounding Fire - Tray Stack CT17 - %TC	U3 CABLE TUNNEL	2.35E-04	0.02	0.2	2.34E-03	2.20E-09
8.2.4	I	Tray CT8T - %TC	U3 CABLE TUNNEL	3.36E-05	1	0.8	1.46E-07	3.92E-12
8.2.4	II	Bounding Fire - Tray Stack CT18 - %TP	U3 CABLE TUNNEL	1.18E-04	0.02	0.2	6.66E-03	3.14E-09
8.2.4	J	Tray CT8M1 - %TC	U3 CABLE TUNNEL	3.36E-05	1	0.8	5.04E-09	1.36E-13
8.2.4	JJ	Bounding Fire - Tray Stack CT18 - %TC	U3 CABLE TUNNEL	1.18E-04	0.02	0.2	2.95E-04	1.39E-10
8.2.4	K	Tray CT8M2 - %TC	U3 CABLE TUNNEL	3.36E-05	1	0.8	1.11E-05	2.98E-10

Table 4-16
Unit 3 Unscreened Fire Compartment Analysis Details

Fire Compartment	Scen	Scenario Description	Fire Compartment Description	Ignition Frequency	NSP	SF	CCDP	CDF
8.2.4	KK	Bounding Fire - Tray Stack CT29/CT38 - %TC	U3 CABLE TUNNEL	1.18E-04	0.02	0.2	1.85E-03	8.73E-10
8.2.4	L	Tray CT8M3 - %TC	U3 CABLE TUNNEL	3.36E-05	1	0.8	1.42E-04	3.81E-09
8.2.4	M	Tray CT8M4 - %TC	U3 CABLE TUNNEL	3.36E-05	1	0.8	8.29E-05	2.23E-09
8.2.4	N	Tray CT8M5 - %TC	U3 CABLE TUNNEL	3.36E-05	1	0.8	2.65E-07	7.13E-12
8.2.4	O	Tray CT8B - %TP	U3 CABLE TUNNEL	3.36E-05	1	0.8	1.06E-03	2.85E-08
8.2.4	P	Tray CT17M3 - %TI	U3 CABLE TUNNEL	5.88E-05	1	0.8	7.37E-04	3.47E-08
8.2.4	Q	Tray CT17M4 - %TC	U3 CABLE TUNNEL	5.88E-05	1	0.8	5.63E-05	2.65E-09
8.2.4	R	Tray CT17M5 - %TC	U3 CABLE TUNNEL	5.88E-05	1	0.8	1.37E-04	6.44E-09
8.2.4	S	Tray CT17B - %TC	U3 CABLE TUNNEL	5.88E-05	1	0.8	7.85E-08	3.69E-12
8.2.4	T	Tray CT18M3 - %TC	U3 CABLE TUNNEL	2.94E-05	1	0.8	1.42E-04	3.34E-09
8.2.4	U	Tray CT18M4 - %TC	U3 CABLE TUNNEL	2.94E-05	1	0.8	2.73E-06	6.42E-11
8.2.4	V	Tray CT18M5 - %TC	U3 CABLE TUNNEL	2.94E-05	1	0.8	2.65E-07	6.24E-12
8.2.4	W	Tray CT18B - %TP	U3 CABLE TUNNEL	2.94E-05	1	0.8	1.06E-03	2.50E-08
8.2.4	X	Tray CT38 - %TC	U3 CABLE TUNNEL	2.35E-05	1	0.8	1.22E-04	2.30E-09
8.2.4	Y	Tray CT29M3 - %TC	U3 CABLE TUNNEL	2.35E-05	1	0.8	9.38E-06	1.76E-10
8.2.4	Z	Tray CT29M4 - %TC	U3 CABLE TUNNEL	2.35E-05	1	0.8	5.46E-05	1.03E-09
8.2.5.A	B	Switchgear Area Fire	U2 NORTH TRACKWAY/SWGR AREA	8.78E-03	1	1	1.66E-05	1.46E-07
8.2.5.A	C	RFP Area - Large, No Suppression	U2 NORTH TRACKWAY/SWGR AREA	9.25E-03	0.02	0.18	6.91E-05	2.30E-09

Table 4-16
Unit 3 Unscreened Fire Compartment Analysis Details

Fire Compartment	Scen	Scenario Description	Fire Compartment Description	Ignition Frequency	NSP	SF	CCDP	CDF
8.2.5.A	D	RFP Area - All with Suppression	U2 NORTH TRACKWAY/SWGR AREA	9.25E-03	1	1	7.50E-06	6.93E-08
8.2.5.A	E	Self-Initiated Cable Fire - %TP	U2 NORTH TRACKWAY/SWGR AREA	2.68E-04	1	1	1.03E-03	2.76E-07
8.2.5.C	B	Instrument Air Compressor 2B Large Fire	2/3 TB CORRIDOR	5.63E-04	1	0.18	2.61E-03	2.64E-07
8.2.5.C	C	Resin Air Compressors	2/3 TB CORRIDOR	1.13E-03	1	1	7.40E-06	8.36E-09
8.2.5.C	D	MCC 27-1	2/3 TB CORRIDOR	4.09E-04	1	1	7.38E-06	3.02E-09
8.2.5.C	E	MCC 37-1	2/3 TB CORRIDOR	4.09E-04	1	1	7.40E-06	3.02E-09
8.2.5.C	F	Cond Demin Panels 2252-11 & 2253-11	2/3 TB CORRIDOR	8.18E-04	1	1	7.40E-06	6.05E-09
8.2.5.C	H	EHC Pumps	2/3 TB CORRIDOR	7.80E-04	1	1	8.18E-06	6.38E-09
8.2.5.C	I	Instrument Air Compressor 2B Small Fire	2/3 TB CORRIDOR	5.63E-04	1	0.82	7.39E-06	3.41E-09
8.2.5.C	J	Self-Initiated Cable Fire - %TP	2/3 TB CORRIDOR	2.10E-04	1	1	2.58E-03	5.42E-07
8.2.5.E	B	Bus 31	U3 WEST CORRIDOR AND TRACKWAY	1.13E-03	1	1	8.69E-06	9.82E-09
8.2.5.E	C	Bus 32	U3 WEST CORRIDOR AND TRACKWAY	1.13E-03	1	1	9.25E-06	1.04E-08
8.2.5.E	D	Swgr 35 and Xfmr	U3 WEST CORRIDOR AND TRACKWAY	1.13E-03	1	1	5.50E-05	6.22E-08

Table 4-16
Unit 3 Unscreened Fire Compartment Analysis Details

Fire Compartment	Scen	Scenario Description	Fire Compartment Description	Ignition Frequency	NSP	SF	CCDP	CDF
8.2.5.E	E	Swgr 36 and Xfmr	U3 WEST CORRIDOR AND TRACKWAY	1.13E-03	1	1	5.50E-05	6.22E-08
8.2.5.E	F	MCC 36-1 and Misc Cond Pumps	U3 WEST CORRIDOR AND TRACKWAY	1.17E-03	1	1	7.77E-04	9.09E-07
8.2.5.E	G	Self-Initiated Cable Fire - %TP	U3 WEST CORRIDOR AND TRACKWAY	1.56E-04	1	1	2.99E-03	4.66E-07
8.2.5.E	H	Self-Initiated Cable Fire - %TI	U3 WEST CORRIDOR AND TRACKWAY	1.56E-04	1	1	7.41E-05	1.16E-08
8.2.5.E	I	RFPs & Compressors - Large Fire w/o Supp	U3 WEST CORRIDOR AND TRACKWAY	6.53E-03	0.02	0.18	8.11E-04	1.91E-08
8.2.5.E	J	RFPs - Large w/Supp & Small w/o Supp	U3 WEST CORRIDOR AND TRACKWAY	4.19E-03	1	1	8.11E-04	3.40E-06
8.2.5.E	K	Compressors - Large w/Supp & Small w/o Supp	U3 WEST CORRIDOR AND TRACKWAY	2.34E-03	1	1	8.11E-04	1.90E-06
8.2.6.A	B	Bounding Fire	CONTROL ROOM BKUP VENTILATION	1.70E-02	1	1	1.41E-05	2.39E-07
8.2.6.A	C	Self-Initiated Cable Fire - %TP	CONTROL ROOM BKUP VENTILATION	2.13E-04	1	1	1.03E-03	2.20E-07
8.2.6.C	C	MCCs 35-1 and 39-2	U2/3 SBGT & TBCCW HX	1.04E-03	1	1	1.41E-04	1.47E-07
8.2.6.C	D	480V Swgr 27	U2/3 SBGT & TBCCW HX	1.79E-03	1	1	7.38E-06	1.32E-08
8.2.6.C	E	480V Swgr 37	U2/3 SBGT & TBCCW HX	1.79E-03	1	1	7.38E-06	1.32E-08
8.2.6.C	F	2/3 Sparging Air Compressors	U2/3 SBGT & TBCCW HX	1.04E-03	1	1	8.73E-06	9.08E-09
8.2.6.C	G	Unit 2 TBCCW Pumps	U2/3 SBGT & TBCCW HX	3.00E-04	1	1	7.38E-06	2.22E-09

Table 4-16
Unit 3 Unscreened Fire Compartment Analysis Details

Fire Compartment	Scen	Scenario Description	Fire Compartment Description	Ignition Frequency	NSP	SF	CCDP	CDF
8.2.6.C	H	Unit 3 TBCCW Pumps	U2/3 SBGT & TBCCW HX	3.00E-04	1	1	8.73E-06	2.62E-09
8.2.6.C	I	Self-Initiated Cable Fire - Severe - %TP	U2/3 SBGT & TBCCW HX	4.20E-05	1	0.2	2.71E-02	2.28E-07
8.2.6.C	J	Self-Initiated Cable Fire - Non-Severe - %TP	U2/3 SBGT & TBCCW HX	4.20E-05	1	0.8	3.47E-03	1.17E-07
8.2.6.D	B	Self-Initiated Cable Fire - Severe - %TP	U3 MEZZANINE FLOOR	4.70E-04	1	0.2	6.19E-03	5.82E-07
8.2.6.D	C	Self-Initiated Cable Fire - %TC	U3 MEZZANINE FLOOR	4.70E-04	1	1	2.91E-04	1.37E-07
8.2.6.D	D	Self-Initiated Cable Fire - Non-Severe - %TP	U3 MEZZANINE FLOOR	4.70E-04	1	0.8	1.89E-04	7.11E-08
8.2.6.E	B	DC Panel Fire	U3 MEZZANINE FLOOR	1.19E-03	1	1	2.26E-03	2.69E-06
8.2.6.E	C	MCC 38-2/38-3 Fire	U3 MEZZANINE FLOOR	1.19E-03	1	1	1.74E-05	2.08E-08
8.2.6.E	D	Bounding Switchgear Fire	U3 MEZZANINE FLOOR	3.29E-03	1	1	2.23E-04	7.33E-07
8.2.7	A	Screening Scenario	VENT ROOM OVER NE SWGR	7.34E-03	1	1	3.26E-05	2.39E-07
9.0.B	A	Screening Scenario	U3 D/G	2.96E-02	1	1	7.40E-06	2.19E-07
					TOTAL UNIT 3 CDF			**3.08E-5**

4.7.3 *Analysis of Multi-Compartment Fires*

The multi-compartment analysis (MCA) evaluated the risks associated with fires involving more than one compartment. The analysis investigated the potential for a fire starting in any single compartment spreading to or damaging equipment in an adjacent compartment. The analysis used a graded screening approach that considered the potential for severe fires that challenge the integrity of barriers, the frequency of occurrence, and, if necessary, the challenge to plant safe shutdown capability assuming loss of equipment in both compartments. The probability of failure of two successive barriers was assumed to be very low and, therefore, not considered in this analysis.

4.7.3.1 Methodology

The MCA methodology used for the Dresden analysis applies the concepts and processes presented in the EPRI Fire PRA Implementation Guide. The analysis steps are summarized below. The analysis focuses on the physical boundaries that separate the various Dresden Fire Areas. It is recognized that many of these areas also have virtual boundaries. The MCA does not address these virtual boundaries. The performance of these virtual barriers are integrated into the individual compartment assessments.

1. Identify the entire population of fire compartments used in the Dresden Fire PRA. Determine the fire detection and suppression features that are present in each fire compartment. Preaction and deluge type automatic fire suppression systems which require a supporting fire detection system are credited only as having an automatic fire suppression system. Separate credit for an automatic detection system is provided only if that system is independent of the suppression system.
2. For each fire compartment, identify the adjacent fire compartments and the general orientation of the adjacent space (up, down, horizontal). Boundaries with the outside (yard areas) areas need not be identified.
3. Screen potential multi-compartment scenarios involving downward propagation pathways.
4. Screen potential multi-compartment scenarios involving the drywell as the initiating (exposing) fire compartment. This is because the drywell is inert which precludes the occurrence of a significant fire event.
5. Screen potential multi-compartment scenarios wherein the exposing fire compartment does not contain any credited Fire PRA equipment. The potential consequences of fire propagation to an adjacent area is bounded by the existing analyses for the 'adjacent' compartment.
6. Screen potential multi-compartment scenarios wherein the exposing fire compartment does not contain any significant ignition sources. The occurrence of a multi-compartment scenario requires the development of a significant fire. The lack

of a suitable ignition source or concentration of combustible material would preclude the occurrence of a multi-compartment scenario.

7. For those fire compartments with area-wide automatic fire suppression system coverage, calculate the multi-compartment scenario initiating event frequency. This frequency is calculated based on the total fire ignition frequency for the entire fire compartment, a severity factor of 0.20, the automatic fire suppression system failure probability, and the barrier failure probability. Multi-compartment fire scenarios with frequencies below 1.0E-6/yr. based on these four factors are screened from further consideration. Compartments which are unscreened are evaluated further as described below.
8. For those fire compartments with an area-wide fire detection system, calculate the multi-compartment scenario initiating event frequency, or modify the value from step 7. The availability of an automatic fire detection system would alert the fire brigade. The probability that the fire brigade fails to prevent a multi-compartment scenario is taken to be 0.10. This frequency is calculated based on the total fire ignition frequency for the entire fire compartment, a severity factor of 0.20, the fire brigade failure probability, and the barrier failure probability. Multi-compartment fire scenarios with frequencies below 1.0E-6/yr. based on these four factors are screened from further consideration.
9. For those cases where the initiating (exposing) compartment has either an automatic fire suppression or detection system, and the adjacent (exposed) compartment has an independent detection or suppression system, respectively, credit for that redundant system can be provided. Treatment of this case is identical to that described in step 8, above.
10. Instances where fire modeling was performed for the Fire PRA analyses, and those analyses show that a damaging compartment wide hot gas layer condition does not occur can be screened. This is consistent with the methodology presented in the EPRI Fire PRA Implementation Guide.

The final screening step involves the treatment of those cases where neither an automatic fire suppression system or detection system is available. The multi-compartment scenario initiating event frequency is calculated based on the total fire ignition frequency for the entire fire compartment, a severity factor of 0.20, and the barrier failure probability. Multi-compartment fire scenarios with initiating event frequencies below 1.0E-6/yr. are screened from further consideration. Compartments which are unscreened following this step require detailed examination for potential risk contribution.

The barrier failure probabilities provided used in the MCA are provided in Table 4-17.

Table 4-17
Barrier Failure Probabilities

Barrier Type	Failure Probability (Ref. 4-5)
Type 1 – fire, security, and water tight doors	7.4E-03
Type 2 – fire and ventilation dampers	2.7E-03
Type 3 – penetration seals, fire walls	1.2E-03

4.7.3.2 Multi-Compartment Analysis Results

Table 4-18 summarizes the multi-compartment analysis results for each of the Dresden fire compartments. The upgraded fire risk analysis concluded that multi-compartment scenarios are not a risk significant concern.

Table 4-18
Multi-Compartment Analysis Results

No.	Exposing Fire Compartment	Compartment Description	Multi-Compartment Analysis Result
1	1.1.1.1	U3 TORUS BASEMENT	Screened - Note 5
2	1.1.1.2	U3 RX BLDG GROUND FLOOR	Screened - Note 7
3	1.1.1.3	U3 SECOND FLOOR RX BLDG	Screened - Note 7
4	1.1.1.4	U3 RX BLDG SWITCHGEAR AREA	Screened - Note 7
5	1.1.1.5.A	U3 ISCO FLOOR	Screened - Note 5
6	1.1.1.5.B	U3 ISCO PIPE CHASE	Screened - Note 5
7	1.1.1.5.C	U3 ISCO PIPE CHASE	Screened - Note 5
8	1.1.1.5.D	U3 SKIMMER SURGE TANK ROOM	Screened - Note 5
9	1.1.1.6	U3 REFUEL FLOOR	Screened - Note 2
10	1.1.2.1	U2 TORUS BASEMENT	Screened - Note 5
11	1.1.2.2	U2 RX BLDG GROUND FLOOR	Screened - Note 7
12	1.1.2.3	U2 SECOND FLOOR RX BLDG	Screened - Note 7
13	1.1.2.4	U2 RX BLDG SWITCHGEAR AREA	Screened - Note 7
14	1.1.2.5.A	U2 ISCO FLOOR	Screened - Note 7
15	1.1.2.5.B	U2 ISCO PIPE CHASE	Screened - Note 7

Table 4-18
Multi-Compartment Analysis Results

No.	Exposing Fire Compartment	Compartment Description	Multi-Compartment Analysis Result
16	1.1.2.5.C	U2 ISCO PIPE CHASE	Screened - Note 7
17	1.1.2.5.D	U2 SKIMMER SURGE TANK ROOM	Screened - Note 7
18	1.1.2.6	U2 REFUEL FLOOR	Screened - Note 2
19	1.2.1	U3 DRYWELL	Screened - Note 3
20	1.2.2	U2 DRYWELL	Screened - Note 3
21	1.3.1	U3 SHUTDOWN COOLING PUMP ROOM	Screened - Note 6
22	1.3.2	U2 SHUTDOWN COOLING PUMP ROOM	Screened - Note 6
23	1.4.1	U3 TIP ROOM	Screened - Note 5
24	2.0	CONTROL ROOM	Screened - Note 7
25	6.1	U3 BATTERY CHARGER ROOM	Screened - Note 6
26	6.2	AUX ELEC EQUIP ROOM	Screened - Note 6
27	7.0.A.1	U2 BATTERY ROOM	Screened - Note 6
28	7.0.A.2	125 VDC BATT. RM.	Screened - Note 6
29	7.0.A.3	250 VDC BATT. RM.	Screened - Note 6
30	7.0.B	U3 STATION BATT. ROOM	Screened - Note 2
31	8.1	CLEAN/DIRTY OIL ROOM	Screened - Note 5
32	8.2.1.A	U2 COND. PP AREA	Screened - Note 6
33	8.2.1.B	U3 COND. PP AREA	Screened - Note 6
34	8.2.2.A	U2 CRD PUMP ROOM	Screened - Note 6
35	8.2.2.B	U3 CRD PUMP ROOM	Screened - Note 6
36	8.2.4	U3 CABLE TUNNEL	Screened - Note 6
37	8.2.5.A	U2 NORTH TRACKWAY/SWGR AREA	Note 8
38	8.2.5.B	U2 LP HEATER BAY	Screened - Note 6
39	8.2.5.C	2/3 TB CORRIDOR	Note 8
40	8.2.5.D	U3 LP HEATER BAY	Screened - Note 6
41	8.2.5.E	U3 WEST CORRIDOR AND TRACKWAY	Note 8
42	8.2.6.A	CONTROL ROOM BKUP VENTILATION	Screened - Note 7
43	8.2.6.B	U2 MEZZANINE	Screened - Note 6
44	8.2.6.C	U2/3 SBGT & TBCCW HX	Screened - Note 6
45	8.2.6.D	U3 MEZZANINE FLOOR	Screened - Note 6

Table 4-18
Multi-Compartment Analysis Results

No.	Exposing Fire Compartment	Compartment Description	Multi-Compartment Analysis Result
46	8.2.6.E	U3 MEZZANINE FLOOR	Screened - Note 7
47	8.2.7	VENT ROOM OVER NE SWGR	Partially Screened - Notes 6, 8
48	8.2.8.A	U2/3 TURBINE OPERATING FLOOR	Screened - Notes 4, 6
49	8.2.8.B	VENT FLOOR	Screened - Note 4
50	8.2.8.C	VENT FLOOR	Screened - Note 6
51	8.2.8.D	VENT FLOOR	Screened - Note 4
52	9.0.A	U2 D/G	Screened - Notes 4, 6
53	9.0.B	U3 D/G	Screened - Notes 4, 6
54	9.0.C	U2/3 DG	Screened - Note 6
55	11.1.1	U3 LPCI PP ROOM DIV II	Screened - Note 6
56	11.1.2	U3 LPCI PP ROOM DIV I	Screened - Note 6
57	11.1.3	U3 HPCI	Screened - Note 6
58	11.2.1	U2 LPCI PP ROOM DIV II	Screened - Note 6
59	11.2.2	U2 LPCI PP ROOM DIV I	Screened - Note 6
60	11.2.3	U2 HPCI	Screened - Note 6
61	11.3	CRIBHOUSE UPPER	Screened - Note 1
62	14.1	RADWASTE BLDG.	Screened - Note 4
63	14.2.A	U2 STEAM JET AIR EJECTOR ROOM	Screened - Note 4
64	14.2.B	U2 OFF GAS RECOMBINER	Screened - Note 5
65	14.2.C	U2 OFF GAS CONDENSER	Screened - Note 5
66	14.3.A	U3 STEAM JET AIR EJECTOR ROOM	Screened - Note 4
67	14.3.B	U3 OFF GAS RECOMBINER	Screened - Note 5
68	14.3.C	U3 OFF GAS CONDENSER	Screened - Note 5
69	14.4	OFF GAS FILTER BUILDING	Screened - Note 1
70	14.5	RADWASTE BLDG.	Screened - Note 4
71	14.6	MAX RECYCLE	Screened - Note 4
72	18.1.1	U3 MPT	Screened - Note 1
73	18.1.2	U2 MPT	Screened - Note 1
74	18.2.1	U3 UAT	Screened - Note 1
75	18.2.2	U2 UAT	Screened - Note 1

Table 4-18
Multi-Compartment Analysis Results

No.	Exposing Fire Compartment	Compartment Description	Multi-Compartment Analysis Result
76	18.3.1	U3 RAT	Screened - Note 1
77	18.3.2	U2 RAT	Screened - Note 1
78	18.4	AUX BOILER HOUSE	Screened - Note 1
79	18.6	SBO BATT RM	Screened - Note 1
80	18.7.1	ISCO PP HOUSE NORTH	Screened - Note 1
81	18.7.2	ISCO PP HOUSE SOUTH	Screened - Note 1
82	Outdoor - 1	H2 STORAGE FACILITY	Screened - Note 1
83	Outdoor - 2	UNIT 1	Screened - Note 5

Notes:

1. Compartment screened based on physical location, configuration, and/or spatial separation from other compartments.
2. Compartment screened based on exposed adjacent compartments being located below.
3. Compartment screened based on inert atmosphere.
4. Compartment screened based on the scope of equipment and cables potentially impacted given a postulated multi-compartment scenario is bounded by the analysis of the exposing compartment alone.
5. Compartment screened based on lack of an ignition source of sufficient magnitude to cause critical HGL conditions.
6. Compartment screened based on calculated initiator frequency being below 1.0E-6/yr. This initiator frequency included consideration of severity factors, automatic suppression system failure if applicable, and barrier failure probability.
7. Compartment screened based on results of fire modeling analyses performed as part of individual compartment analyses which showed no HGL formation.
8. Compartment has one or more multi-compartment scenarios that initially could not be screened when viewed from an area-wide basis. Additional reviews were performed which analyzed specific scenarios/ignition sources and showed that the cumulative multi-compartment scenario initiating event frequency remained below the 1.0E-06/yr. screening criteria.

4.7.4 Analysis of Control Room Fires

This section documents the analysis of Control Room fires at Dresden. A fire in the Control Room has the potential to result in risk in two distinctive ways. In the most severe case, a fire can develop and fail to be suppressed before sufficient concentrations of smoke develop, thus requiring abandonment of the Control Room. In this case the remote shutdown capability of the plant would be used for safe shutdown.

In less severe cases, a fire is successfully suppressed before abandonment of the Control Room becomes necessary. In these cases, fire may have damaged controls in one or more cabinets. Operators may attempt to shut down the reactor using the remaining capability that can be operated from the Control Room, or using the remote shutdown capability of the plant.

The analysis of these scenarios, which followed the guidelines in Appendix M of the Fire PRA Implementation Guide (Ref. 4-5), is described in the following section.

4.7.4.1 Analysis

The main control room (Fire Zone 2.0) is located at the 534 foot elevation of the turbine building. From the floor slab to the ceiling slab, the compartment height is approximately 15 feet. However, throughout most of the zone a suspended ceiling reduces the compartment height to approximately 12 ft. A locker room and kitchen are included in this fire zone. However, a one-hour fire barrier separates the kitchen from the main control room. Three-hour rated concrete block walls separate the locker room from the main control room. The floor area of the main control room (not including the locker room and the kitchen) measures approximately 4900 ft^2. The ventilation system for the main control room provides 13,600 cubic feet per minute (approximately 13 room changes per hour). The ventilation system ducts are provided with smoke detectors which, upon detection of smoke, automatically switch to a smoke purge operating mode. In addition, the designed airflow pattern is such that air is exhausted from within the MCB area which tends to minimize the impact of MCB fires on control room habitability.

The updating of the original Fire IPEEE analysis for the Control Room involved a general modification of the overall analysis approach to simplify the analysis. It also resolved an apparent non-conservatism in the original analysis wherein the HVAC system operation was credited to preclude control room abandonment scenarios for certain postulated cabinet fires. The principal steps of the analysis were:

A. Identify the plant system functions associated with each of the Control Room cabinets;

B. Apportion the Control Room ignition frequency to individual Control Room cabinets;

C. Determine the scope of plant system functions impacted by postulated Control Room fires that are successfully suppressed and did not require abandoning of the Control Room;

D. Quantify CDF contribution for unscreened fire scenarios that did not require abandoning the Control Room; and

E. Quantify CDF contribution for scenarios requiring abandoning the Control Room.

These steps are discussed in detail in Section 4.7.4.2.2. Section 4.7.4.1.1 presents the key assumptions for the analysis.

4.7.4.1.1 Key Assumptions and Bases

The following are key assumptions made in the analysis:

A. **Assumption:** Abandonment of the Control Room was assumed to occur at the time smoke visibly obscured the control panels.

A1. **Basis:** General discussions with operations staff indicated a preference to remain in the Control Room to continue safe shutdown activities. Sandia National Laboratories (SNL's) tests (Ref. 4-14 and 4-15) indicate that eventually smoke will descend to a level that affects operators' ability to perform their tasks (page 3 of Ref. 4-13 (Executive Summary)). Even if an air breathing apparatus is used, smoke will accumulate in a sufficient concentration to affect the ability to see controls on the MCB. Temperatures in the Control Room will remain below 150°F throughout this time (page 2 of Ref. 4-14, Executive Summary), thereby not causing widespread effects on solid state controls.

B. **Assumption:** The abandonment time was assumed to be similar to representative SNL cabinet fire tests as reported in the Fire PRA Implementation Guide (Ref. 4-5).

B1. **Basis:** The free volume of the Control Room is about the same size as the free volume of the SNL test facility. The smoke ejection rate of the HVAC system is about twice the highest rate used in the SNL tests. In addition, the ventilation system ducts are provided with smoke detectors. In the event of a fire, smoke detectors automatically switch the air handling unit to the smoke purge mode. During this mode 100% outdoor air is provided, recirculation of smoke into occupied areas is prevented, and 100% of the return air is exhausted to the outdoors. Therefore, the abandonment time is the same or longer than the representative SNL tests.

C. **Assumption:** Each cabinet was assumed to contain sufficient cable or combustible loading so that enough smoke could be generated to cause Control Room abandonment if suppression was not successful.

C1. **Basis:** This assumption is bounding because the representative SNL tests used cabinets with combustible loads that appeared larger than those observed in the Dresden Control Room cabinets including the MCB. Those Dresden cabinets with light cable loading may not be capable of generating enough smoke to cause abandonment.

D. **Assumption:** Postulated cabinet fires can be screened as non-risk significant if the fire is suppressed prior to causing loss of Control Room habitability, and the functional loss of the cabinets and associated circuits does not impact any systems credited in the fire PRA or does not cause a plant trip.

D1. **Basis:** Multiple safe shutdown paths are addressed in the fire PRA model. The availability of all paths would lead to a CCDP of less than 1.0E-6. Combining this with the fire ignition frequency of less than 1.93E-2 for any fire in the Control Room results in a CDF contribution of approximately 2E-8/yr.

E. **Assumption:** Fire propagation to an adjacent cabinet is prevented if the fire is suppressed within the time frame associated with Control Room abandonment and there is a double wall and intervening air gap that separates adjacent cabinets.

E1. **Basis:** SNL cabinet fire tests indicate damage will not occur to meters, relays, and switches in adjacent cabinets when a double wall separates them (Ref. 4-16). Damage could occur to solid state equipment because temperatures sometimes exceed 150°F. However, SNL tests 21 through 24 of Ref. 4-14 indicate that the smoke obscuration of the enclosure occurs at or before temperatures in the adjacent cabinet reach 150°F.

F. **Assumption:** Fire damage to the RPS circuits result in a scram.

F1. **Basis:** All RPS circuits are normally energized, thereby requiring a hot short to prevent a single circuit from de-energizing. Therefore, RPS redundancy requires multiple hot shorts to preclude reactor scram. Therefore, fire-induced Anticipated Transient Without Scram (ATWS) is considered unlikely with negligible contribution to fire risk.

4.7.4.1.2 Analysis Summary

The examination of the Dresden main control room panels was performed using the plant simulator and the actual control room itself. The simulator was used to obtain information necessary to establish the scope of postulated plant system failures that should be considered given a panel fire. The determination of plant system failures was based on an examination of the controls and indications that were present. A walkdown of the actual control room panels was also performed to support the determination of the extent of a postulated fire event. The walkdown also determined a weighting factor based generally on panel or cabinet length to support the partitioning of the total control room fire ignition frequency. The guidance provided in the Fire PRA Implementation Guide was used to determined whether a postulated control room panel fire would remain confined within the panel boundaries. A postulated fire which is not severe or is suppressed before control room becomes uninhabitable is not assumed to propagate to an adjacent panel if they are separated by a substantial solid metal barrier. Small penetrations or openings in these barriers that do not contain significant combustible materials are not assumed to compromise the adequacy of the boundary. The results of these walkdowns are presented in Table 4-19.

Table 4-19
Control Room Panel/Cabinet Walkdown Results

Panel ID.	Description	Space Units	Plant Trip	Impacted Fire IPEEE PSA Systems
902-3	Main Control Board Subsection 1	1 ½	Y	Div. II LPCI and CS, Hard Pipe Vent, Isolation Condenser, and Main Condenser (MSIV closure)
	Subsection 2	½	Y	HPCI
	Subsection 3	½	Y	ADS, Div. I LPCI and CS
902-4	Main Control Board	2	Y	SDC
902-5	Main Control Board	3	Y	CRD
902-6	Main Control Board	1 ½	Y	Main Condenser, FW, and Condensate
902-7	Main Control Board	2	Y	Main Condenser (Circ. Water and EHC)
902-8	Main Control Board Subsection 1	1	Y	AC Power - Loss of Offsite Power, 4kV buses 24 and 24-1, and EDG 2
	Subsection 2	½	Y	AC Power - Loss of 4kV buses 23 and 23-1, and EDG 2/3
903-3	Main Control Board Subsection 1	1 ½	Y	Div. II LPCI and CS, Hard Pipe Vent, Isolation Condenser, and Main Condenser (MSIV closure)
	Subsection 2	½	Y	HPCI
	Subsection 3	½	Y	ADS, Div. I LPCI and CS
903-4	Main Control Board	2	Y	SDC
903-5	Main Control Board	3	Y	CRD
903-6	Main Control Board	1 ½	Y	Main Condenser, FW, and Condensate
903-7	Main Control Board	2	Y	Main Condenser (Circ. Water and EHC)
903-8	Main Control Board Subsection 1	1	Y	AC Power - Loss of Offsite Power, 4kV buses 34 and 34-1, and EDG 3
	Subsection 2	½	Y	AC Power - Loss of 4kV buses 33 and 33-1, and EDG 2/3
902-2	Radiation Monitoring	½	N	None
902-10	Radiation Monitoring	1	N	None
902-11	Radiation Monitoring	½	N	None
902-13	TIP	1	N	None

Table 4-19
Control Room Panel/Cabinet Walkdown Results

Panel ID.	Description	Space Units	Plant Trip	Impacted Fire IPEEE PSA Systems
902-15	RPS/PCIS – A	2	Y	None
902-16	CRD Scram Testing	½	Y	None
902-17	RPS/PCIS – B	2	Y	None
902-18	FW Controls	1	Y	FW
902-19	ECCS Instrumentation	½	Y	CRD Hyd - PS 2-340-11, Sqr 2-302-88, 89, 90 2A LPCI/CCSW 2-1540-9A (recorder), 2-1540-10A, 2-1540-6A – dP modulator (valve control) 2B LPCI/CCSW – same Misc - 2-263-103 (jet pumps), 2-1290-17 (RWCU), 2-2340-3 , 2-1040-5, 2-1340-6 (iso cond), 2-1450-6 (ind only) Pressure Suppression 2-1602-5, 2-1602-6 Loss of 2-1501-3A and 3B
902-20	Termination Cabinet	1 ½	Y	None
902-21	Leak Detection and Acoustical Monitoring	1	N	None
902-36	IRM/SRM	1 ½	Y	None
902-37	PRM	3	Y	None
902-54	Offgas Monitoring	1	N	None
902-55	ACAD	½	N	None
902-56	ACAD	½	N	None
903-2	Radiation Monitoring	½	N	None
903-10	Radiation Monitoring	1	N	None
903-11	Radiation Monitoring	½	N	None
903-13	TIP	1	N	None
903-15	RPS/PCIS – A	2	Y	None
903-16	CRD Scram Testing	½	Y	None
903-17	RPS/PCIS – B	2	Y	None
903-18	FW Controls	1	Y	FW

Table 4-19
Control Room Panel/Cabinet Walkdown Results

Panel ID.	Description	Space Units	Plant Trip	Impacted Fire IPEEE PSA Systems
903-19	ECCS Instrumentation	½	Y	CRD Hyd - PS 2-340-11, Sqr 2-302-88, 89, 90 2A LPCI/CCSW 2-1540-9A, 2-1540-10A, 2-1540-6A - dP modulator 2B LPCI/CCSW – same Misc - 2-263-103, 2-1290-17, 2-2340-3, 2-1040-5, 2-1340-6, 2-1450-6 Pressure Suppression 2-1602-5, 2-1602-6
903-20	Termination Cabinet	1 ½	Y	None
903-21	Leak Detection and Acoustical Monitoring	1	N	None
903-36	IRM/SRM	1 ½	Y	None
903-37	PRM	3	Y	None
903-54	Offgas Monitoring	1	N	None
903-55	ACAD	½	N	None
903-56	ACAD	½	N	None
923-1	RBCCW, TBCCW, SW, FP, IA, SA, and Isolation Condenser MU Controls	1 ½	N	Loss of indicated systems, closure of Diesel Driven FP valve 2/3-4101, and 2/3A and 2/3B Isolation Condenser MU Pumps
923-2	345 kV Switchyard Control	2	Y	Loss of Unit 3 RAT
923-4	Reactor Building Drains and Sumps	½	N	None
923-5	HVAC	2	N	Loss of LPCI/CS and HPCI Room Coolers, RB, TB, and CRM Fan Control
923-5A	Torus/Drywell O_2 Monitoring	½	N	None
923-6	Lift Station CRT	½	N	None
923-7	Radiation Monitoring	1	N	None
923-74	SBO Diesel	1	N	SBO Diesel Generators
901-B1	Unit 1 AC Power	2	N	Unit 1 Impact Only
901-2	Unit 1 Controls	4	N	None
901-P18	138 kV Switchyard Control	2	Y	Loss of Unit 2 RAT

Table 4-19
Control Room Panel/Cabinet Walkdown Results

Panel ID.	Description	Space Units	Plant Trip	Impacted Fire IPEEE PSA Systems
TOTAL NUMBER OF SPACE UNITS		78		

Note : The space units column is used to weight each of the cabinets and panels solely for the purposes of parsing the total control room ignition frequency.

The walkdown results indicate that the control room could be treated based on a total of 78 panel space units. The total ignition frequency contribution from the control room panels and cabinets is 1.93E-2/yr. Therefore, the ignition frequency contribution for each cabinet or panel space unit is 1.93E-02/78 = 2.47E-04

This contribution was multiplied by the number of panel 'units' identified in the Table 4-19. Table 4-20 provides the description for each of the fire scenarios that were analyzed. Those cases where a fire was determined to cause a plant trip required a quantification. A separate quantification for a bounding fire was also performed and addressed failure to suppress the fire leading to loss of control room habitability. The ignition frequency for this last case was the total ignition frequency for the control room so that those cases that did not cause a plant trip were properly treated for their potential challenge to control room habitability.

Two of the control room panels contained internal barriers which substantially subdivided them into smaller sections. These panels are 902-3/903-3 and 902-8/903-8.

902-3/903-3 - the analysis of this panel was performed using three subsections. Subsection 1 was treated based on loss of Division II of ECCS, Isolation Condenser, and the Hard Pipe Vent. Subsection 2 was treated based on loss of HPCI. Subsection 3 was treated based on loss of Division I of ECCS and ADS. Subsection 3 was also evaluated for a postulated fire induced spurious actuation of ADS.

902-8/903-8 - the analysis of this panel was performed using two subsections. Subsection 1 was treated based on loss of offsite power and Division II of the onsite AC power distribution system Buses 24/24-1 (34/34-1). Subsection 2 was treated based on loss of Division I of the onsite AC power distribution system Buses 23/23-1 (33/33-1).

Table 4-20
Control Room Fire Scenarios

Panel ID.	Description	Comments	CDF
902-3	Main Control Board Subsection 1	Failure of ECCS Div. II	1.56E-08
	Subsection 2	Failure of HPCI	4.05E-11
	Subsection 3 – Case 1	Failure of ECCS Div. I, ADS functional failure	1.29E-08
	Subsection 3 – Case 2	Same as case 1 above, except fire induced spurious actuation of all ADS valves is assumed (see Note 4)	3.84E-08
902-4	Main Control Board	Failure of SDC and CRD	5.13E-11
902-5	Main Control Board	Panels 902-5, 902-6, and 902-7 are combined because of identical fire-induced consequences. Scenario considers failure of CRD, Feedwater, Condensate, and the Main Condenser	1.01E-08
902-6	Main Control Board	See 902-5 above	--
902-7	Main Control Board	See 902-5 above	--
902-8	Main Control Board Subsection 1	Non-recoverable loss of offsite power and loss of Div. II AC power	4.98E-07
	Subsection 2	Loss of Div. I AC power	1.79E-10
903-3	Main Control Board Subsection 1	Failure of ECCS Div. II	1.56E-08
	Subsection 2	Failure of HPCI	4.05E-11
	Subsection 3 – Case 1	Failure of ECCS Div. I, ADS functional failure	1.28E-08
	Subsection 3 – Case 2	Same as case 1 above, except fire induced spurious actuation of all ADS valves is assumed (see Note 4)	3.84E-08
903-4	Main Control Board	Failure of SDC and CRD	5.13E-11
903-5	Main Control Board	Panels 903-5, 903-6, and 903-7 are combined because of identical fire-induced consequences. Scenario considers failure of CRD, Feedwater, Condensate, and the Main Condenser	1.01E-08
903-6	Main Control Board	See 903-5 above	--
903-7	Main Control Board	See 903-5 above	--
903-8	Main Control Board Subsection 1	Non-recoverable loss of offsite power and loss of Div. II AC power	4.66E-07

Table 4-20
Control Room Fire Scenarios

Panel ID.	Description	Comments	CDF
	Subsection 2	Loss of Div. I AC power	1.79E-10
902-2	Radiation Monitoring	Note 1	--
902-10	Radiation Monitoring	Note 1	--
902-11	Radiation Monitoring	Note 1	--
902-13	TIP	Note 1	--
902-15	RPS/PCIS – A	Note 2	--
902-16	CRD Scram Testing	Note 2	--
902-17	RPS/PCIS – B	Note 2	--
902-18	FW Controls	Loss of Feedwater	1.55E-09
902-19	ECCS Instrumentation	Note 2	--
902-20	Termination Cabinet	Note 2	--
902-21	Leak Detection and Acoustical Monitoring	Note 1	--
902-36	IRM/SRM	Note 2	--
902-37	PRM	Note 2	--
902-54	Offgas Monitoring	Note 1	--
902-55	ACAD	Note 1	--
902-56	ACAD	Note 1	--
903-2	Radiation Monitoring	Note 1	--
903-10	Radiation Monitoring	Note 1	--
903-11	Radiation Monitoring	Note 1	--
903-13	TIP	Note 1	--
903-15	RPS/PCIS – A	Note 2	--
903-16	CRD Scram Testing	Note 2	--
903-17	RPS/PCIS – B	Note 2	--
903-18	FW Controls	Loss of Feedwater	1.55E-09
903-19	ECCS Instrumentation	Note 2	--
903-20	Termination Cabinet	Note 2	--
903-21	Leak Detection and Acoustical Monitoring	Note 1	--
903-36	IRM/SRM	Note 2	--

Table 4-20
Control Room Fire Scenarios

Panel ID.	Description	Comments	CDF
903-37	PRM	Note 2	--
903-54	Offgas Monitoring	Note 1	--
903-55	ACAD	Note 1	--
903-56	ACAD	Note 1	--
923-1	RBCCW, TBCCW, SW, FP, IA, SA, and Isolation Condenser MU Controls	Loss of support systems – Instrument Air, Service Air, Service Water, Reactor Building Closed Cooling Water, Turbine Building Closed Cooling Water, and Isolation Condenser Makeup	9.83E-09 Each Unit
923-2	345 kV Switchyard Control	Loss of 345kV Switchyard Connection	Unit 2 – 1.30E-10 Unit 3 – 3.52E-12
923-4	Reactor Building Drains and Sumps	Note 1	--
923-5	HVAC	Note 1	--
923-5A	Torus/Drywell O_2 Monitoring	Note 1	--
923-6	Lift Station CRT	Note 1	--
923-7	Radiation Monitoring	Note 1	--
923-74	SBO Diesel	Note 3	--
901-B1	Unit 1 AC Power	Note 1	--
901-2	Unit 1 Controls	Note 1	--
901-P18	138 kV Switchyard Control	Loss of 138kV Switchyard Connection	Unit 2 – 3.77E-09 Unit 3 – 1.27E-10
n/a	Bounding Control Room Fire	Suppression failure probability of 3.4E-3 and severity factor of 0.10 applied. The severity factor is incorporates a panel severity factor of 0.20 and a CCDP for shutdown from outside the control room of 0.50.	6.56E-06 Each Unit

Note 1: Postulated fire does not cause a plant trip and no post fire safe shutdown functions are immediately disabled. These scenarios are qualitatively screened. Treatment of this fire for potential challenges to control room habitability is addressed in final bounding fire scenario.

Note 2: Postulated fire may cause a plant trip, but no post fire safe shutdown functions are immediately disabled. These scenarios are qualitatively screened. Treatment of this fire for potential challenges to control room habitability is addressed in final bounding fire scenario.

Note 3: Postulated fire does not cause a plant trip, but post fire safe shutdown functions are impacted. The affected system is not a risk significant system with respect to the specific scenario being considered and is qualitatively screened on that basis. Treatment of this fire for potential challenges to control room habitability is addressed in final bounding fire scenario.

Note 4: The two scenarios for this panel are mutually exclusive. A postulated fire will either cause the functional failure of ADS or cause spurious actuation of ADS. Only the greater of the two values need be included in the CDF total. However, to simplify the overall process, both scenarios are numerically included in the reported values.

4.7.5 Summary of the Fire-Induced Core Damage Results

The following is a summary of the fire-induced core damage results for Dresden Units 2 and 3. These results include analysis findings from fire modeling of single fire compartments and the Control Room analysis. Since the multi-compartment analysis screened all scenarios, there is no multi-compartment contribution to the CDF presented below.

The Core Damage Frequencies (CDFs) resulting from these analyses for Dresden Station are:

Unit 2 CDF	Unit 3 CDF
1.69E-5	3.08E-5

The dominant fire scenarios which constitute 90% of the reported total are provided in Tables 4-21 and 4-22 for Units 2 and 3, respectively. As shown in these tables, the dominant contributor for each unit is a severe Control Room fire requiring evacuation and shutdown from outside the Control Room. The second largest contributor for each unit involves a large RFP fire with successful actuation of the suppression system.

One other notable scenario for Unit 3 is the bounding Cable Tunnel fire involving tray stack CT7. Based on the fire-induced equipment failures identified, this fire results in a CCDP of 1.0. The complement fire involving tray stack CT8 results in a CCDP of 2.34E-01. Cables for Unit 2 are not routed through the Cable Tunnel, therefore no corresponding scenarios were identified in the Unit 2 analysis.

Table 4-21 Unit 2 Fire Scenarios - Top 90% - Sorted by Descending CDF Contribution

Fire Compartment	Scen	Scenario Description	Fire Compartment Description	CDF	% Contrib	Total Contrib
2.0	N	Severe Fire w/evacuation	CONTROL ROOM	6.56E-06	38.9%	38.9%
8.2.5.A	N	RFP C - Large Fire w/Suppression	U2 NORTH TRACKWAY/SWGR AREA	2.48E-06	14.7%	53.6%
8.2.5.A	P	Misc Cond Pumps - MCC 26-1 - Large Fire	U2 NORTH TRACKWAY/SWGR AREA	1.68E-06	10.0%	63.5%
1.1.2.3	D	RWCU Pumps	U2 SECOND FLOOR RX BLDG	1.44E-06	8.5%	72.1%
8.2.6.B	B	Tray 3BA - %TP	U2 MEZZANINE	6.72E-07	4.0%	76.0%
8.2.6.A	E	4 kV Bus 24 - Large Fire	CONTROL ROOM BKUP VENTILATION	5.38E-07	3.2%	79.2%
2.0	H	Control Panel 902-8 Subsection 1 - LOOP + Div II	CONTROL ROOM	4.98E-07	3.0%	82.2%
8.2.5.A	F	480 V Bus/Xfmr 25 - Large	U2 NORTH TRACKWAY/SWGR AREA	3.11E-07	1.8%	84.0%
11.3	A	Screening Scenario	CRIBHOUSE UPPER	2.45E-07	1.5%	85.5%
8.2.5.A	R	Self-Initiated Cable Fire - Severe - %TP	U2 NORTH TRACKWAY/SWGR AREA	2.37E-07	1.4%	86.9%
8.2.5.A	O	RFP C - Small Fire w/o Suppression	U2 NORTH TRACKWAY/SWGR AREA	2.31E-07	1.4%	88.2%
8.2.5.A	M	RFP Area - Large Fire w/o Suppression	U2 NORTH TRACKWAY/SWGR AREA	1.52E-07	0.9%	89.1%
8.2.5.C	B	Instrument Air Compressor 2B Large Fire w/o Supp	2/3 TB CORRIDOR	1.48E-07	0.9%	90.0%

Table 4-22 Unit 3 Fire Scenarios - Top 90% - Sorted by Descending CDF Contribution

Fire Compartment	Scen	Scenario Description	Fire Compartment Description	CDF	% Contrib	Total Contrib
2.0	N	Severe Fire w/evacuation	CONTROL ROOM	6.56E-06	21.3%	21.3%
8.2.5.E	J	RFPs - Large w/Supp & Small w/o Supp	U3 WEST CORRIDOR AND TRACKWAY	3.4E-06	11.1%	32.4%
8.2.6.E	B	DC Panel Fire	U3 MEZZANINE FLOOR	2.69E-06	8.7%	41.1%
1.1.1.3	G	Self-Initiated Cable Fire - Trays 959-973 - %TP	U3 SECOND FLOOR RX BLDG	1.92E-06	6.2%	47.4%
8.2.5.E	K	Compressors - Large w/Supp & Small w/o Supp	U3 WEST CORRIDOR AND TRACKWAY	1.9E-06	6.2%	53.5%
6.2	U	Propagation to Tray 655B - %TP	AUX ELEC EQUIP ROOM	1.28E-06	4.2%	57.7%
8.2.4	CC	Bounding Fire - Tray Stack CT7 - %TP	U3 CABLE TUNNEL	9.4E-07	3.1%	60.8%
8.2.5.E	F	MCC 36-1 and Misc Cond Pumps	U3 WEST CORRIDOR AND TRACKWAY	9.09E-07	3.0%	63.7%
1.1.1.3	F	Self-Initiated Cable Fire - %TI	U3 SECOND FLOOR RX BLDG	8.78E-07	2.9%	66.6%
8.2.4	D	Tray CT7M2 - %TP	U3 CABLE TUNNEL	7.71E-07	2.5%	69.1%
8.2.6.E	D	Bounding Switchgear Fire	U3 MEZZANINE FLOOR	7.33E-07	2.4%	71.5%
6.2	T	Propagation to Trays 650B & 671B - %TP	AUX ELEC EQUIP ROOM	6.51E-07	2.1%	73.6%
8.2.6.D	B	Self-Initiated Cable Fire - Severe - %TP	U3 MEZZANINE FLOOR	5.82E-07	1.9%	75.5%
8.2.5.C	J	Self-Initiated Cable Fire - %TP	2/3 TB CORRIDOR	5.42E-07	1.8%	77.2%
8.2.1.B	A	Screening Scenario	U3 COND. PP AREA	4.85E-07	1.6%	78.8%
2.0	H	Control Panel 903-8 Subsection I -- LOOP + Div II	CONTROL ROOM	4.66E-07	1.5%	80.3%
8.2.5.E	G	Self-Initiated Cable Fire - %TP	U3 WEST CORRIDOR AND TRACKWAY	4.66E-07	1.5%	81.8%
1.1.1.2	G	Pump Back Air Compressor Lg Fire	U3 RX BLDG GROUND FLOOR	4.04E-07	1.3%	83.2%
1.1.1.3	H	Self-Initiated Cable Fire - Trays 974-987 - %TP	U3 SECOND FLOOR RX BLDG	3.12E-07	1.0%	84.2%

Table 4-22 Unit 3 Fire Scenarios - Top 90% - Sorted by Descending CDF Contribution

Fire Compartment	Scen	Scenario Description	Fire Compartment Description	CDF	% Contrib	Total Contrib
8.2.5.A	E	Self-Initiated Cable Fire - %TP	U2 NORTH TRACKWAY/SWGR AREA	2.76E-07	0.9%	85.1%
1.1.1.2	I	Self-Initiated Cable Fire - %TI	U3 RX BLDG GROUND FLOOR	2.73E-07	0.9%	86.0%
1.1.1.3	E	Bus 34-1	U3 SECOND FLOOR RX BLDG	2.67E-07	0.9%	86.8%
8.2.5.C	B	Instrument Air Compressor 2B Large Fire	2/3 TB CORRIDOR	2.64E-07	0.9%	87.7%
8.2.6.A	B	Bounding Fire	CONTROL ROOM BKUP VENTILATION	2.39E-07	0.8%	88.5%
8.2.7	A	Screening Scenario	VENT ROOM OVER NE SWGR	2.39E-07	0.8%	89.2%
11.3	A	Screening Scenario	CRIBHOUSE UPPER	2.38E-07	0.8%	90.0%

4.8 ANALYSIS OF CONTAINMENT PERFORMANCE

Containment performance was evaluated because of its importance in preventing the release of radioactive material. Generic Letter 88-20, Supplement 4, and NUREG-1407 provide guidance for the review of containment performance for fire induced core damage accidents.

A. Generic Letter 8-20, Supplement 4:

> *The evaluation of containment performance for external events should be directed toward a systematic examination of whether there are sequences that involve containment failure modes distinctly different from those found in the IPE internal events evaluation or contribute significantly to the likelihood of functional failure of the containment (i.e., loss of containment barrier independent of core melt).*

B. NUREG-1407, Section 4.1.5:

> *Perform containment analysis if containment failure modes differ significantly from those found in the IPE internal events evaluation.*

Primary containment is constructed of a steel pressure vessel, i.e., the Drywell, downcomers, and wetwell (torus). The containment is penetrated to allow the passage of pipe and cable necessary for power operation and accident prevention or mitigation. Hatches in the drywell and torus provide access for maintenance and inspection. "Containment" is provided by the containment pressure vessel, the hatches, penetration seals, piping and associated isolation valves.

Fire impact on the containment itself is expected to be minimal. The containment hatches at Dresden do not rely on active means (air, electricity) to function. Because of the type of construction and the low combustible loading nearby, hatches are not expected to sustain fire damage. The piping and cable penetrations are fabricated of steel. Fire is not expected to fail these penetrations.

The containment fire area was eliminated during the screening phase of the fire analysis according to the approach suggested by FIVE (Ref. 4-4). FIVE references the EPRI Fire Events Database (Ref. 4-13) which provides evidence that containment fires at power have been few. Also, except for brief periods after a reactor startup, before a shutdown, or for infrequent drywell entries at power, the primary containment is kept inert with nitrogen. As a result, a fire inside containment at power is not expected to present a significant risk.

The information summarized below supports the conclusion that Dresden fire induced core damage events have an insignificant influence on the reliability of containment in that fire core damage sequences, (a) progress in the same manner as the core damage sequences assessed in the interval events analysis, i.e., they do not involve containment failure modes distinctly different from those found in the internal events, (b) do not introduce containment performance insights beyond those already identified in the internal events analysis, and (c) do not contribute significantly to the likelihood of functional failure of the containment.

The details of the fire results are further discussed below relative to the discussion of new containment failure modes or significant increases in containment functional failure independent of core melt. These aspects include:

- containment bypass
- containment isolation failures
- other failure modes

4.8.1 Containment Bypass

A review was performed of the high pressure/low pressure interfacing systems LOCA (ISLOCA) paths identified in the Dresden IPE for a determination of the possible impact due to fire. The interfacing systems LOCA paths are analyzed in the IPE Interfacing Systems LOCA Assessment (Ref. 4-17) In accordance with Appendix N of the EPRI Fire PRA Implementation Guide (Ref. 4-5), any path that contains two or more non-fire-susceptible closed valves can be screened from further evaluation. All of the interfacing system LOCA paths, except the following, have at least two non-fire-susceptible closed valves and were screened from further evaluation.

C. LPCI injection lines (normally closed MOVs MO2(3)-1501-22A & B and inboard check valves), and

D. Core Spray injection lines (MO2(3)-1402-25A & B and inboard check valves).

These scenarios include one random failure (leakage/rupture of the check valve) and one spurious actuation of a closed MOV.

The fire ignition frequencies for the zones containing cables for LPCI and Core Spray MOVs of concern are all less than 2.0E-2 per reactor year. Probability of a hot short given damage to a circuit is estimated as 0.068 in NUREG/CR-2258, page 112. A random failure probability for a check valve failing to close of 8.0E-5 per demand from Table 4.4.1-6 in the Dresden IPE Submittal Report (Ref. 4-10). Pipe rupture given over pressure is estimated at less than 0.1 based on the schedule of pipe and methods developed by Wesley in NUREG/CR-5603 (Ref. 4-23). Combining the upper bound fire ignition frequency of 2.0E-2 per reactor year, spurious actuation of a cable given fire damage to the cable (0.068), the random failure of a check valve (8.0E-5/d), and pipe rupture probability (0.1/d) results in an upper bound CDF value for the fire induced ISLOCA event of 1E-8 per reactor year, per line.

This estimate is considered conservative because the fire frequency used to estimate ISLOCA core damage frequency assumed any fire anywhere in the zone damaged the cable(s) associated with applicable MOV(s). Additionally, an analysis was not performed to determine if a temporary spurious actuation could be recovered or a seal-in circuit exists which would keep the valve open after circuit failure progressed to an open circuit.

4.8.2 Containment Isolation

Dresden normally operates with containment inerted with nitrogen. The Dresden IPE concluded that containment isolation failures are not considered likely failure modes leading to core damage and were not considered in the Dresden Level 1 event trees.

The scope of the containment isolation pathways considered here is the same as that evaluated in the Level 2 PSA. This scope includes containment penetration paths larger than 2" in diameter.

Fire-induced impacts on automatic primary containment isolation valves (PCIVs) may be postulated from hot shorts. The Primary Containment Isolation System is equipped with the following design features, which minimize the likelihood of containment isolation failure.

1. The PCIS is designed to fail to a safe mode given loss of electric power. The PCIS sensor and logic circuitry provides both automatic and remote manual isolation capabilities.

2. The control logic for the closure of the PCIVs is designed to assure that once an isolation signal has been initiated, the valves continue to close until full closure is achieved. Once full closure occurs, the valves will not automatically re-open even if the closure signal ceases.

3. The valves are solenoid-controlled; air-operated valves that are designed to fail closed on loss of air/power. One exception to this design is the torus vacuum relief AOVs, which fail open on loss of power/pneumatic pressure. However, highly reliable backup check valves provide containment isolation of this penetration.

4. Two in-series (redundant) isolation valves protect all penetration paths.

The Appendix R Analysis similarly concluded that the probability of both isolation valves in a line being affected by a fire such that they spuriously open was too low to warrant further consideration.

4.8.3 Other Methods of Containment Failure Prior to Core Melt

Section 6.3.9 of FIVE (Ref. 4-4) provides guidance on the evaluation for potential impact of a fire on containment heat removal and isolation. The fire effects on containment performance must be evaluated if the likelihood of loss of safe shutdown capability for a fire compartment is greater than 1.0E-6 per reactor year. Fire compartments that did not screen (see Section 4.7) were reviewed for potential impact on containment performance.

There are additionally accident sequences that could fail containment prior to core damage. These sequences are related to failure to scram and loss of decay heat removal. These heat removal mismatch sequences can be induced by fires. However, the impact on containment performance for these can be summarized as follows:

- Failure to Scram: These sequences are so low in frequency to be screened from consideration.

- Loss of Decay Heat Removal: The character of these sequences is exactly the same as in the internal events analysis. No new insights related to containment performance are derived from these sequences.

4.8.4 Summary

In summary, the characteristics of the accidents identified in the fire induced scenarios are similar to those already evaluated in the internal events IPE. No potential containment performance issues were identified.

4.9 ANALYSIS RESULTS AND INSIGHTS

The upgraded Dresden Fire analysis provided results that are significantly different than the original analysis in terms of both the reported CDF contributions and risk insights. The total calculated CDF contribution due to fires from the upgraded fire analysis is almost an order of magnitude lower than that presented in the original submittal. In addition, several fire scenarios which were previously determined to have relatively low risk significance in the original analysis were identified as important fire scenarios in the upgraded analysis. The combination of all of these factors results in a significant altering of the fire risk insights from that portrayed in the original Fire IPEEE submittal. The following sections discuss the key differences in the analysis results.

4.9.1 Reduction in Overall CDF

The upgraded fire analysis produced a calculated CDF contribution due to fires that was approximately an order of magnitude lower than that presented in the original Fire IPEEE submittal. A comparison of the upgraded fire analysis results and the original Fire IPEEE results is provided in Tables 4-23 and 4-24. Only the dominant contributors to the original and upgraded analysis results are presented. In addition, the analysis results have been grouped to be consistent with the presentation of results in the original Fire IPEEE submittal.

The significant reduction is due to the incorporation of detailed cable spatial information and the more realistic treatment of fire consequences into the upgraded analysis. The availability of comprehensive spatial information for critical cables allowed the fire modeling results to provide a more realistic characterization of the consequences of postulated fire events. The more realistic treatment of fire consequences eliminated the bounding assumption of large scale target failures applied in some of the original fire scenarios.

The original Fire IPEEE analysis for some of the fire compartments suffered from the lack of detailed cable spatial information. While the data did associate critical cables with fire compartments, it did not provide additional details necessary to locate the cable within the compartment. Because of this limitation, the integration of the fire modeling results into the analysis could not distinguish between individual circuit and equipment failures. As a consequence, the original Fire IPEEE assumed that if any postulated fire damaged any circuit, all circuits within the fire compartment were considered to be damaged.

The upgraded fire analysis had the benefit of more detailed cable spatial information. This detailed information allowed the analyst to determine the location of any given cable within a fire compartment based on conduit or tray node. In addition, the upgraded analysis performed supplemental cable function reviews to determine whether the scope of circuits potentially impacted by the postulated fire event would actually cause the equipment or systems failures indicated by the data relationships. Using this information, the upgraded analysis was able to confirm that the scope of cable damage did not result in the same scope of systems failures considered in the original analysis.

Table 4-23 Comparison of Unit 2 Original and Revised Dresden Fire CDF Results

Fire Zone	Fire Area	Fire Zone Description	Original CDF	Contribution	Revised CDF	Contribution
2.0	TB-V	CONTROL ROOM	1.66E-06	0.7%	7.15E-06	42.4%
8.2.5.A	TB-I	U2 NORTH TRACKWAY/SWGR AREA	1.57E-05	6.3%	5.38E-06	31.9%
6.2	TB-V	AUX ELEC EQUIP ROOM	3.07E-06	1.2%	5.36E-07	3.2%
1.1.2.3	RB2-II	U2 SECOND FLOOR RX BLDG	2.34E-05	9.4%	1.65E-06	9.8%
1.1.2.2	RB2-II	U2 RX BLDG GROUND FLOOR	8.76E-06	3.5%	4.69E-08	0.3%
8.2.6.A	TB-I	CONTROL ROOM BKUP VENTILATION	6.16E-05	24.6%	5.86E-07	3.5%
8.2.5.C	TB-II	2/3 TB CORRIDOR	1.32E-05	5.3%	2.52E-07	1.5%
7.0.A.1	TB-I	U2 BATTERY ROOM	1.04E-05	4.2%	5.89E-08	0.3%
1.1.2.4	RB2-II	U2 RX BLDG SWITCHGEAR AREA	9.11E-06	3.6%	1.37E-08	0.1%
8.2.6.C	TB-II	U2/3 SBGT & TBCCW HX	5.87E-05	23.5%	5.29E-09	0.0%
8.2.6.B	TB-I	U2 MEZZANINE			6.74E-07	4.0%
		Cumulative CDF of Listed Items	2.06E-04	82.2%	1.64E-05	96.9%
		Total CDF Reported in Submittal	2.50E-04		1.69E-05	

Table 4-24 Comparison of Unit 3 Original and Revised Dresden Fire CDF Results

Fire Zone	Fire Area	Fire Zone Description	Original CDF	Contribution	Revised CDF	Contribution
8.2.5.E	TB-III	U3 WEST CORRIDOR AND TRACKWAY	5.27E-05	18.8%	6.85E-06	22.3%
1.1.1.3	RB3-II	U3 SECOND FLOOR RX BLDG	5.06E-05	18.1%	3.54E-06	11.5%
2.0	TB-V	CONTROL ROOM	2.65E-06	0.9%	7.11E-06	23.1%
1.1.1.4	RB3-II	U3 RX BLDG SWITCHGEAR AREA	1.78E-05	6.4%	2.06E-08	0.1%
6.2	TB-V	AUX ELEC EQUIP ROOM	1.12E-05	4.0%	2.53E-06	8.2%
8.2.6.E	TB-III	U3 MEZZANINE FLOOR	7.27E-06	2.6%	3.44E-06	11.2%
8.2.4	TB-III	U3 CABLE TUNNEL	1.38E-05	4.9%	2.12E-06	6.9%
8.2.5.C	TB-II	2/3 TB CORRIDOR	2.15E-05	7.7%	8.36E-07	2.7%
8.2.6.A	TB-I	CONTROL ROOM BKUP VENTILATION	5.54E-06	2.0%	4.59E-07	1.5%
8.2.6.C	TB-II	U2/3 SBGT & TBCCW HX	5.89E-05	21.0%	5.32E-07	1.7%
1.1.1.2	RB3-II	U3 RX BLDG GROUND FLOOR	7.39E-06	2.6%	7.16E-07	2.3%
7.0.A.1	TB-I	U2 BATTERY ROOM	2.37E-06	0.8%	1.59E-08	0.1%
6.1	TB-III	U3 BATTERY CHARGER ROOM	3.52E-06	1.3%	1.07E-09	0.0%
8.2.6.D	TB-III	U3 MEZZANINE FLOOR			7.90E-07	2.6%
		Cumulative CDF of Listed Items	2.55E-04	91.2%	2.90E-05	94.2%
		Total CDF Reported in Submittal	2.80E-04		3.08E-05	

4.9.1.1 Fire Compartment 8.2.6.A

The original Fire IPEEE evaluation of fire compartment 8.2.6.A resulted in a cumulative CDF contribution of 6.16E-05 per year and 5.54E-06 per year for Units 2 and 3, respectively. This is in contrast with the upgraded fire analysis results of 5.86E-07 per year and 4.59E-06 per year for Units 2 and 3, respectively. The upgraded results are more than two orders of magnitude lower than the original analysis.

The original analysis for these areas included consideration of potential transient fires. These fires contribute approximately 72% to the cumulative Unit 2 CDF and 56% to the cumulative Unit 3 CDF for these areas. The methodology applied to transient fires in the original Fire IPEEE assumed all circuits within the fire compartment were damaged. Additionally, the original analysis did not credit the Core Spray System. For all three areas, the Isolation Condenser is an important system for providing decay heat removal and mitigating the effects of the fire-induced equipment failures.

The dominant contributor in both analyses for 8.2.6.A involve a fire originating from 4160V Switchgear 24. The original analysis assumed fire-induced loss of LPCI, among other systems. The upgraded fire analysis had the benefit of more detailed cable information, and it credits the Core Spray System for low pressure injection. The review of this information determined that the postulated Switchgear 24 fires would not result in the loss of LPCI Train A nor Core Spray Train A.

Additionally, the upgraded analysis provided a more realistic treatment of transient fires by adding a portion of the transient initiating event frequency contribution to the initiating event frequency for each discrete scenario evaluated. The realistic treatment of the transient fires and the fire-induced failures resulted in a two order of magnitude reduction in the CDF. The upgraded analysis also explicitly treated non-severe fires as compared to the original analysis where they were implicitly screened.

4.9.1.2 Fire Compartment 8.2.6.C

The original Fire IPEEE evaluation of fire compartment 8.2.6.C resulted in a cumulative CDF contribution of 5.89E-05 per year and 5.87E-05 per year for Units 2 and 3, respectively. This is in contrast with the upgraded fire analysis results of 5.32E-07 per year and 5.29E-09 per year for Units 2 and 3, respectively. The upgraded results are more than two orders of magnitude lower than the original analysis.

The dominant contributor from the original analysis involved postulated fires originating from the Sparging Air Compressors, 2/3-4603A and 2/3-4603B. These fires were conservatively modeled as damaging all equipment in the compartment. Additionally, 58% of the cumulative CDF is contributed by transient fires, which were also modeled as damaging all equipment in the compartment.

The upgraded fire analysis had the benefit of more detailed cable information and spatial information. The review of this information determined that the postulated air compressor fires would result in the loss of a minimal amount of equipment, primarily 480V Switchgear 35 and 37. Additionally, the upgraded analysis provided a more realistic treatment of transient fires by adding a portion of the transient initiating event frequency contribution to the initiating event frequency for each discrete scenario evaluated. The realistic treatment of the transient fires and the fire-induced failures resulted in a two order of magnitude reduction in the CDF.

4.9.2 Analysis Insights

The upgraded fire analysis produced significantly different results as compared to the original Fire IPEEE. These differences are evidenced by the much lower calculated CDF contribution due to postulated fire events and the distribution of the risk contributors amongst the fire compartments. A review of the upgraded analysis results provides risk insights that are considered to be a much more accurate characterization of the Dresden station.

The upgraded analysis highlighted eleven insights.

1. The calculated CDF contribution due to postulated fire events is consistent with other Boiling Water Reactor (BWR) plants.
2. A large oil fire involving Unit 2 Reactor Feedwater Pump C or a fire involving MCC 26-1 is a dominant contributor to the Unit 2 CDF. This is because of the location of the cables needed for the Unit 2 DC power system. The Unit 3 DC power feed to one train of the Unit 2 DC system as well as the Unit 2 AC power cable to the battery charger for the redundant DC train are exposed to a common hazard. Although the circuits are located in separate trays, they are stacked vertically. The occurrence of a postulated large fire event requires an operator action to either align the spare battery charger or to connect the spare Unit 2 battery bank.
3. Excluding the control room severe fire, the dominant core damage sequence is loss of decay heat removal. Opportunities to recover decay heat removal functions will have significant risk reduction potential.
4. The configuration of the ADS system at Dresden is such that there are unprotected (no fire wrapping) circuits located outside of the auxiliary electric equipment and main control rooms whose fire induced failure could cause the spurious actuation of ADS valves. The spurious opening of the ADS valve(s) for this postulated case would not be precluded by the ADS inhibit switch. However, these cables are routed in such a fashion that they are not exposed to any significant external fire threat and were not a dominant risk contributor.
5. The most risk significant control room fire scenario which did not require the control room to be abandoned involves a postulated fire in panel 902-8/903-8. Such a fire

results in a loss of offsite power and Division II of the onsite AC power distribution system (Buses 24/24-1, 34/34-1).

6. The lower damage threshold for non-IEEE 383 qualified cables limited the effectiveness of installed automatic fire suppression systems. Although a specific sensitivity study was not performed, it is expected that the results of the dominant risk contributor (Reactor Feed Pump oil fire) would be reduced if IEEE 383 qualified cables had been installed.

4.10 TREATMENT OF FIRE RISK SCOPING STUDY ISSUES

The purpose of this section is to provide Dresden Station responses to the Fire Risk Scoping Issues addressed in EPRI FIVE (Ref. 4-4).

The EPRI FIVE documentation discusses the following five issues:

- Seismic/fire interactions;
- Fire barrier qualification;
- Manual fire fighting effectiveness;
- Total environment equipment survival; and
- Control systems interaction.

These issues, which were originally taken from the Fire Risk Scoping Study (NUREG/CR-5088) (Ref. 4-3) performed by Sandia National Laboratories, are discussed below.

4.10.1 Method

The Dresden response to the Fire Risk Scoping Study (FRSS) issues was developed from a variety of sources including:

- Interviews with Dresden staff
- Review of Dresden procedures
- Review of Dresden fire protection documentation

The responses are provided below. For each issue, the question from FIVE is repeated, followed by the Dresden response.

4.10.2 Response to Issues

4.10.2.1 Seismic/Fire Interactions

The issue of seismic/fire interactions centers on the following three areas of interest:

- Seismically-induced fires. In particular, this concern centers on fires caused by flammable gas or liquid storage containers or systems that could rupture during a seismic event.
- Seismic actuation of fire suppression systems. In particular, this concern centers on the failure of electrical or other components due to water sprays.
- Seismic degradation of fire suppression systems. In particular, this concern reviews the plant design for fragility of fire suppression systems to a seismic event.

Each of these areas is described in detail below.

4.10.2.1.1 Seismically-Induced Fires

As part of the seismic assessment walkdown, verify hydrogen or other flammable gas or liquid storage vessels in areas with seismic safe shutdown or safety related equipment are not subject to leakage under seismic conditions. Examples would be improperly anchored hydrogen or oxygen bottles, hydrogen tanks used for primary coolant chemistry control, etc.

Response The fire-seismic walkdowns evaluated fixed plant systems including piping and vessels for storage of hydrogen and combustible liquid and determined, with few exceptions, that they are not subject to leakage under seismic conditions. The following items were noted during the fire-seismic walkdowns and have been evaluated in the fire evaluation:

1. Oil Filled Step-Down Transformers
   Step-down transformers associated with switchgear 25, 26, 27, 35, 36 and 37 were found not to be anchored and potentially subject to tipping which could release their oil. These transformers have been specifically fire modeled considering their potential as ignition sources and assuming that transformer oil is released to the surrounding bermed area.

2. Hydrogen Seal Oil Panel and Hydrogen Monitors
   The hydrogen seal oil control panel located in the Unit 3 turbine building at elevation 538', and the shelf mounted Turbine Generator Hydrogen Monitors are unanchored. In addition, the hydrogen seal oil control panel located in the Unit 2 Turbine Building is anchored but welds are of questionable quality, and the shelf mounted hydrogen monitor is not positively anchored. Hydrogen lines are routed through these cabinets so the potential for a hydrogen gas release in these areas exists.

3. Flammable Liquid Storage Cabinets in Reactor Building
The flammable liquids storage cabinets were determined not to be subject to tipping, and, therefore, were not considered an exposed combustible in the fire evaluation.

4. PCB Holding Tanks
The PCB holding tanks located behind Switchgear 23-1 and 34-1 were identified as release hazards (for fire) due to the sight glass at the bottom of each tank which could break. It was determined that their use is temporary in nature (e.g., during an outage) and that they are normally empty. Since the tanks are normally empty, they are not considered a fire hazard.

5. Hydrogen Tanks in the Tank Farm
The tank farm is substantially removed from safety-related structures and equipment, and does not represent a fire risk to them.

4.10.2.1.2 Seismic Actuation of Fire Suppression Systems

As part of the seismic assessment, verify that the design of the water suppression system considers the effects, if appropriate, of inadvertent suppression system actuation and discharge on that equipment credited as part of the seismic safe shutdown path in a margins assessment that was not previously reviewed relative to the internal flooding analysis or concerns such as those discussed in NRC I&E Information Notice 83-41.

Response An analysis of the Effects of Fire Suppression System Actuation on Nuclear Safety Related Equipment was performed in response to IN 83-41 for Dresden in 1985.

In support of IPEEE, inadvertent actuation of fire suppression systems was studied by means of a walkdown. During the walkdown, care was taken to observe potential for spray-down or release of fire suppression media due to seismic interaction. No such instances were observed at Dresden. In addition, fire control equipment (panels and cabinets) were walked down to ensure they were properly anchored and not subject to potential seismic interactions.

A review of relays which could potentially lead to inadvertent suppression system actuation (i.e., "bad actor" relays), determined that no such relays exist at Dresden Station.

4.10.2.1.3 Seismic Degradation of Fire Suppression Systems

As part of the seismic assessment walkdown, verify that plant fire suppression systems have been structurally installed in accordance with good industrial practice and reviewed for seismic considerations, such that suppression system piping and components will not fail and damage safe shutdown path components, nor is it likely that leaking or cascading of the suppressant will result.

Response Seismic degradation of fire suppression systems was reviewed by walking down fire piping and looking for poor structural design features or potential interactions with safe shutdown path components. No such potential interactions were noted except for the piping in the vicinity of panels D02 and D03-2203-073A&B which has been analyzed as summarized in Table 3.3 I.D.A37 of the seismic portion of the Dresden IPEEE response.

As a result, fire protection system piping is not expected to fail in a seismic event and safe shutdown path components will not be damaged.

4.10.2.2 Fire Barrier Qualifications

The concern for fire barrier qualification centers on the following four (4) areas of interest:

- Fire barrier surveillance program;
- Inspection and maintenance of fire doors;
- Installation, inspection, surveillance and maintenance of penetration seal assemblies; and
- Inspection, testing and maintenance of fire dampers.

Each of these areas of interest is described in detail below.

4.10.2.2.1 Fire Barriers

Fire barriers and components such as fire dampers, fire penetration seals and fire doors for fire barriers are included in the plant surveillance program.

Response Fire rated barriers are visually inspected every 18 months, as required by Dresden Administrative Technical Requirements (DATR) 3/4.1.6, (Fire Rated Assemblies), and implemented under procedures DFPS 4175-02 (Operating Fire Stop/Break Surveillance) and DFPS 4175-03 (Shutdown Fire Stop/Break Surveillance) to verify that they are operable. Additionally, specific surveillances are performed on fire doors, penetration seals, fire dampers and structural steel fire proofing under DFPS series 4175 procedures.

In the unlikely event that the condition of a fire barrier (or any of its components) is found to be unsatisfactory, compensatory measures (fire watches) are established as required by DATR 3/4.1.6 (Fire Rated Assemblies) in accordance with DFPP 4175-01 (Fire Barrier Integrity and Maintenance). The fire watches remain in place until the barrier is restored to a satisfactory condition.

Fire barriers credited in the Dresden Fire PRA have been visually inspected during the development of the analysis to verify that they are adequate to prevent the spread of fire and hot gases.

To ensure that the fire propagation beyond a barrier has been considered in the analysis, a detailed multi-compartment analysis (MCA) has been performed which postulates barrier failure and fire propagation into adjacent compartments. The contribution to Core Damage Frequency from this analysis is incorporated in the overall plant fire risk.

4.10.2.2.2 Fire Doors

A fire door inspection and maintenance program should be implemented at the plant.

Response Fire door surveillance and inspection is performed as specified in the DATR 3/4.1.6 (Fire Rated Assemblies) and implemented under DFPS 4175-07 (Fire Door/Spill Barrier Surveillance).

4.10.2.2.3 Penetration Seal Assemblies

A. A penetration seal inspection and surveillance program should be implemented at the plant.

Response Penetration seal surveillance and inspection is performed once every 18 months as specified in DATR 3/4.1.6 (Fire Rated Assemblies) and implemented under DFPS 4175-02 (Operating Fire Stop/Break Surveillance) and DFPS 4175-03 (Shutdown Fire Stop/Break Surveillance). A minimum of 10% of the seal population is inspected on an 18 month interval, with all penetration seals being inspected at least once every 15 years.

B. Fire barrier penetration seals have been installed and maintained to address concerns such as those identified in NRC Information Notice 88-04.

Response Dresden fire barrier penetration seals have been evaluated for numerous concerns identified in various NRC Information Notices including IN 88-04. Penetration seals are maintained in accordance with station procedures and corporate guidelines and procedures.

4.10.2.2.4 Fire Dampers

A. An inspection and maintenance program for fire dampers should be implemented at the plant.

Response Fire damper installation was evaluated per NFPA code reviews. Fire dampers are tested and maintained in accordance with approved station procedures.

B. Damper installations address concerns such as those identified in NRC Information Notice 89-52, "Potential Fire Damper Operational Problems," dated June 8, 1989, and NRC Information Notice 83-69, "Improperly Installed Fire Dampers at Nuclear Power Plants," dated October 21, 1983.

Response Fire damper installation was evaluated per NFPA code reviews. Fire dampers are tested and maintained in accordance with approved station procedures.

4.10.2.3 Manual Fire Fighting Effectiveness

The concern for manual fire fighting effectiveness centers on the following six (6) areas of interest.

- Fire reporting, including the use and availability of portable fire extinguishers and plant procedures for reporting fires, including plant communication.
- Fire brigade makeup and equipment.
- Fire brigade training in the classroom.
- Fire brigade practice in hands on structural fire training and in the use of equipment.
- Fire brigade drills.
- Fire brigade training records.

Each of these areas of interest is described in detail below.

4.10.2.3.1 Reporting Fires

A. Appropriate plant personnel are knowledgeable in the use of portable fire extinguishers.

Response The fire brigade is trained in accordance with the Commonwealth Edison Production Training Center, Fire Brigade Initial and Continuing Training Program, Administration and Course Management Information (ACMI).

B. Portable extinguishers are located throughout the plant.

Response Portable fire extinguishers are located throughout the plant. These locations are identified in the Dresden Nuclear Units 2&3 Fire Pre-Plans. The extinguishers are maintained in accordance with DFPS 4114-04 (Fire Extinguisher Maintenance Inspection).

C. A plant procedure is in use for reporting fires in the plant.

Response All plant personnel are directed, during Nuclear General Employee Training, to contact the Control Room in the event of a fire.

Additionally, DHP 0210-01 (Responding to Station Fire Alarms) and DOA 0010-10 (Fire/Explosion) provide specific instructions for fire events.

D. A plant communication system that includes contact to the control room is operable at the plant.

Response Systems available to communicate with the Control Room at Dresden include:

1. Telephone System
2. Public Address (PA)
3. Sound Powered Phones
4. Two-Way Radios

4.10.2.3.2 Fire Brigade Makeup and Equipment

A. A fire brigade that is made up of at least five (5) trained people on each shift should be maintained at the plant.

Response Dresden has a dedicated Safety & Property Loss Prevention (S&PLP) department. Per DATR 6.1, a five man brigade will be maintained on site. As specified in DFPP 4100-01 (Fire Protection Program), the brigade will be made up of a brigade lead and four brigade members.

B. The fire brigade leader and at least two other brigade members on each brigade shift should be knowledgeable in plant systems and operations.

Response Per Dresden procedure DFPP 4100-01(Fire Protection Program), a licensed operator is assigned to advise the brigade leader in the event of a fire.

C. Each brigade member should receive an annual review of physical condition to evaluate his ability to perform fire fighting activities.

Response Per Dresden procedure DFPP 4100-01 (Fire Protection Program), physical examinations and physical activity are incorporated in the Fire Brigade Training Program. Brigade members receive annual physical examinations.

D. Personal protective equipment should be provided such as SCBA, turnout coats, boots, gloves, and hard hats.

Response Dresden procedures DFPS 4114-14 (MSA SCBA Inspection) and DFPS 4114-16, (ISI Magnum Self Contained Breathing Apparatus Inspection) address the fire brigades SCBA equipment.

There is no specific procedure for inspecting turn-out gear, however, Dresden fire brigade members are trained to use and ensure the availability of their safety clothing/equipment in accordance with the ACMI training. It is the responsibility of each brigade member to inspect his own equipment. Each brigade member is provided turnout coats, boots, gloves and hard-hats.

E. Emergency communications equipment should be provided for fire brigade use.

Response Systems available to communicate with the Control Room during a fire include:

1. Telephone system
2. Public Address (PA)
3. Sound Powered Phones
4. Fire Brigade Radios

F. Portable lights should be provided for fire brigade use.

Response Portable lights are included on the three fire equipment carts maintained under DFPS 4114-12 (Fire Equipment Cart Inspection).

G. Portable ventilation equipment should be provided for fire brigade use.

Response Smoke ejectors are included on the three fire equipment carts maintained under DFPS 4114-12 (Fire Equipment Cart Inspection).

H. Portable extinguishers should be provided for fire brigade use.

Response Portable fire extinguishers are located throughout the plant as described in the Dresden Fire Preplans. Extinguishers are also maintained on the three fire equipment carts under DFPS 4114-12 (Fire Equipment Cart Inspection).

4.10.2.3.3 Fire Brigade Training

Brigade members should receive an initial classroom instruction program consisting of the following:

- A review of the plant fire fighting plan and identification of each individual's responsibilities.
- Identification of typical fire hazards and associated types of fires that may occur in the plant.
- Identification of the location of fire fighting equipment and familiarization with the layout of the plant, including access and egress routes.
- Training on the proper use of available fire fighting equipment and the correct method of fighting each type of fire. The types of fires covered should include fires in energized electrical equipment, fires in cables and cable trays, and fires involving flammable and combustible liquids and gases.
- Training on the proper use of communication, lighting, ventilation and emergency breathing equipment.
- Training on techniques for fighting fires inside buildings and confined spaces.
- A review of fire fighting strategies and procedures.

Response Dresden fire brigade training is performed in accordance with the Commonwealth Edison Production Training Center, Fire Brigade Initial and Continuing Training Program, Administration and Course Management Information (ACMI). Training requirements are outlined in Dresden procedure DFPP 4100-01, Fire Protection Program.

Items a) through g) above are included in the training course modules identified in Attachment I (Lesson Plan List) to the ACMI.

4.10.2.3.4 Fire Brigade Practice

Fire brigade members should receive hands-on structural fire fighting training at least once a year to provide experience in actual fire extinguishment and the use of emergency breathing apparatus.

<u>Response</u> As addressed in DFPP 4100-01(Fire Protection Program) and in accordance with the ACMI, brigade member training includes annual structural fire fighting with fire extinguishment and use of SCBA.

4.10.2.3.5 Fire Brigade Drills

A. Fire brigade drills are performed in the plant so that each fire brigade shift can practice as a team.

<u>Response</u> In accordance with DFPP 4100-01 (Fire Protection Program), the Site Quality Verification Department is responsible for continually assessing the effectiveness of the Fire Protection Program. Specific fire drill activities are consistent with the 1977 Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance(FRACQuA) guidance. Additionally, fire drills are held during NML's biannual inspections of Dresden Station.

Fire drills are staged in the plant so the team can practice as a team.

B. Drills should be performed at regular intervals for each shift fire brigade.

<u>Response</u> Consistent with DFPP 4100-01's adaptation of the FRACQuA guidelines, drills are performed at least once per quarter for each shift fire brigade.

C. At least one unannounced fire drill for each shift fire brigade should be performed per year.

<u>Response</u> Consistent with DFPP 4100-01's adaptation of the FRACQuA guidelines, at least one fire drill per year will be unannounced for each shift fire brigade.

D. At least one drill per year should be performed on a "backshift" for each shift fire brigade.

<u>Response</u> Consistent with DFPP 4100-01's adaptation of the FRACQuA guidelines, one fire drill per year will be performed on a "backshift" for each shift fire brigade.

E. Drills should be preplanned to establish training objectives and critiqued to determine how well the training objectives were met.

<u>Response</u> Consistent with DFPP 4100-01's adaptation of the FRACQuA guidelines, drills are preplanned to establish training objectives of the drill. The drills are critiqued to determine how well the objectives are met.

F. At least triennially, an unannounced drill should be performed for and critiqued by qualified individuals, independent of the licensee's staff.

<u>Response</u> As required by DFPP 4100-01 (Fire Protection Program), an inspection and audit of the fire protection program is performed by a qualified fire protection individual once every three years. Included in this audit is development of a fire drill critique.

G. Pre-fire plans should be developed for safety related areas of the plant (at a minimum).

<u>Response</u> Dresden Fire Preplans have been developed for the main power block and plant support structures such as the administration building.

H. The pre-fire plans should be updated and used as part of the brigade training.

<u>Response</u> Fire Preplans are developed and maintained in accordance with DFPP 4100-01, (Fire Protection Program). The brigade is trained on the Fire Preplans in accordance with ACMI. Included in the ACMI Lesson Plan List is training on the Fire Preplans, and, for continuing training, is a lesson plan which includes plant modifications that would impact the Fire Preplans.

I. Fire brigade equipment is maintained in good condition and ready for use by the fire brigade.

<u>Response</u> Brigade equipment is maintained under DFPS 4114-12 (Fire Equipment Cart Inspection), DFPS 4114-14 (MSA SCBA Inspection), and DFPS 4114-16 (ISI Magnum Self Contained Breathing Apparatus Inspection). Fire brigade members are responsible for maintaining their own protective clothing (boots, gloves, turnout coats, etc.) and are trained in their maintenance in accordance with ACMI.

4.10.2.3.6 Fire Brigade Training Records

Records are provided for each fire brigade member, demonstrating the minimum level of training and refresher training has been provided.

Response Training records for individual fire brigade members are maintained as specified in DFPP 4100-01 (Fire Protection Program).

4.10.2.4 Total Environment Equipment Survival

The general issue of total environmental equipment survival centers on the following three (3) areas of interest:

- Adverse effects of combustion products on plant equipment;
- Spurious or inadvertent fire suppression system actuation; and
- Impact on effectiveness of operator actions.

Each of these areas of interest is discussed in detail below.

4.10.2.4.1 Potential Adverse Effects on Plant Equipment by Combustion Products (Smoke)

A. The FIVE methodology does not currently provide for an evaluation of non-thermal environmental effects of smoke on equipment. See Section 4.23.2 of EPRI TR-100370, Fire Induced Vulnerability Evaluation (FIVE) (Ref. 4-4).

Response For the purposes of this evaluation, the potential detrimental short- or long-term effects of combustion products on the ability of safe shutdown equipment to continue to function in smoke filled environments or the evaluation of smoke transport throughout the building is not being considered. The present state of knowledge regarding the actual effects of combustion products is inadequate to allow any specific treatment of the issue at this time. However, the detrimental short-term effects of smoke on equipment are not believed to be significant.

B. Plant staff should be aware of and sensitive to the potential impact of smoke and products of combustion on human performance in safe shutdown operations in application of FIVE.

Response N-GET training, which is required annually, ensures plant operators are trained on the use of SCBA.

4.10.2.4.2 Spurious or Inadvertent Fire Suppression Actuation

Verify that the design of fire suppression systems considers the effects, if appropriate, of inadvertent suppression system actuation and discharge on equipment credited for safe shutdown for concerns such as those discussed in NRC I&E Information Notice 83-41.

Response Suppression effects analyses were performed and as a result modifications were implemented to ensure safety-related equipment could not be damaged from the perils described in IN 83-41.

4.10.2.4.3 Operator Action Effectiveness

A. There are safe shutdown procedures that identify the steps for planned shutdown when necessary, in the event of a fire.

Response Plant shutdown in the event of a fire involving extensive damage is performed in accordance with Dresden procedure DSSP 0010-01 (Determining Safe Shutdown Paths for Extensive Plant Damage) and accompanying DSSP series procedures.

B. Operators should receive training on the safe shutdown procedures.

Response Dresden Reactor Operators receive training on plant safe shutdown procedures under Dresden Lesson Plan ILT 299L-S4

C. If, in performance of these procedures, operators are expected to pass through or perform actual actions in areas that may contain fire or smoke, suitable SCBA equipment and other protective equipment are available for operators to perform their functions.

Response All operators at Dresden receive N-GET training annually which includes use of SCBA and other protective equipment. SCBA equipment is maintained in the main control room, and on safe shutdown and fire equipment carts stored throughout the plant.

4.10.2.5 Control Systems Interactions

This issue centers on the concern that safe shutdown circuits are physically independent of, or can be isolated from, the control room for a fire in the control room fire area.

Response As described in the Fire Protection Report (Appendix R Conformance/ Safe Shutdown Report), safe shutdown circuits which are not independent of the Control Room are manually isolated in the event of a Control Room fire. DSSP 0100-CR defines the operator actions required.

4.10.3 Conclusion

The FRSS issues have been addressed at Dresden Station. The plant fire protection features, staff qualifications and fire fighting equipment support the assumptions made in performing the Dresden fire PRA analyses.

4.11 USI A-45 AND OTHER SAFETY ISSUES

4.11.1 USI A-45 Decay Heat Removal

The Decay Heat Removal (DHR) function is discussed in Section 4.6.4 of the Dresden IPE Submittal Report (Ref. 4-10), and in the Dresden Safe Shutdown Analysis (Ref. 4-8).

During transient events decay heat may be removed via the following systems and associated operator actions:

a) Main Condenser
b) Isolation Condenser (IC) System
c) Automatic Depressurization System Valves (ADSV)
d) High Pressure Coolant Injection (HPCI)
e) Auxiliary Steam Loads (Steam Jet Air Ejectors, Gland Seal Steam, Offgas Preheater, Max Recycle Reboiler)
f) Main Steam Line Drains
g) IC Vent Line
h) Reactor Water Clean-Up (RWCU) in the Recirculation Mode
i) RWCU in the Blowdown Mode
j) Shutdown Cooling System (After the high suction temperature interlock clears.)

The Condensate / Feedwater, Control Rod Drive (CRD), HPCI, Low Pressure Coolant Injection (LPCI) and Core Spray (CS) systems would be used to maintain Reactor Pressure Vessel (RPV) inventory when employing “feed and bleed” or steaming for decay heat removal. Emergency operating procedures provide for alternate injection sources if these systems are not available.

The IC is a passive system requiring only one valve to open for operation. The IC can remove all decay heat produced within minutes of reactor shutdown, and is capable of taking the reactor to cold shutdown. The IC system has enough shell side water to operate for 20 minutes before makeup is needed. Makeup to the IC shell can be provided with multiple sources including clean demineralized water, fire water, and contaminated condensate.

The IC system and selected makeup sources are defined as a part of the safe shutdown paths that are used for a majority of the fire areas at Dresden. The HPCI system is used for the remaining fire areas. The Dresden Safe Shutdown Analysis (SSA) and related procedures document the availability of the DHR by demonstrating availability of equipment needed for IC and HPCI operation. The fire IPEEE models these safe shutdown methods as well as others if available (i.e., subject to availability offsite power). The results of the fire IPEEE demonstrate availability of DHR for any fire leading to a non-LOCA event. That is, no fire scenario leads to a conditional core damage probability of 1.0.

A fire-induced plant trip is postulated to occur on low RPV water level due to an inadvertent open relief valve (IORV). Under such an event, decay heat is removed by providing RPV makeup with HPCI and operating the LPCI and Containment Cooling Service Water (CCSW) systems in the Suppression Pool Cooling (SPC) mode, or by use of the IC. The SSA documents the availability of HPCI, LPCI, CCSW and IC system components as part of the safe shutdown methods. The fire IPEEE models these safe shutdown methods as well as others if available (i.e., subject to the availability of offsite power). The results of the fire IPEEE demonstrate availability of DHR for any fire leading to a IORV event. That is, no fire scenario leads to a conditional core damage probability of 1.0.

In conclusion, it was determined that the DHR will be available, with associated manual actions, following a fire in any location at Dresden. This finding is based on the CDF results for the compartments containing DHR equipment, on the redundancy of the methods and equipment to achieve the DHR function, and on the implementation time required to achieve various stages of the DHR functions.

4.11.2 GI-57 Effects of Fire Protection System Actuation on Safety-Related Equipment

GI-57 was investigated in 1985 with an analysis of the effects of fire suppression system actuation on nuclear safety related equipment in response to IN 83-41. This issues was evaluated again as part of the IPEEE under seismically induced fires and is discussed in Section 4.10.2.1.1 of this report.

4.12 REFERENCES

4-1 Generic Letter 88-20, Supplement 4, "Individual Plant Examination of External Events for Severe Accident Vulnerabilities - 10CFR50.54(f)," United States Nuclear Regulatory Commission, June 28, 1991.

4-2 NUREG-1407, *Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities*, United States Nuclear Regulatory Commission, June 1991.

4-3 NUREG/CR-5088, *Fire Risk Scoping Study*, Sandia National Laboratories, January 1989.

4-4 *Fire-Induced Vulnerability Evaluation (FIVE) Methodology Plant Screening Guide*, Professional Loss Control, EPRI TR-100370, April 1992.

4-5 W. J. Parkinson, et. al., *Fire PRA Implementation Guide*, TR-105928, Electric Power Research Institute, Palo Alto, CA, Final Report, December 1995.

4-6 NUREG/CR-2815, *Probabilistic Safety Assessment Procedures Guide*, Brookhaven National Laboratory, August 1985.

4-7 NUREG/CR-4840, *Recommended Procedures for the Simplified External Event Risk Analyses for NUREG-1150*, Sandia National Laboratories, September 1989.

4-8 Dresden Station Fire Protection Report

4-9 Sargent & Lundy Interactive Cable Engineering (SLICE) Cable Database

4-10 Commonwealth Edison Co., Individual Plant Examination Submittal Report, Revision 01, June 1996.

4-11 Dresden Station transient combustible control procedure, DHP 023-02, FIRE PROTECTION FOR TRANSIENT COMBUSTIBLES.

4-12 Dresden Station housekeeping procedure, DAP 03-12, PLANT CLEANING PROGRAM AND IN-PLANT PERMANENT STORAGE AREAS.

4-13 EPRI NSAC-178L, Fire Events Database for U. S. Nuclear Power Plants, W. Parkinson, et al, Electric Power Research Institute, December 1991.

4-14 J. M. Chavez, *An Experimental Investigation of Internally Ignited Fires in Nuclear Power Plant Control Cabinets, Part I, Cabinet Effects Tests*, NUREG/CR-4527 Vol.1, U. S. Nuclear Regulatory Commission, Washington DC, April 1988.

4-15 J. M. Chavez, S. P. Nowlen, *An Experimental Investigation of Internally Ignited Fires in Nuclear Power Plant Control Cabinets, Part II, Room Effects Tests*, NUREG/CR-4527 Vol. 2, U. S. Nuclear Regulatory Commission, Washington DC, November, 1988.

4-16 NUREG/CR-5384, S.P. Nowlen, *A Summary of Nuclear Power Plant Fire Safety Research at Sandia National Laboratories, 1975 - 1987*, Albuquerque, New Mexico: U.S. Government Printing Office, Sandia National Laboratories, SAND89-1359, December 1989.

4-17 Individual Plant Evaluation Partnership, Dresden Nuclear Power Station Units 2&3, Initiating Events Notebook, Appendix A, Dresden Interfacing Systems LOCA Assessment, Revision 0, February 1993.

4-18 Individual Plant Evaluation Partnership, Dresden Nuclear Power Station Unit 2 and 3 Success Criteria Notebook, Transient Event, Revision 0, February 1993.

4-19 J. M. Heffley (ComEd) letter, JMHLTR #0129.97, to USNRC dated December 30, 1997, "Final Report – Individual Plant Examination of External Events (IPEEE)".

4-20 Fire Risk Analysis Code (FRANC), Version 2.2ax, Science Applications International Corporation.

4-21 Fully Optimized Risk & Reliability Quantification Engine (FORTE), Version 1.1, 1998, Dr. Woo Sik Jung, Korea Power Engineering Company (KOPEC).

4-22 NUREG/CR-2258, M. Kazarians, G. Apostolakis, *Fire Risk Analysis for Nuclear Power Plants*, September 1981.

4-23 NUREG/CR-5603, D. Wesley, et. al., *Pressure-Dependent Fragilities for Piping Components*, October 1990.

ATTACHMENT 2

RESPONSE TO SEISMIC QUESTIONS

NRC Question #1.

In Section 3.4 (crib house masonry walls), Section 3.6 (IPEEE-only relays), and Section 3.8 (Open Items Pending Resolution) of the submittal, ComEd identifies items that had not been evaluated and had been classified as outliers, for tracking purposes. Consequently, the IPEEE submittal is incomplete, and the plant high confidence, low probability of failure (HCLPF) is undetermined at this time.

In addition, the submittal is somewhat confusing with respect to disposition of (1) identified items with HCLPF capacity <0.2g peak ground acceleration (PGA) (e.g., cable tray supports), which are designated " potential design basis issues", and (2) items with HCLPF capacity >0.2g but <0.3g PGA. These items are listed on p.1-3 (and again on p. 3-4) of the submittal.

Please provide the following information in order to establish the plant HCLPF:

(a) Results of the evaluation for the crib house masonry walls, IPEEE-only relays, and "Open Items Pending Resolution".

(b) What, if any, plant improvements have been implemented or are scheduled (please provide schedule) for the items listed in (a) above.

(c) HCLPF capacities, after any improvements, for the items listed in (a) above.

(d) Tabulation of the HCLPF capacities for items listed on p.1-3 of the submittal, following any implemented or scheduled plant improvements. For scheduled improvements, please provide the implementation schedule.

ComEd Response:

(a) Evaluation of the Cribhouse masonry walls is not required per the following:

Cribhouse masonry walls were included in the IPEEE review because equipment such as MCCs, and switchgear associated with the refuse pit pumps are located in the cribhouse, floor elevation 517' 6". The refuse pit pumps and isolation condenser were added to the IPEEE Success Path Equipment List (SPEL) as part of the success path chosen to address decay heat removal in case of a Dresden Lock and Dam failure due to a design basis earthquake. Subsequent to the submittal of the IPEEE report to the NRC, ComEd identified the Unit 2 diesel generator cooling water system as a more viable and reliable success path (ComEd (J. M. Heffley) letter to

USNRC dated September 9, 1998 regarding Failure of the Dresden Lock and Dam). This path would eliminate the use of the refuse pit pumps and the associated electrical equipment. Therefore, the cribhouse walls will not have to remain in the IPEEE program and as such will be deleted.

The IPEEE-only relays had been evaluated in the original submittal as is described in Section 3.6. Four low ruggedness relays per unit were identified and were replaced by modifications.

As described above, the isolation condenser and its associated equipment including relays were added to the SPEL as part of the success path chosen to address decay heat removal in case of a Dresden Lock and Dam failure. Initially, approximately 30% of the relay contacts associated with the isolation condenser have been screened out as chatter acceptable or not vulnerable. Subsequently, an additional 5% of the contacts were found acceptable based on walkdowns and evaluations. The evaluation of the remaining relays is pending walkdowns during an outage to allow access to the associated cabinets.

Appendix F2 is revised to contain only the isolation condenser relays as open items. Appendix F2 is included in Attachment 4, SQUG /IPEEE Relay Screening and Evaluation Tabulation. It should be noted that this list includes the contacts that have been screened out as chatter acceptable or not vulnerable.

The following table from Section 3.8 (Open Items Pending Resolution) of the original submittal is updated to show the evaluation results.

Equipment ID	Description	Outlier Finding	Potential Resolution / Evaluation Result
D02-7820-02-M05, D02-7820-03-M05	Motor Control Centers	Inadequate anchorage, potential masonry wall interaction	Improve anchorage, evaluate potential interaction. **Evaluation not required. Equipment deleted from IPEEE program.**
D02-8501-0005AV05, D02-8501-0005BV05	Air Operated Valves	AOVs supported off 1/2" diameter tubing	Perform more rigorous analysis and/or modify support of valve. **Evaluation qualified ½" tubing. Capacity 0.3g.**

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RESPONSE TO SEISMIC QUESTIONS

Equipment ID	Description	Outlier Finding	Potential Resolution / Evaluation Result
Cable Tray & Conduit	Bus duct supported by threaded rod.	Unknown weight of duct - LAR009	Determine weight and assess supports. **Actual weight of the duct is $<$ 20 lb/ft compared to 98 lb/ft capacity of supports. Capacity $>$0.3g.**
Cable Tray	Cable tray supported off masonry wall	Wall capacity unknown - RACE010	Assess wall for cable tray loading. **Cable tray is found to be not supported off the masonry wall. The wall has been evaluated under 80-11 effort. Capacity $\geq$ 0.2g.**
Remaining Isolation Condenser Relays	Relays	Inaccessible relays need to be walked down	Walk down relays when plant status permits. **Partial walkdowns have been completed. Approximately 35% of the total isolation condenser relay contacts have been evaluated and found acceptable.**
D02-0902-0040-P05, D02-0902-0008-P05, 2-0902-18, D03-0903-0040-P05, 3-0903-18	Control Panels	Walkdowns and assessments not yet performed	Perform walkdowns and assessments. **Assessments have been completed. Capacity $>$ 0.3g.**
D03-7330---S35	480 VAC – SWGR 30	Walkdown and assessment not yet performed	Perform walkdown and assessment. **Evaluation not required. Equipment deleted from IPEEE program.**

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RESPONSE TO SEISMIC QUESTIONS

Equipment ID	Description	Outlier Finding	Potential Resolution / Evaluation Result
D02-2202-0028-P05, D02-2202-0076-P05, D03-2203-0076-P05	Instrument Panels	Walkdowns and assessments not yet performed	Perform walkdowns and assessments. **Assessments have been completed. Capacity > 0.3g.**
D02-1301-0003-V20, D03-1301-0003-V20	Condensate return gate valves	Walkdowns and assessments not yet performed	Perform walkdowns and assessments. **Assessments have been completed. Capacity is 0.3g.**
D02-1340-0002-LI, D03-1340-0002-LI	Level Indicators	Walkdowns and assessments not yet performed	Perform walkdowns and assessments. **Assessments have been completed. Capacity is 0.3g.**
Refuse Pit Pump Piping	Piping	Walkdowns and assessments not yet performed	Perform walkdowns and assessments. **Evaluation not required. Equipment deleted from IPEEE program.**

Items with HCLPF capacity < 0.2g peak ground acceleration (PGA), which were designated as "potential design basis issues" were evaluated under operability determination 97-113. All items were shown to meet or exceed the design basis requirement of 0.2g and hence no longer are considered design basis issues.

As for items with HCLPF capacity >0.2g but < 0.3g, Dresden Station is modifying the anchorage of items which were also USI A-46 outliers, such as Cabinets D02-2252-0010 and D02-2252-0021 and Control Panels D02-0902-0004, 0015, 0017, 0019 and 36, and D03-903-004, 0015, 0017, 0019 and 0036. With modifications, these items will have a HCLPF value of greater than 0.3g PGA. All other items meet Dresden Station's intention to ensure that all IPEEE components have a seismic capacity that complies with design basis requirements. This activity will be completed with the resolution of USI A-46 outliers and adhere to the USI A-46 outlier resolution schedule.

(b) As is seen from the table above, evaluation of all open items other than the isolation condenser relays has been completed, and their HCLPF capacities are found to be ≥ 0.3g PGA except for the masonry wall for which it is ≥ 0.2g PGA, based on its 80-11 evaluation. Therefore, based on Dresden Station's intention to ensure that all IPEEE components have a seismic capacity that complies with design basis requirements, no improvements are required. Upon completion of the relay evaluation, any outliers found will be resolved and if necessary improvements will be implemented. Dresden Station intends to resolve all IPEEE outliers in conjunction with the resolution of USI A-46 outliers and adhere to the approved USI A-46 outlier resolution schedule.

(c) HCLPF capacities for the open items are provided in the table in (a) above.

(d) The following table shows the new HCLPF capacity of items listed on page 1-3 and 1-4 of the original submittal. The new HCLPF capacities are based on additional evaluations and proposed/scheduled improvements.

Original Capacity (pga)	New Capacity (pga)	Description	Basis for New Capacity
0.15g	>0.3g	Cable Trays-Turbine, Reactor & Service Bldgs., El. 517' (GIP LAR 007)	More rigorous evaluation.
0.17g	0.202g	Buses - D03-8303B-----M05, D02-8302B-----M05, D03-8303A---M05, Dist. Panel D03-83125----P06	Additional evaluation.
0.17g	0.27g	Distribution Panel - D02-83125----P06 and Bus D02-8302A---M05	Additional evaluation.
0.17g	>0.3g	Cabinet - D02-2252-0010	Anchorage modification pending.
0.20g	No change	Condensate Storage Tanks - D00-3303-A----T05, D00-3303-B----T05	Original evaluation
0.22g	>0.3g	Control Panels D02-0902-0004, 0015, 0017, 0019 & 0036, D03-0903-0004, 0015, 0017, 0019 & 0036	Modification pending.
0.23g	>0.3g	Control Panels D02-0902-0028 & -0003, D03-0903-0028	Additional evaluation.
0.26g	No change	Diesel Fuel Oil Storage Day Tank D00-5202-T05	Original evaluation.
0.27g	No change	Battery Charger - D02-8300-2A---B05	Original evaluation.
0.27g	No change	Distribution Panels - D02-9802-A & B---P06	Original evaluation.

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Original Capacity (pga)	New Capacity (pga)	Description	Basis for New Capacity
0.27g	No change	Switchgear - D02-7328---S35 & D02-7329----S35	Original evaluation.
0.27g	No change	Bus #2A-1 - D02-8302A1----P06	Original evaluation.
0.27g	No change	125V DC/TB Battery Bus #2 D02-83125-2-P06	Original evaluation.
0.27g	No change	125V DC/Battery Charger #2 D02-8300-2---B05	Original evaluation.
0.28g	No change	125V DC Battery Charger – D03-8300-3A---B05	Original evaluation.
0.28g	No change	Unit 2&3 Torus Suppression Chambers	Original evaluation.
0.28g	>0.3g	Cabinet - D02-2252-0021	Anchorage modification pending.
0.29g	No change	Motor Control Centers D02-83250-----M05 & D02-7826-4---M05	Original evaluation.
0.29g	No change	Bus #2B-1 - D02-8302B-1---P06	Original evaluation.
0.29g	No change	125V DC/TB Res Bus #2 D02-83125-1---P06	Original evaluation.

NRC Question #2

The submittal does not provide numerical details of the seismic loading utilized for the IPEEE nor the original design basis seismic loading. To evaluate the appropriateness of the reported HCLPF capacities, review of this numerical data is necessary.

In addition, ComEd states, without providing a quantitative basis, that equipment screening during the A-46 program is assigned a HCLPF capacity of at least 0.3g PGA.

Please provide the following information to expedite completion of the review:

(a) The ground and in-structure response spectra used in the IPEEE.

(b) The corresponding safe shutdown earthquake (SSE) design basis ground and in-structure response spectra.

(c) If different from (b), the A-46 in-structure response spectra.

(d) A quantitative basis for assigning a HCLPF capacity of at least 0.3g PGA for equipment screened in the A-46 program.

ComEd Response:

(a) The ground and in-structure response spectra used in the IPEEE are included in Attachment 5.

(b) The corresponding safe shutdown earthquake (SSE) design basis ground and in-structure response spectra are included in Attachment 6.

(c) The A-46 program in-structure response spectra are the same as the spectra provided in Attachment 6.

(d) The basis for assigning a HCLPF capacity of 0.3g PGA for equipment screened in the A-46 program is discussed in Section 3.4.4.2 of the submittal. An item of equipment, except for atmospheric storage tanks and equipment supported on vibration isolators, that passes an A-46 evaluation satisfies the requirements for the first earthquake level in Table 2-4 of EPRI NP-6041. Items of equipment meeting the requirements of the first earthquake level as discussed in Appendix A of EPRI NP-6041 are assigned a HCLPF capacity of at least 0.3g PGA as recommended by the Expert Panel on the Quantification of Seismic Margins in NUREG/CR-4334. Further quantification of seismic margins of items, which have been screened out per the guidelines of EPRI NP-6041 would be unnecessary.

NRC Question #3:

The re-evaluation of masonry walls for IPEEE (discussed in Section 3.4.3 of the submittal) utilizes 7% damping, compared to 2% damping in the I.E Bulletin 80-11 evaluation. An increase in the allowable mortar tensile stress from 23 psi to 32 psi is not a sufficient technical basis for this increase in damping, because only an extremely low level of response is achieved in both analyses. Scaling should be limited to the ratio of allowable tensile stress in the mortar.

Please provide the following additional information concerning the masonry wall evaluations for IPEEE:

(a) The HCLPF calculations for the two (2) masonry walls with the lowest capacities, for 7% damping and for 2% damping. Please discuss the implications of using 2% damping on the HCLPF capacities in Table 3.1.

(b) HCLPF capacities for the crib house masonry walls, based on 2% damping. (As noted in Question 1, the evaluation of the crib house masonry walls was left as an open item in the submittal.

ComEd Response:

(a) The use of 7% of critical damping as the estimated "median" damping is in accordance with the EPRI Report NP-6041-SL, *A Methodology for Assessment of Nuclear Power Plant Seismic Margin (Revision 1).*

As background, it is important to understand that 2% damping was conservatively utilized for the original design basis calculations because the site design basis earthquake ground spectrum was based on the Housner spectrum. The use of the Housner spectrum requires lower damping values than those recommended, for example, when using the USNRC Regulatory Guide 1.60 or NUREG/CR-0098, "Development of Criteria for Seismic Review of Selected Nuclear Power Plants," spectra.

The seismic margin is an assessment for capacity "beyond design basis". The Seismic Margin Earthquake (SME) utilized in the Dresden study is the NUREG/CR-0098 spectral shape with a peak ground acceleration (PGA) of 0.30g. Referring to page 2-48 of the NP-6041 report, it is noted that damping value ranges in NUREG/CR-0098 are considered appropriate so long as the structure is significantly stressed by the SME. The NP-6041 report goes on to state that a range of damping from 7% to 10% for concrete structures is appropriate when linear analysis is performed. As the NP-6041 report also states on that same page that:

> "The important point is that the aim should be to use a median damping and not intentionally introduce conservatism at this point".

Since this is not a design basis evaluation, but rather a beyond design basis study with an input SME ground spectrum (NUREG/CR-0098) much greater than the Housner spectrum, the use of 7% damping is justified.

As stated in the IPEEE report, no additional calculations were performed. The original design basis calculations for the resolution of the IEB 80-11 issue for seismic adequacy of masonry block walls were utilized. The seismic (SMA) capacity of the walls was determined using the ratio of the SME at 7% damping to the DBE at 2% damping along with increased mortar allowable based on the ratio of the SME allowable stress to the IEB 80-11 allowable stress.

(b) At the time of the submittal, selected equipment located in the cribhouse were included in the study to mitigate the effects of a nearby Dresden Lock and Dam failure and the resulting loss of cooling water inventory. A conceptual alternate success path has since been developed which will allow removal of that equipment from the success path equipment list. The cribhouse is a Class II structure and contains only the emergency diesel generator cooling water pumps (from the SPEL seismic review list) below grade. Therefore, the performance (structural integrity) of the crib house masonry walls above grade is not an issue. As stated in the UFSAR, Section 3.8.5, the concrete structure (below grade) of the cribhouse would not be affected by tornado and earthquakes.

NRC Question #4

Decay heat removal is achieved in the success paths by the use of the Low Pressure Coolant Injection (LPCI) system in the torus cooling mode with the Containment Cooling Service Water (CCSW) system providing cooling water to the LPCI heat exchangers. The pumps of the CCSW system take suction from Bay 13 of the crib house.

Regarding the availability of an ultimate heat sink, it is stated in the submittal (Section 3.4.2, page 3-20) that "The NRC safety evaluation for SEP Topic 1-4.E concluded that the plant is designed so that it can be safely shut down in the event of failure of the Dresden dam and the loss of the pool impounded by it. Part of the basis for this conclusion was that there is enough water impounded in the intake and discharge canals below their high point elevations to allow a safe shutdown of Dresden Units 2 and 3. Based on the SEP evaluation, the failure of the dam or dike will not impact the ability to safely shut down Unit 2 and 3", and that "Upon dam failure, the water level in Bay 13 is maintained by the screen wash refuse pit pumps. This requires inserting stop logs and some valving."

The above statements are not sufficient to indicate that there is sufficient cooling water to support the success paths identified in the IPEEE. The systems available for a safe shutdown in the IPEEE may not be the same as those used in the safety evaluation for SEP Topic 11-4.E. As discussed above, the only systems included in the IPEEE for DHR are the LPCI in the torus cooling mode and the CCSW system. DHR by isolation condenser or shut down cooling is not available. The service water system and the fire protection system are also not available in the IPEEE.

(a) Please discuss in more detail the effect of dam failure on the success paths. Please provide the analysis results that show that the water remaining in the intake and discharge canals is sufficient to support the operation of the cooling water pumps and the cooling needs for all units after dam failure. Please include in the discussion the potential for (e.g., the HCLPF values) and the effects of failure of other structures that may affect water availability, in particular the walls of the intake and discharge canals.

(b) Please discuss the operator actions required to assure adequate cooling. Please include in the discussion the systems required (e.g., the availability of the screen wash refuse pit pumps and associated components after a seismic margin earthquake, SME), the time allowed for operator actions, the procedures for operator actions, and the consideration of the adverse effect of the seismic event on operator actions.

(c) In a recent letter to the NRC (J. M. Heffley to USNRC letter dated September 9, 1998 regarding Failure of the Dresden Lock & Dam), ComEd states that "to satisfy Generic Letter 87-02 program requirements and enhance plant safety using the Seismic Qualification Utility Group (SQUG) methodology, ComEd searched for a method of supply make-up water to the shell of the isolation condensers through piping and components that are seismically qualified or that can be seismically verified using the Generic Implementation Procedure (GIP) of the SQUG program." If such a plant modification is implemented, this will add a success path for decay heat removal, but torus cooling may still be needed for the small LOCA case required in the IPEEE. Please include in your response the status of the above-mentioned program and its effect on IPEEE results.

It is noted that only equipment in the Success Path Equipment List (SPEL) should be considered as available in the above discussions (e.g., service water and shutdown cooling may not be available) and that concurrent demand of cooling water from all units needs to be considered (e.g., may need the operation of more than one CCSW pumps).

ComEd Response:

Response to Question 4(a):

As noted by the NRC in question 4, the success path identified for decay heat removal on page 3-20 of the subject report was the Low Pressure Coolant Injection (LPCI) system in the torus cooling mode with the Containment Cooling Service Water (CCSW) providing cooling to the LPCI heat exchangers. Following a dam failure, the level in the intake canal will drop to elevation 495'. This elevation is the high point of the invert that exists near the inlet of the intake canal. Because the center line of the CCSW intake pipes is at elevation 498', stop logs must be placed where screens normally exist in the openings to the CCSW intake bay (Bay 13). The screen wash refuse pit pumps would then be used to reflood Bay 13 so that the CCSW pumps could be started.

The refuse pit pumps were included on the Success Path Equipment List (SPEL) and were evaluated as having a seismic capacity of 0.3g. However, the motor control centers and switchgear for these pumps were identified as outliers because of potential interactions with the cribhouse block wall. Because of the high cost to resolve these outliers, the refuse pit pumps, CCSW, and LPCI containment cooling mode will not be used for decay heat removal for the specific case of a dam failure. For a dam failure, the isolation condenser for each unit will be used as the means of decay heat removal. As noted in question 4(c), a modification is being considered to develop a seismically qualified or verified make-up path to supply water from the ultimate heat sink to the shell of the isolation condenser.

Response to Question 4(b):

As discussed in the response to question 4(a), the isolation condenser will be used as the means of decay heat removal following a dam failure in lieu of CCSW and LPCI mode of torus cooling. Operator actions required for the proposed seismically qualified/verified make-up path to the isolation condenser will be submitted to the NRC when they are developed.

Response to Question 4(c):

The concept of a seismically qualified/verified make-up path to the isolation condensers that was submitted to the NRC in a letter dated September 9, 1998 is being developed. The design changes needed to implement this concept will be presented to station committees that review the technical adequacy and the projected cost of major design changes. Before the design changes are presented to the committees, a study will be performed to ensure that a small break LOCA, with no torus cooling but with the isolation condenser in operation, does not result in unacceptable torus temperatures. Unacceptable torus temperatures are those that would challenge structural integrity of the torus or prevent the HPCI system from performing its inventory control function.

The design changes required to implement this concept will be completed in conjunction with the approved schedule for resolution of USI A-46 outliers.

ATTACHMENT 3

RESPONSE TO FIRE QUESTIONS

NRC Question #1

The automatic suppression failure analysis used reliability values from the Fire Induced Vulnerability Evaluation (FIVE) methodology. This data is acceptable for systems that have been designed, installed, and maintained in accordance with appropriate industry standards, such as those published by National Fire Protection Agency (NFPA).

Please verify that automatic fire suppression systems at Dresden meet NFPA standards.

ComEd Response:

The upgraded Dresden fire risk assessment credited the automatic fire suppression systems in the Reactor Feedwater Pump Areas, over the Station Air Compressor in the Unit 2 ground floor switchgear area, over the Instrument Air Compressor in the ground floor Turbine Building common area, and the Unit 3 Cable Tunnel. These and all Dresden Station automatic fire suppression systems comply with applicable NFPA codes of record except where deviations have been identified. Technical justifications have been provided for these deviations. The code compliance reviews identifying deviations and corresponding technical justifications are contained in the Dresden Fire Protection Program Documentation Package.

NRC Question #2:

Fire severity factors were used in the analyses of many fire compartments. At issue is the use of severity factors in scenarios where fire suppression was explicitly credited. The severity factors used appear to be determined, in part, by eliminating fires successfully suppressed from contributing to the fire frequency. Thus, the potential for a large fire that the severity factor represents, is dependent upon the success of fire suppression. There appears to be a significant possibility that the use of a fire severity factor when fire suppression is explicitly modeled credits suppression efforts twice.

Please describe the instances in the Dresden fire assessment in which automatic fire suppression was credited explicitly in conjunction with the use of a fire severity factor. For each case explain why such credit does not constitute redundant credit for suppression. Please reanalyze the core damage frequency (CDF) contribution from each scenario where redundant credit for suppression is identified.

ComEd Response:

The upgraded Dresden fire risk assessment credits automatic fire suppression systems in conjunction with the use of fire severity factors in several areas. The method used to develop the fire severity factors explicitly included criteria to ensure the Generic Fire IPEEE RAI issue related to 'double' credit for suppression did not occur. The application of the severity factor was based on a review of fire incidents in the EPRI Fire Events Database. The methodology for partitioning the fire events in the EPRI Fire Events Database required that any fire that caused the actuation of an automatic fire suppression system be treated as a large, or severe, fire regardless of the actual consequences of the fire event. As such, fires that could have become severe events if the suppression system had failed were properly counted as severe events. As such no reanalysis is required.

NRC Question #3:

It appears that the Dresden IPEEE fire analysis assumed that the plant cables are not IEEE-383 qualified. However, the Dresden fire analysis assumed a cable ignition temperature of 932°F (see page 4-16 of the submittal) and cites the *EPRI Fire PRA Implementation Guide* (FPRAIG) as the basis for this value. This value is significantly optimistic in comparison to piloted ignition temperature observed in tests by Sandia National Laboratories (SNL) (Ref. NUREG/CR-5546). The SNL tests show that the piloted ignition temperature for cables will be as low or lower than the thermal damage threshold; hence, use of a piloted ignition temperature of no greater than 425°F would be appropriate for unqualified cable. The assumed temperature of 932°F may have resulted in the optimistic treatment of cable fire growth behavior.

If a cable ignition temperature of 932°F was used, please describe the fire scenarios, associated cables, and analysis results for those cases in which it was applied. Include a specific basis for the assumption that the cables at Dresden are consistent with this temperature. Alternatively, provide an assessment of the impact on the analysis results (CDF) if it is assumed that the flammability and/or appropriate non-qualified cable damage properties, including a piloted ignition temperature of 425°F, as appropriate.

ComEd Response:

The upgraded Dresden fire risk assessment did not use the 932°F cable ignition temperature value recommended in the EPRI Fire PRA Implementation Guide. Instead, all fire modeling analysis was based on a temperature of 425°F. No further assessment is required.

ATTACHMENT 3

RESPONSE TO FIRE QUESTIONS

NRC Question #4:

In computing the extent of fire propagation and equipment damage for a given scenario, it is important that experimental results not be used out of context. Inappropriate use of experimental results (e.g., employing propagation times specific to a particular cable tray separation to fires involving cable trays with lesser separation (can lead to improper assessments of scenario importance. In one case [R.1], rather than performing fire model calculation and using the results, experimental data from a test performed to model cable tray fire propagation in the absence of an exposure fire was used to model cable response to an exposure fire, which led to over an order of magnitude reduction in predicted fire-induced core damage frequency.

The submittal apparently assumes a fixed fire spread geometry (35%) for at least one cable tray scenario and fixed propagation delay times between the involvement of subsequent stack trays in the fire. The submittal does not provide a basis for expecting the results of limited experimental observation to be reproduced in the plant fire scenarios.

For each fire scenario in which experimental data were used to estimate the rate and extent of fire propagation, please describe the scenario and how the experimental results were used in the analysis. In those cases where the analysis that was used is found to be unjustified, analyze the scenario using FIVE (or a similar methodology) and provide the results (equipment damaged) of these calculations. Indicate which experimental results were used and how they were utilized in the reanalysis, and justify the applicability of these experimental results to the scenario being analyzed. The discussion on results applicability should compare the geometries, ignition sources, fuel type and loadings, ventilation characteristics, and compartment characteristics of the experimental setup(s) with those of the scenario of interest.

ComEd Response:

The upgraded fire analysis has no scenarios in which experimental data were used to estimate the rate and extent of fire propagation. The extent of fire propagation considered in the upgraded fire analysis relied on the fire modeling relationships developed in the EPRI FIVE Methodology. The analysis applied a simplified approach that assumed no delay in fire propagation from the source to targets, except in those cases where suppression system actuation is credited. Refer to Sections 4.4 and 4.6 for additional details.

ATTACHMENT 3

RESPONSE TO FIRE QUESTIONS

NRC Question #5:

The heat loss factor is defined as the fraction of energy released by a fire that is transferred to the enclosure boundaries. This is a key parameter in the prediction of component damage, as it determines the amount of heat available to the hot gas layer. A larger heat loss factor means that a larger amount of heat (due to a more severe fire, a longer burning time, or both) is needed to cause a given temperature rise. It can be seen that if the value assumed for the heat loss factor is unrealistically high, fire scenarios can be improperly screened out. Figure R.1 provides a representative example of how hot gas layer temperature predictions can changes assuming different heat loss factors. Note that: 1) the curves are computed for a 1000 kW fire in a 10m x 5m x 4m compartment with a forced ventilation rate of 1130 cfm; 2) the FIVE-recommended damage temperature for qualified cable is 700°F for qualified cable and 450°F for unqualified cable; and, 3) the SFPE curve in the figure is generated from a correlation provided in the Society for Fire Protection Engineers (SFPE) Handbook [R.2].

Based on evidence provided by a 1982 paper by Cooper et al. [R.3], the *EPRI Fire PRA Implementation Guide* recommends a heat loss factor of 0.94 for fires with durations greater than five minutes and 0.85 for "exposure fires away from a wall and quickly developing hot gas layers." However, as a general statement, this appears to be a misinterpretation of the results. Reference R.3, which documents the results of multi-compartment fire experiments, states that the higher heat loss factors are associated with the movements of the hot gas layer from the burning compartment to adjacent, cooler compartments. Earlier in the experiments, where the hot gas layer is limited to the burning compartment, Reference R.3 reports much lower heat loss factors (on the order of 0.51 to 0.74). These lower heat loss factors are more appropriate when analyzing a single compartment fire.

In summary, (a) hot gas layer predictions are very sensitive to the assumed value of the heat loss factor; and (b) large heat loss factors cannot be justified for a single-room scenarios based on the information referenced in the *EPRI Fire PRA Implementation Guide*.

Figure R.1 Sensitivity of the hot gas layer temperature predictions to the assumed heat loss factor.

For each multi-compartment or single area scenario analyzed where the hot gas layer temperature was calculated, please describe the scenario and specify the heat loss factor value used in the analysis. In light of the preceding discussion, please either. a) justify the value used and discuss its effect on the identification of fire vulnerabilities, or b) repeat the analysis using a more justifiable value *(such as the 0.7 value recommended by the FIVE methodology). Please provide the resulting changes in scenario contributions to core damage frequency.*

ComEd Response:

All single area scenarios assumed a heat loss factor of 0.7 as recommended by the FIVE methodology. The analysis of multi-compartment scenarios, as described in Section 4.7.3, did not require the calculation of a hot gas layer temperature.

NRC Question #6:

The *EPRI Fire PRA Implementation Guide* methodology for evaluating the effectiveness of suppression efforts treats manual recovery of automatic suppression systems as being independent of subsequent manual efforts to suppress the fire. This assumption is optimistic, since the fire conditions (e.g., heat, smoke) that lead to the failure of recovery efforts can also influence the effectiveness of later suppression efforts. Such an approach, therefore, can overlook plant-specific vulnerabilities.

It is important that all relevant factors be considered in an evaluation of the effectiveness of fire suppression. These factors include: (a) the delay between ignition and detector/suppression system actuation (which is specific to the configuration being analyzed); (b) the time-to-damage for the critical component(s) (which is specific to the fuel type and loading as well as to the configuration being modeled); (c) the response time of the fire brigade (which is plant-specific and fire-location-specific); (d) the time required by the fire brigade to diagnose that automatic suppression has failed and to take manual action to recover the automatic suppression system; and, (e) performance shaping factors (PSF) affecting fire brigade actions. These PSFs could include factors such as perseverance (persistent efforts made to recover a failed automatic suppression system), smoke obscuration, and impaired communications [R.1].

Finally, it should be noted that the Nuclear Regulatory Commission (NRC) staff's evaluation of the FIVE methodology [R.4] specifically stated that licensees need to assess the effectiveness of manual fire-fighting teams by using plant-specific data from fire brigade training to determine the response time of the fire fighters.

Please identify those scenarios for which credit is taken for both manual recovery of automatic suppression systems and manual suppression of the fires (if manual recovery efforts are unsuccessful), and please indicate the plant equipment that may be affected by the fires. In the analysis of these scenarios, how are dependencies between manual actions treated? Please justify the treatment, considering the expected fire environment, the recovery actions required, and the manual fire suppression actions required.

ComEd Response:

The upgraded Dresden fire risk assessment has no scenarios in which credit is taken for both manual recovery of automatic suppression systems and manual suppression of the fires. The upgraded assessment did not credit recovery of any automatic fire suppression systems failures. Refer to Sections 4.6 and 4.7 for additional details.

NRC Question #7:

The treatment of manual suppression appears to be derived from the curves that indicate manual suppression success as a function of fire-fighting time. The submittal does not provide a basis for a quantitative assessment of manual suppression effectiveness at Dresden. An acceptable approach to the assessment of manual suppression success compares the damage time to the time required for suppression. The suppression time includes the time to detect the fire, the brigade response time, fire assessment time, and the extinguishment time.

Please provide a comparison of the manual suppression time to the damage time for those compartments where manual suppression was credited. Include in this assessment any adjustments resulting from responses to questions above addressing ignition and damage temperatures, propagation delay assumptions, or model parameters.

ComEd Response:

The upgraded Dresden fire risk assessment did not credit manual suppression in the individual compartment fire assessments. The exception involved the postulated main control room fire wherein operator action to suppress the fire is credited. Refer to Sections 4.6.2 and 4.7.4 for additional details.

NRC Question #8:

Assumptions concerning the effectiveness of unrated fire barriers can have a major impact on the screening of multi-compartment fires. The potential for fire barrier failure due to fires in high-hazard areas (e.g., large spills of oil or other liquid fuel, oil-filled transformers, large turbine fires) can also be important.

a) *Section 4.7.3.3.2 implies that unrated fire barriers have been credited in the fire study's multi-compartment analysis. Please discuss the impact of eliminating the credit for such barriers in the multi-compartment analysis. (In the analysis, a damage temperature of 425°F for unqualified cable* *should be used. If a higher temperature is used, such as the 700°F referenced in Section 4.7.3.3.2 of the submittal, please provide justification.)*

b) *From the discussions provided for multi-compartment fire scenarios, it can not be determined that the impact resulting from all barriers in high hazard fire areas failing has been considered. Please evaluate the effect of such barrier failures and describe the resulting CDF contributions from the associated scenarios.*

ComEd Response:

The multi-compartment analysis and detailed results summary are provided in Section 4.7.3.

a) The multi-compartment, as well as the overall upgraded Dresden fire risk assessment, relies on the fire zones definitions established in the plant Appendix R related studies. These fire zones were examined during the course of the upgraded assessment effort and the adequacy of their boundaries with respect to minimizing the likelihood of fire propagation was considered. While a formal re-examination of the multi-compartment analysis to eliminate credit for unrated barriers has not been performed, three qualitative insights can be readily discerned which indicate that no significant change in the overall conclusions of the multi-compartment analysis would result.

1. The presence of an unrated barrier does not necessarily mean that a fire will readily propagate across the boundary. In many cases, the barrier is unrated because of unsealed openings. The barrier would otherwise be 'rated'. In these cases, the assessment of 'eliminating' credit for the barrier would involve only examining the unsealed opening. In most cases, these openings involve ventilation and piping penetrations.

2. Many of the plant fire compartments do not have a sufficient concentration of combustible materials and/or ignition sources to cause a compartment wide hot gas layer condition to occur. For these compartments, fire modeling would typically show that a target in the adjacent fire compartment would not be affected by a credible fire event.

3. Dresden Station has automatic fire detection and suppression installed in many areas of the plant. These systems would tend to minimize the likelihood of a severe fire developing and challenging the functionality of the compartment boundaries.

b) The upgraded Dresden fire risk assessment modified the multi-compartment analysis such that it includes an initiating event frequency estimation. The treatment of multiple boundary failures is indirectly addressed by these frequency estimates. Refer to Section 4.7.3 for additional details.

NRC Question #9:

Section 4.2 and Appendix C of NUREG-1407, and GL 88-20, Supplement 4 [R.6], request that documentation be submitted with the IPEEE submittal with regard to the Fire Risk Scoping Study (FRSS) [R.7] issues, including the basis and assumptions used to address these issues, and a discussion of the findings and conclusions. NUREG-1407 also requests that evaluation results and potential improvements be specifically highlighted. Control system interactions involving a combination of fire-induced failures and high probability random equipment failures were identified in the FRSS as potential contributors to fire risk.

The issue of control systems interactions is associated primarily with the potential that a fire in the plant (e.g., the main control room (MCR) might lead to potential control systems vulnerabilities. Given a fire in the plant, the likely sources of control systems interactions could happen between the control room, the remote shutdown panels, and shutdown systems. Specific areas that have been identified as requiring attention in the resolution of this issue include:

(a) Electrical independence of the remote shutdown control systems: The primary concern of control systems Interactions occurs at plants that do not provide independent remote shutdown control systems. The electrical independence of the remote shutdown panels and the evaluation of the level of indication and control of remote shutdown control and monitoring circuits need to be assessed.

(b) Loss of control equipment or power before transfer: The potential for loss of control power for certain control circuits as a result of hot shorts and/or blown fuses before transferring control from the MCR to remote shutdown locations needs to be assessed.

(c) Spurious actuation of components leading to component damage, loss-of-coolant accident (LOCA), or interfacing systems LOCA: The spurious actuation of one or more safety-related to safe-shutdown-related components as a result of fire-induced cable faults, hot shorts, or component failures leading to component damage, LOCA, or interfacing systems LOCA, prior to taking control from the remote shutdown panels, needs to be assessed. This assessment also needs to include the spurious starting and running of pumps as well as the spurious repositioning of valves.

(d) Total loss of system function: The potential for total loss of system function as a result of fire-induced redundant component failures or electrical distribution system (power source) failure needs to be described.

Please provide a description of the control and instrumentation functions that are provided at each remote shutdown station. For each such function indicate whether or not it can be isolated from damage in the main control room. Has the IPEEE identified or considered any scenarios that might not be mitigated by the remote stations? Provide an evaluation of the reliability of the remote shutdown stations that includes consideration of spurious actuations that might result from fire-induced cable faults, hot shorts, or component failures. Include in this evaluation the potential for such faults to lead to component damage (including damage to MOVs per Information Notice 92-18), a LOCA or Interfacing system LOCA, prior to taking control from the remote shutdown panels, spurious starting and running of pumps, and repositioning of valves. Describe how your procedures provide for transfer of control to the remote station(s). Provide an evaluation of whether loss of control power due to hot shorts and/or blown fuses could occur prior to transferring control to the remote shutdown locations and identify the risk contribution of these types of failures (if these failures are screened, please provide the basis for the screening).

ComEd Response:

The information requested is in preparation and will be provided under a separate transmittal by July 31, 2000. The risk contribution of failure of the remote shutdown station control and instrumentation functions has not been specifically identified as part of the updated fire risk analysis. Instead, the upgraded assessment assumes a scenario specific conditional failure probability of 1.0 for all potential fire induced cable failure modes (hot shorts, short circuits, and open circuits). A key feature of the new analysis is that spurious equipment actuation is considered for all three failure modes. Therefore the assumption is that the spurious actuation would occur prior to transferring control to the remote shutdown location and would require action to mitigate its effect. The operator actions that would mitigate the effects of these spurious actuations have been given a probability of 0.5.

NRC Question #10:

The submittal indicates that some fire areas contain elements of both units. The general concern is that the CDFs resulting from fires that impact both units could be significant. Except for LOOP, fires that could affect both units were not considered.

For multi-unit sites, there are three issues of potential interest and, for Dresden, a specific concern. Hence, please answer the following:

(a) A fire in a shared area might cause a simultaneous trip demand for more than one unit. This may considerably complicate the response of operators to the fire event, and may create conflicting demands on plant systems which are shared between units.

Please provide the following information regarding this issue: (1) identify all fire areas that are shared between units and the potentially risk-important systems/components for each unit that are housed in each such area, (2) for each area identified in (1), provide an assessment of the associated multi-unit fire risk, (3) for the special case of control rooms, assess the likelihood of a fire or smoke-induced evacuation with subsequent shutdown of both units from remote shutdown panels, and (4) provide an assessment of the risk contribution of any such multi-unit scenario.

(b) At some sites, the safe shutdown path for a given unit may call for cross-connects to a sister unit in the event of certain fires. In the event of a dual unit LOOP at Dresden, the submittal states only that "operator actions incorporate the added risk." The fire analysis should include the unavailability of all cross-connected equipment due to outages at the sister unit (e.g., routine in-service maintenance outages and/or the potential that normally available equipment may be unavailable during extended or refueling outages at the sister unit).

Please provide the following relevant information regarding this issue: (1) indicate whether any fire response safe shutdown procedures call for unit cross-connects, (2) the operator actions associated with these procedures, and (3) if any such cross-connects are required, the impact on fire risk if the total unavailability of the sister unit equipment is included in the assessment.

(c) Propagation of fire, smoke, and suppressants between fire zones containing equipment for one unit to fire zones containing equipment for the other unit also can result in multi-unit scenarios. Hence, the fire assessment for each unit should include analyses of scenarios involving propagation of smoke, fire, and suppressants to and from fire zones containing equipment for the other unit. From the information in the submittal, it is not clear if these types of scenarios are possible.

Please provide an assessment of the risk contribution of any such multi-unit scenarios.

ComEd Response:

All fires that could affect both units have been considered.

a) The upgraded Dresden fire risk assessment consisted of a comprehensive analysis of CDF contributors for both Units 2 and 3. The analysis for each unit examined the entire plant site. For example, the calculation of the Unit 2 CDF contribution due to fires included explicit treatment of the Unit 3 Reactor Feedwater Pump area, as well as all other common and Unit 3 areas. Similarly, the Unit 3 CDF calculation included explicit treatment of all common and Unit 2 areas. This treatment included the consideration that a fire in one unit adversely affects the functionality or ability of an operator to perform actions in the opposite unit. As such, the potential for a multi-unit trip and multi-unit scenarios due to a single fire event is addressed and reported with the CDF for each unit. See Section 4.7 for details.

b) With respect to the issue of cross-connects, there are a limited number of such features credited in the fire risk assessment. These cross-connects consist of designed common shared equipment, such as the common Emergency Diesel Generator, and cases where a designated opposite unit system is credited. In both cases, the fire risk assessment model explicitly treats these features. The logic is structured such that the 'opposite' unit system is credited only if both redundant trains are available. This ensures that should a dual unit trip occur, the 'opposite' unit is capable of responding to that challenge. See Section 4.7 for details.

c) The upgraded Dresden fire risk assessment provides the risk contribution of all multi-unit scenarios. See Section 4.7 for details.

NRC Question #11:

As a result of the seismic/fire walkdown, a hydrogen seal oil control panel and a turbine generator hydrogen monitor were found to be unanchored or inadequately anchored. Hydrogen lines are routed through these cabinets so the potential for a gas release exists. The submittal did not assess the potential risk associated with these lines.

Please evaluate the risk associated with a seismic/fire event due to inadequate seismic anchoring of the hydrogen seal oil control panel and a turbine generator hydrogen monitor. Alternatively, describe any existing systems or procedures *which could mitigate the impact of such hydrogen system failure during a seismic event. Discuss the results, and any plant modifications that might reduce the potential risk as appropriate.*

ComEd Response:

The hydrogen seal oil control panel and the turbine generator hydrogen monitor are being modified such that they will be seismically mounted. Design Change Packages DCP 9900204 and DCP 9900205 have been issued to the Station for installation. DCP 990204 is scheduled for installation beginning April 3, 2000 and DCP 9900205 will be installed prior to the completion of the next Unit 2 Refueling outage (D2R17).

NRC Question #12:

The Dresden procedures for the use of alternate shutdown in the event of main control room abandonment following a fire seems to indicate that the procedures call for reliance on an emergency diesel generator (EDG) whether or not off-site power remains available. Hence, there is a potential that a station blackout (SBO) could result if the controls for the diesel are impacted by fire, the load sequencer fails to perform its function properly, and/or if manual recovery of the diesel generator is compromised by an inability to isolate either the connected load sequencers or the damaged control circuits.

Describe how this aspect of the Dresden alternate shutdown procedures was considered in the IPEEE analysis of fires leading to main control room abandonment and reliance on remote shutdown. If it has not been addressed, provide an assessment of the risk significance of potential SBO scenarios associated with remote shutdown.

ComEd Response:

The upgraded Dresden fire risk assessment credited the Dresden Safe Shutdown Procedures (DSSPs) for only the bounding main control room fire scenario which caused a demand to abandon the control room. It is recognized that the occurrence of such a severe event would place the plant in a configuration wherein the designated safe shutdown success path would involve the systems and plant features explicitly addressed in the DSSPs. The overall reliability of this success path and the potential challenges that are presented given the need to perform all of the necessary action outside of the main control room is treated by the application of a bounding Conditional Core Damage Probability (CCDP) estimate of 0.50. The application of this relatively high failure probability is judged to bound potential uncertainty in the HRA estimation, and the failure probability of the available safe shutdown path equipment including any potential SBO scenarios.

Attachment 4

SQUG/IPEEE Relay Screening and Evaluation Tabulation

SYSTEM ISOCND

SQUG/IPEEE RELAY SCREENING AND EVALUATION TABULATION
DRESDEN NUCLEAR POWER STATION: UNIT 2 OR COMMON

EQUIP ID D02-1001-0001-BV20

MFGR	TYPE	MODEL	CONTACT NO	NOTES	ENER	NO/ NC	SCHEMATIC DWG NO REV	PANEL	BUILDING	ELEVATION	SAT
			OL				12E-2508A R/J	MCC 28-1 (E2)	RB-L/38	517'-6"	
			42/O				12E-2508A R/J	MCC 28-1 (E2)	RB-L/38	517'-6"	
			42/C				12E-2508A R/J	MCC 28-1 (E2)	RB-L/38	517'-6"	
			LS				12E-2508A R/J	D02-1001-0001-BV20	RB	517'-6"	NV
			TS				12E-2508A R/J	D02-1001-0001-BV20	RB	517'-6"	NV
			595-124				12E-2508A R/J	902-4	MCR	534'-0"	CA
GE	HFA	12HFA 151A9F	595-105A				12E-2508 R/P	902-15	MCR	534'-0"	
GE	HFA	12HFA 151A9F	595-105B				12E-2508 R/P	902-17	MCR	534'-0"	
GE	HFA	12HFA 151A9F	595-105C				12E-2508 R/P	902-15	MCR	534'-0"	
GE	HFA	12HFA 151A9F	595-105D				12E-2508 R/P	902-17	MCR	534'-0"	
			595-110A				12E-2508 R/P	902-4	MCR	534'-0"	CA
			595-114				12E-2508 R/P	902-4	MCR	534'-0"	
			595-123				12E-2508 R/P	902-4	MCR	534'-0"	CA
GE			260-12				12E-2508 R/P	902-18	MCR	534'-0"	
GE	CR290	US202A	595-306				12E-2502A R/G	902-5	MCR	534'-0"	NV
GE	SBM	175A9245	595-315B				12E-2508A R/J	902-4	MCR	534'-0"	NV

MOTIVE POWER MCC 28-1

CONTROL POWER MCC 28-1

SYSTEM ISOCND

SQUG/IPEEE RELAY SCREENING AND EVALUATION TABULATION

DRESDEN NUCLEAR POWER STATION: UNIT 2 OR COMMON

EQUIP ID D02-1301-0001-V20

MFGR	TYPE	MODEL	CONTACT NO	NOTES	ENER	NO/ NC	SCHEMATIC DWG NO REV	PANEL	BUILDING	ELEVATION	SAT
GE	SBM		595-310				12E-2507B R/M	902-3	MCR	534'-0"	NV
GE	NEMA	CR109C0	42/O				12E-2507B R/M	MCC 28-1 (J4)	RB-L/38	517'-6"	
GE	NEMA	CR109C0	42/C				12E-2507B R/M	MCC 28-1 (J4)	RB-L/38	517'-6"	
		CR124K028	OL				12E-2507B R/M	MCC 28-1 (J4)	RB-L/38	517'-6"	
GE	HFA	12HFA151A2H	595-116A				12E-2507B R/M	902-40	AEER	517'-6"	
			TS				12E-2507B R/M	D02-1301-0001-V20	RB-DW	570'-0"	NV
			LS				12E-2507B R/M	D02-1301-0001-V20	RB-DW	570'-0"	NV
W	AR		2R1/2A		N	NO/NC	12E-2507B R/M	2202-75	RB-J/39	517'-6"	
			A1		N	NO/NC	12E-2507B R/M	2202-75	RB-J/39	517'-6"	
			N1		Y	NO	12E-2507B R/M	2202-75	RB-J/39	517'-6"	
W	AR		2R1/2B		N	NO/NC	12E-2507B R/M	2202-75	RB-J/39	517'-6"	
			IS				12E-2507B R/M	2202-75	RB-J/39	517'-6"	NV
			SS				12E-2507B R/M	2202-76	RB-N/44	517'-6"	NV
			A4				12E-2507B R/M	2202-75	RB-J/39	517'-6"	CA
			42R/O				12E-2507B R/M	MCC 38-1 (D1)	RB-L/44	517'-6"	
			42R/C				12E-2507B R/M	MCC 38-1 (D1)	RB-L/44	517'-6"	
GE	HFA	12HFA151A2H	595-115A				12E-2506, SHT. 1 R/AK	902-40	AEER	517'-6"	
GE	HFA	12HFA151A2H	595-115B				12E-2506, SHT. 1 R/AK	902-41	AEER	517'-6"	
GE	CR	120A04002AA	595-109A				12E-2506, SHT. 1 R/AK	902-40	AEER	517'-6"	
			1349A				12E-2506, SHT. 1 R/AK	2202-28	RB-M/40	517'-6"	
			1350A				12E-2506, SHT. 1 R/AK	2202-28	RB-M/40	517'-6"	
			595-109B				12E-2506, SHT. 2 R/AH	902-41	AEER	517'-6"	

MOTIVE POWER MCC 28-1

CONTROL POWER MCC 28-1
MCC 38-1

SYSTEM ISOCND

SQUG/IPEEE RELAY SCREENING AND EVALUATION TABULATION
DRESDEN NUCLEAR POWER STATION: UNIT 2 OR COMMON

EQUIP ID D02-1301-0001-V20

MFGR	TYPE	MODEL	CONTACT NO	NOTES	ENER	NO/ NC	SCHEMATIC DWG NO REV	PANEL	BUILDING	ELEVATION	SAT
			1349B				12E-2506, SHT. 2 R/AH	2202-28	RB-M/40	517'-6"	
			1350B				12E-2506, SHT. 2 R/AH	2202-28	RB-M/40	517'-6"	
GE	CR2940	US202A	595-305				12E-2502A, R/G	902-5	MCR	534'-0"	

MOTIVE POWER MCC 28-1

CONTROL POWER MCC 28-1
MCC 38-1

SYSTEM ISOCND

SQUG/IPEEE RELAY SCREENING AND EVALUATION TABULATION

DRESDEN NUCLEAR POWER STATION: UNIT 2 OR COMMON

EQUIP ID D02-1301-0002-V20

MFGR	TYPE	MODEL	CONTACT NO	NOTES	ENER	NO/ NC	SCHEMATIC DWG NO REV	PANEL	BUILDING	ELEVATION	SAT
			595-311				12E-2507A R/S	902-3	MCR	534'-0"	NV
GE	HFA	12HFA151A2H	595-116B				12E-2507A R/S	902-41	AEER	517'-6"	
CUTLER	NEME	6002H352B	42/O				12E-2507A R/S	250VDC MCC 2A (H02)	RB-N/42	570'-0"	
CUTLER	NEME	6002H352B	42/C				12E-2507A R/S	250VDC MCC 2A (H02)	RB-N/42	570'-0"	
CUTLER	NEME	6002H342B	42/M				12E-2507A R/S	250VDC MCC 2A (H02)	RB-N/42	570'-0"	
			LS				12E-2507A R/S	D02-1301-0002-V20	RB-M/41	570'-0"	NV
			TS				12E-2507A R/S	D02-1301-0002-V20	RB-M/41	570'-0"	NV
CUTLER		789	OL				12E-2507A R/S	250VDC MCC 2A (H02)	RB-N/42	570'-0"	
			PB/STOP				12E-2507A R/S	D02-1301-0002-V20	RB-M/41	570'-0"	NV
			PB/OPEN				12E-2507A R/S	D02-1301-0002-V20	RB-M/41	570'-0"	NV
			PB/CLOSE				12E-2507A R/S	D02-1301-0002-V20	RB-M/41	570'-0"	NV
GE	HFA	12HFA151A2H	595-115A				12E-2506, SHT. 2 R/AH	902-40	AEER	517'-6"	
GE	HFA	12HFA151A2H	595-115B				12E-2506, SHT. 2 R/AH	902-41	AEER	517'-6"	
			595-109B				12E-2506, SHT. 2 R/AH	902-41	AEER	517'-6"	
			1349B				12E-2506, SHT. 2 R/AH	2202-28	RB-M/40	517'-6"	
			1350B				12E-2506, SHT. 2 R/AH	2202-28	RB-M/40	517'-6"	
			595-109A				12E-2506, SHT. 1 R/AK	902-41	AEER	517'-6"	
			595-305				12E-2502A, R/G	902-5	AEER	517'-6"	
			1349A				12E-2506, SHT. 1 R/AK	2202-28	RB-M/40	517'-6"	
			1350A				12E-2506, SHT. 1 R/AK	2202-28	RB-M/40	517'-6"	

MOTIVE POWER 250VDC MCC 2A

CONTROL POWER 250VDC MCC 2A

SYSTEM ISOCND

SQUG/IPEEE RELAY SCREENING AND EVALUATION TABULATION
DRESDEN NUCLEAR POWER STATION: UNIT 2 OR COMMON

EQUIP ID D02-1301-0003-V20

MFGR	TYPE	MODEL	CONTACT NO	NOTES	ENER	NO/ NC	SCHEMATIC DWG NO REV	PANEL	BUILDING	ELEVATION	SAT
			595-312				12E-2507 R/AG	902-3	MCR	534'-0"	NV
GE			595-151				12E-2507 R/AG	902-3	MCR	534'-0"	CA
GE		CR2820B	595-117A				12E-2507 R/AG	902-41	AEER	517'-6"	CA
GE	TDR	CR2820B	595-117B		Y	NO	12E-2507 R/AG	902-41	AEER	517'-6"	
GE		CR2820B	595-117C				12E-2507 R/AG	902-41	AEER	517'-6"	CA
GE	TDR	CR2820B	595-117D		Y	NO	12E-2507 R/AG	902-41	AEER	517'-6"	
GE	HFA	12HFA151A2H	595-118A				12E-2507 R/AG	902-41	AEER	517'-6"	CA
GE	HFA-2NC	12HFA151A2H	595-118B		Y	NC	12E-2507 R/AG	902-41	AEER	517'-6"	
GE	HFA	12HFA151A2H	595-116B				12E-2507 R/AG	902-41	AEER	517'-6"	
			TS				12E-2507 R/AG	D02-1301-0003-V20	RB-M/41	545'-6"	NV
			LS				12E-2507 R/AG	D02-1301-0003-V20	RB-M/41	545'-6"	NV
			OL				12E-2507 R/AG	250VDC MCC 2A (H01)	RB-N/42	570'-0"	CA
CUTLER		6002H3	42/O				12E-2507 R/AG	250VDC MCC 2A (H01)	RB-N/42	570'-0"	
			42/C				12E-2507 R/AG	250VDC MCC 2A (H01)	RB-N/42	570'-0"	
			42/M				12E-2507 R/AG	250VDC MCC 2A (H01)	RB-N/42	570'-0"	CA
			PB/OPEN				12E-2507 R/AG	D02-1301-0003-V20	RB-M/41	545'-6"	NV
			PB/CLOSE				12E-2507 R/AG	D02-1301-0003-V20	RB-M/41	545'-6"	NV
			PB/STOP				12E-2507 R/AG	D02-1301-0003-V20	RB-M/41	545'-6"	NV
BARKSDALE	PRESS.S	B2T-M12SSGE	PS-263-53B				12E-2506-2 R/AH	2202-5	RB-E/39	545'-6"	
BARKSDALE	PRESS.S	B2T-M12SS-GE	PS-263-53D				12E-2506-2 R/AH	2202-6	RB-M/41	545'-6"	
			595-312A				12E-2506, SHT. 1 R/AK	902-3	MCR	534'-0"	NV
			263-53A				12E-2506, SHT. 1 R/AK	2202-5	RB-E/39	545'-6"	

MOTIVE POWER 250VAC MCC 2A

CONTROL POWER 125VDC BUS 2B-1

SYSTEM ISOCND

SQUG/IPEEE RELAY SCREENING AND EVALUATION TABULATION
DRESDEN NUCLEAR POWER STATION: UNIT 2 OR COMMON

EQUIP ID D02-1301-0003-V20

MFGR	TYPE	MODEL	CONTACT NO	NOTES	ENER	NO/ NC	SCHEMATIC DWG NO REV	PANEL	BUILDING	ELEVATION	SAT
			595-109B				12E-2506, SHT. 1 R/AK	902-41	AEER	517'-6"	
			595-305				12E-2502A, R/G	902-5	MCR	534'-0"	
			263-53C				12E-2506, SHT. 1 R/AK	2202-6	RB-E/39	545'-6"	
GE	HFA	12HFA151A2H	595-115A				12E-2506, SHT. 2 R/AH	902-40	AEER	517'-6"	
GE	HFA	12HFA151A2H	595-115B				12E-2506, SHT. 2 R/AH	902-41	AEER	517'-6"	
GE	CR	120A04002AA	595-109A				12E-2506, SHT. 1 R/AK	902-40	AEER	517'-6"	
			1349A				12E-2506, SHT. 1 R/AK	2202-28	RB-M/40	517'-6"	
			1350A				12E-2506, SHT. 1 R/AK	2202-28	RB-M/40	517'-6"	
			1349B				12E-2506, SHT. 2 R/AH	2202-28	RB-M/40	517'-6"	
			1350B				12E-2506, SHT. 2 R/AH	2202-28	RB-M/40	517'-6"	

MOTIVE POWER 250VAC MCC 2A

CONTROL POWER 125VDC BUS 2B-1

SYSTEM ISOCND

SQUG/IPEEE RELAY SCREENING AND EVALUATION TABULATION
DRESDEN NUCLEAR POWER STATION: UNIT 2 OR COMMON

EQUIP ID D02-1301-0004-V20

MFGR	TYPE	MODEL	CONTACT NO	NOTES	ENER	NO/ NC	SCHEMATIC DWG NO REV	PANEL	BUILDING	ELEVATION	SAT
			595-313				12E-2507B R/M	902-3	MCR	534'-0"	NV
GE	NEMA	CR109C0	42/O				12E-2507B R/M	MCC 28-1 (J3)	RB-L/38	517'-6"	
GE	NEMA	CR109C0	42/C				12E-2507B R/M	MCC 28-1 (J3)	RB-L/38	517'-6"	
		CR124K028	OL				12E-2507B R/M	MCC 28-1 (J3)	RB-L/38	517'-6"	
GE	HFA	12HFA151A2H	595-116A				12E-2507B R/M	902-40	AEER	517'-6"	
			TS				12E-2507B R/M	D02-1301-0004-V20	RB-DW	589'-0"	NV
			LS				12E-2507B R/M	D02-1301-0004-V20	RB-DW	589'-0"	NV
W	AR		2R4/2A		N	NO/NC	12E-2507B R/M	2202-75	RB-J/39	517'-6"	
			A4		N	NO/NC	12E-2507B R/M	2202-75	RB-J/39	517'-6"	
			N4		Y	NO	12E-2507B R/M	2202-75	RB-J/39	517'-6"	
W	AR		2R4/2B		N	NO/NC	12E-2507B R/M	2202-75	RB-J/39	517'-6"	
			IS				12E-2507B R/M	2202-75	RB-J/39	517'-6"	NV
			SS				12E-2507B R/M	2202-76	RB-N/44	517'-6"	NV
			A1				12E-2507B R/M	2202-75	RB-J/39	517'-6"	CA
GE	HFA	12HFA151A2H	595-115A				12E-2506, SHT. 1 R/AK	902-40	AEER	517'-6"	
GE	HFA	12HFA151A2H	595-115B				12E-2506, SHT. 1 R/AK	902-41	AEER	517'-6"	
GE	CR	120A04002AA	595-109A				12E-2506, SHT. 1 R/AK	902-40	AEER	517'-6"	
			1349A				12E-2506, SHT. 1 R/AK	2202-28	RB-M/40	517'-6"	
			1350A				12E-2506, SHT. 1 R/AK	2202-28	RB-M/40	517'-6"	
			595-109B				12E-2506, SHT. 2 R/AH	902-41	AEER	517'-6"	
			1349B				12E-2506, SHT. 2 R/AH	2202-28	RB-M/40	517'-6"	
			1350B				12E-2506, SHT. 2 R/AH	2202-28	RB-M/40	517'-6"	

MOTIVE POWER MCC 28-1

CONTROL POWER MCC 28-1
MCC 38-1

SYSTEM ISOCND

SQUG/IPEEE RELAY SCREENING AND EVALUATION TABULATION
DRESDEN NUCLEAR POWER STATION: UNIT 2 OR COMMON

EQUIP ID D02-1301-0004-V20

MFGR	TYPE	MODEL	CONTACT NO	NOTES	ENER	NO/ NC	SCHEMATIC DWG NO REV	PANEL	BUILDING	ELEVATION	SAT
			595-305				12E-2502A R/AH	902-5	MCR	534'-0"	NV

MOTIVE POWER MCC 28-1

CONTROL POWER MCC 28-1
MCC 38-1

SYSTEM ISOCND

SQUG/IPEEE RELAY SCREENING AND EVALUATION TABULATION
DRESDEN NUCLEAR POWER STATION: UNIT 2 OR COMMON

EQUIP ID D02-1301-0010-V20

MFGR	TYPE	MODEL	CONTACT NO	NOTES	ENER	NO/ NC	SCHEMATIC DWG NO REV	PANEL	BUILDING	ELEVATION	SAT
			M				12E-2484 R/P	MCC 2B (M01)	RB-N/42	570'-0"	
			O				12E-2484 R/P	MCC 2B (M01)	RB-N/42	570'-0"	
			C				12E-2484 R/P	MCC 2B (M01)	RB-N/42	570'-0"	
			LS				12E-2484 R/P	D02-1301-0010-V20	RB	570'-0"	NV
			TS				12E-2484 R/P	D02-1301-0010-V20	RB	570'-0"	NV
			PB/OPEN				12E-2484 R/P	D02-1301-0010-V20	RB	570'-0"	NV
			PB/CLOSE				12E-2484 R/P	D02-1301-0010-V20	RB	570'-0"	NV
			PB/STOP				12E-2484 R/P	D02-1301-0010-V20	RB	570'-0"	NV
			1301-383				12E-2484 R/P	902-3	MCR	534'-0"	

MOTIVE POWER 250 VDC MCC 2B

CONTROL POWER 250 VDC MCC 2B

SYSTEM ISOCND

SQUG/IPEEE RELAY SCREENING AND EVALUATION TABULATION
DRESDEN NUCLEAR POWER STATION: UNIT 2 OR COMMON

EQUIP ID D02-1301-0017-V27

MFGR	TYPE	MODEL	CONTACT NO	NOTES	ENER	NO/ NC	SCHEMATIC DWG NO REV	PANEL	BUILDING	ELEVATION	SAT
GE	HFA	12HFA151A2H	595-116A				12E-2506, SH. 1 R/AK	902-40	AEER	517'-6"	
GE	HFA	12HFA151A2H	595-111A				12E-2506, SH. 1 R/AK	902-40	AEER	517'-6"	
GE		120A04002AA	595-112A				12E-2506, SH. 1 R/AK	902-40	AEER	517'-6"	
GE	HFA	12HFA151A2H	595-118A				12E-2506, SH. 1 R/AK	902-41	AEER	517'-6"	
GE	HFA	12HFA151A2H	595-118B				12E-2506, SH. 1 R/AK	902-41	AEER	517'-6"	
			595-312				12E-2506, SH. 1 R/AK	902-3	MCR	534'-0"	
			595-314				12E-2506, SH. 1 R/AK	902-3	MCR	534'-0"	
			1301-20R				12E-2506, SH. 1 R/AK	902-62	TB-E/32	549'-0"	
			1301-17R				12E-2506, SH. 1 R/AK	902-61	AEER	517'-6"	
GE	HFA	12HFA151A9F	595-106A				12E-2504, SH. 1 R/T	902-15	MCR	534'-0"	
GE	HFA	12HFA151A9F	595-106B				12E-2504, SH. 1 R/T	902-17	MCR	534'-0"	
GE	HFA	12HFA151A9F	595-106C				12E-2504, SH. 1 R/T	902-15	MCR	534'-0"	
GE	HFA	12HFA151A9F	595-106D				12E-2504, SH. 1 R/T	902-17	MCR	534'-0"	
GE		120A04002AA	595-108A				12E-2504, SH. 1 R/T	902-40	AEER	517'-6"	
GE		CR2820B	595-117B				12E-2506, SH. 2 R/AH	902-41	AEER	517'-6"	
GE		CR2820B	595-117D				12E-2506, SH. 2 R/AH	902-41	AEER	517'-6"	
GE	HFA	12HFA151A2H	595-115A				12E-2506, SH. 1 R/AK	902-40	AEER	517'-6"	
GE	HFA	12HFA151A2H	595-115B				12E-2506, SH. 1 R/AK	902-41	AEER	517'-6"	
GE		CR2820B	595-117A				12E-2506, SH. 1 R/AK	902-41	AEER	517'-6"	
GE		CR2820B	595-117C				12E-2506, SH. 1 R/AK	902-41	AEER	517'-6"	
			1349B				12E-2506, SH. 2 R/AH	2202-28	RB-M/40	517'-6"	
			1350B				12E-2506, SH. 2 R/AH	2202-28	RB-M/40	517'-6"	

MOTIVE POWER INST. BUS PANEL 902-50 CONTROL POWER INST. BUS PANEL 902-50

SYSTEM ISOCND

SQUG/IPEEE RELAY SCREENING AND EVALUATION
TABULATION
DRESDEN NUCLEAR POWER STATION: UNIT 2 OR COMMON

EQUIP ID D02-1301-0017-V27

MFGR	TYPE	MODEL	CONTACT NO	NOTES	ENER	NO/ NC	SCHEMATIC DWG NO REV	PANEL	BUILDING	ELEVATION	SAT
			595-109B				12E-2506, SH. 2 R/AH	902-41	AEER	517'-6"	
GE	CR	120A04002AA	595-109A				12E-2506, SH. 1 R/AK	902-40	AEER	517'-6"	
			1349A				12E-2506, SH. 1 R/AK	2202-28	RB-M/40	517'-6"	
			1350A				12E-2506, SH. 1 R/AK	2202-28	RB-M/40	517'-6"	
			263-53A				12E-2506, SH. 1 R/AK	2202-5(A)	RB-E/39	545'-6"	
			263-53C				12E-2506, SH. 1 R/AK	2202-6(A)	RB-M/41	545'-6"	
			263-53B				12E-2506, SH. 2 R/AH	2202-5(A)	RB-E/39	545'-6"	
			263-53D				12E-2506, SH. 2 R/AH	2202-6(A)	RB-M/41	545'-6"	
			595-304				12E-2502A, R/G	902-5	MCR	534'-0"	NV
			595-305				12E-2502A, R/G	902-5	MCR	534'-0"	NV

MOTIVE POWER INST. BUS PANEL 902-50

CONTROL POWER INST. BUS PANEL 902-50

SYSTEM ISOCND

SQUG/IPEEE RELAY SCREENING AND EVALUATION TABULATION
DRESDEN NUCLEAR POWER STATION: UNIT 2 OR COMMON

EQUIP ID D02-1301-0020-V27

MFGR	TYPE	MODEL	CONTACT NO	NOTES	ENER	NO/ NC	SCHEMATIC DWG NO REV	PANEL	BUILDING	ELEVATION	SAT
GE	HFA	12HFA151A2H	595-116B				12E-2506, SH. 1 R/AK	902-41	AEER	517'-6"	
GE	HFA	12HFA151A2H	595-111B				12E-2506, SH. 1 R/AK	902-41	AEER	517'-6"	
GE	CR	120A04002AA	595-112B				12E-2506, SH. 1 R/AK	902-41	AEER	517'-6"	
GE	HFA	12HFA151A2H	595-118A				12E-2506, SH. 1 R/AK	902-41	AEER	517'-6"	
GE	HFA	12HFA151A2H	595-118B				12E-2506, SH. 1 R/AK	902-41	AEER	517'-6"	
			595-312				12E-2506, SH. 1 R/AK	902-3	MCR	534'-0"	
			595-314				12E-2506, SH. 1 R/AK	902-3	MCR	534'-0"	
			1301-20R				12E-2506, SH. 1 R/AK	902-62	TB-E/32	549'-0"	
			1301-17R				12E-2506, SH. 1 R/AK	902-61	AEER	517'-6"	
GE	HFA	12HFA151A9F	595-106A				12E-2504, SH. 2 R/V	902-15	MCR	534'-0"	
GE	HFA	12HFA151A9F	595-106B				12E-2504, SH. 2 R/V	902-17	MCR	534'-0"	
GE	HFA	12HFA151A9F	595-106C				12E-2504, SH. 2 R/V	902-15	MCR	534'-0"	
GE	HFA	12HFA151A9F	595-106D				12E-2504, SH. 2 R/V	902-17	MCR	534'-0"	
GE	CR	120A04002AA	595-108B				12E-2504, SH. 2 R/V	902-41	AEER	517'-6"	
GE	HFA	12HFA151A2H	595-115A				12E-2506, SH. 2 R/AH	902-40	AEER	517'-6"	
GE	HFA	12HFA151A2H	595-115B				12E-2506, SH. 2 R/AH	902-41	AEER	517'-6"	
GE		CR2820B	595-117A				12E-2506, SH. 1 R/AK	902-41	AEER	517'-6"	
GE		CR2820B	595-117C				12E-2506, SH. 1 R/AK	902-41	AEER	517'-6"	
GE		CR2820B	595-117B				12E-2506, SH. 2 R/AH	902-41	AEER	517'-6"	
GE		CR2820B	595-117D				12E-2506, SH. 2 R/AH	902-41	AEER	517'-6"	
			1349A				12E-2506, SH. 1 R/AK	2202-28	RB-M/40	517'-6"	
			1350A				12E-2506, SH. 1 R/AK	2202-28	RB-M/40	517'-6"	

MOTIVE POWER INST. BUS PANEL 902-50 CONTROL POWER INST. BUS PANEL 902-50

SYSTEM ISOCND

SQUG/IPEEE RELAY SCREENING AND EVALUATION TABULATION

DRESDEN NUCLEAR POWER STATION: UNIT 2 OR COMMON

EQUIP ID D02-1301-0020-V27

MFGR	TYPE	MODEL	CONTACT NO	NOTES	ENER	NO/ NC	SCHEMATIC DWG NO REV	PANEL	BUILDING	ELEVATION	SAT
GE	CR	120A04002AA	595-109A				12E-2506, SH. 1 R/AK	902-40	AEER	517'-6"	
			1349B				12E-2506, SH. 2 R/AH	2202-28	RB-M/40	517'-6"	
			1350B				12E-2506, SH. 2 R/AH	2202-28	RB-M/40	517'-6"	
			595-109B				12E-2506, SH. 2 R/AH	902-41	AEER	517'-6"	
			263-53A				12E-2506, SH. 1 R/AK	2202-5(A)	RB-E/39	545'-6"	
			263-53C				12E-2506, SH. 1 R/AK	2202-6(A)	RB-M/41	545'-6"	
			263-53B				12E-2506, SH. 2 R/AH	2202-5(A)	RB-E/39	545'-6"	
			263-53D				12E-2506, SH. 2 R/AH	2202-6(A)	RB-M/41	545'-6"	
			595-304				12E-2502A, R/G	902-5	MCR	534'-0"	
			595-305				12E-2502A, R/G	902-5	MCR	534'-0"	

MOTIVE POWER INST. BUS PANEL 902-50

CONTROL POWER INST. BUS PANEL 902-50

SYSTEM ISOCND

SQUG/IPEEE RELAY SCREENING AND EVALUATION
TABULATION
DRESDEN NUCLEAR POWER STATION: UNIT 3

EQUIP ID D03-1001-0001-BV20

MFGR	TYPE	MODEL	CONTACT NO	NOTES	ENER	NO/ NC	SCHEMATIC DWG NO REV	PANEL	BUILDING	ELEVATION	SAT
			42/O				12E-3508A R/L	MCC 38-1 (D3)	RB-L/44	517'-6"	
			42/C				12E-3508A R/L	MCC 38-1 (D3)	RB-L/44	517'-6"	
			LS				12E-3508A R/L	D03-1001-0001-BV20	RB	517'-6"	NV
			TS				12E-3508A R/L	D03-1001-0001-BV20	RB	517'-6"	NV
			595-124				12E-3508A R/L	903-4	MCR	534'-0"	CA
GE	HFA	12HFA151A9F	595-105A				12E-3508 R/F	903-15	MCR	534'-0"	
GE	HFA	12HFA151A9F	595-105B				12E-3508 R/F	903-17	MCR	534'-0"	
GE	HFA	12HFA151A9F	595-105C				12E-3508 R/F	903-15	MCR	534'-0"	
GE	HFA	12HFA151A9F	595-105D				12E-3508 R/F	903-17	MCR	534'-0"	
			595-114				12E-3508 R/F	903-4	MCR	534'-0"	
			595-110A				12E-3508 R/F	903-4	MCR	534'-0"	CA
			595-123				12E-3508 R/F	903-4	MCR	534'-0"	CA
GE			260-12				12E-3508 R/F	903-18	MCR	534'-0"	
			595-306				12E-3502A R/J	903-5	MCR	534'-0"	NV
			595-315B				12E-3508A R/L	903-4	MCR	534'-0"	NV
			OL				12E-3508A R/L	MCC 38-1 (D3)	RB-L/44	517'-6"	NV

MOTIVE POWER MCC 38-1

CONTROL POWER MCC 38-1

SYSTEM ISOCND

SQUG/IPEEE RELAY SCREENING AND EVALUATION
TABULATION
DRESDEN NUCLEAR POWER STATION: UNIT 3

EQUIP ID D03-1301-0001-V20

MFGR	TYPE	MODEL	CONTACT NO	NOTES	ENER	NO/NC	SCHEMATIC DWG NO REV	PANEL	BUILDING	ELEVATION	SAT
			42/O				12E-3507B R/M	MCC 38-1 (H2)	RB-L/44	517'-6"	
			42/C				12E-3507B R/M	MCC 38-1 (H2)	RB-L/44	517'-6"	
			OL				12E-3507B R/M	MCC 38-1 (H2)	RB-L/44	517'-6"	
GE	HFA	12HFA151A2H	595-116A				12E-3507B R/M	903-40	AEER	517'-6"	
			TS				12E-3507B R/M	D03-1301-0001-V20	RB-DW	570'-0"	NV
			LS				12E-3507B R/M	D03-1301-0001-V20	RB-DW	570'-0"	NV
W	AR		3R1/2A		N	NO/NC	12E-3507B R/M	2203-75	RB-H/45	517'-6"	
			A1		N	NO/NC	12E-3507B R/M	2203-75	RB-H/45	517'-6"	
			N1		Y	NO	12E-3507B R/M	2203-75	RB-H/45	517'-6"	
W	AR		3R1/2B		N	NO/NC	12E-3507B R/M	2203-75	RB-H/45	517'-6"	
			IS				12E-3507B R/M	2203-75	RB-H/45	517'-6"	NV
			SS				12E-3507B R/M	2203-76	RB-N/44	517'-6"	NV
			A4				12E-3507B R/M	2203-75	RB-H/45	517'-6"	CA
GE	HFA	12HFA151A2H	595-115A				12E-3506, SHT. 1 R/Z	903-40	AEER	517'-6"	
GE	HFA	12HFA151A2H	595-115B				12E-3506, SHT. 1 R/Z	903-41	AEER	517'-6"	
AGA		E7022PB002	595-115C				12E-3506, SHT. 1 R/Z	903-40	AEER	517'-6"	
AGA		E7022PB002	595-115D				12E-3506, SHT 2 R/AA	903-41	AEER	517'-6"	
			42R/0				12E-3507B R/M	MCC 38-1 (K4)	RB-L/44	517'-6"	
			42R/C				12E-3507B R/M	MCC 38-1 (K4)	RB-L/44	517'-6"	
GE	SBM		595-310				12E-3507B R/M	903-3	MCR	534'-0"	NV

MOTIVE POWER MCC 38-1

CONTROL POWER MCC 38-1
MCC 28-1

SYSTEM ISOCND

SQUG/IPEEE RELAY SCREENING AND EVALUATION
TABULATION
DRESDEN NUCLEAR POWER STATION: UNIT 3

EQUIP ID D03-1301-0002-V20

MFGR	TYPE	MODEL	CONTACT NO	NOTES	ENER	NO/ NC	SCHEMATIC DWG NO REV	PANEL	BUILDING	ELEVATION	SAT
			595-311				12E-3507A R/R	903-3	MCR	534'-0"	NV
GE	HFA	12HFA151A2H	595-116B				12E-3507A R/R	903-41	AEER	517'-6"	
			O				12E-3507A R/R	250VDC MCC 3A (H02)	RB-N/47	570'-0"	
			C				12E-3507A R/R	250VDC MCC 3A (H02)	RB-N/47	570'-0"	
			M				12E-3507A R/R	250VDC MCC 3A (H02)	RB-N/47	570'-0"	
			LS				12E-3507A R/R	D03-1301-0002-V20	RB-M/47	570'-0"	NV
			TS				12E-3507A R/R	D03-1301-0002-V20	RB-M/47	570'-0"	NV
			OL				12E-3507A R/R	250VDC MCC 3A (H02)	RB-N/47	570'-0"	
			PB/OPEN				12E-3507A R/R	D03-1301-0002-V20	RB-M/47	570'-0"	NV
			PB/CLOSE				12E-3507A R/R	D03-1301-0002-V20	RB-M/47	570'-0"	NV
			PB/STOP				12E-3507A R/R	D03-1301-0002-V20	RB-M/47	570'-0"	NV
GE	HFA	12HFA151A2H	595-115A				12E-3506, SHT 2 R/AA	903-40	AEER	517'-6"	
GE	HFA	12HFA151A2H	595-115B				12E-3506, SHT 2 R/AA	903-41	AEER	517'-6"	
		E7022PB002	595-115D				12E-3506, SHT 2 R/AA	903-41	AEER	517'-6"	
AGA		E7022PB002	595-115C				12E-3506, SHT 1 R/Z	903-40	AEER	517'-6"	

MOTIVE POWER 250VDC MCC 3A

CONTROL POWER 250VDC MCC 3A

SYSTEM ISOCND

SQUG/IPEEE RELAY SCREENING AND EVALUATION

TABULATION

DRESDEN NUCLEAR POWER STATION: UNIT 3

EQUIP ID D03-1301-0003-V20

MFGR	TYPE	MODEL	CONTACT NO	NOTES	ENER	NO/ NC	SCHEMATIC DWG NO REV	PANEL	BUILDING	ELEVATION	SAT
AGA			595-312				12E-3507 R/X	903-3	MCR	534'-0"	NV
			595-151				12E-3507 R/X	903-3	MCR	534'-0"	CA
GE		CR2820B	595-117A				12E-3507 R/X	903-41	AEER	517'-6"	CA
GE	TDR	CR2820	595-117B		Y	NO	12E-3507 R/X	903-41	AEER	517'-6"	
GE		CR2820B	595-117C				12E-3507 R/X	903-41	AEER	517'-6"	CA
GE	TDR	CR2820	595-117D		Y	NO	12E-3507 R/X	903-41	AEER	517'-6"	
GE	HFA	12HFA151A2H	595-118A				12E-3507 R/X	903-41	AEER	517'-6"	CA
GE	HFA-2NC	12HFA151A2H	595-118B		Y	NC	12E-3507 R/X	903-41	AEER	517'-6"	
GE	HFA	12HFA151A2H	595-116B				12E-3507 R/X	903-41	AEER	517'-6"	CA
			TS				12E-3507 R/X	D03-1301-0003-V20	RB-M/47	545'-6"	NV
			LS				12E-3507 R/X	D03-1301-0003-V20	RB-M/47	545'-6"	NV
			OL				12E-3507 R/X	250VDC MCC 3A (H01)	RB-N/47	570'-0"	CA
			42/O				12E-3507 R/X	250VDC MCC 3A (H01)	RB-N/47	570'-0"	
			42/C				12E-3507 R/X	250VDC MCC 3A (H01)	RB-N/47	570'-0"	
			42/M				12E-3507 R/X	250VDC MCC 3A (H01)	RB-N/47	570'-0"	CA
			PB/STOP				12E-3507 R/X	D03-1301-0003-V20	RB-M/47	545'-6"	NV
			PB/OPEN				12E-3507 R/X	D03-1301-0003-V20	RB-M/47	545'-6"	NV
			PB/CLOSE				12E-3507 R/X	D03-1301-0003-V20	RB-M/47	545'-6"	NV
BARKSDALE	PRESS.S	B2T-M12SS GE	PS-263-53B				12E-3506-2 R/AA	2203-5	RB-M/47	545'-6"	
BARKSDALE	PRESS.S	B2T-M12SS-TC	PS-263-53D				12E-3506-2 R/AA	2203-6	RB-K/49	545'-6"	
			595-312A				12E-3506, SHT. 3 R/AB	903-3	MCR	534'-0"	NV
			263-53A				12E-3506, SHT. 1 R/Z	2203-5	RB-M/47	545'-6"	

MOTIVE POWER 250VDC MCC 3A

CONTROL POWER 125VDC BUS 3B-1

SYSTEM ISOCND

SQUG/IPEEE RELAY SCREENING AND EVALUATION
TABULATION
DRESDEN NUCLEAR POWER STATION: UNIT 3

EQUIP ID D03-1301-0003-V20

MFGR	TYPE	MODEL	CONTACT NO	NOTES	ENER	NO/ NC	SCHEMATIC DWG NO REV	PANEL	BUILDING	ELEVATION	SAT
			595-109B				12E-3506, SHT. 1 R/Z	903-41	AEER	517'-6"	CA
			263-53C				12E-3506, SHT. 1 R/Z	2203-6	RB-K/49	545'-6"	
			595-305				12E-3502A R/Z	903-5	MCR	534'-0"	NV
GE	HFA	12HFA151A2H	595-115A				12E-3506, SHT. 2 R/AA	903-40	AEER	517'-6"	CA
GE	HFA	12HFA151A2H	595-115B				12E-3506, SHT. 2 R/AA	903-41	AEER	517'-6"	CA
AGA		E7022PB002	595-115D				12E-3506, SHT. 2 R/AA	903-41	AEER	517'-6"	CA
AGA		E7022PB002	595-115C				12E-3506, SHT. 1 R/Z	903-40	AEER	517'-6"	CA

MOTIVE POWER 250VDC MCC 3A

CONTROL POWER 125VDC BUS 3B-1

SYSTEM ISOCND

SQUG/IPEEE RELAY SCREENING AND EVALUATION

TABULATION

. DRESDEN NUCLEAR POWER STATION: UNIT 3

EQUIP ID D03-1301-0004-V20

MFGR	TYPE	MODEL	CONTACT NO	NOTES	ENER	NO/ NC	SCHEMATIC DWG NO REV	PANEL	BUILDING	ELEVATION	SAT
			42/O				12E-3507B R/M	MCC 38-1 (H1)	RB-L/44	517'-6"	
			42/C				12E-3507B R/M	MCC 38-1 (H1)	RB-L/44	517'-6"	
			OL				12E-3507B R/M	MCC 38-1 (H1)	RB-L/44	517'-6"	
GE	HFA	12HFA151A2H	595-116A				12E-3507B R/M	903-40	AEER	517'-6"	
			TS				12E-3507B R/M	D03-1301-0004-V20	RB-DW	589'-0"	NV
			LS				12E-3507B R/M	D03-1301-0004-V20	RB-DW	589'-0"	NV
W	AR		3R4/2A		N	NO/NC	12E-3507B R/M	2203-75	RB-H/45	517'-6"	
			A4		N	NO/NC	12E-3507B R/M	2203-75	RB-H/45	517'-6"	
			N4		Y	NO	12E-3507B R/M	2203-75	RB-H/45	517'-6"	
W	AR		3R4/2B		N	NO/NC	12E-3507B R/M	2203-75	RB-H/45	517'-6"	
			IS				12E-3507B R/M	2203-75	RB-H/45	517'-6"	NV
			SS				12E-3507B R/M	2203-76	RB-N/44	517'-6"	NV
			A1				12E-3507B R/M	2203-75	RB-H/45	517'-6"	CA
			42R/O				12E-3507B R/M	MCC 38-1 (K4)	RB-L/44	517'-6"	
			42R/C				12E-3507B R/M	MCC 38-1 (K4)	RB-L/44	517'-6"	
GE	HFA	12HFA151A2H	595-115A				12E-3506, SHT. 1 R/Z	903-40	AEER	517'-6"	CA
GE	HFA	12HFA151A2H	595-115B				12E-3506, SHT. 1 R/Z	903-41	AEER	517'-6"	CA
GE		E7022PB002	595-115C				12E-3506, SHT. 1 R/Z	903-40	AEER	517'-6"	CA
GE		E7022PB002	595-115D				12E-3506, SHT. 2 R/AA	903-41	AEER	517'-6"	CA
			595-313				12E-3507B R/M	903-3	MCR	534'-0"	NV

MOTIVE POWER MCC 38-1

CONTROL POWER MCC 38-1
MCC 28-1

SYSTEM ISOCND

SQUG/IPEEE RELAY SCREENING AND EVALUATION
TABULATION
DRESDEN NUCLEAR POWER STATION: UNIT 3

EQUIP ID D03-1301-0010-V20

MFGR	TYPE	MODEL	CONTACT NO	NOTES	ENER	NO/ NC	SCHEMATIC DWG NO REV	PANEL	BUILDING	ELEVATION	SAT
			O				12E-3484 R/M	MCC 3B (M01)	RB-N/47	570'-0"	
			C				12E-3484 R/M	MCC 3B (M01)	RB-N/47	570'-0"	
			LS				12E-3484 R/M	D03-1301-0010-V20	RB	570'-0"	NV
			TS				12E-3484 R/M	D03-1301-0010-V20	RB	570'-0"	NV
			PB/OPEN				12E-3484 R/M	D03-1301-0010-V20	RB	570'-0"	NV
			PB/CLOSE				12E-3484 R/M	D03-1301-0010-V20	RB	570'-0"	NV
			HS/STOP				12E-3484 R/M	D03-1301-0010-V20	RB	570'-0"	NV
			1301-383				12E-3484 R/M	903-3	MCR	534'-0"	
			M				12E-3484 R/M	MCC 3B (M01)	RB-N/47	570'-0"	

MOTIVE POWER 250VDC MCC 3B

CONTROL POWER 250VDC MCC 3B

SYSTEM ISOCND

SQUG/IPEEE RELAY SCREENING AND EVALUATION TABULATION

DRESDEN NUCLEAR POWER STATION: UNIT 3

EQUIP ID D03-1301-0017-V27

MFGR	TYPE	MODEL	CONTACT NO	NOTES	ENER	NO/ NC	SCHEMATIC DWG NO REV	PANEL	BUILDING	ELEVATION	SAT
GE	HFA	12HFA151A2H	595-116A				12E-3506, SH. 3 R/AB	903-40	AEER	517'-6"	
GE	HFA	12HFA151A2H	595-111A				12E-3506, SH. 3 R/AB	903-40	AEER	517'-6"	
GE		120A04002AA	595-112A				12E-3506, SH. 3 R/AB	903-40	AEER	517'-6"	
GE	HFA	12HFA151A2H	595-118A				12E-3506, SH. 3 R/AB	903-41	AEER	517'-6"	
GE	HFA	12HFA151A2H	595-118B				12E-3506, SH. 3 R/AB	903-41	AEER	517'-6"	
			595-312				12E-3506, SH. 3 R/AB	903-3	MCR	534'-0"	
			595-314				12E-3506, SH. 3 R/AB	903-3	MCR	534'-0"	
GE	HGA	12HGA17S63	1301-20R				12E-3506, SH. 3 R/AB	903-62	AEER	517'-6"	
			1301-17R				12E-3506, SH. 3 R/AB	903-61	TB-G/31	549-0"	
GE	HFA	12HFA151A9F	595-106A				12E-3504, SH. 1 R/R	903-15	MCR	534'-0"	
GE	HFA	12HFA151A9F	595-106B				12E-3504, SH. 1 R/R	903-17	MCR	534'-0"	
GE	HFA	12HFA151A9F	595-106C				12E-3504, SH. 1 R/R	903-15	MCR	534'-0"	
GE	HFA	12HFA151A9F	595-106D				12E-3504, SH. 1 R/R	903-17	MCR	534'-0"	
GE	HFA	12HFA151A2H	595-115A				12E-3506, SH. 1 R/Z	903-40	AEER	517'-6"	
GE	HFA	12HFA151A2H	595-115B				12E-3506, SH. 1 R/Z	903-41	AEER	517'-6"	
AGA		E7022PB002	595-115D				12E-3506, SH. 2 R/AA	903-41	AEER	517'-6"	
GE	CR	120A04002AA	595-108A				12E-3504, SH. 1 R/R	903-40	AEER	517'-6"	
GE		CR2820B	595-117A				12E-3506, SH. 1 R/Z	903-41	AEER	517'-6"	
GE		CR2820B	595-117C				12E-3506, SH. 1 R/Z	903-41	AEER	517'-6"	
GE		CR2820B	595-117B				12E-3506, SH. 2 R/AA	903-41	AEER	517'-6"	
GE		CR2820B	595-117D				12E-3506, SH. 2 R/AA	903-41	AEER	517'-6"	
AGA		E7022PB002	595-115C				12E-3506, SH. 1 R/Z	903-40	AEER	517'-6"	

MOTIVE POWER INST. BUS PANEL 903-50 CONTROL POWER INST. BUS PANEL 903-50

SYSTEM ISOCND

SQUG/IPEEE RELAY SCREENING AND EVALUATION
TABULATION
DRESDEN NUCLEAR POWER STATION: UNIT 3

EQUIP ID D03-1301-0017-V27

MFGR	TYPE	MODEL	CONTACT NO	NOTES	ENER	NO/ NC	SCHEMATIC DWG NO REV	PANEL	BUILDING	ELEVATION	SAT
			263-53A				12E-3506, SH. 1 R/Z	2203-5	RB-M/47	545'-6"	
			595-109B				12E-3506, SH. 1 R/Z	903-41	AEER	517'-6"	
			263-53C				12E-3506, SH. 1 R/Z	2203-6	RB-K/49	545'-6"	
			263-53B				12E-3506, SH. 2 R/AA	2203-5	RB-M/47	545'-6"	
			263-53D				12E-3506, SH. 2 R/AA	2203-6	RB-K/49	545'-6"	
			595-304				12E-3502A, R/J	903-5	MCR	534'-0"	
			595-305				12E-3502A, R/J	903-5	MCR	534'-0"	

MOTIVE POWER INST. BUS PANEL 903-50

CONTROL POWER INST. BUS PANEL 903-50

SYSTEM ISOCND

SQUG/IPEEE RELAY SCREENING AND EVALUATION
TABULATION
DRESDEN NUCLEAR POWER STATION: UNIT 3

EQUIP ID D03-1301-0020-V27

MFGR	TYPE	MODEL	CONTACT NO	NOTES	ENER	NO/ NC	SCHEMATIC DWG NO REV	PANEL	BUILDING	ELEVATION	SAT
GE	HFA	12HFA151A2H	595-116B				12E-3506, SH. 3 R/AB	903-41	AEER	517'-6"	
GE	HFA	12HFA151A2H	595-111B				12E-3506, SH. 3 R/AB	903-41	AEER	517'-6"	
GE		120A04002AA	595-112B				12E-3506, SH. 3 R/AB	903-41	AEER	517'-6"	
GE	HFA	12HFA151A2H	595-118A				12E-3506, SH. 3 R/AB	903-41	AEER	517'-6"	
GE	HFA	12HFA151A2H	595-118B				12E-3506, SH. 3 R/AB	903-41	AEER	517'-6"	
			595-312				12E-3506, SH. 3 R/AB	903-3	MCR	534'-0"	
			595-314				12E-3506, SH. 3 R/AB	902-3	MCR	534'-0"	
GE	HGA	12HGA17S63	1301-20R				12E-3506, SH. 3 R/AB	903-62	AEER	517'-6"	
			1301-17R				12E-3506, SH. 3 R/AB	903-61	TB-G/31	549'-0"	
GE	HFA	12HFA151A2H	595-115A				12E-3506, SH. 2 R/AA	903-40	AEER	517'-6"	
GE	HFA	12HFA151A2H	595-115B				12E-3506, SH. 2 R/AA	903-41	AEER	517'-6"	
AGA		E7022PB002	595-115D				12E-3506, SH. 2 R/AA	903-41	AEER	517'-6"	
AGA		E7022PB002	595-115C				12E-3506, SH. 1 R/Z	903-40	AEER	517'-6"	
GE	HFA	12HFA151A9F	595-106A				12E-3504A, SH. 1 R/Q	903-15	MCR	534'-0"	
GE	HFA	12HFA151A9F	595-106B				12E-3504A, SH. 1 R/Q	903-17	MCR	534'-0"	
GE	HFA	12HFA151A9F	595-106C				12E-3504A, SH. 1 R/Q	903-15	MCR	534'-0"	
GE	HFA	12HFA151A9F	595-106D				12E-3504A, SH. 1 R/Q	903-17	MCR	534'-0"	
GE		120A04002AA	595-108B				12E-3504A, SH. 1 R/Q	903-41	AEER	517'-6"	
GE		CR2820B	595-117A				12E-3506, SH. 1 R/Z	903-41	AEER	517'-6"	
		CR2820B	595-117C				12E-3506, SH. 1 R/Z	903-41	AEER	517'-6"	
			263-53A				12E-3506, SH. 1 R/Z	2203-5	RB-M/47	545'-6"	
GE		120A04002AA	595-109B				12E-3506, SH. 1 R/Z	903-41	AEER	517'-6"	

MOTIVE POWER INST. BUS PANEL 903-50 CONTROL POWER INST. BUS PANEL 903-50

SYSTEM ISOCND

SQUG/IPEEE RELAY SCREENING AND EVALUATION

TABULATION

DRESDEN NUCLEAR POWER STATION: UNIT 3

EQUIP ID D03-1301-0020-V27

MFGR	TYPE	MODEL	CONTACT NO	NOTES	ENER	NO/ NC	SCHEMATIC DWG NO REV	PANEL	BUILDING	ELEVATION	SAT
			263-53C				12E-3506, SH. 1 R/Z	2203-6	RB-K/49	545'-6"	
GE		CR2820B	595-117B				12E-3506, SH. 2 R/AA	903-41	AEER	517'-6"	
GE		CR2820B	595-117D				12E-3506, SH. 2 R/AA	903-41	AEER	517'-6"	
			263-53B				12E-3506, SH. 2 R/AA	2203-5	RB-M/47	545'-6"	
			263-53D				12E-3506, SH. 12 R/AA	2203-6	RB-K/49	545'-6"	
			595-304				12E-3502A, R/7	903-5	MCR	534'-0"	
			595-305				12E-3502A, R/7	903-5	MCR	534'-0"	

MOTIVE POWER INST. BUS PANEL 903-50

CONTROL POWER INST. BUS PANEL 903-50

Attachment 5

Ground and In-structure Spectra for Dresden Nuclear Power Station Units 2 and 3

CALC NO. 8.11.6-2

SAFETY RELATED

PROJECT 9630-16 REV 0

A2-1

SARGENT & LUNDY ENGINEERS

05/04/95

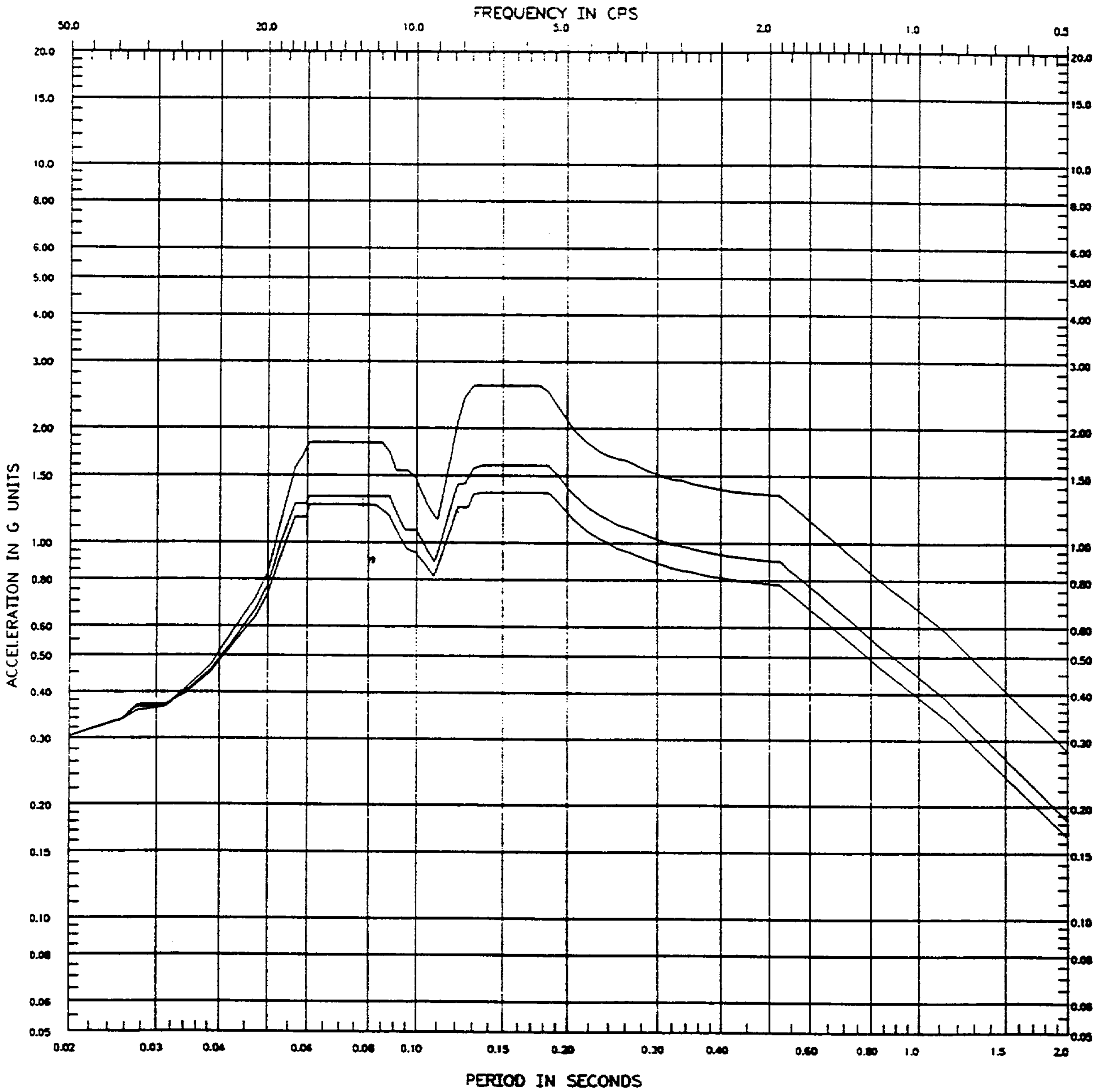
SME SPECTRA - Widened

NODE 9

PEAKS WIDENED ON EACH SIDE BY 15 %

NS DI DAMPING 0.005 0.03 0.05

Dresden Units 2&3



		SPECTRA NO.	SME
NODE	9	ELEVATION	517
DIRECTION	NS	LOCATION	REACTOR BUILDING

Envelope For Entire Floor

6 :

CALC NO. 8.11.6-2

A2-2

SAFETY RELATED

SARGENT & LUNDY ENGINEERS

PROJECT 9630-16 REV 0

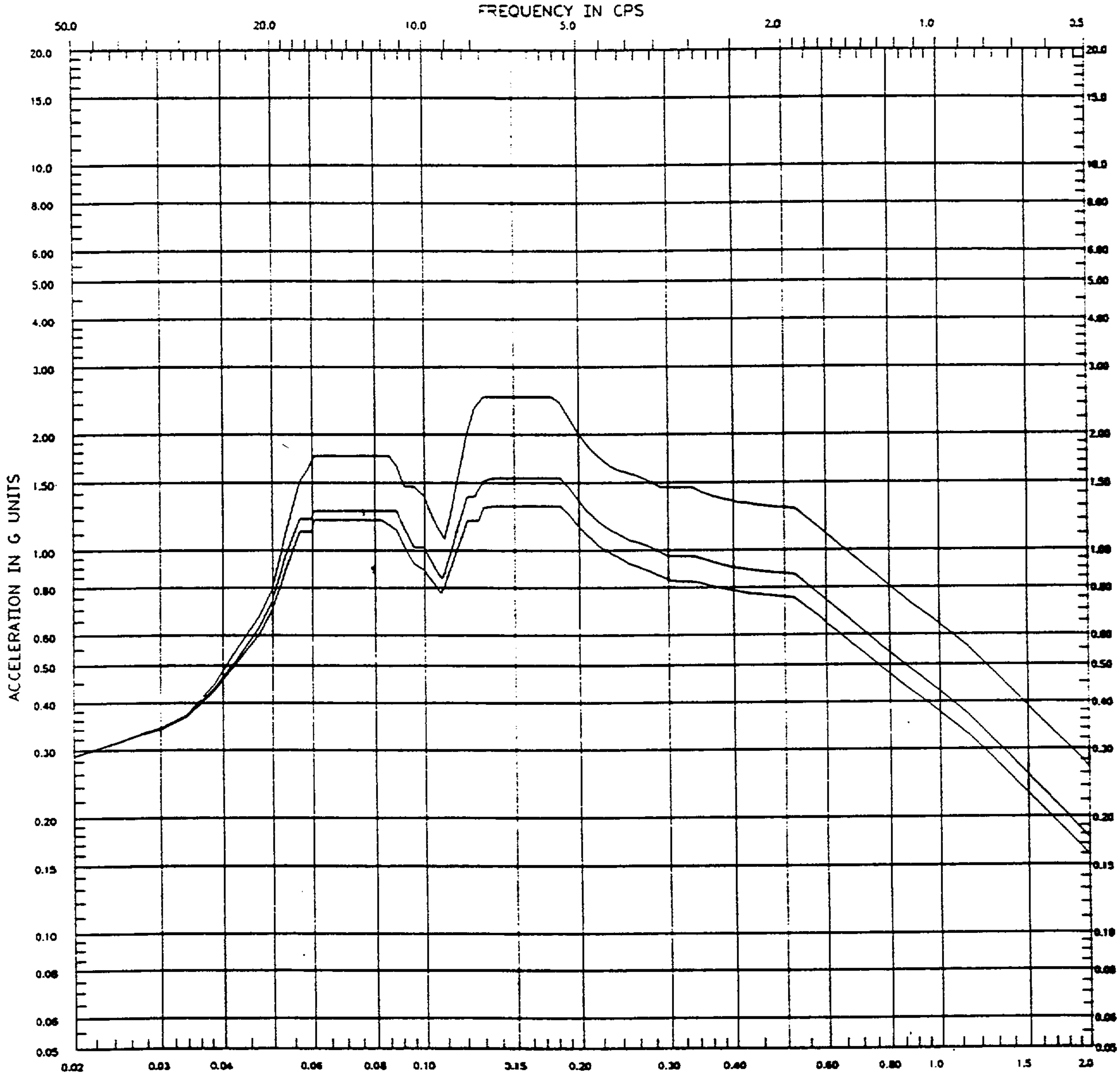
05/04/95

PEAKS WIDENED ON EACH SIDE BY 15 %

SME SPECTRA **Widened** NS DIRECTION DAMPING 0.005 0.03 0.05

NODE 9

Dresden Units 2&3



SPECTRA NO. SME

NODE 9 ELEVATION 517

6 DIRECTION NS **Mass Center of Floor** LOCATION REACTOR BUILDING

CALC NO. 8.11.6-2

SAFETY RELATED

PROJECT 9630-16 REV 0

A2-3

SARGENT & LUNDY ENGINEERS

05/04/95

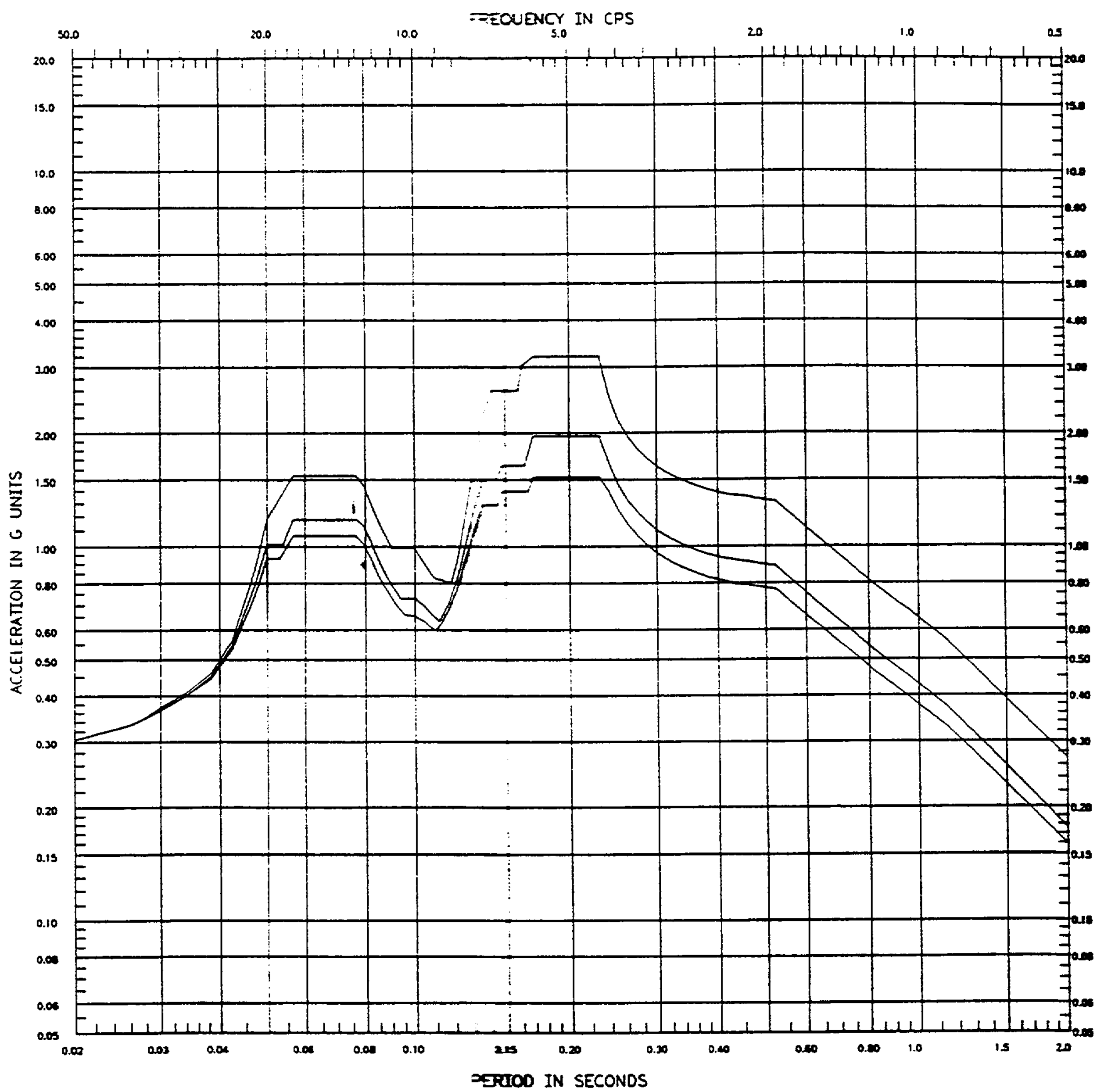
SME SPECTRA - **Widened**

NODE 9

PEAKS WIDENED ON EACH SIDE BY 15 %

EW DI DAMPING 0.005 0.03 0.05

Dresden Units 2&3



SPECTRA NO. SME

NODE 9 ELEVATION 517

6 DIRECTION EW **Envelope For Entire Floor** LOCATION REACTOR BUILDING

CALC NO. 8.11.6-2 A2-4

SAFETY RELATED

SARGENT & LUNDY ENGINEERS

PROJECT 9630-16 REV 0

05/04/95

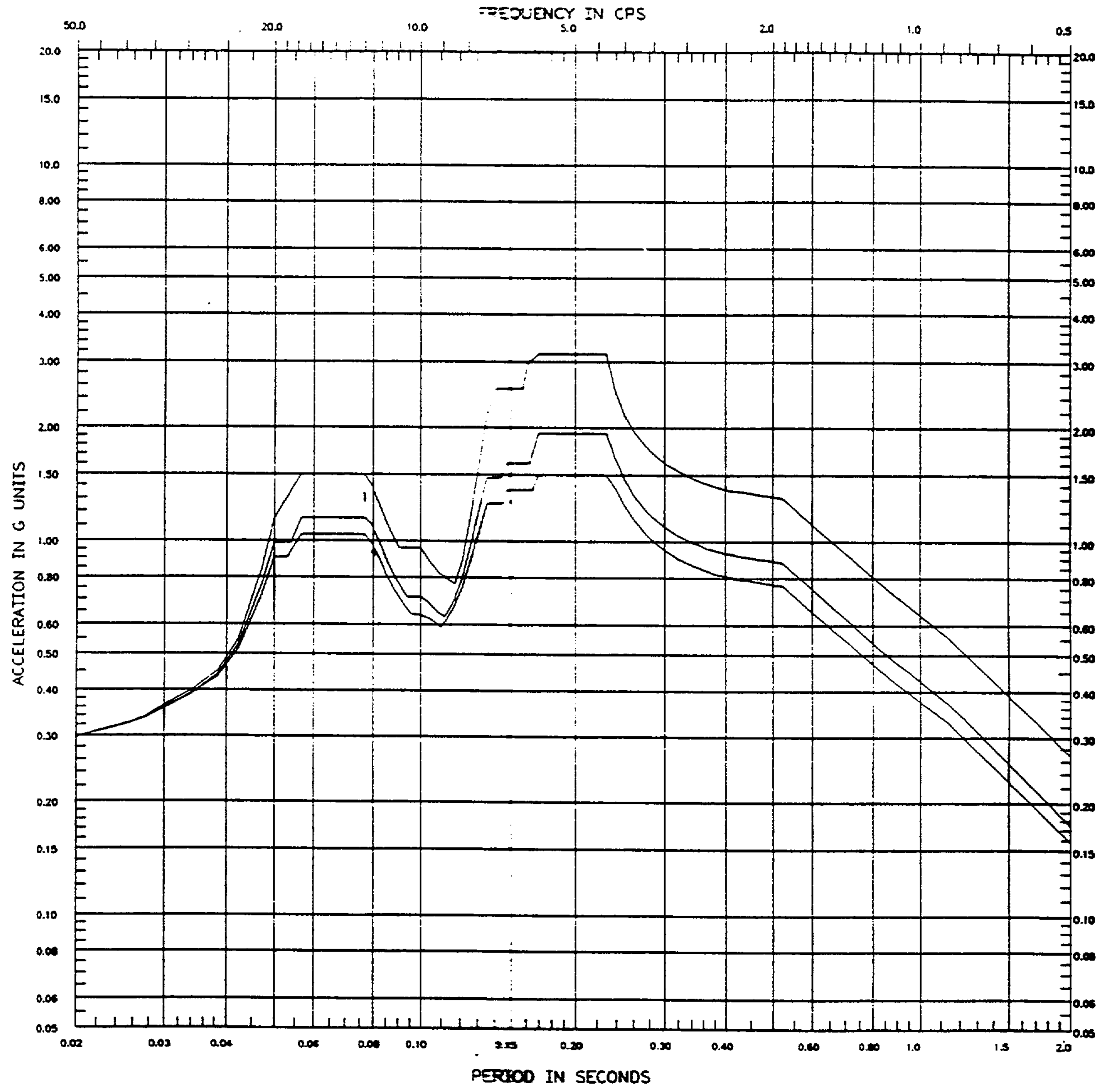
PEAKS WIDENED ON EACH SIDE BY 15 %

SME SPECTRA **Widened** EW DIRECTION

DAMPING 0.005 0.03 0.05

NODE 9

Dresden Units 2&3



SPECTRA NO. SME

NODE 9

ELEVATION 517

6 DIRECTION EW **Mass Center of Floor**

LOCATION REACTOR BUILDING

SARGENT & LUNDY ENGINEERS

CALC NO. 8.11.6-2

SAFETY RELATED

PROJECT 9630-16 REV 0

05/10/95

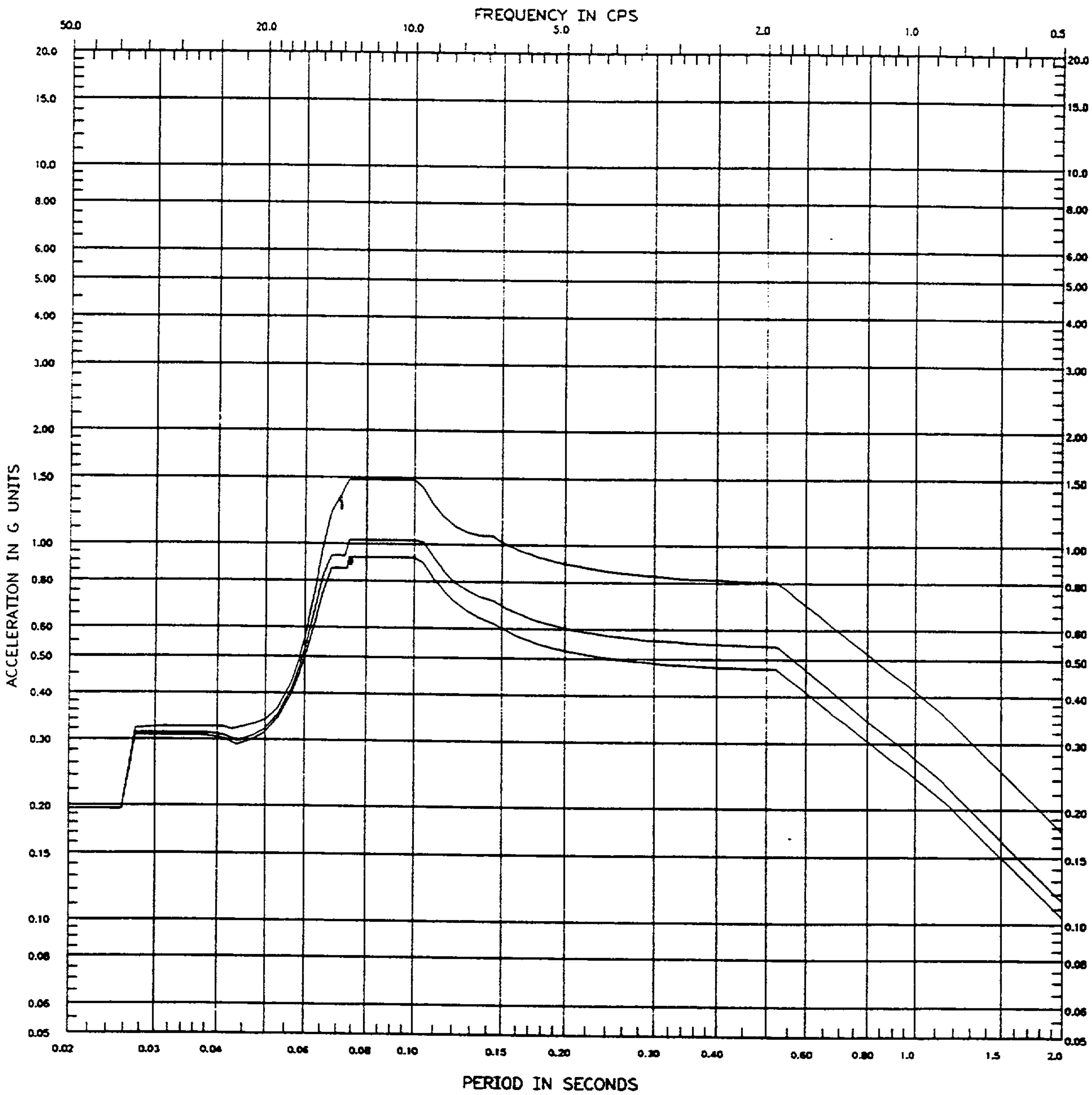
SME SPECTRA - **Widened** DIR.

NODE 9

PEAKS WIDENED ON EACH SIDE BY 15 %

DAMPING 0.005 0.03 0.05

Dresden Units 2&3



SPECTRA NO. SME

NODE 9

ELEVATION 517 **Wall and Slabs >12" Thick**

DIRECTION VERT ANGLE

LOCATION REACTOR BUILDING

CALC NO. 8.11.6-2

SAFETY RELATED

PROJECT 9630-16 REV 0

SARGENT & LUNDY ENGINEERS

05/11/95

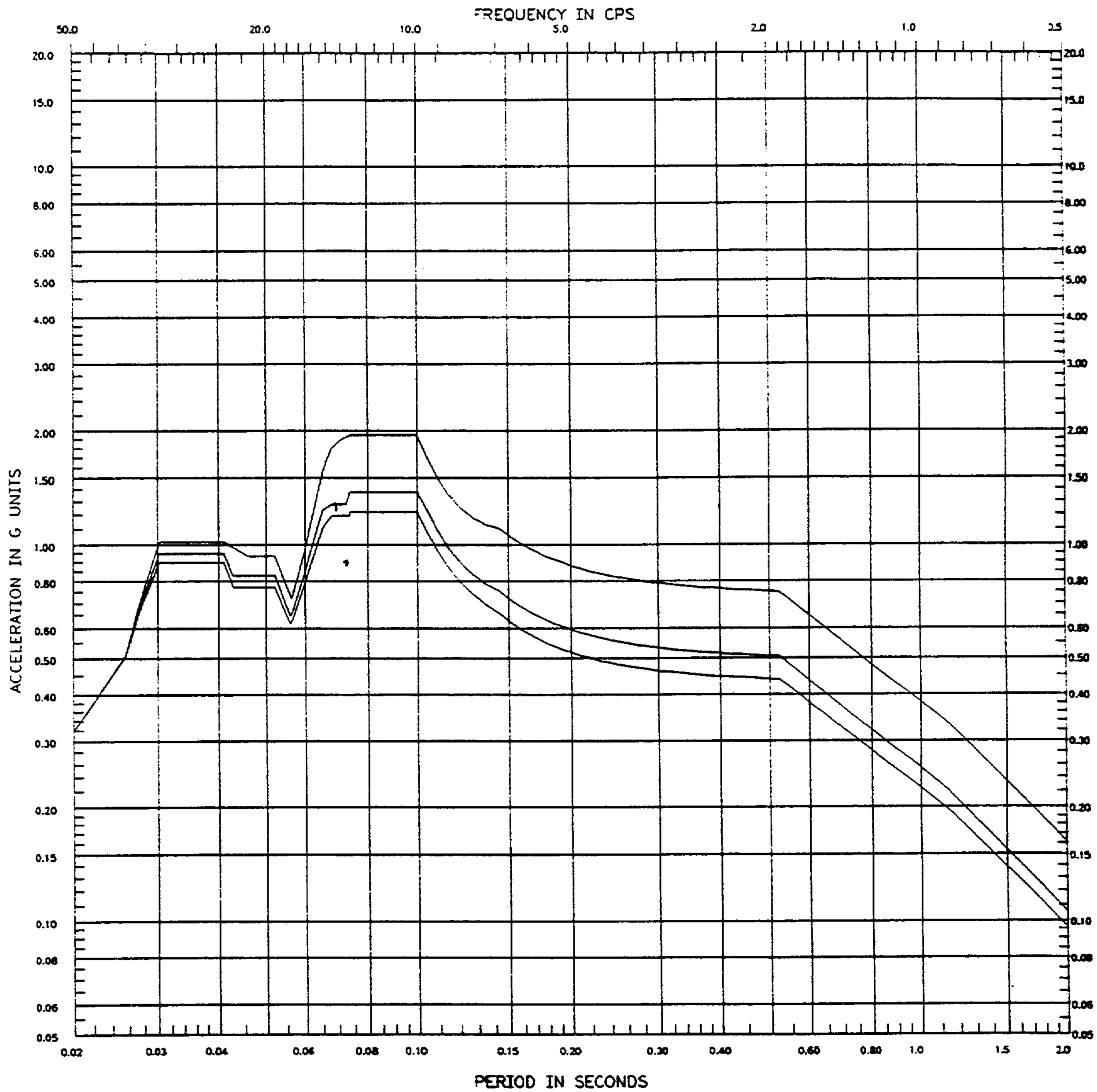
SME SPECTRA - **Widened** SLAB

NODE 47

PEAKS WIDENED ON EACH SIDE BY 15 %

DAMPING 0.005 0.03 0.05

Dresden Units 2&3



SPECTRA NO. SME

NODE 47 ELEVATION 517 **Slab ≤ 12" Thick**

5 DIRECTION VERT ANGLE LOCATION REACTOR BUILDING

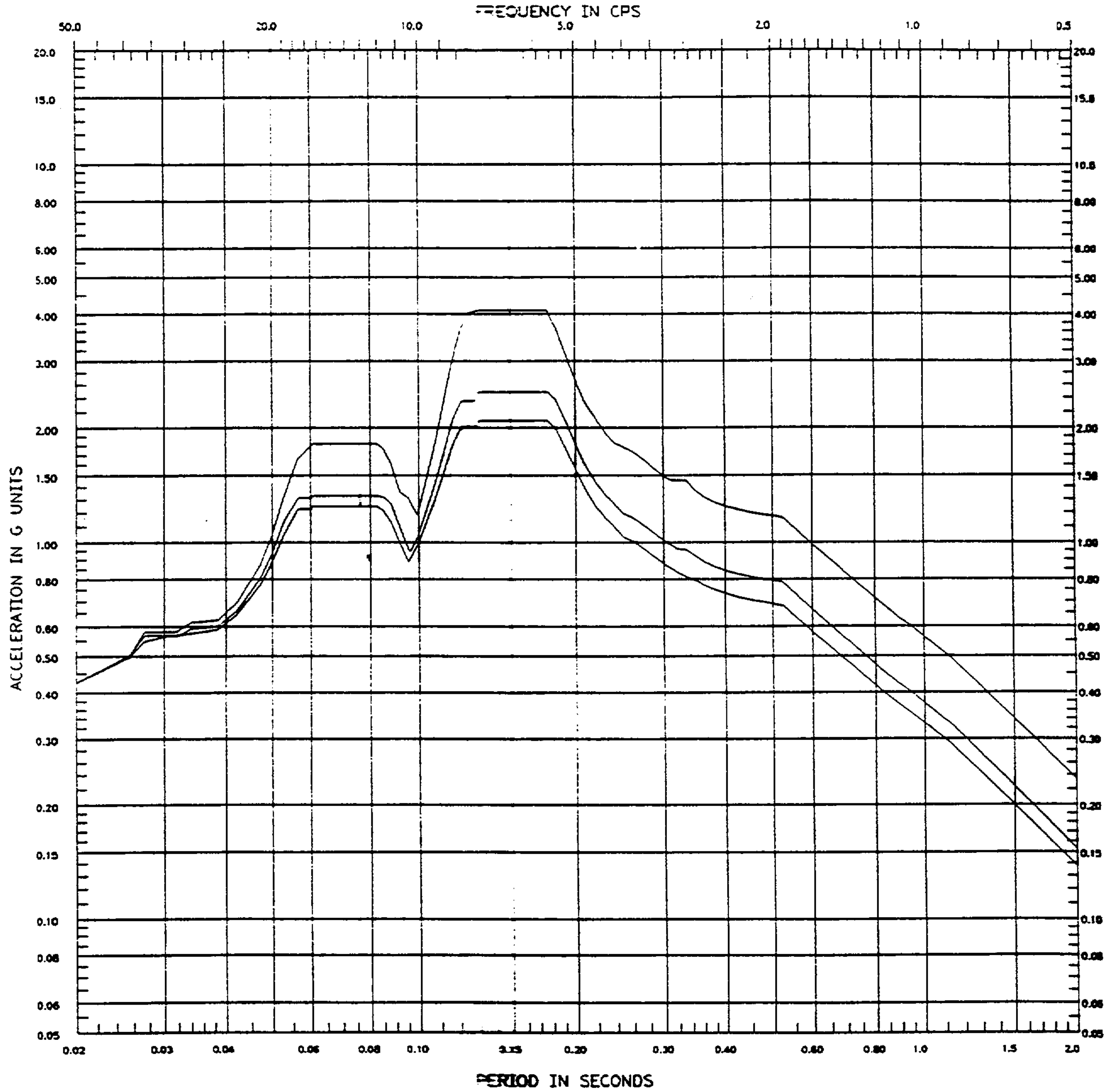
CALC NO. 5.11.6-2
SAFETY RELATED
PROJECT 9630-16 REV 0

SARGENT & LUNDY
ENGINEERS

05/04/95
SME SPECTRA - Widened
NODE 8

PEAKS WIDENED ON EACH SIDE BY 15 %
NS DI DAMPING 0.005 0.03 0.05

Dresden Units 2&3



		SPECTRA NO.	SME
NODE	8	ELEVATION	545
DIRECTION	NS	LOCATION	REACTOR BUILDING

Envelope For Entire Floor

CALC NO. 8.11.6-2

SAFETY RELATED

PROJECT 9630-16 REV 0

A2-8

SARGENT & LUNDY ENGINEERS

05/04/95

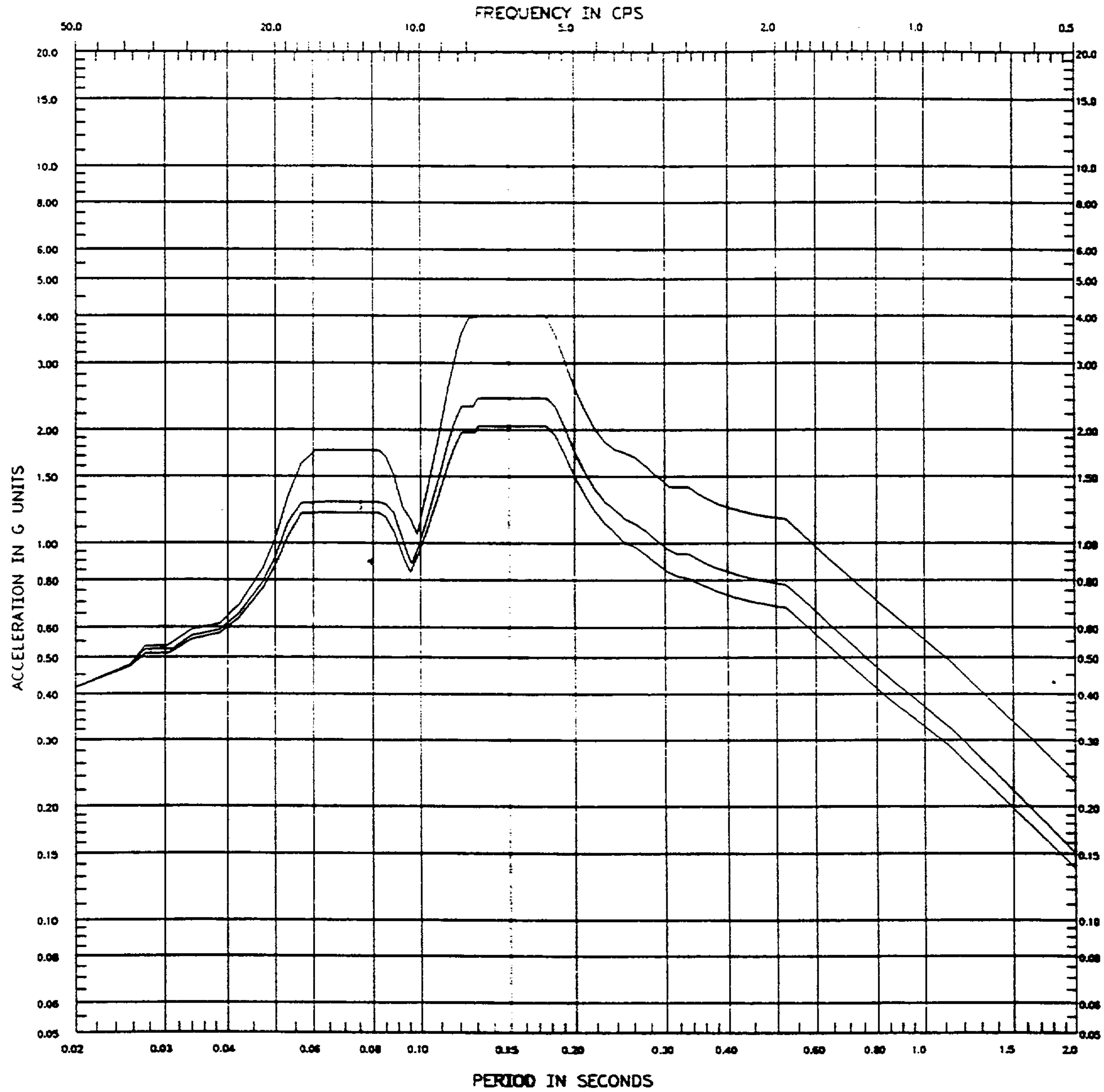
SME SPECTRA **Widened** NS DIRECTION

NODE 8

PEAKS WIDENED ON EACH SIDE BY 15 %

DAMPING 0.005 0.03 0.05

Dresden Units 2&3



SPECTRA NO. SME

NODE 8 ELEVATION 545

5 DIRECTION NS **Mass Center of Floor** LOCATION REACTOR BUILDING

CALC NO. 8.11.6-2
SAFETY RELATED
PROJECT 9630-16 REV 0

SARGENT & LUNDY ENGINEERS

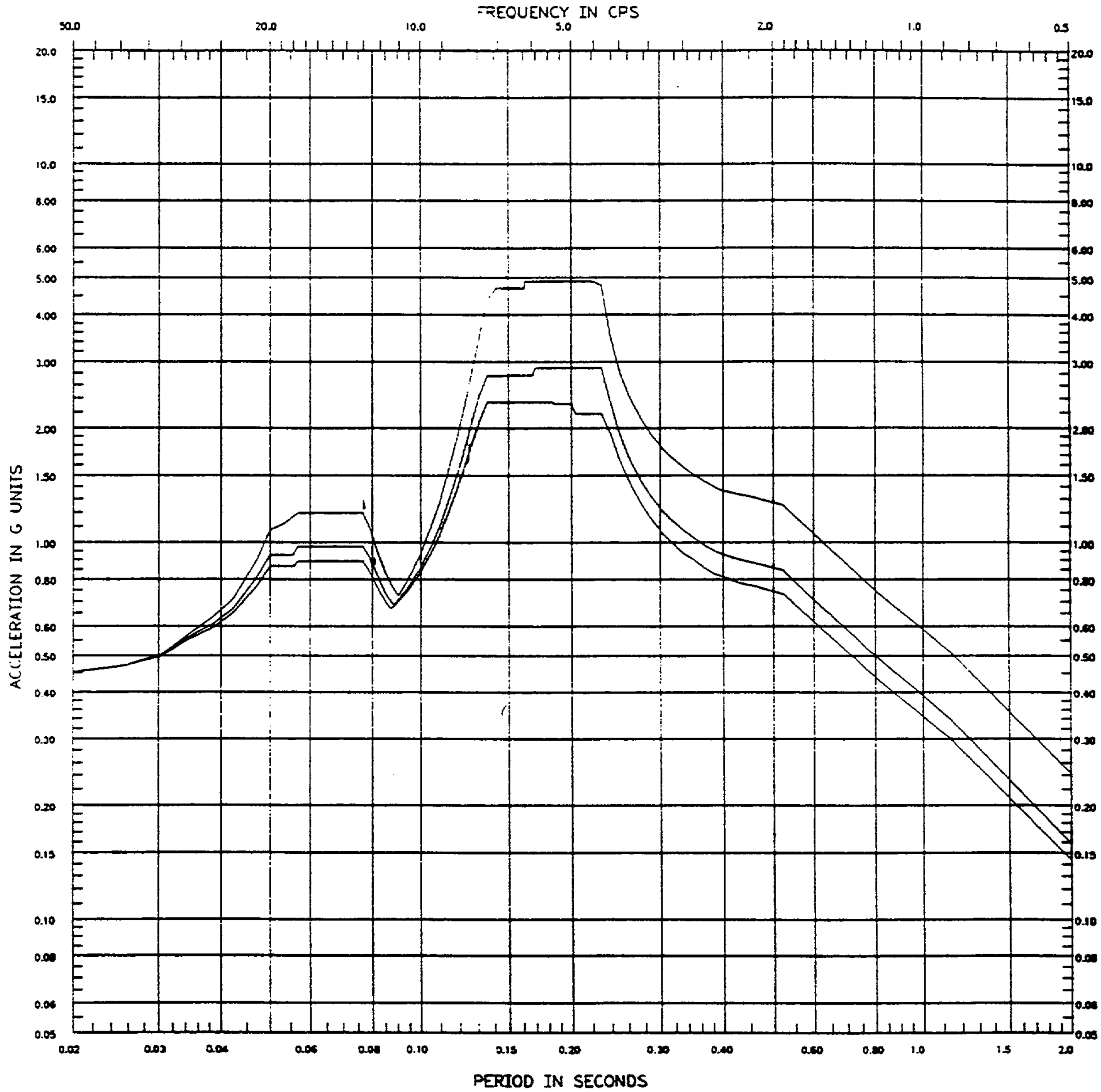
05/04/95

PEAKS WIDENED ON EACH SIDE BY 15 %

SME SPECTRA **Widened** EW DI DAMPING 0.005 0.03 0.05

NODE 8

Dresden Units 2&3



		SPECTRA NO.	SME
NODE	8	ELEVATION	545
DIRECTION	EW **Envelope For Entire Floor**	LOCATION	REACTOR BUILDING

5

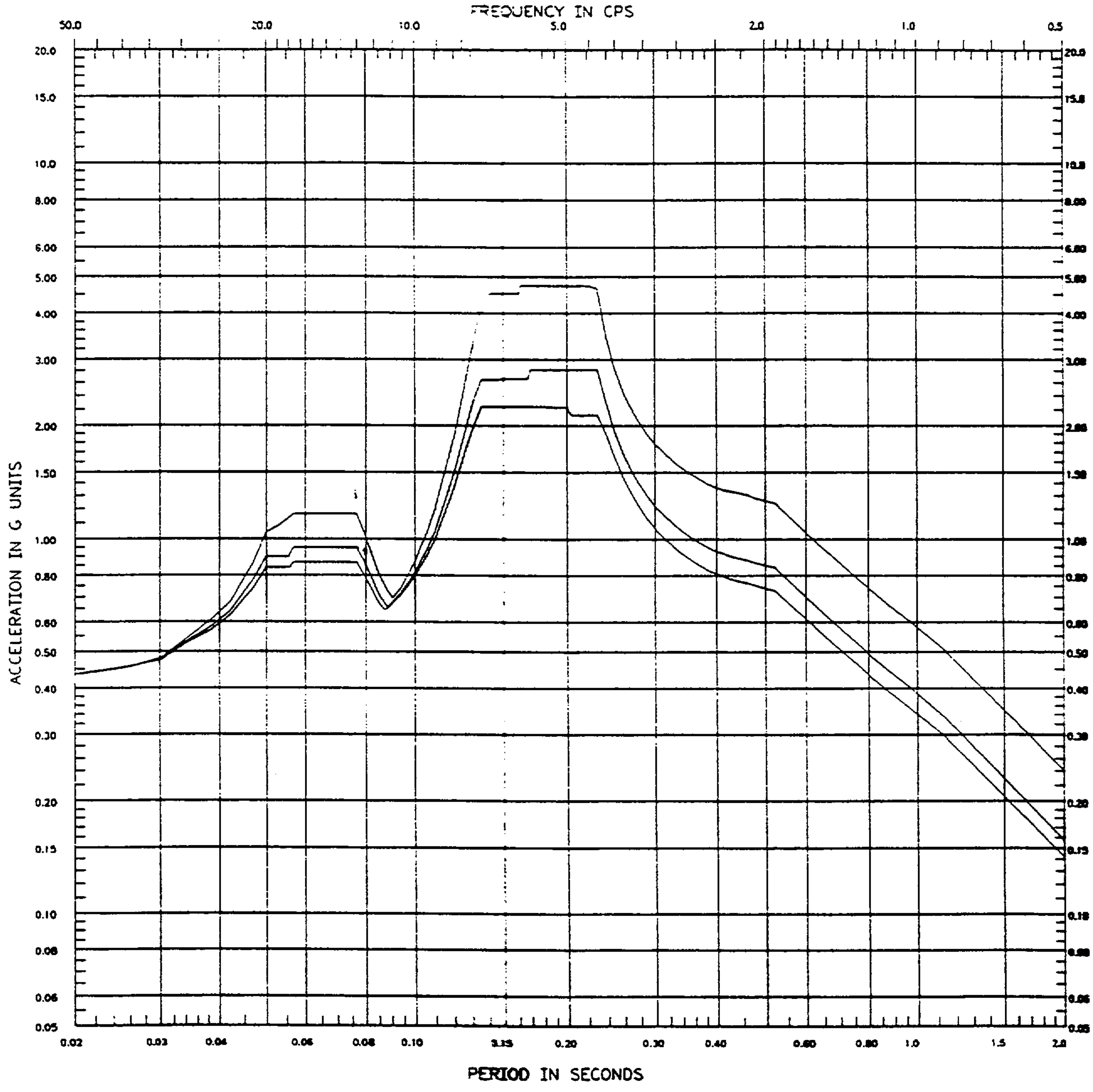
CALC NO. 8.11.6-2
SAFETY RELATED
PROJECT 9630-16 REV 0

SARGENT & LUNDY
ENGINEERS

05/04/95
SME SPECTRA **Widened** EW DIRECTION
NODE 8

PEAKS WIDENED ON EACH SIDE BY 15 %
DAMPING 0.005 0.03 0.05

Dresden Units 2&3



			SPECTRA NO.	SME
NODE	8		ELEVATION	545
DIRECTION	EW	**Mass Center of Floor**	LOCATION	REACTOR BUILDING

CALC NO. 8.11.6-2

SAFETY RELATED

PROJECT 9630-16 REV 0

A2-11

SARGENT & LUNDY
ENGINEERS

05/10/95

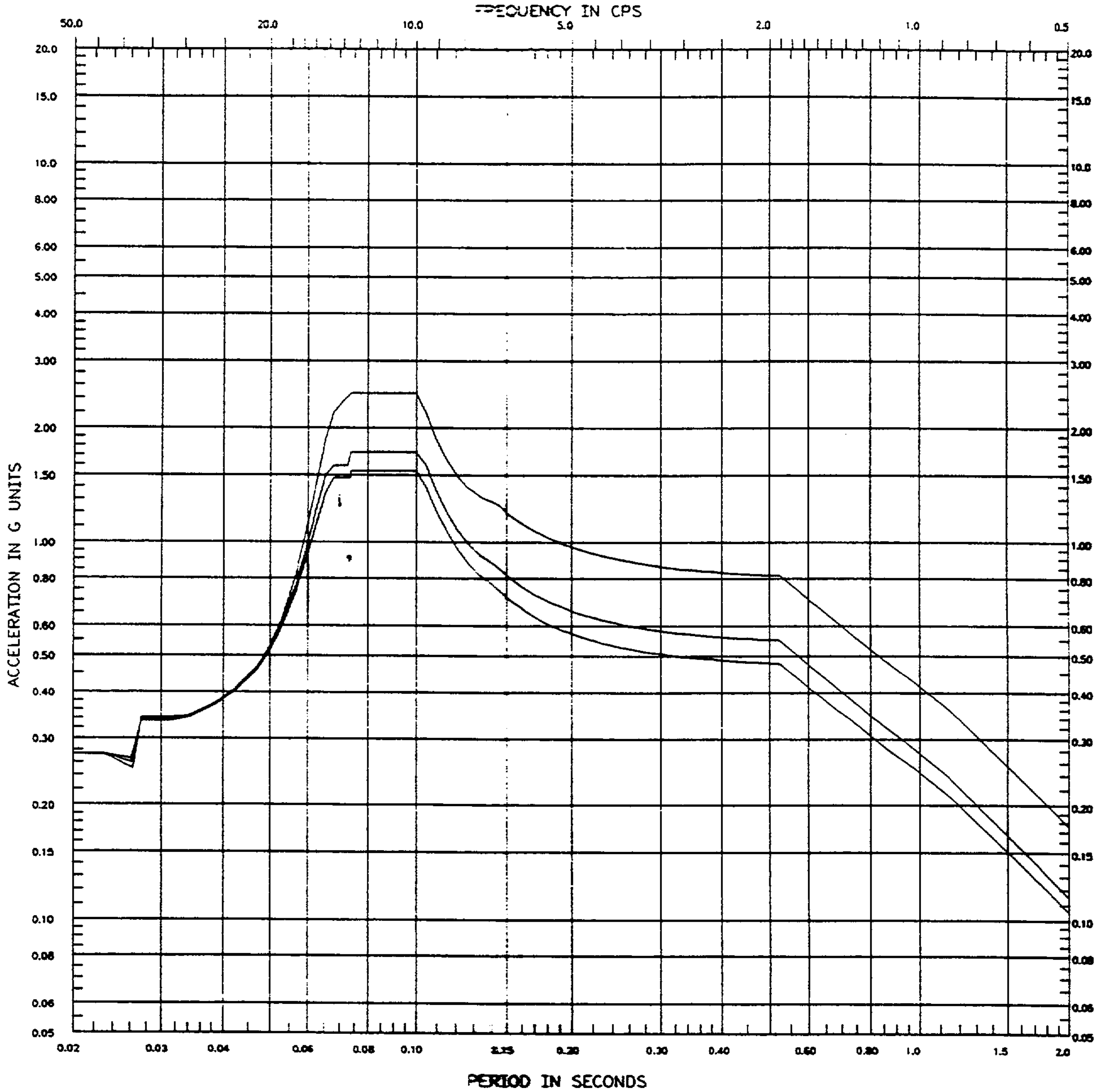
SME SPECTRA - Widened

NODE 8

PEAKS WIDENED ON EACH SIDE BY 15 %

DAMPING 0.005 0.03 0.05

Dresden Units 2&3



		SPECTRA NO.	SME
NODE	8	ELEVATION	545 **Wall and Slabs >12" Thick**
DIRECTION	VERT	ANGLE	LOCATION	REACTOR BUILDING

5

CALC NO. 8.11.6-2

SAFETY RELATED

PROJECT 9630-16 REV 0

A2-12

SARGENT & LUNDY ENGINEERS

05/11/95

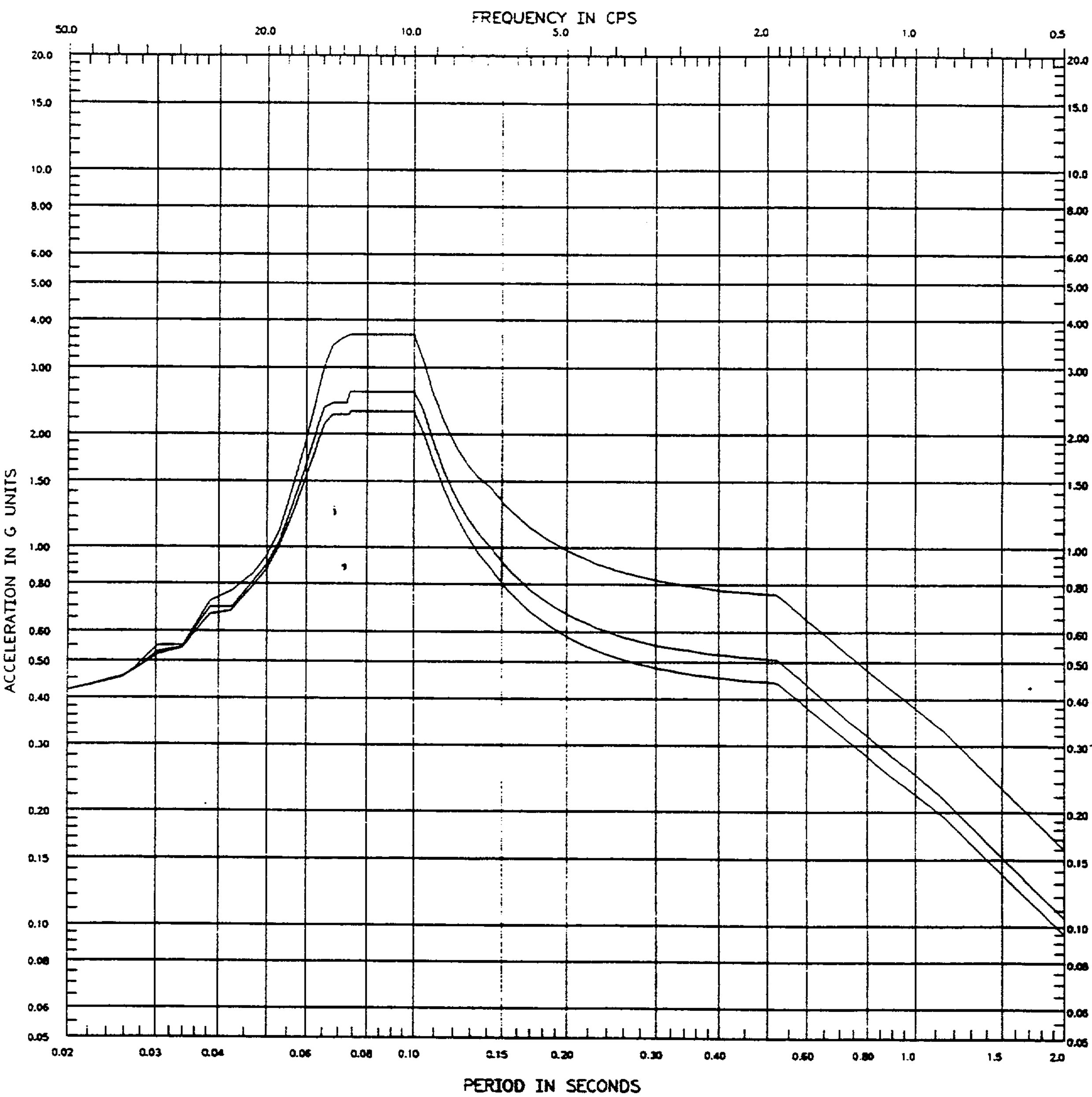
SME SPECTRA - Widened SLAB

NODE 45

PEAKS WIDENED ON EACH SIDE BY 15 %

DAMPING 0.005 0.03 0.05

Dresden Units 2&3



SPECTRA NO. SME

NODE 45

ELEVATION 545 **Slab ≤ 12" Thick**

4 DIRECTION VERT ANGLE

LOCATION REACTOR BUILDING

CALC NO. 8.11 6-2

SAFETY RELATED

SARGENT & LUNDY ENGINEERS

PROJECT 9630-15 REV 0

05/04/95

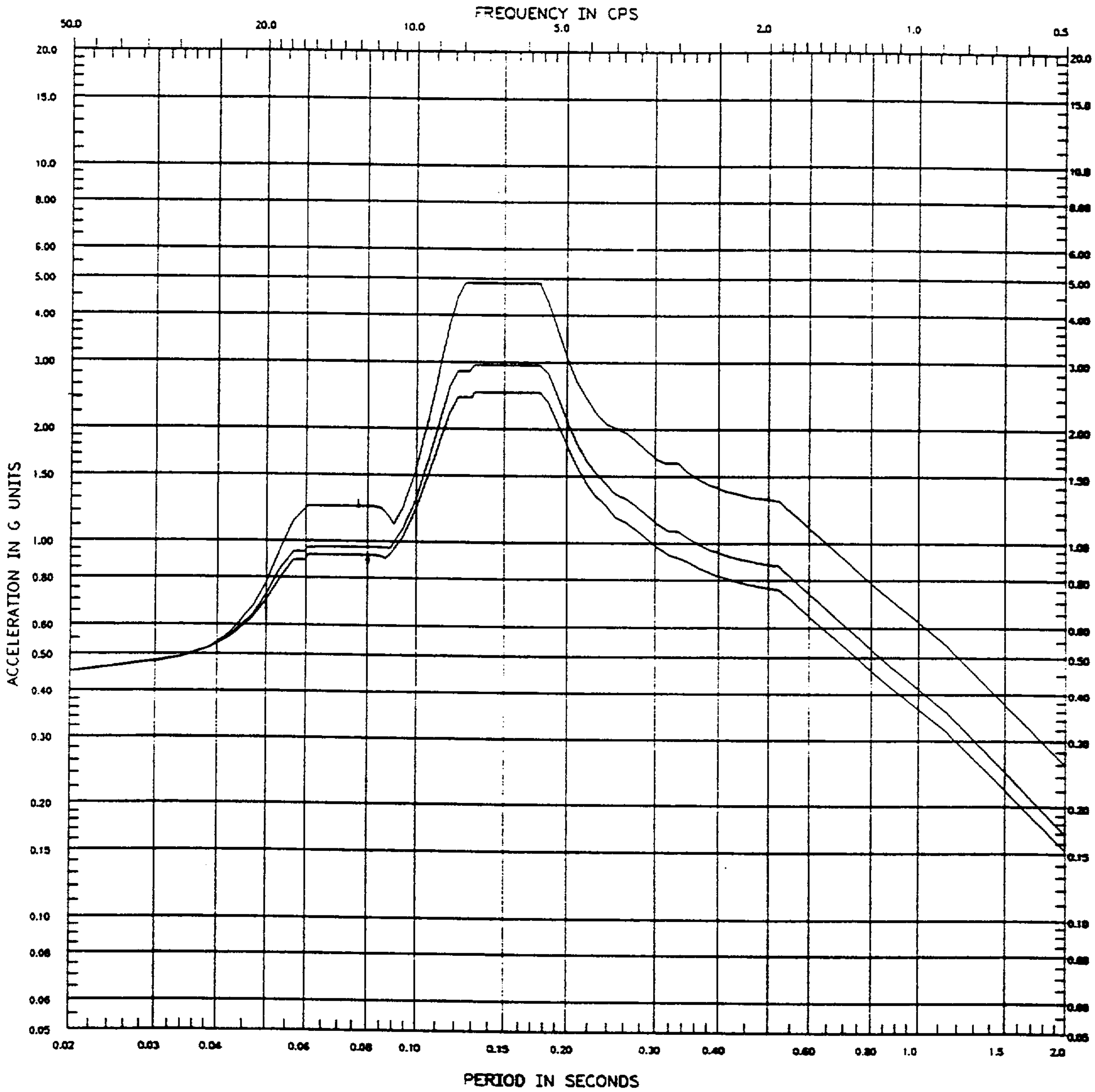
PEAKS WIDENED ON EACH SIDE BY 15 %

SME SPECTRA - **Widened**

NS DI DAMPING 0.005 0.03 0.05

NODE 7

Dresden Units 2&3



			SPECTRA NO.	SME
NODE	7		ELEVATION	561
DIRECTION	NS	**Envelope For Entire Floor**	LOCATION	REACTOR BUILDING

4

CALC NO. 8.11.6-2
SAFETY RELATED
PROJECT 9630-16 REV 0

SARGENT & LUNDY
ENGINEERS

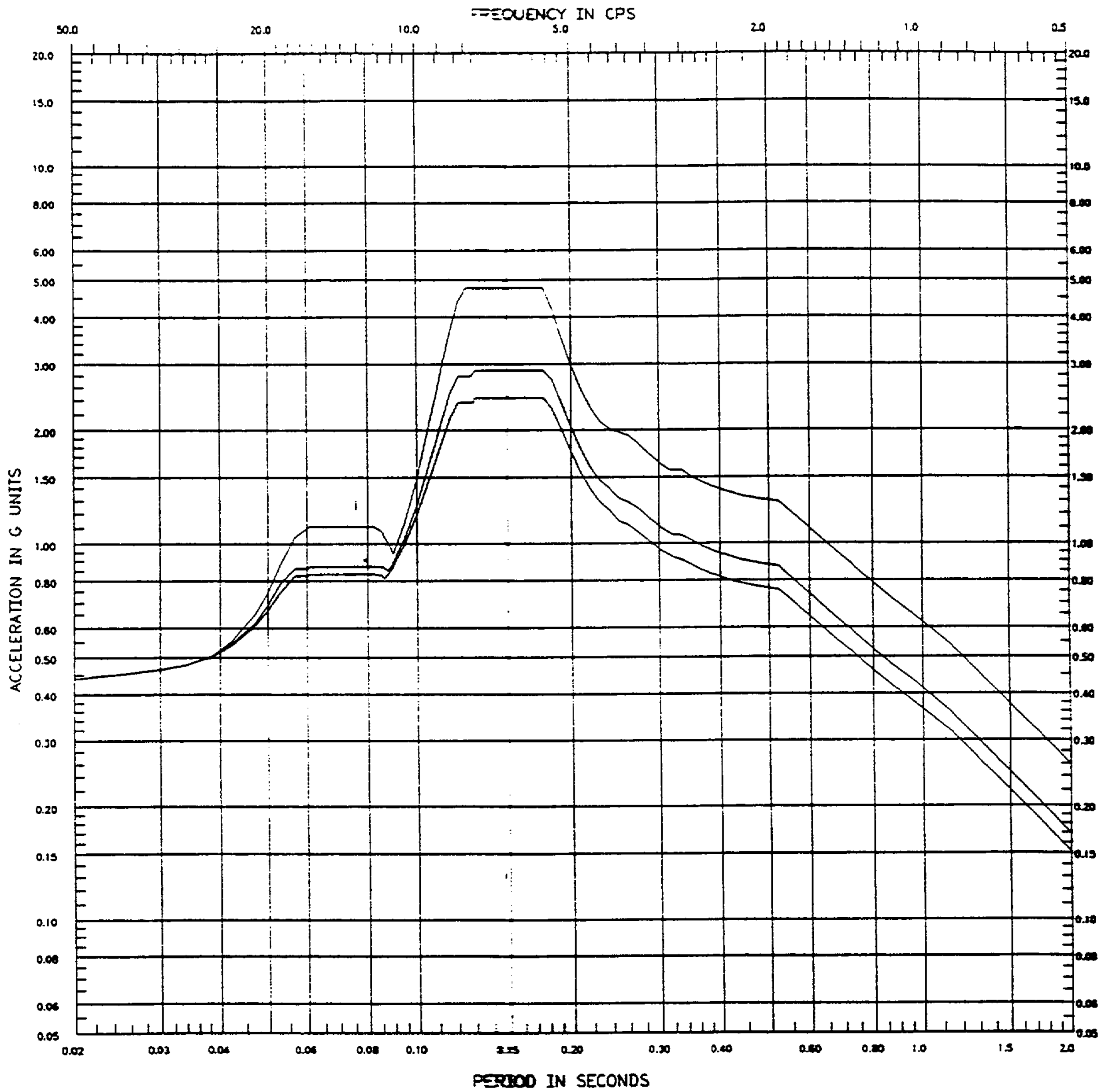
PEAKS WIDENED ON EACH SIDE BY 15 %

05/04/95

SME SPECTRA Widened NS DIRECTION DAMPING 0.005 0.03 0.05

NODE 7

Dresden Units 2&3



		SPECTRA NO.	SME
NODE	7	ELEVATION	561
DIRECTION	NS Mass Center of Floor	LOCATION	REACTOR BUILDING

SARGENT & LUNDY
ENGINEERS

CALC NO. 8.11.6-2
SAFETY RELATED
PROJECT 9630-16 REV 0

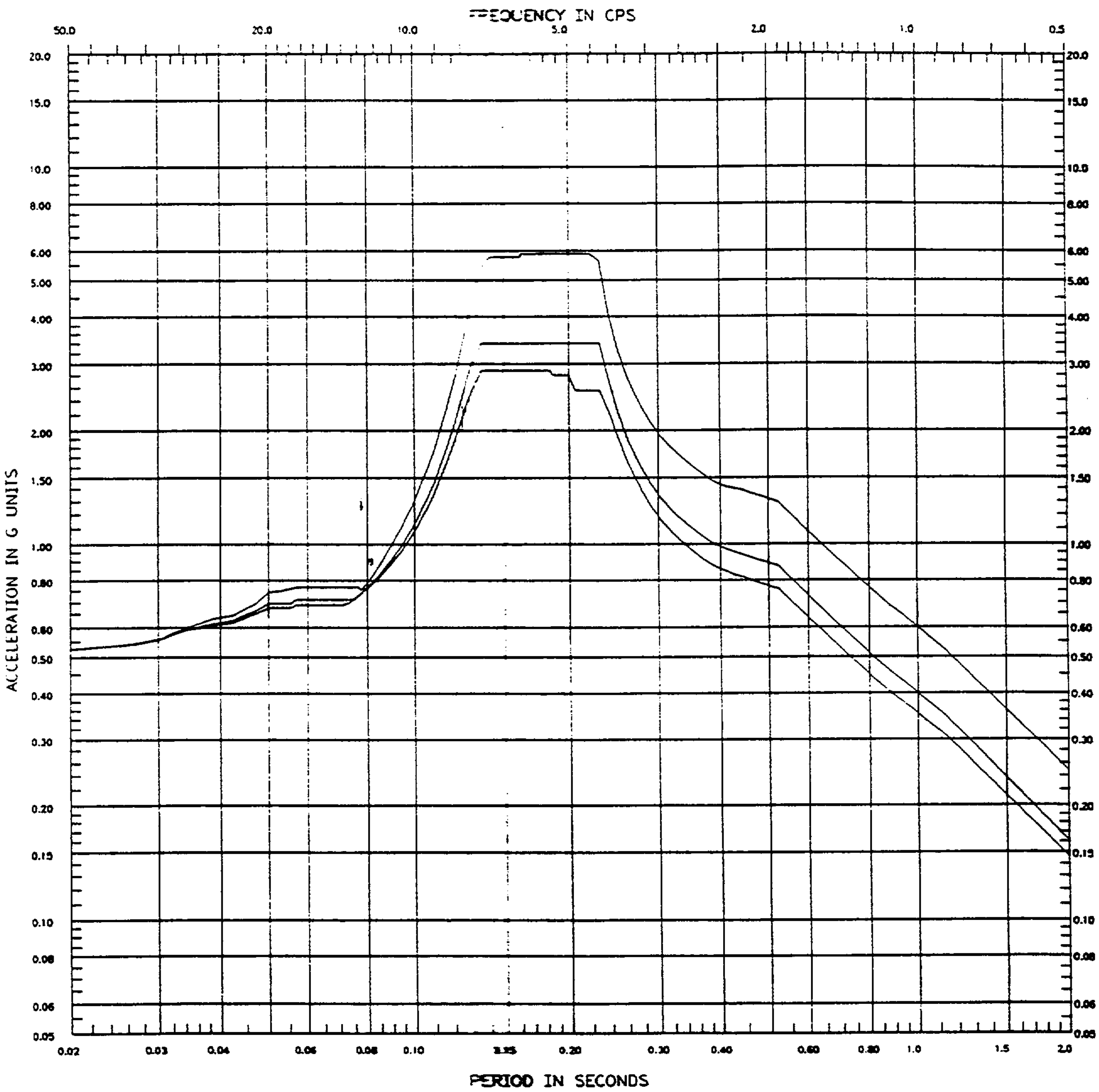
05/04/95

PEAKS WIDENED ON EACH SIDE BY 15 %

SME SPECTRA - **Widened** EW DI DAMPING 0.005 0.03 0.05

NODE 7

Dresden Units 2&3



			SPECTRA NO.	SME
NODE	7		ELEVATION	561
DIRECTION	EW	**Envelope For Entire Floor**	LOCATION	REACTOR BUILDING

CALC NO. 8.11.6-2

SAFETY RELATED

PROJECT 9630-16 REV 0

SARGENT & LUNDY ENGINEERS

05/04/95

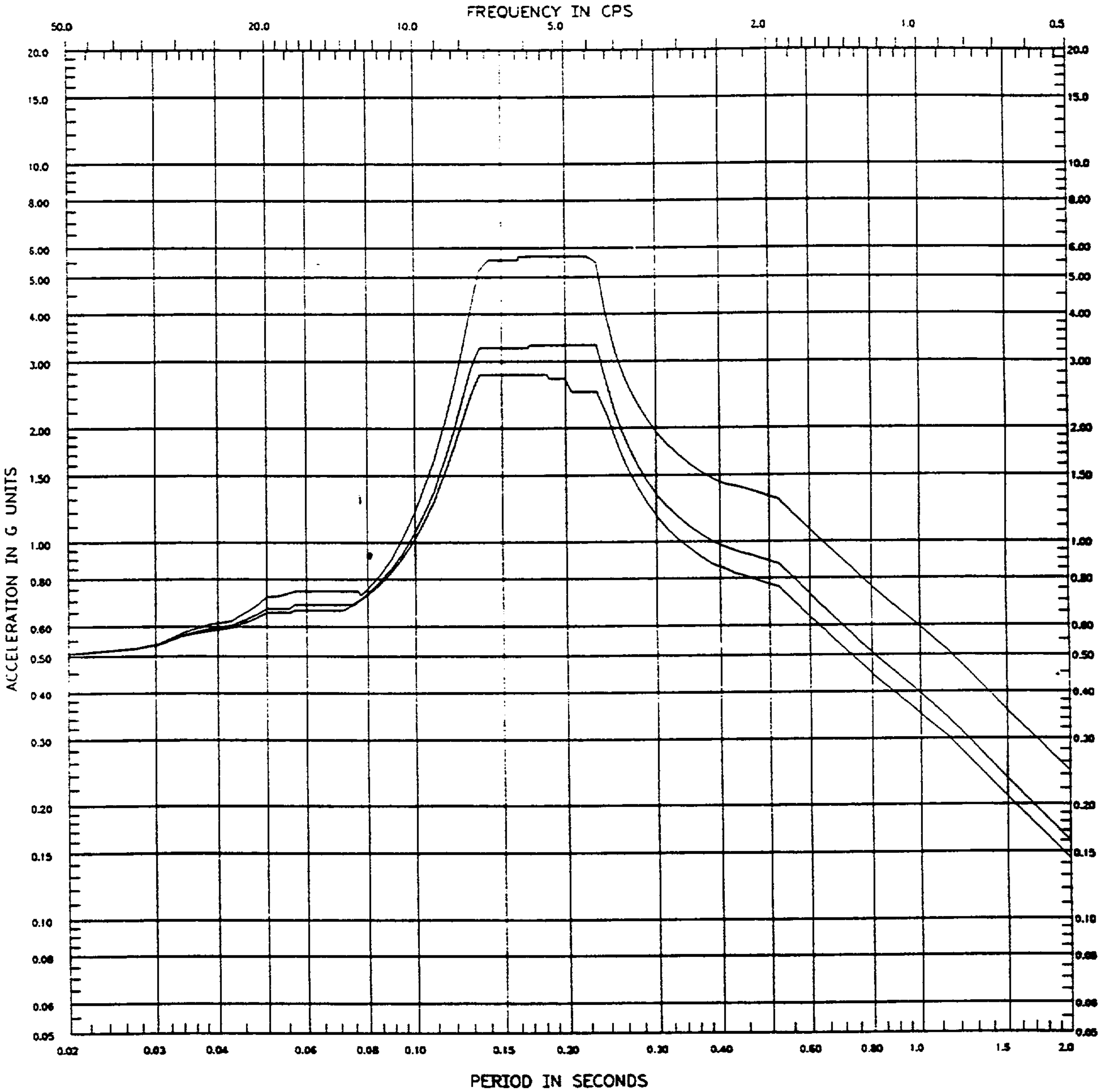
SME SPECTRA **Widened** EW DIRECTION

NODE 7

PEAKS WIDENED ON EACH SIDE BY 15 %

DAMPING 0.005 0.03 0.05

Dresden Units 2&3



SPECTRA NO. SME

NODE 7 ELEVATION 561

4 DIRECTION EW **Mass Center of Floor** LOCATION REACTOR BUILDING

CALC NO. 8.11.6-2

SAFETY RELATED

PROJECT 9630-16 REV 0

A2-17

SARGENT & LUNDY ENGINEERS

05/10/95

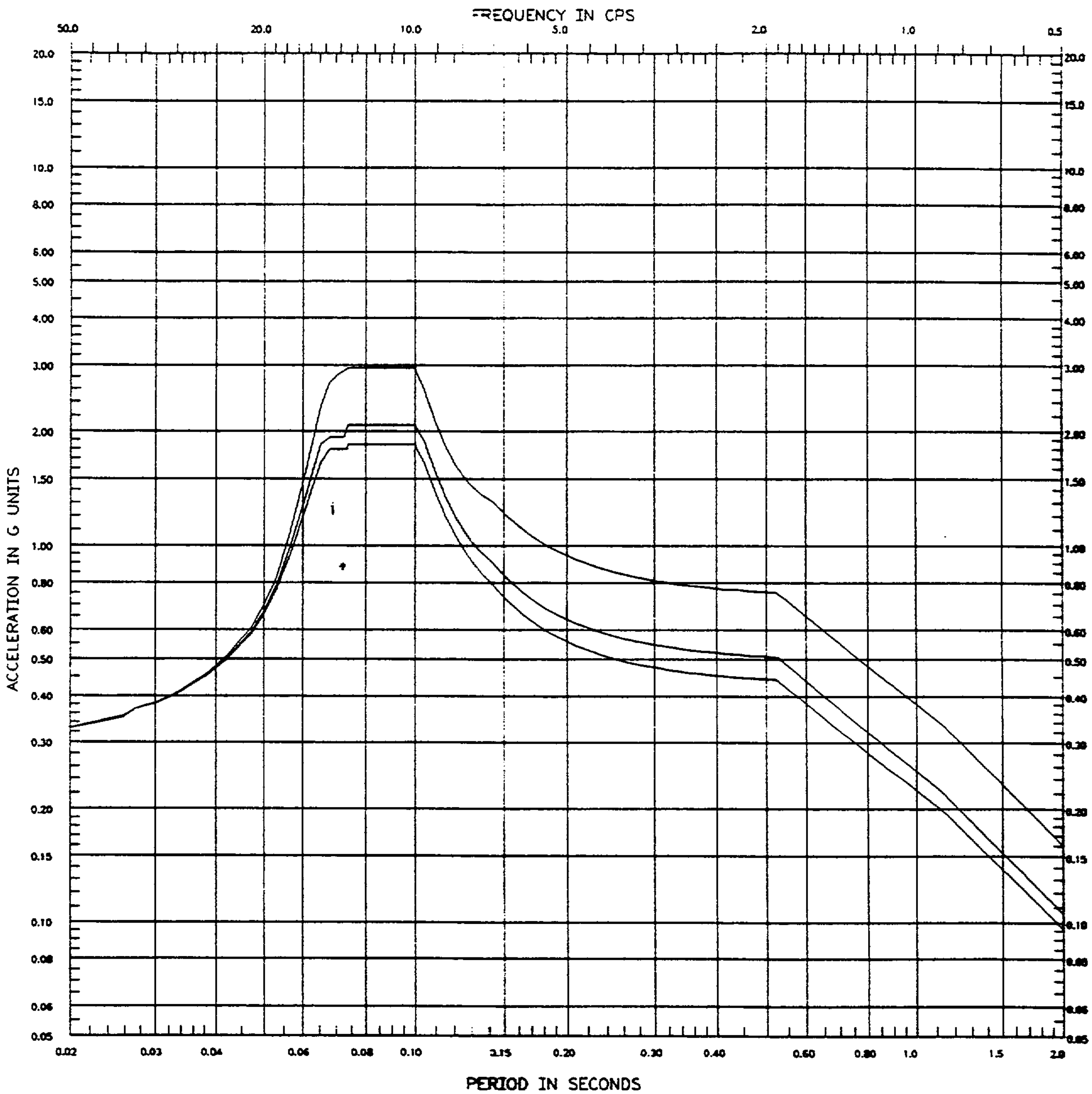
SME SPECTRA - Widened DIR.

NODE 7

PEAKS WIDENED ON EACH SIDE BY 15 %

DAMPING 0.005 0.03 0.05

Dresden Units 2&3



SPECTRA NO. SME

NODE 7 ELEVATION 561 Wall and Slabs >12" Thick

4 DIRECTION VERT ANGLE LOCATION REACTOR BUILDING

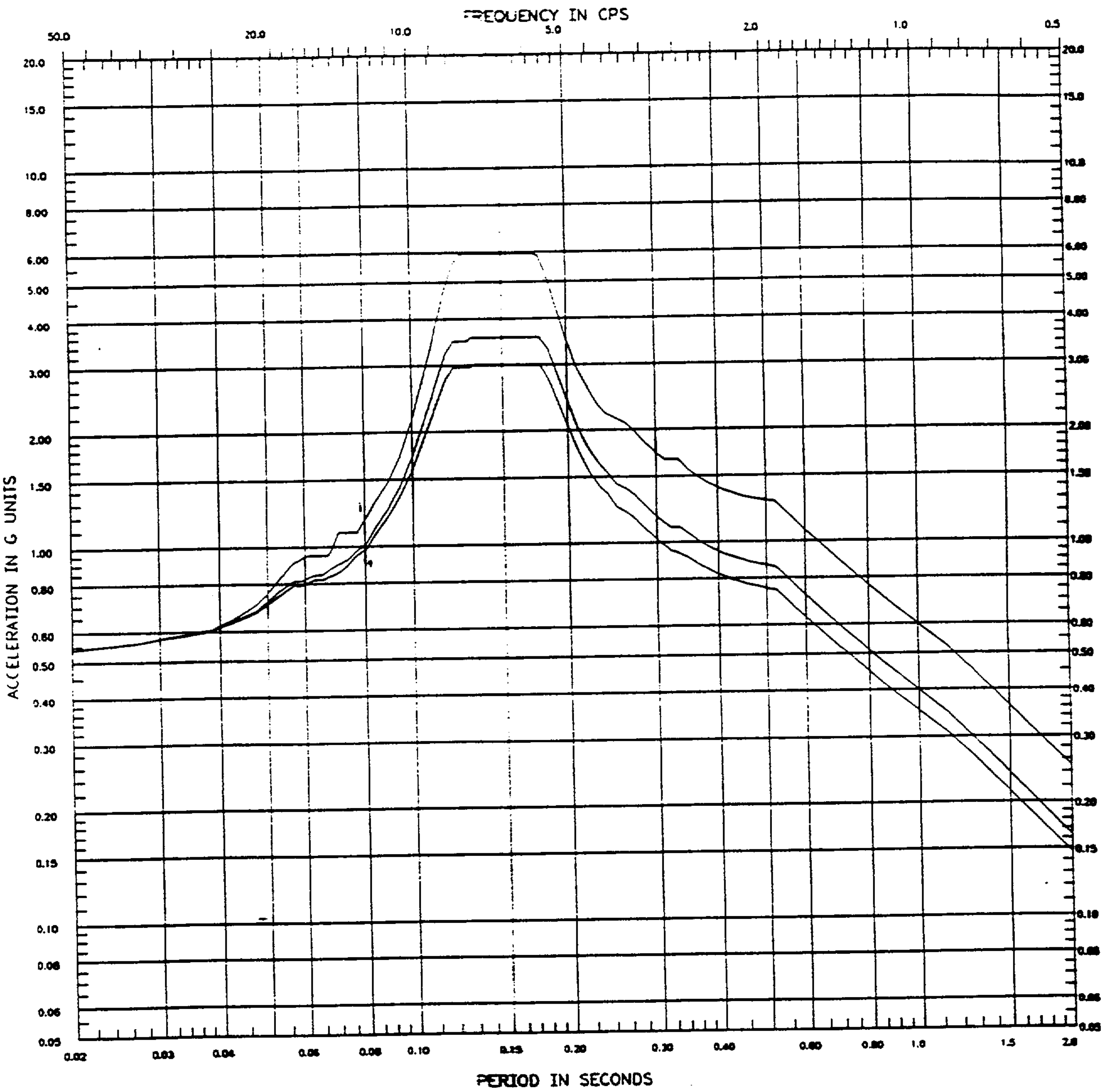
CALC NO. 8.11.6-2
SAFETY RELATED
PROJECT 9630-16 REV 0

SARGENT & LUNDY
ENGINEERS

05/04/95
SME SPECTRA - Widened
NODE 6

PEAKS WIDENED ON EACH SIDE BY 15 %
NS DI DAMPING 0.005 0.03 0.05

Dresden Units 2&3



			SPECTRA NO.	SME
NODE	6		ELEVATION	570
DIRECTION	NS	Envelope For Entire Floor	LOCATION	REACTOR BUILDING

CALC NO. 8.11.6-2 A2-19

SAFETY RELATED

SARGENT & LUNDY ENGINEERS

PROJECT 9630-16 REV 0

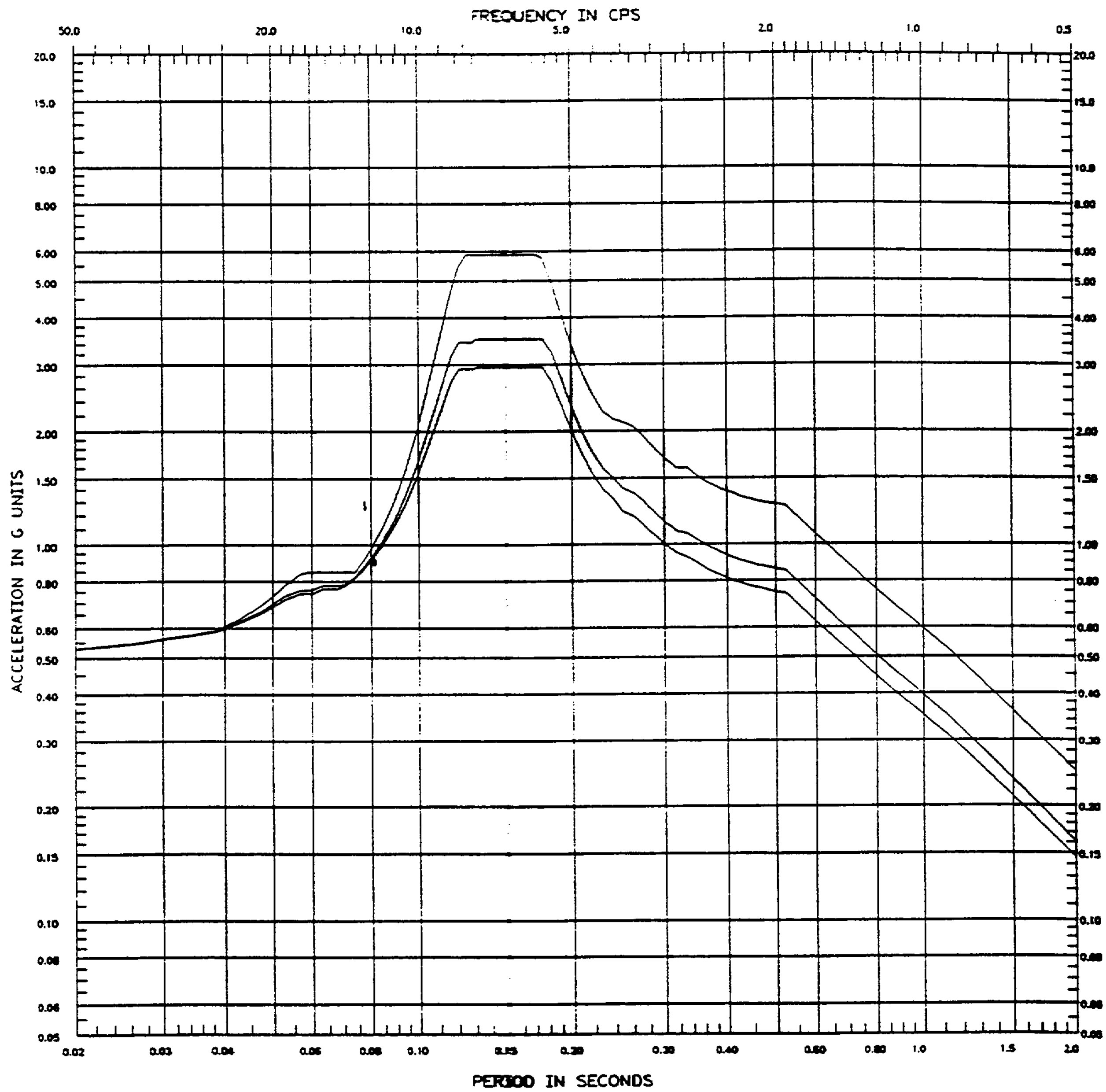
05/04/95

PEAKS WIDENED ON EACH SIDE BY 15 %

SME SPECTRA **Widened** NS DIRECTION DAMPING 0.005 0.03 0.05

NODE 6

Dresden Units 2&3



SPECTRA NO. SME

NODE 6 ELEVATION 570

3 DIRECTION NS **Mass Center of Floor** LOCATION REACTOR BUILDING

CALC NO. 3.11.6-2
SAFETY RELATED
PROJECT #630-16 REV 0

SARGENT & LUNDY ENGINEERS

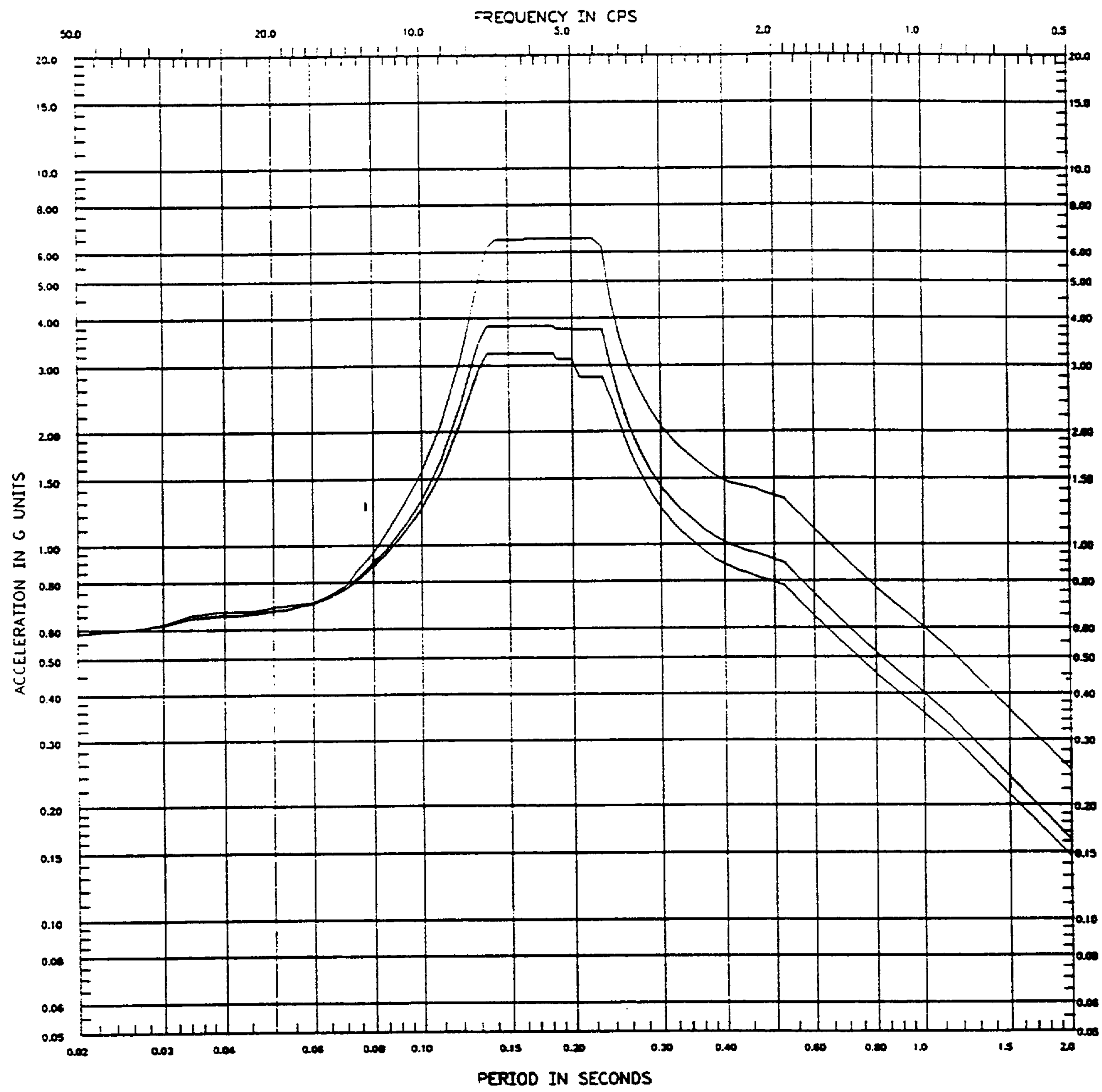
05/04/95

PEAKS WIDENED ON EACH SIDE BY 15 %

SME SPECTRA - Widened EW DI DAMPING 0.005 0.03 0.05

NODE 6

Dresden Units 2&3



SPECTRA NO. SME

NODE 6 ELEVATION 570

3 DIRECTION EW Envelope For Entire Floor LOCATION REACTOR BUILDING

CALC NO. 8.11.6-2

SAFETY RELATED

PROJECT 9630-16 REV 0

SARGENT & LUNDY ENGINEERS

05/04/95

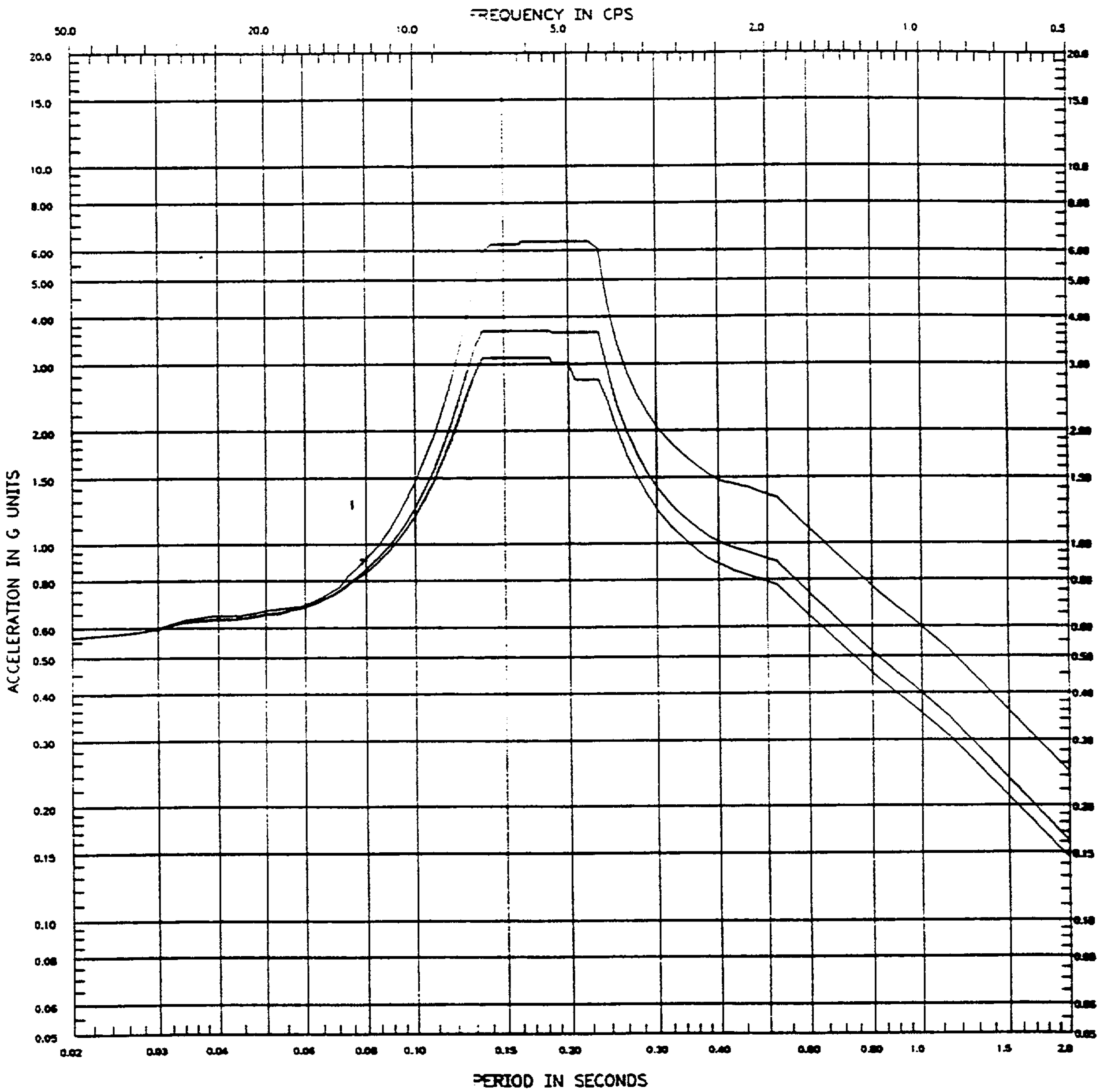
SME SPECTRA **Widened** EW DIRECTION

NODE 6

PEAKS WIDENED ON EACH SIDE BY 15 %

DAMPING 0.005 0.03 0.05

Dresden Units 2&3



		SPECTRA NO.	SME
NODE	6	ELEVATION	570
DIRECTION	EW **Mass Center of Floor**	LOCATION	REACTOR BUILDING

3

CALC NO. 8.11.6-2

SAFETY RELATED

PROJECT 9630-16 REV 0

SARGENT & LUNDY

ENGINEERS

05/10/95

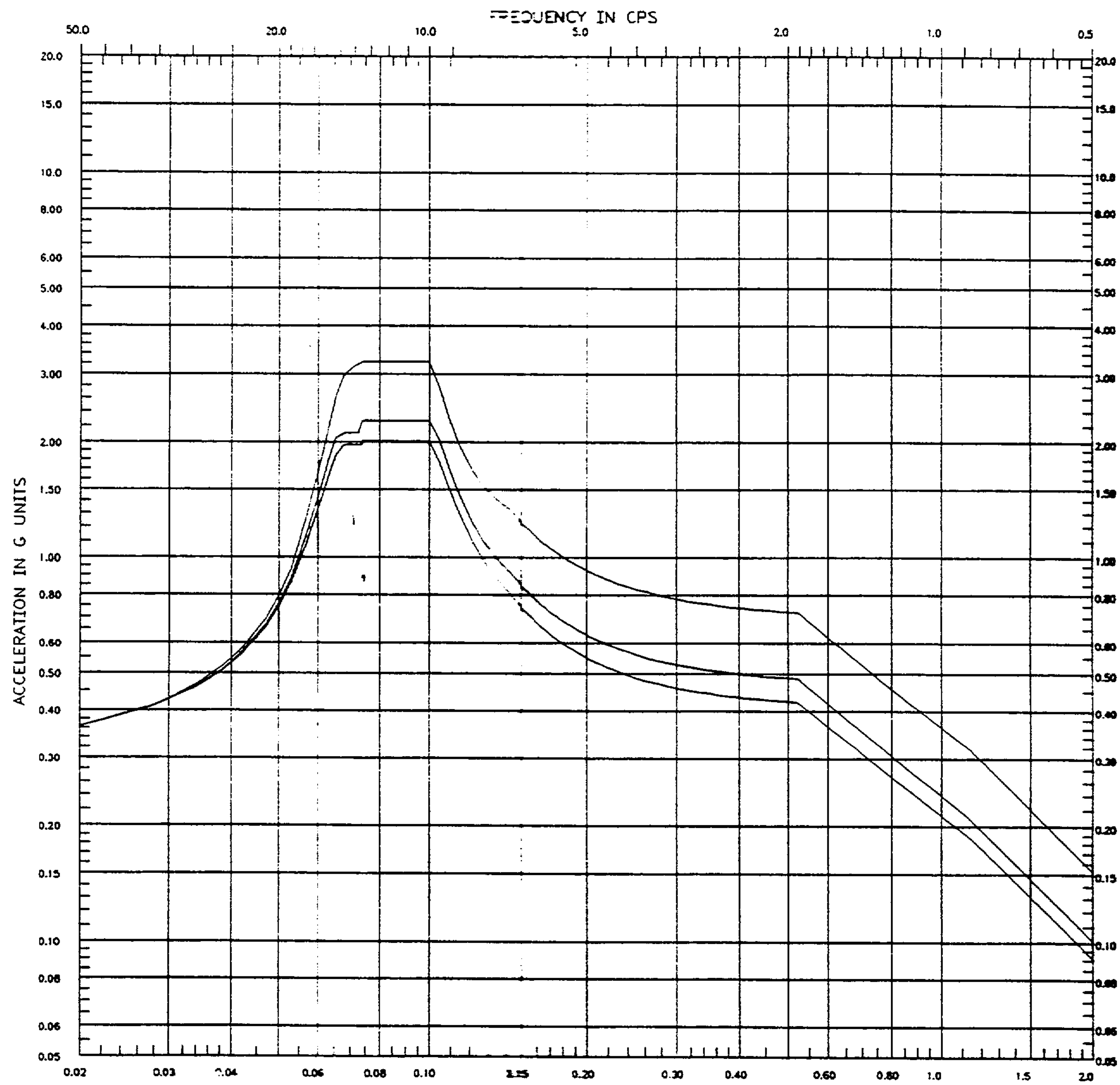
SME SPECTRA - **Widened**

NODE 6

PEAKS WIDENED ON EACH SIDE BY 15 %

DAMPING 0.005 0.03 0.05

Dresden Units 2&3



			SPECTRA NO.	SME
NODE	6		ELEVATION	570 **Wall and Slabs >12" Thick**
DIRECTION	VERT	ANGLE	LOCATION	REACTOR BUILDING

CALC NO. 8.11.6-2

SAFETY RELATED

PROJECT 9630-16 REV 0

SARGENT & LUNDY ENGINEERS

05/11/95

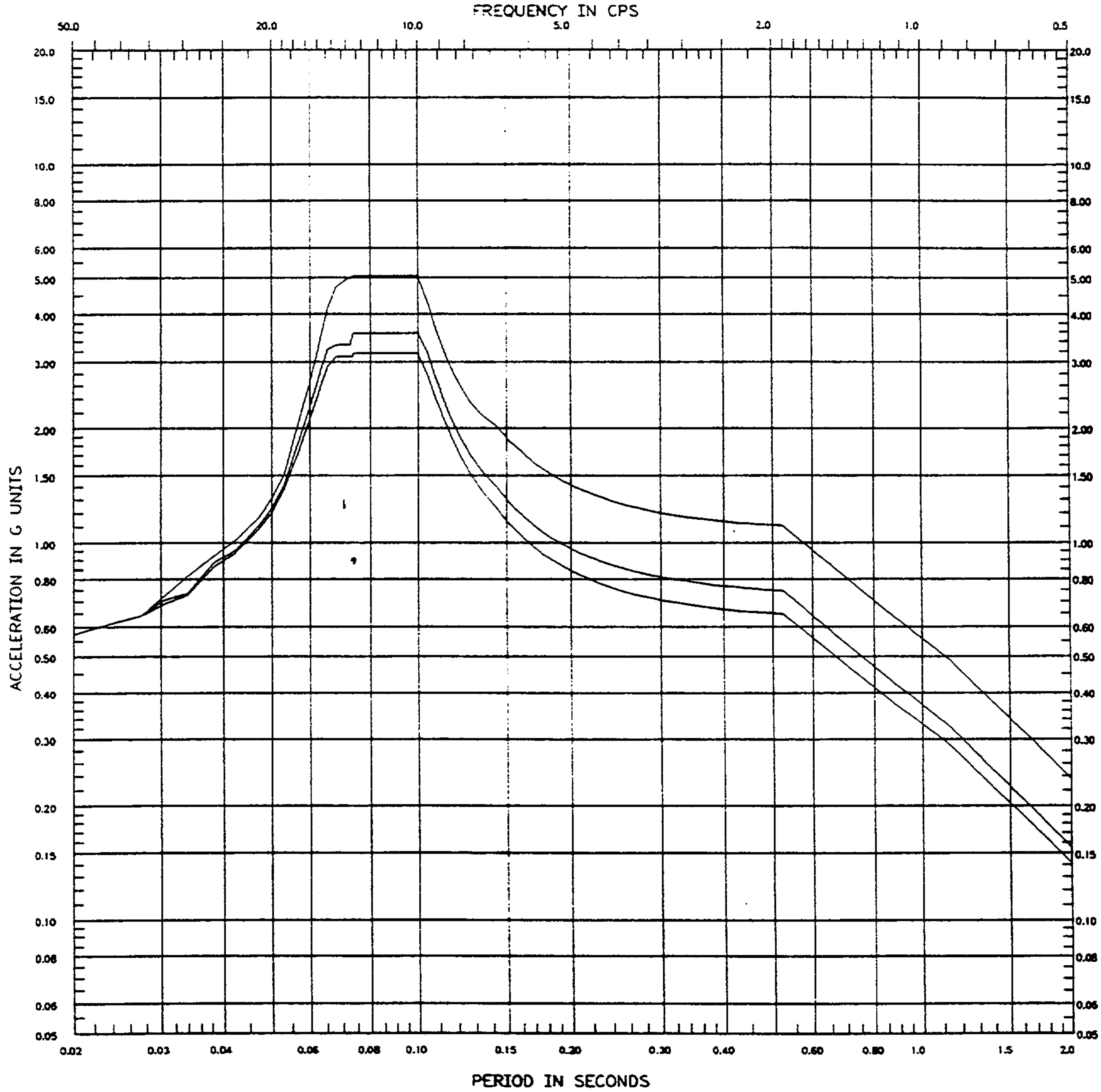
SME SPECTRA - **Widened** SLAB

NODE 43

PEAKS WIDENED ON EACH SIDE BY 15 %

DAMPING 0.005 0.03 0.05

Dresden Units 2&3



		SPECTRA NO.	SME	
NODE	43	ELEVATION	570	**Slab ≤ 12" Thick**
DIRECTION	VERT ANGLE	LOCATION	REACTOR BUILDING	

CALC NO. 8.11 6-2

SAFETY RELATED

SARGENT & LUNDY ENGINEERS

PROJECT 9630-16 REV 0

05/04/95

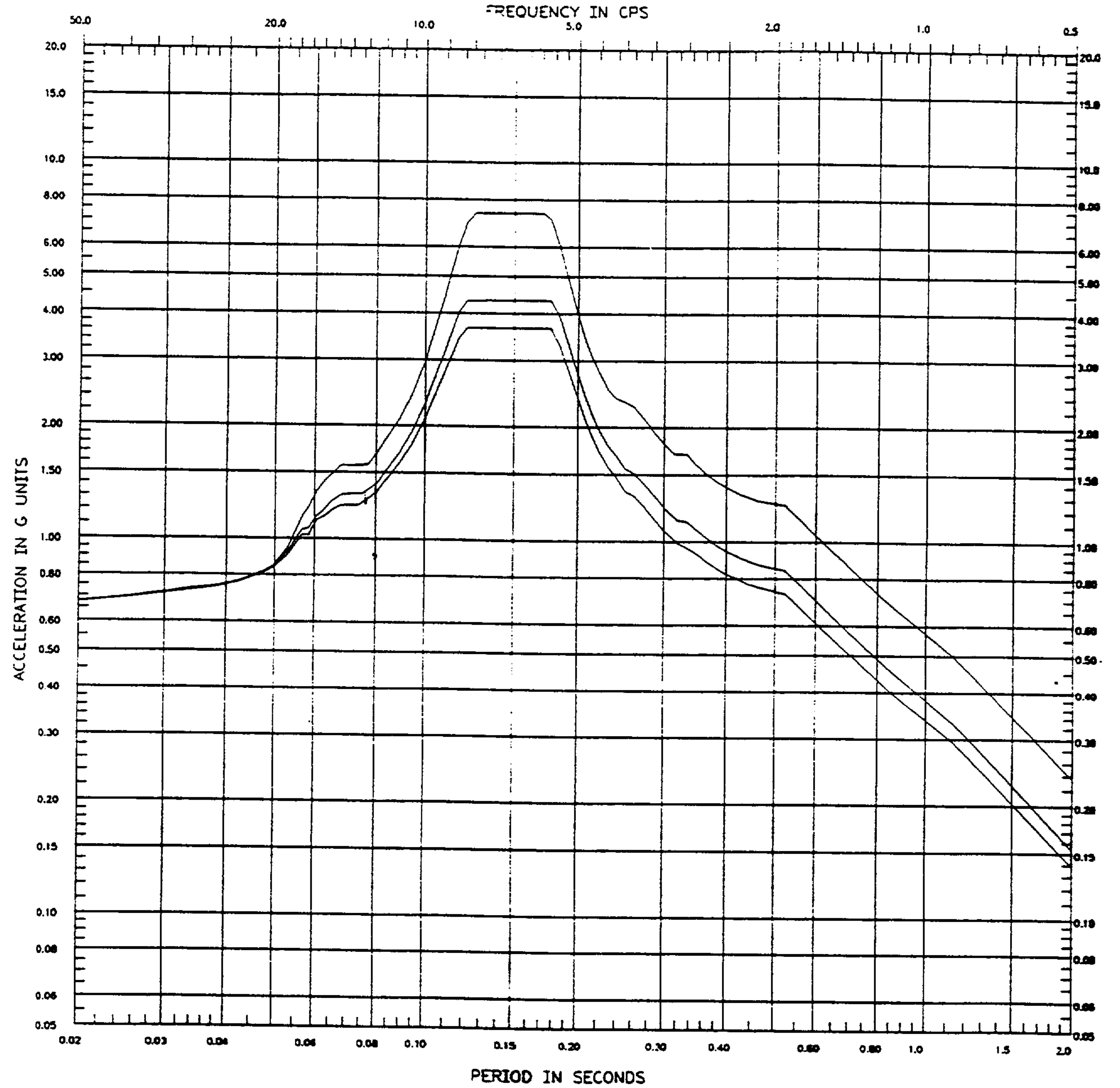
SME SPECTRA - **Widened**

NODE 5

PEAKS WIDENED ON EACH SIDE BY 15 %

NS DI DAMPING 0.005 0.03 0.05

Dresden Units 2&3



		SPECTRA NO.	SME
NODE	5	ELEVATION	589
DIRECTION	NS **Envelope For Entire Floor**	LOCATION	REACTOR BUILDING

2

CALC NO. 8.11.6-2

SAFETY RELATED

PROJECT 9630-16 REV 0

SARGENT & LUNDY ENGINEERS

05/04/95

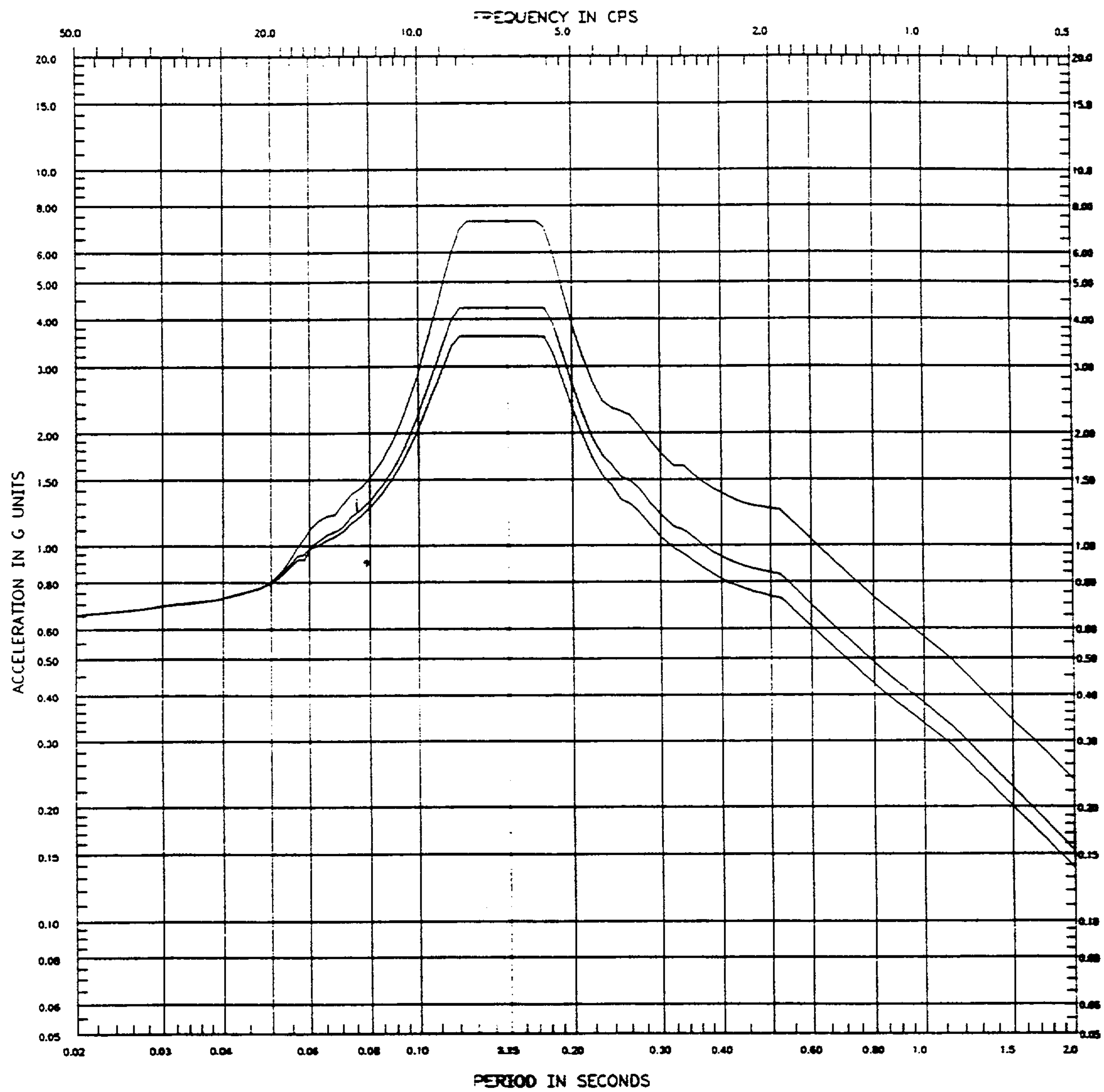
SME SPECTRA **Widened** NS DIRECTION

NODE 5

PEAKS WIDENED ON EACH SIDE BY 15 %

DAMPING 0.005 0.03 0.05

Dresden Units 2&3



SPECTRA NO. SME

NODE 5 ELEVATION 589

2 DIRECTION NS **Mass Center of Floor** LOCATION REACTOR BUILDING

CALC NO. E.11.6-2

SAFETY RELATED

PROJECT 9630-16 REV 0

SARGENT & LUNDY ENGINEERS

05/04/95

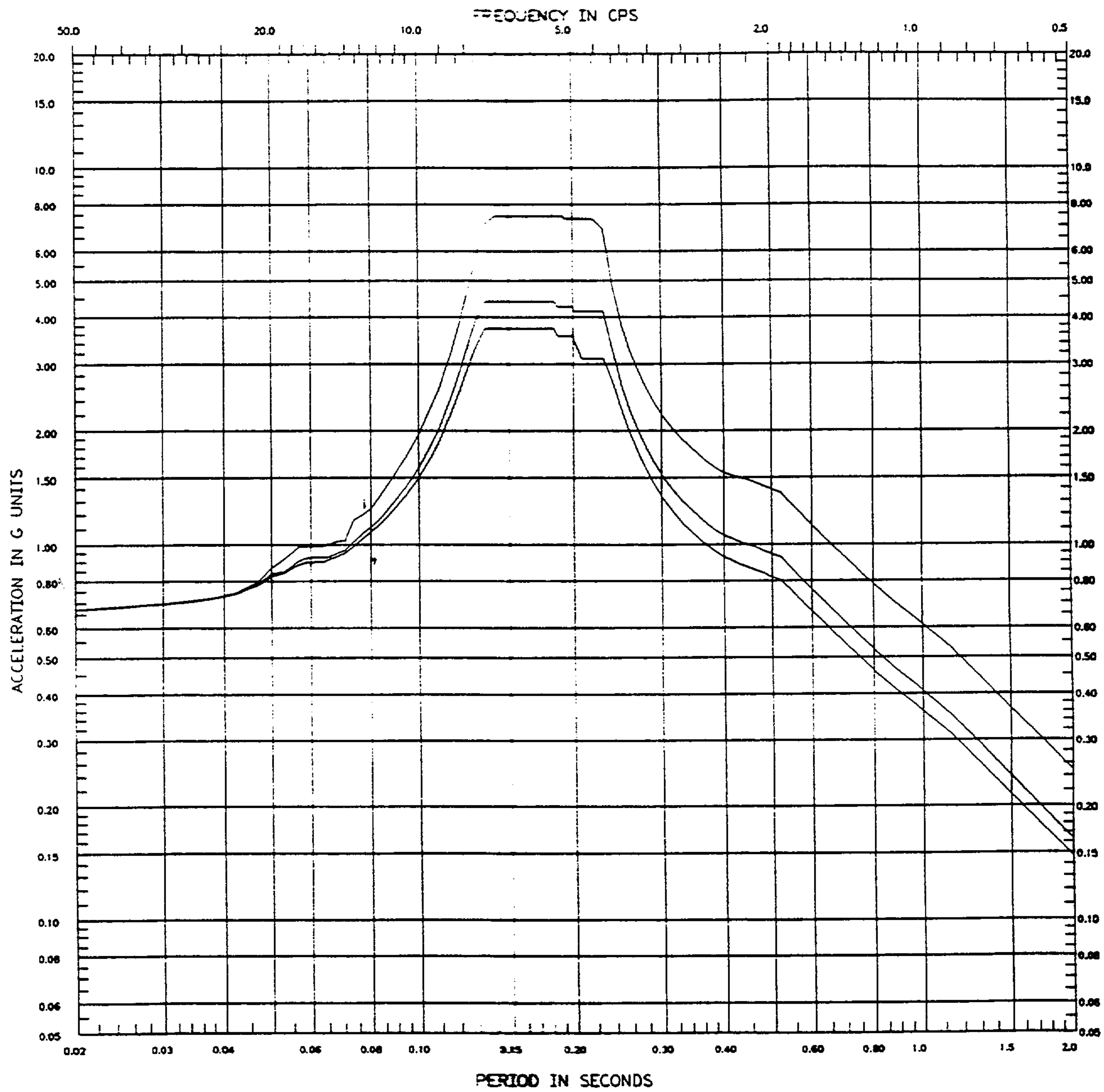
SME SPECTRA - **Widened** EW II

NODE 5

PEAKS WIDENED ON EACH SIDE BY 15 %

DAMPING 0.005 0.03 0.05

Dresden Units 2&3



		SPECTRA NO.	SME
NODE	5	ELEVATION	589
DIRECTION	EW **Envelope For Entire Floor**	LOCATION	REACTOR BUILDING

2

SARGENT & LUNDY
ENGINEERS

CALC NO. 8.11.6-2
SAFETY RELATED
PROJECT 9630-16 REV 0

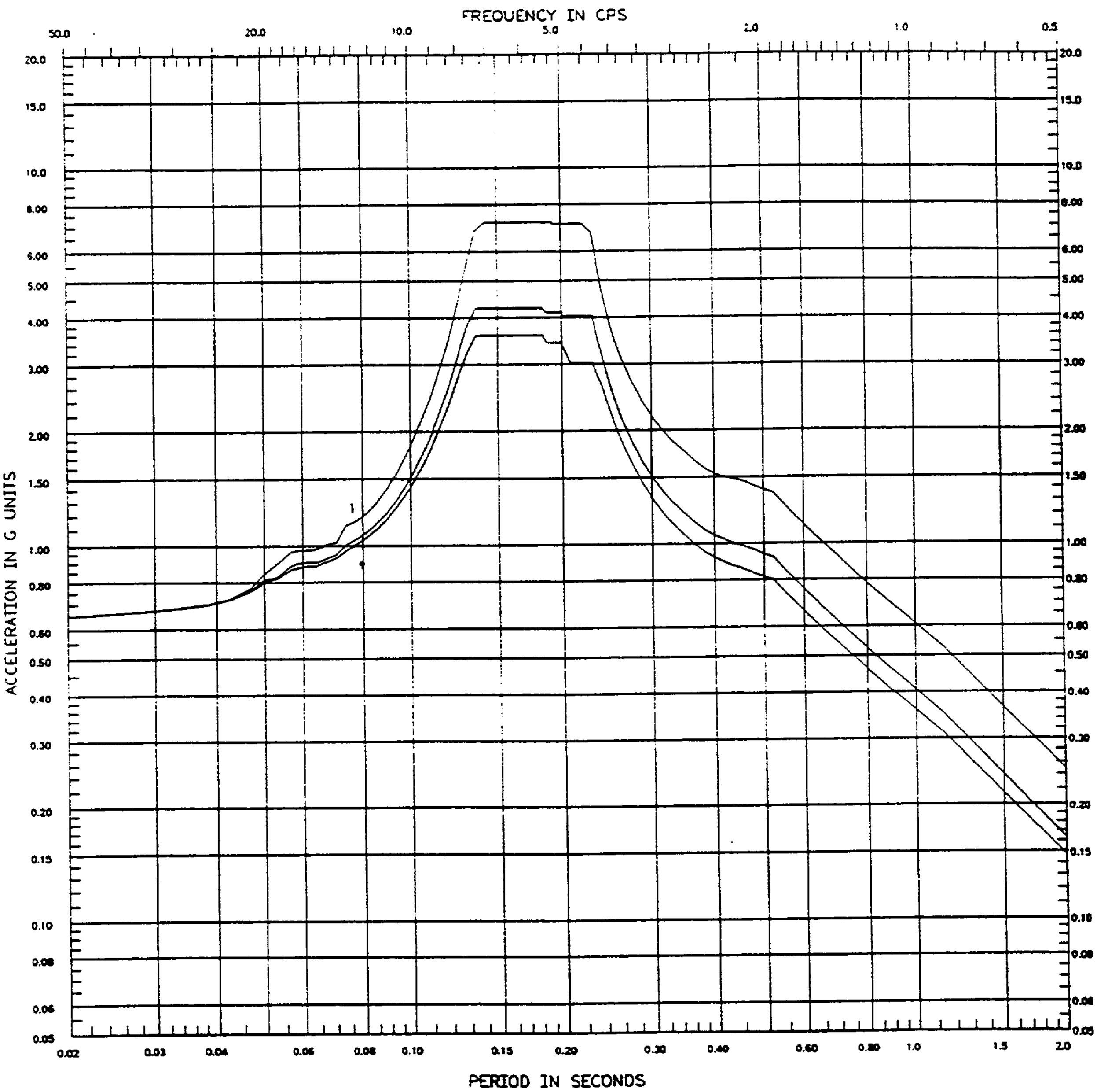
05/04/95

PEAKS WIDENED ON EACH SIDE BY 15 %

SME SPECTRA **Widened** EW DIRECTION DAMPING 0.005 0.03 0.05

NODE 5

Dresden Units 2&3



		SPECTRA NO.	SME
NODE	5	ELEVATION	589
DIRECTION	EW **Mass Center of Floor**	LOCATION	REACTOR BUILDING

2

CALC NO. 8.11.6-2

SAFETY RELATED

SARGENT & LUNDY ENGINEERS

PROJECT 9630-16 REV 0

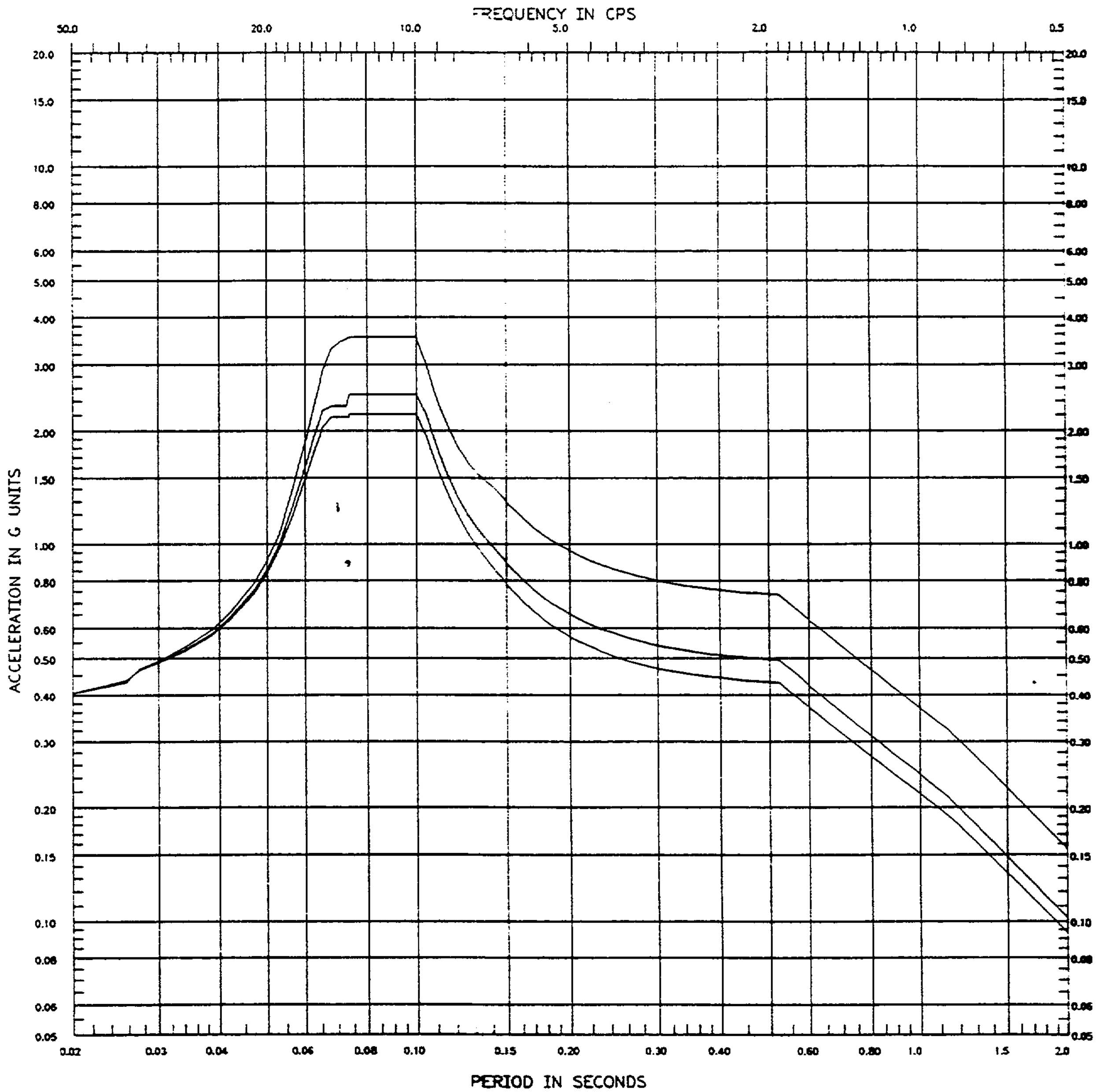
05/10/95

PEAKS WIDENED ON EACH SIDE BY 15 %

SME SPECTRA - **Widened** DIR. DAMPING 0.005 0.03 0.05

NODE 5

Dresden Units 2&3



			SPECTRA NO.	SME
NODE	5		ELEVATION	589 **Wall and Slabs >12" Thick**
DIRECTION	VERT	ANGLE	LOCATION	REACTOR BUILDING

CALC NO. 8.11.6-2

SAFETY RELATED

PROJECT 9630-16 REV 0

A2-29

SARGENT & LUNDY ENGINEERS

05/11/95

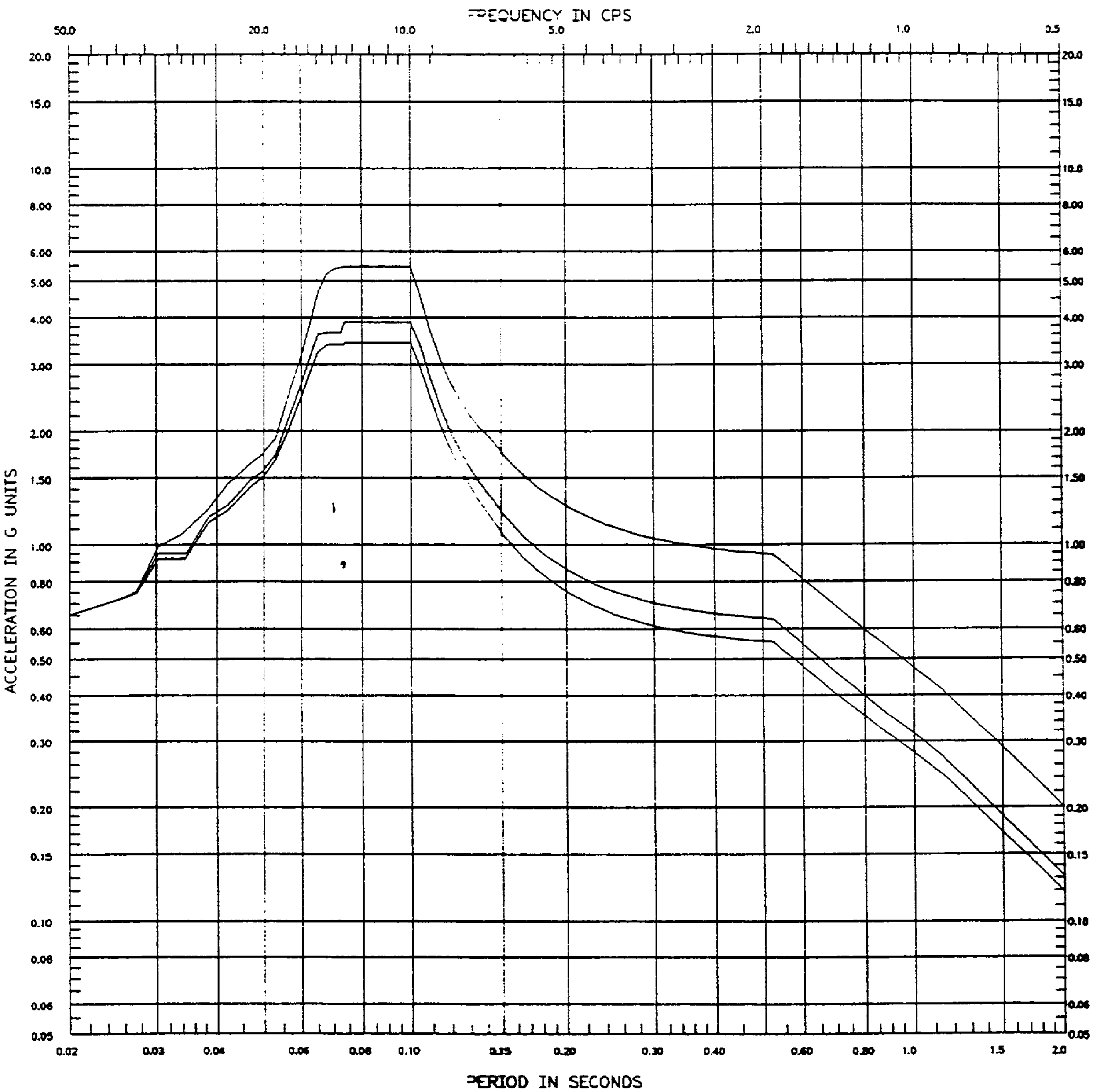
SME SPECTRA - **Widened** SLAB

NODE 41

PEAKS WIDENED ON EACH SIDE BY 15 %.

DAMPING 0.005 0.03 0.05

Dresden Units 2&3



SPECTRA NO. SME

NODE 41

ELEVATION 589 **Slab ≤ 12" Thick**

2 DIRECTION VERT ANGLE

LOCATION REACTOR BUILDING

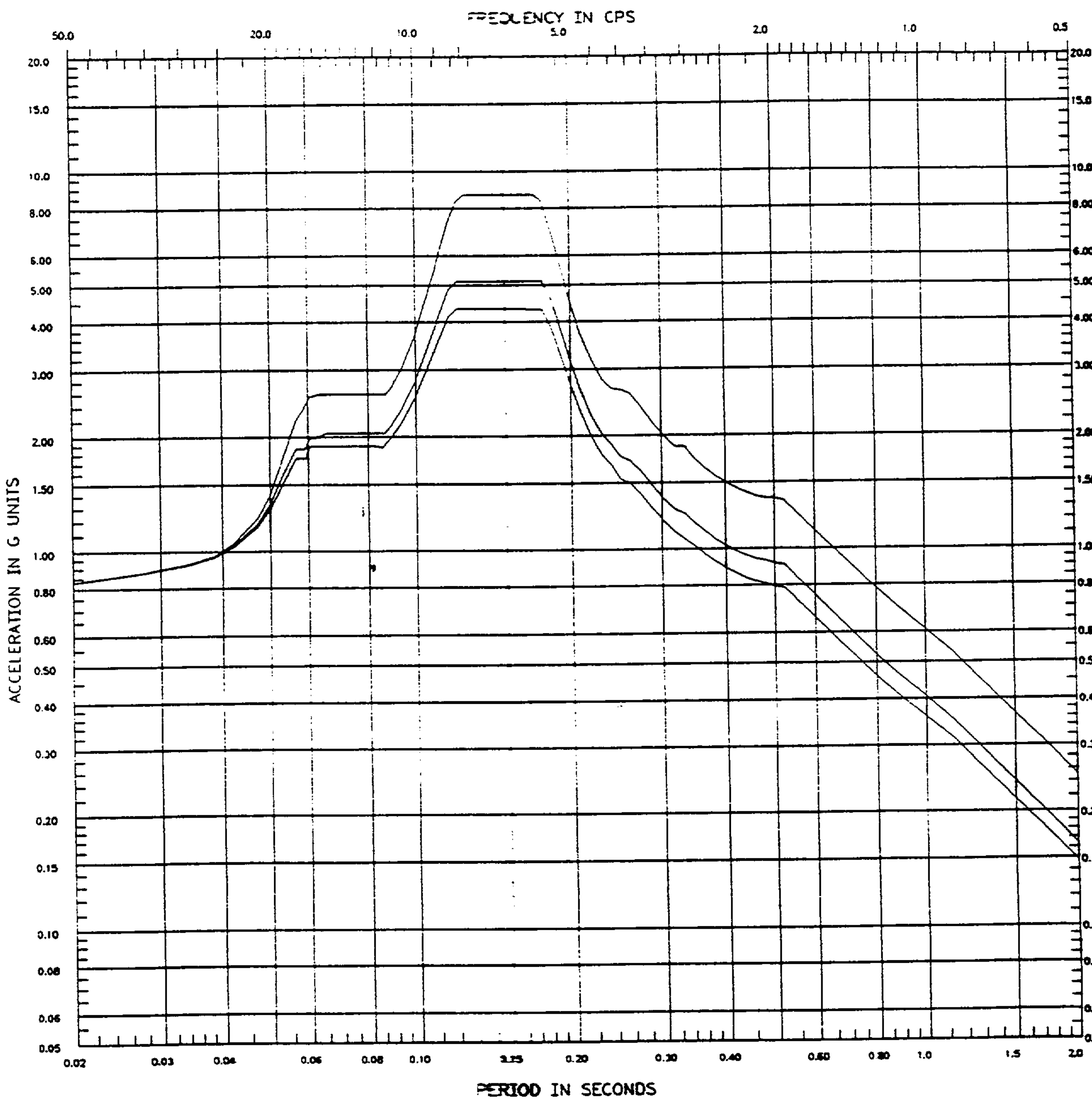
CALC NO. 8.11.6-2
SAFETY RELATED
PROJECT 9630-16 REV 0

SARGENT & LUNDY ENGINEERS

05/04/95
SME SPECTRA - **Widened**
NODE 4

PEAKS WIDENED ON EACH SIDE BY 15 %
NS DI DAMPING 0.005 0.03 0.05

Dresden Units 2&3



		SPECTRA NO.	SME
NODE	4	ELEVATION	613
DIRECTION	NS **Envelope For Entire Floor**	LOCATION	REACTOR BUILDING

CALC NO. 8.11 6-2
SAFETY RELATED
PROJECT 9630-16 REV 0

SARGENT & LUNDY
ENGINEERS

PEAKS WIDENED ON EACH SIDE BY 15 %

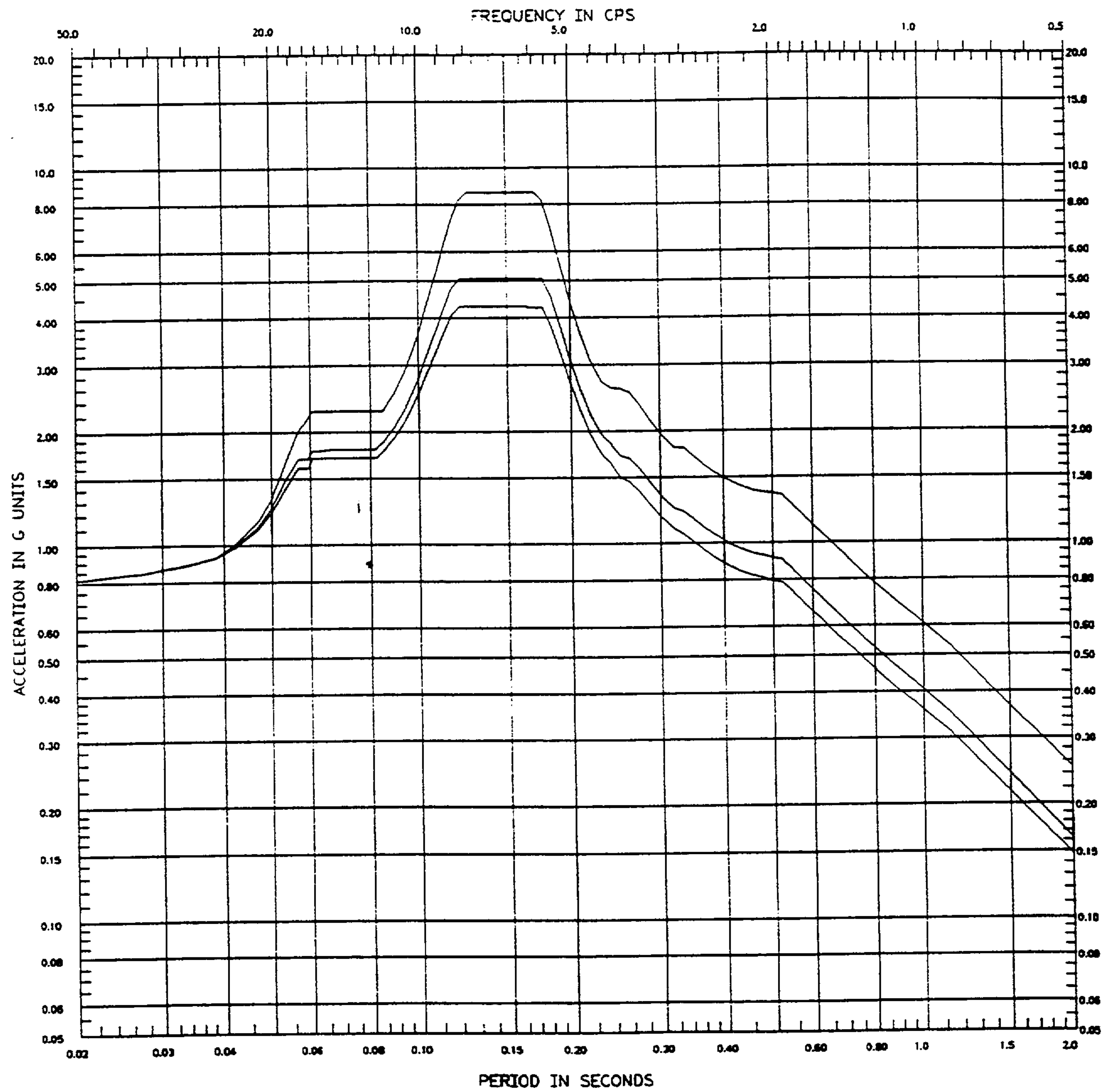
05/04/95

SME SPECTRA - **Widened** NS DIRECTION

DAMPING 0.005 0.03 0.05

NODE 4

Dresden Units 2&3



SPECTRA NO. SME

NODE 4

ELEVATION 613

DIRECTION NS **Mass Center of Floor**

LOCATION REACTOR BUILDING

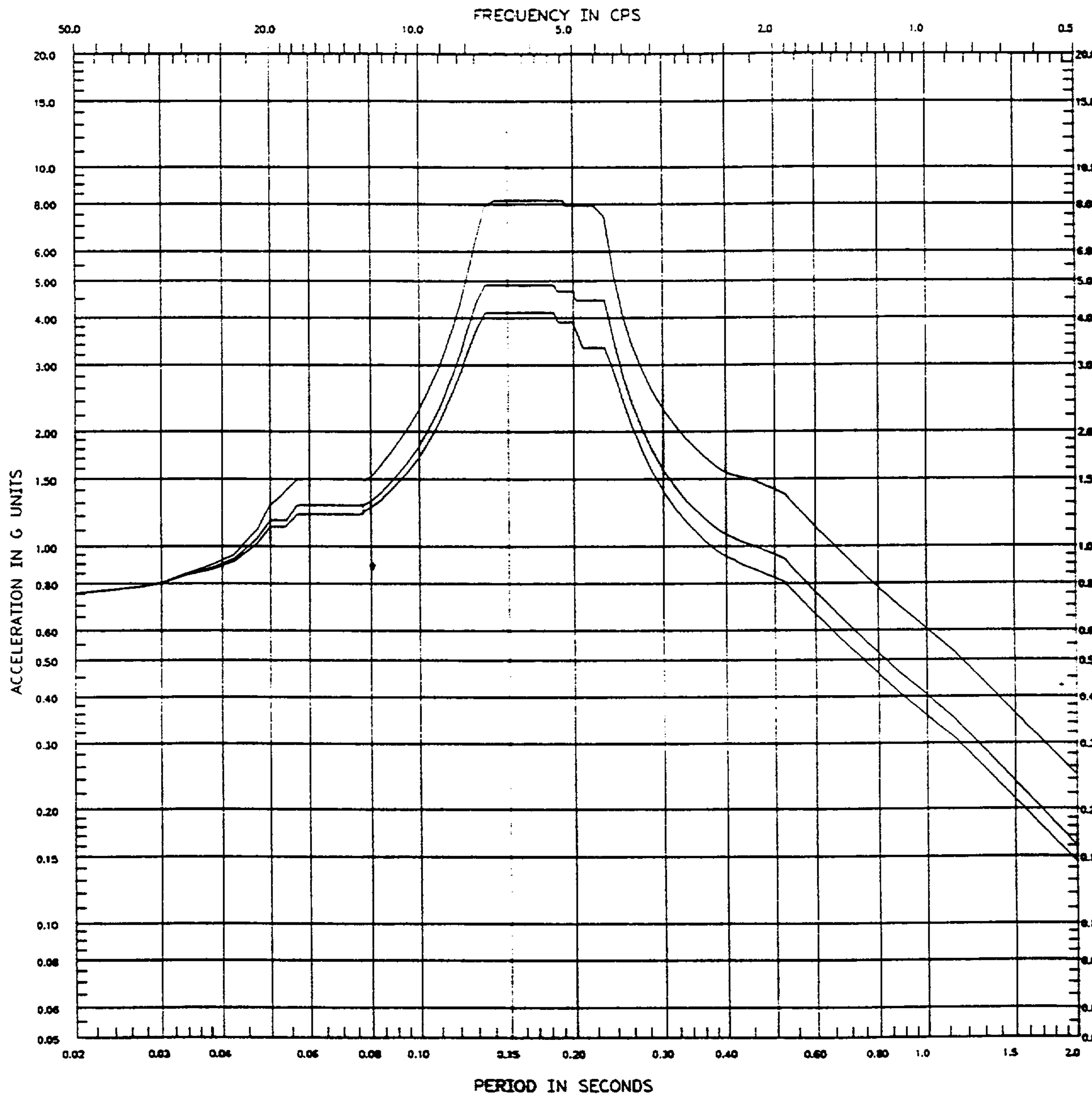
CALC NO. 8.11.6-2
SAFETY RELATED
PROJECT 9630-16 REV 0

SARGENT & LUNDY ENGINEERS

05/04/95
SME SPECTRA Widened
NODE 4

PEAKS WIDENED ON EACH SIDE BY 15 %
EW DI DAMPING 0.005 0.03 0.05

Dresden Units 2&3



		SPECTRA NO.	SME
NODE	4	ELEVATION	613
DIRECTION	EW Envelope For Entire Floor	LOCATION	REACTOR BUILDING

CALC NO. 8.11 6-2

SAFETY RELATED

PROJECT 9630-16 REV 0

SARGENT & LUNDY ENGINEERS

05/04/95

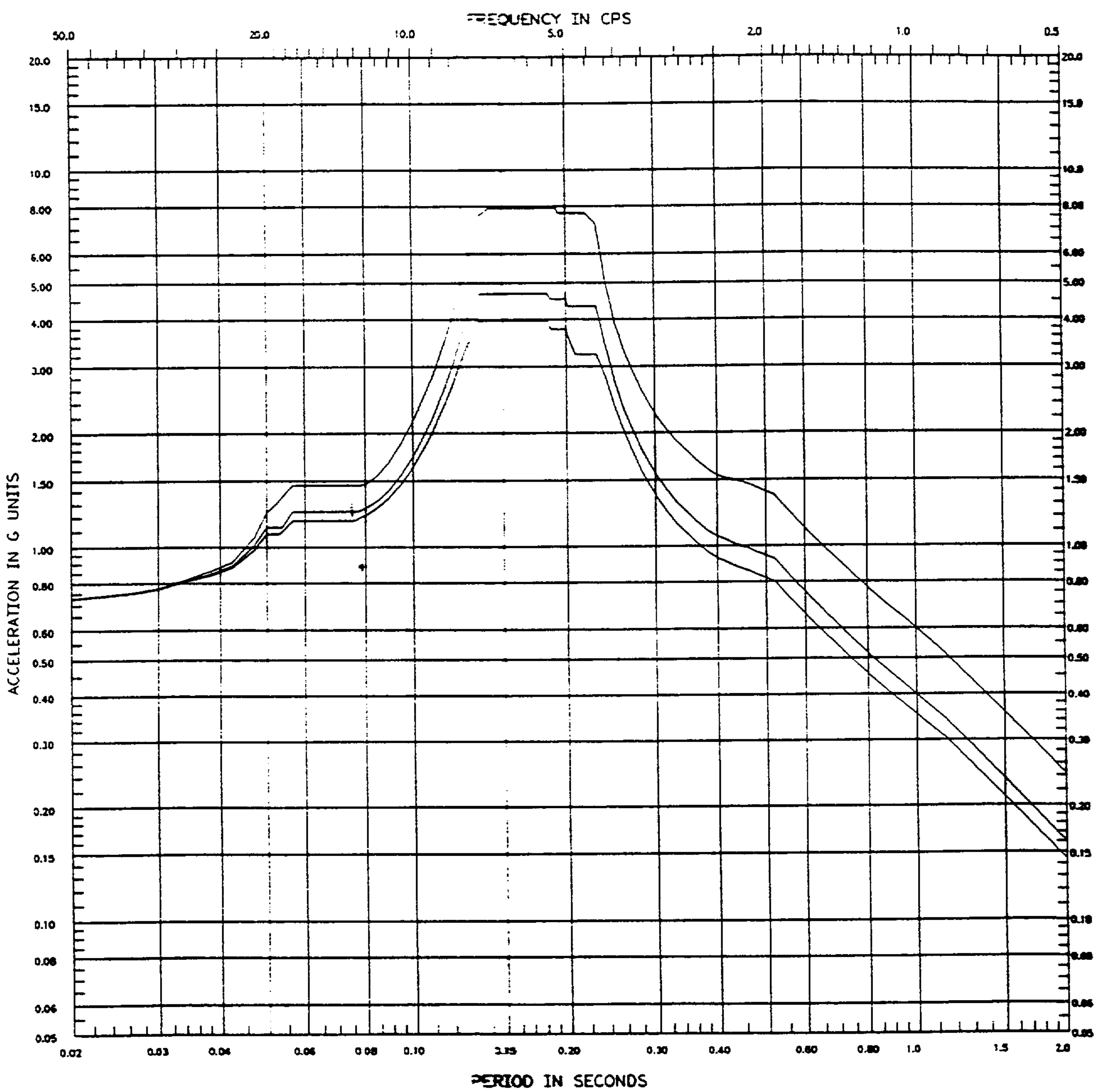
SME SPECTRA **Widened** EW DIRECTION

NODE 4

PEAKS WIDENED ON EACH SIDE BY 15 %.

DAMPING 0.005 0.03 0.05

Dresden Units 2&3



		SPECTRA NO.	SME
NODE	4	ELEVATION	613
DIRECTION	EW **Mass Center of Floor**	LOCATION	REACTOR BUILDING

1

CALC NO. 8.11.6-2

SAFETY RELATED

PROJECT 9630-16 REV 0

SARGENT & LUNDY ENGINEERS

05/10/95

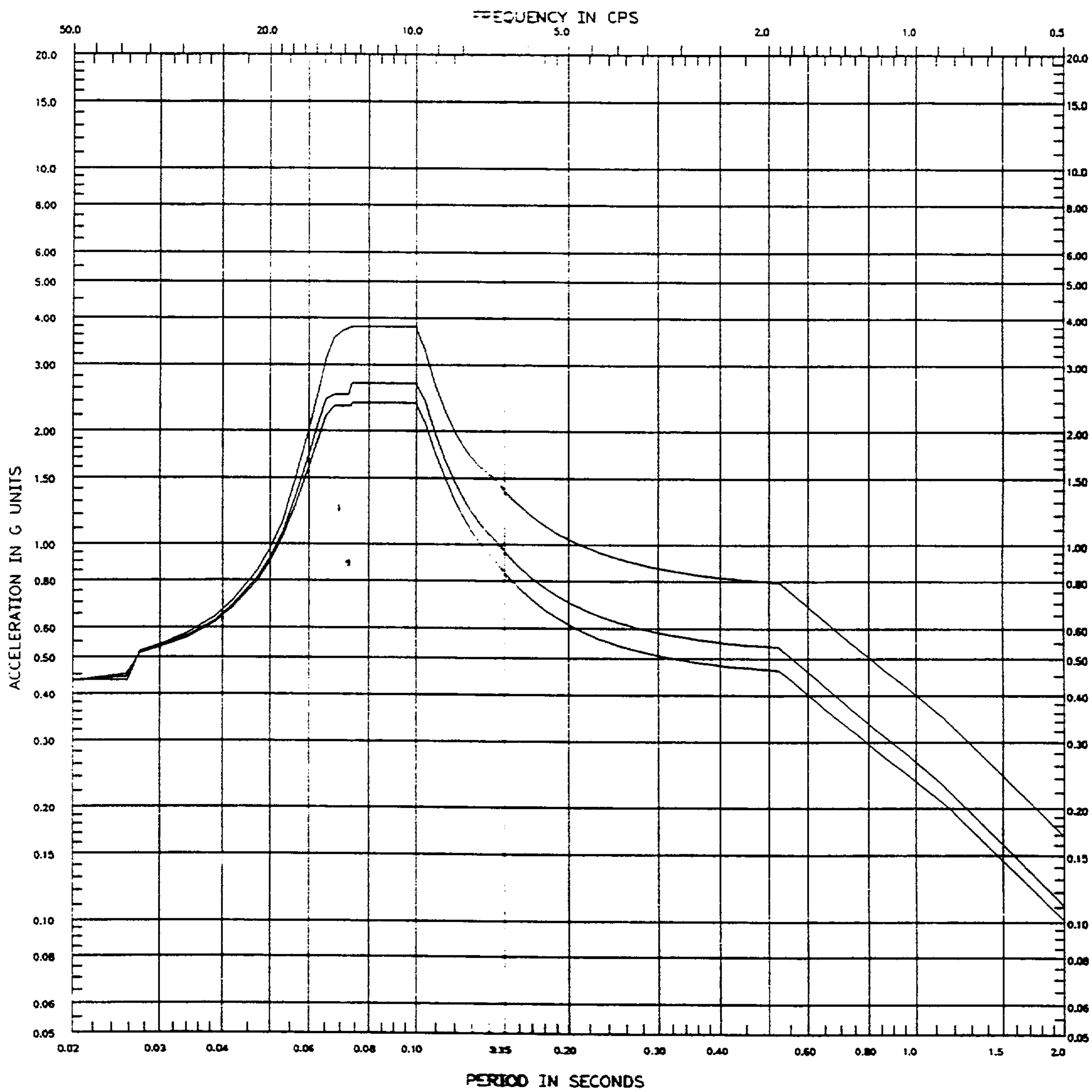
SME SPECTRA - Widened DIR.

NODE 4

PEAKS WIDENED ON EACH SIDE BY 15 %.

DAMPING 0.005 0.03 0.05

Dresden Units 2&3



SPECTRA NO. SME

NODE 4

ELEVATION 613 Wall and Slabs >12" Thick

DIRECTION VERT ANGLE

LOCATION REACTOR (SLAB (= 12")

CALC NO. 8.11.6-2

SAFETY RELATED

PROJECT 9630-16 REV 0

SARGENT & LUNDY ENGINEERS

05/11/95

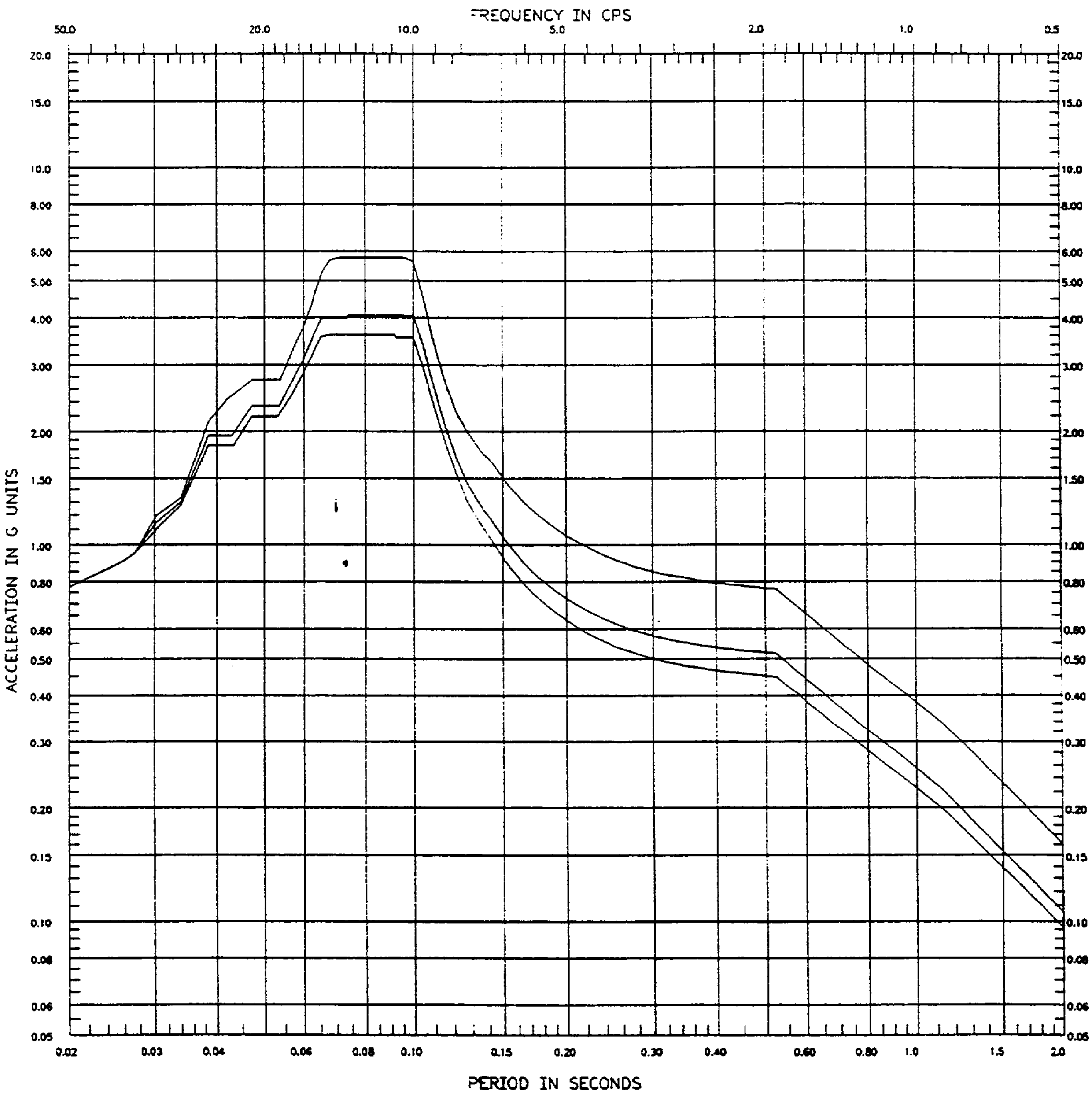
SME SPECTRA - **Widened** SLAB

NODE 20

PEAKS WIDENED ON EACH SIDE BY 15 %.

DAMPING 0.005 0.03 0.05

Dresden Units 2&3



SPECTRA NO. SME

NODE 20 ELEVATION 613 ' **Slab ≤ 12" Thick**

1 DIRECTION VERT ANGLE LOCATION REACTOR BUILDING

CALC NO. 8.11.6-2

SAFETY RELATED

SARGENT & LUNDY ENGINEERS

PROJECT 9630-16 REV 0

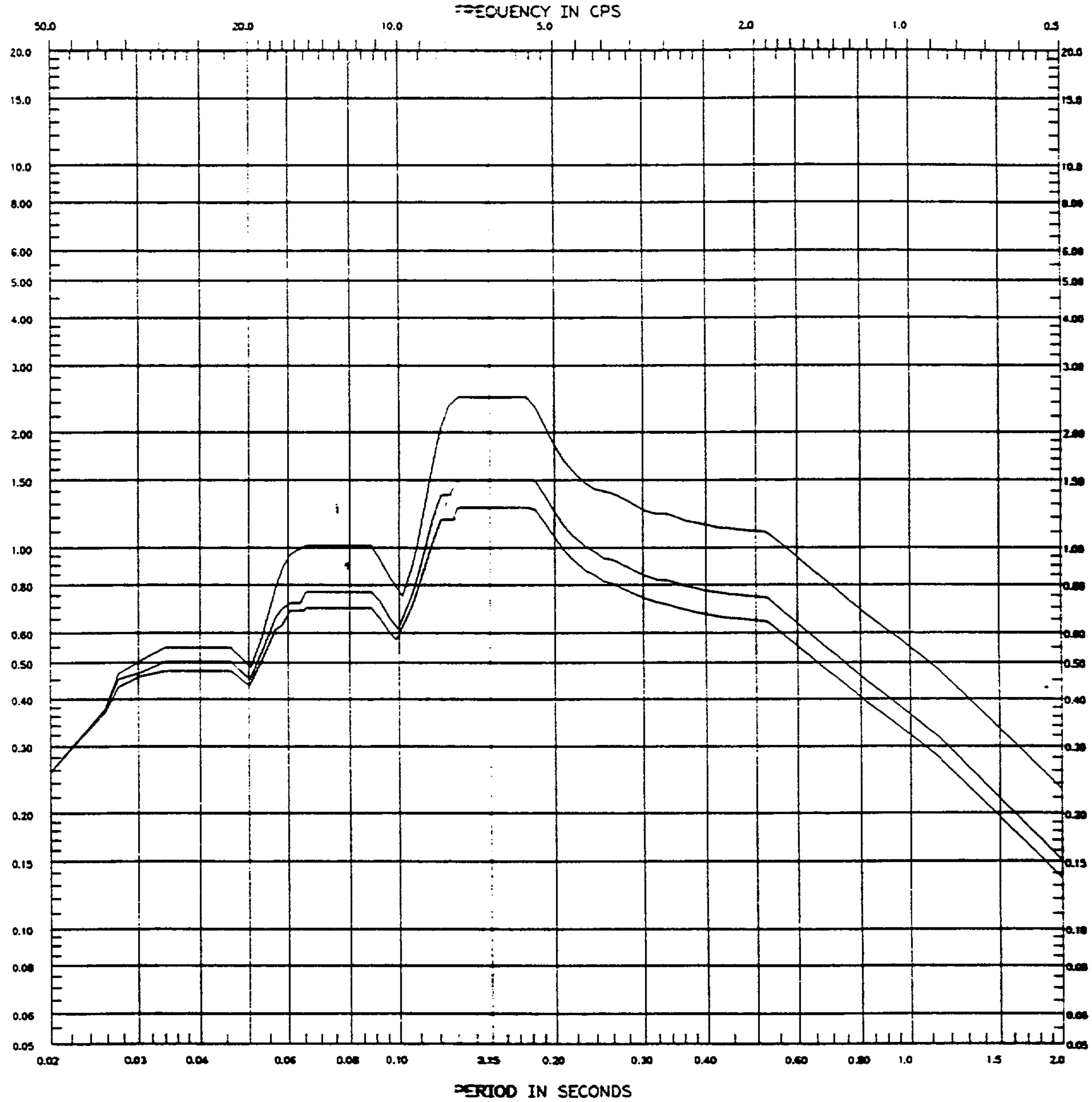
05/04/95

PEAKS WIDENED ON EACH SIDE BY 15 % -

SME SPECTRA - Widened .S DI DAMPING 0.005 0.03 0.05

NODE 61

Dresden Units 2&3



		SPECTRA NO.	SME
NODE	61	ELEVATION	534
DIRECTION	NS **Envelope For Entire Floor**	LOCATION	CONTROL BUILDING

CALC NO. 8.11.6-2

SAFETY RELATED

PROJECT 9630-16 REV 0

SARGENT & LUNDY ENGINEERS

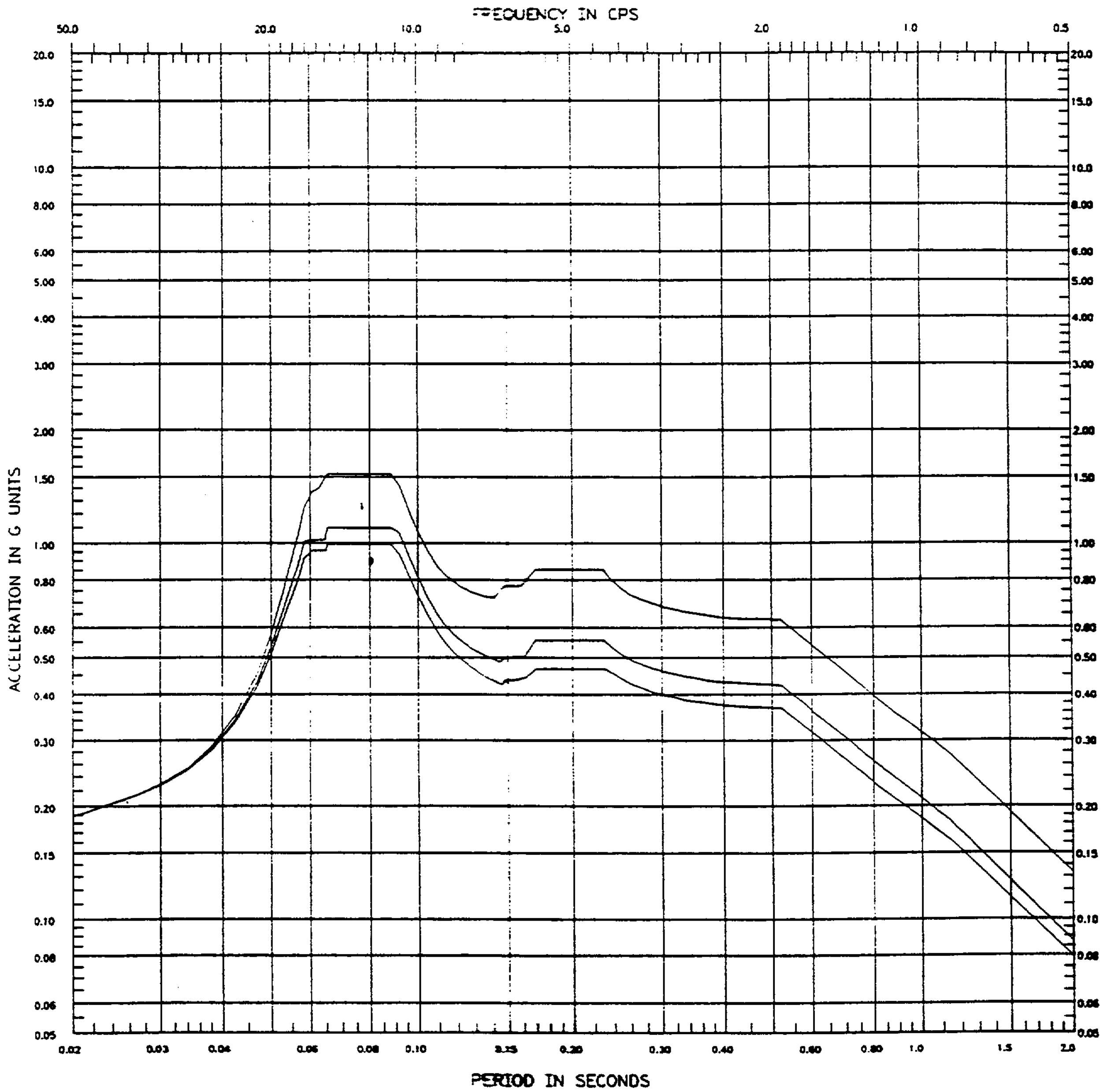
05/04/95

PEAKS WIDENED ON EACH SIDE BY 15 %

SME SPECTRA - **Widened** EW DI DAMPING 0.005 0.03 0.05

NODE 61

Dresden Units 2&3



		SPECTRA NO.	SME
NODE	61	ELEVATION	534
DIRECTION	EW **Envelope For Entire Floor**	LOCATION	CONTROL BUILDING

CALC NO. 8.11.6-2

SAFETY RELATED

PROJECT 9630-16 REV 0

SARGENT & LUNDY ENGINEERS

05/12/95

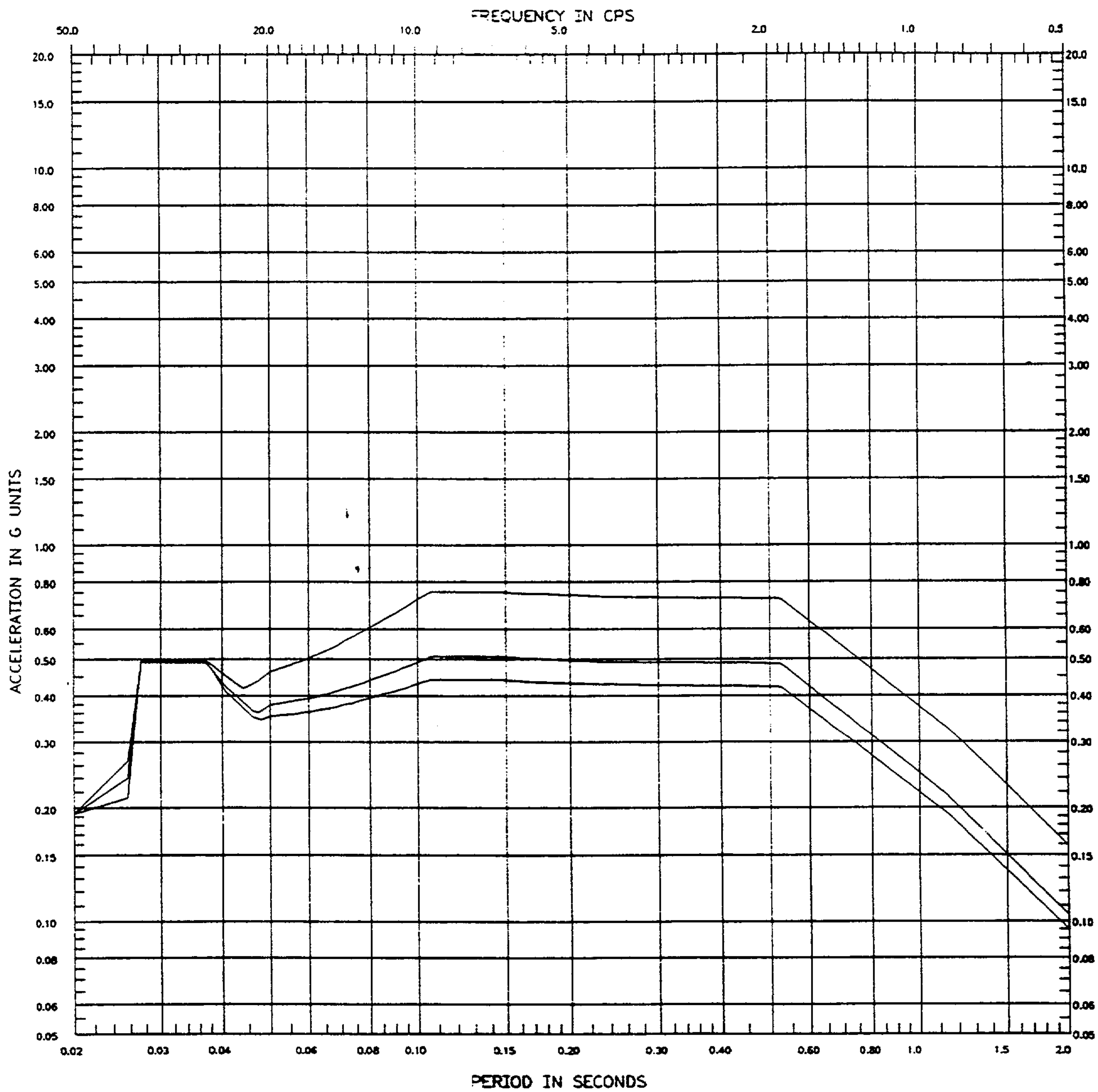
SME SPECTRA - **Widened**

NODE 16

PEAKS WIDENED ON EACH SIDE BY 15 %.

DAMPING 0.005 0.03 0.05

Dresden Units 2&3



			SPECTRA NO.	SME
NODE	16		ELEVATION	534 **Wall and Slabs >12" Thick**
DIRECTION	VERT	ANGLE	LOCATION	CONTROL ROOM

CALC NO. 8.11.6-2

SAFETY RELATED

PROJECT 9630-16 REV 0

SARGENT & LUNDY ENGINEERS

05/11/95

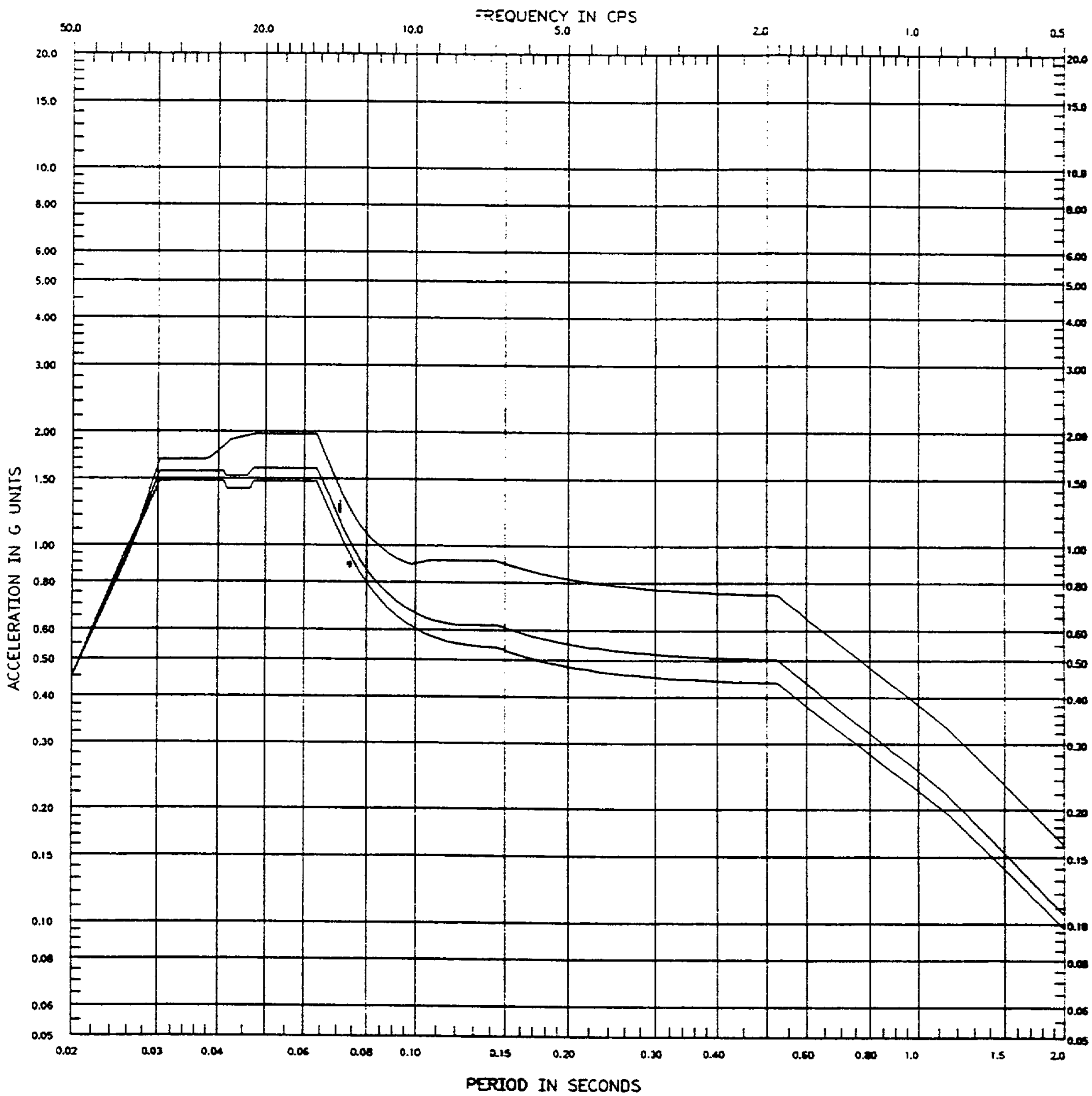
SME SPECTRA - Widened SLAB

NODE 51

PEAKS WIDENED ON EACH SIDE BY 15 %.

DAMPING 0.005 0.03 0.05

Dresden Units 2&3



			SPECTRA NO.	SME
NODE	51		ELEVATION	534 Slab ≤ 12" Thick
DIRECTION	VERT	ANGLE	LOCATION	CONTROL ROOM

7

A2-40

SARGENT & LUNDY
ENGINEERS

CALC NO. E.11 6-2
SAFETY RELATED
PROJECT 9630-16 REV 0

05/04/95

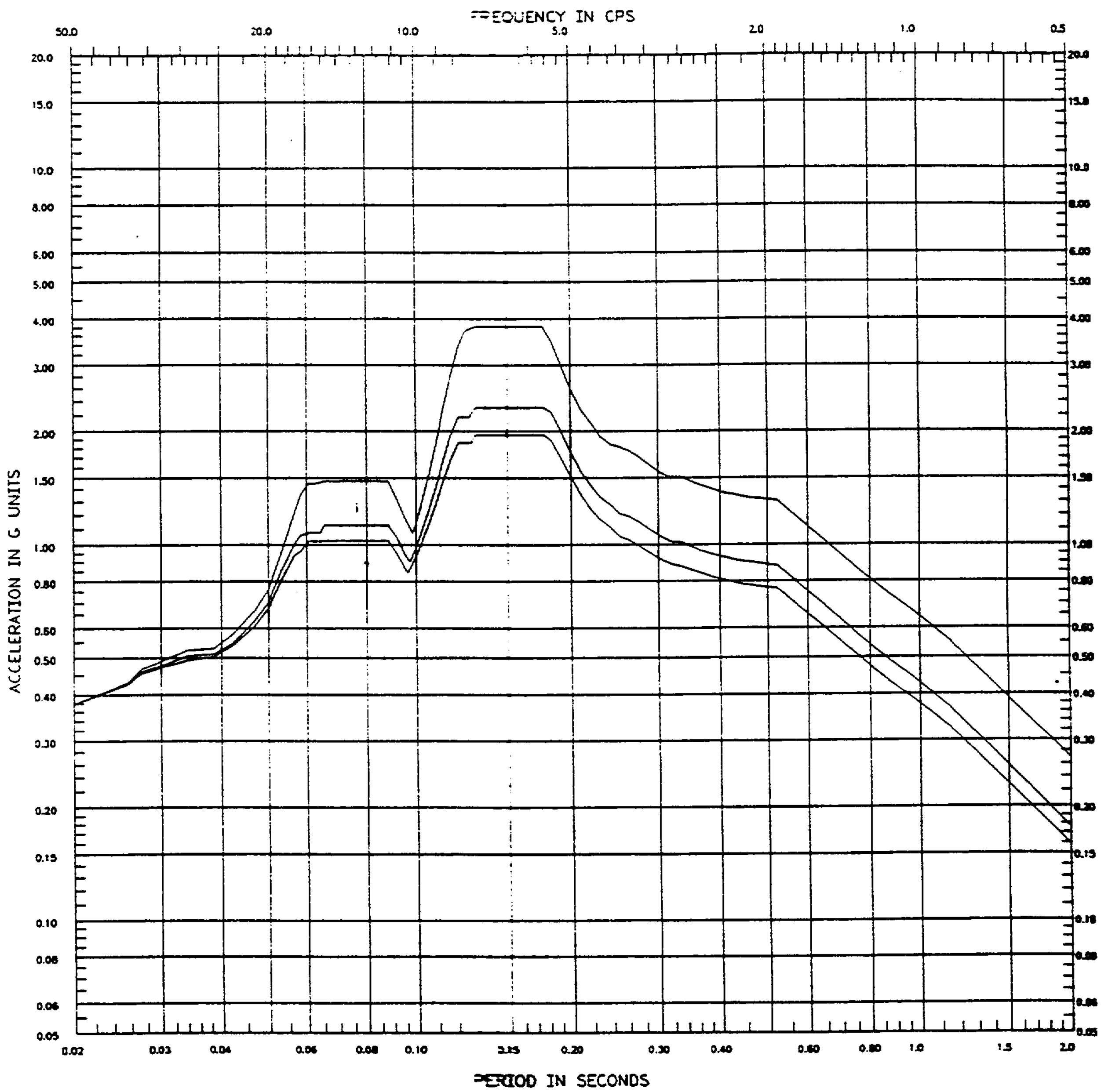
SME SPECTRA - Widened . NS DI

NODE 63

PEAKS WIDENED ON EACH SIDE BY 15 %

DAMPING 0.005 0.03 0.05

Dresden Units 2&3



			SPECTRA NO.	SME
NODE	63		ELEVATION	549
DIRECTION	NS	Envelope For Entire Floor	LOCATION	CONTROL BUILDING

9

CALC NO. 8.11.6-2

SAFETY RELATED

PROJECT 9630-16 REV 0

SARGENT & LUNDY ENGINEERS

05/04/95

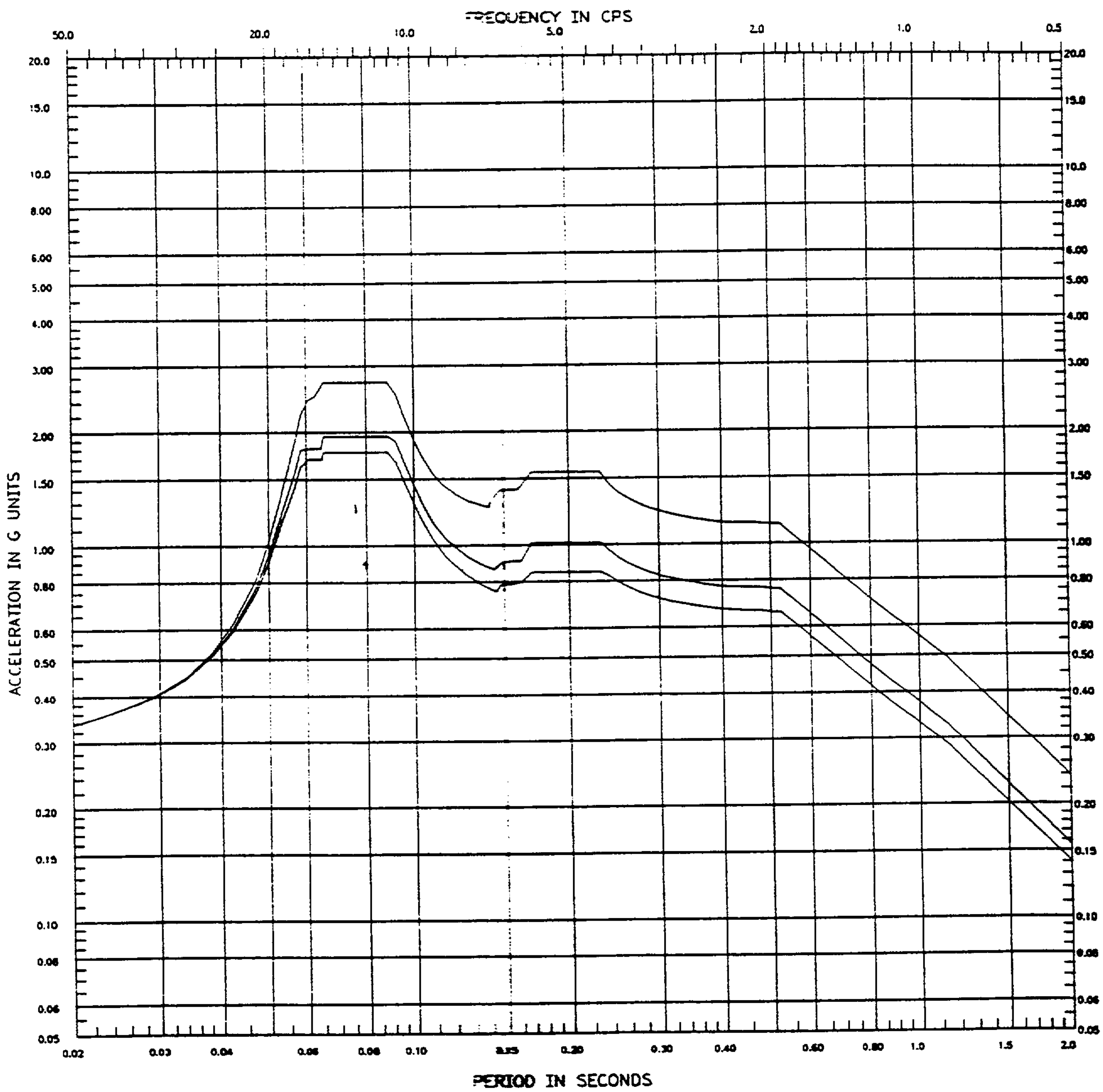
SME SPECTRA · Widened

NODE 63

PEAKS WIDENED ON EACH SIDE BY 15 % .

EW DI DAMPING 0.005 0.03 0.05

Dresden Units 2&3



		SPECTRA NO.	SME
NODE	63	ELEVATION	549
DIRECTION	EW Envelope For Entire Floor	LOCATION	CONTROL BUILDING

9

CALC NO. 8.11.6-2

SAFETY RELATED

PROJECT 9630-16 REV 0

SARGENT & LUNDY ENGINEERS

05/10/95

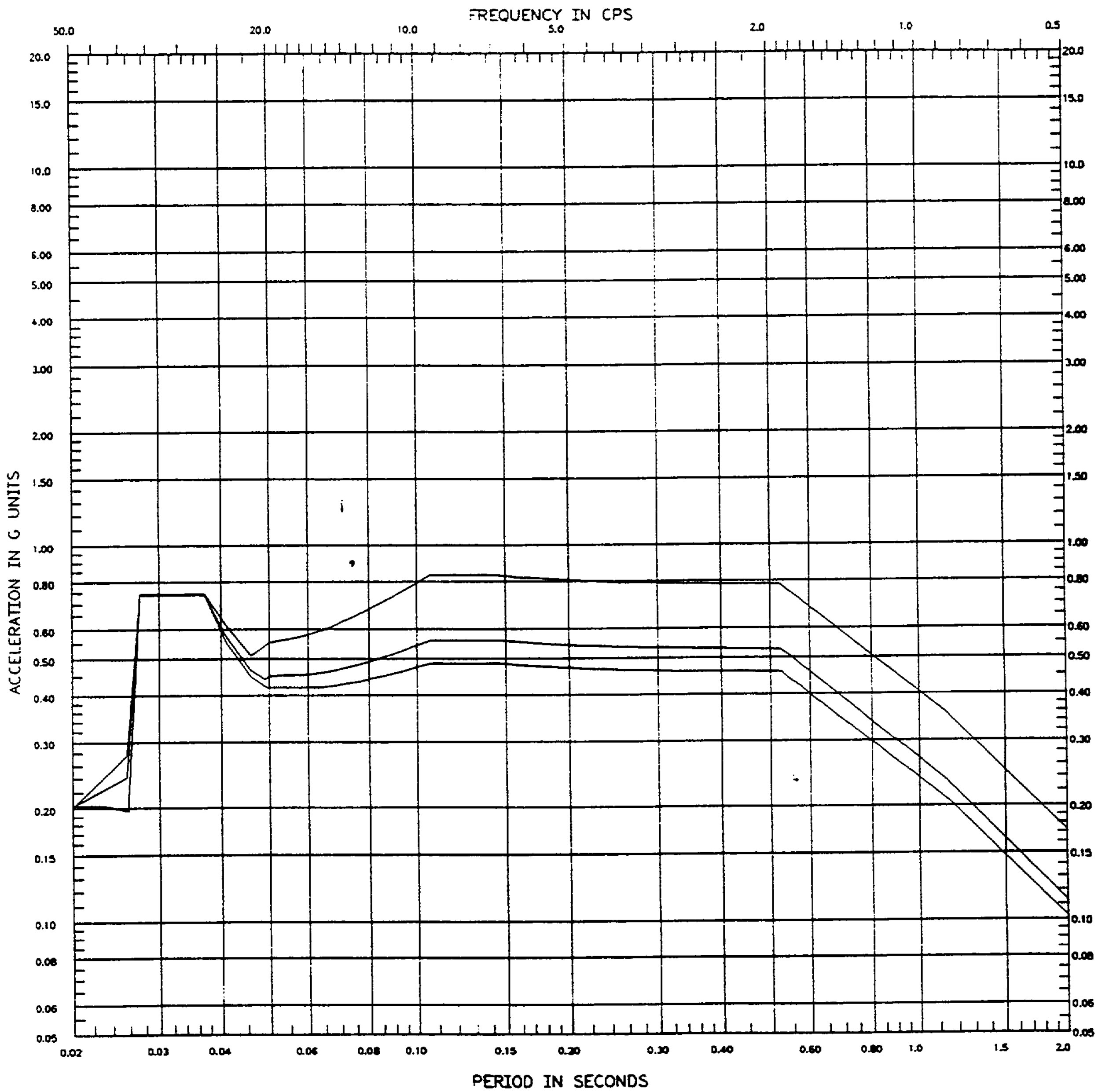
SME SPECTRA - Widened DIR.

NODE 17

PEAKS WIDENED ON EACH SIDE BY 15 %.

DAMPING 0.005 0.03 0.05

Dresden Units 2&3



SPECTRA NO. SME

NODE 17

ELEVATION 549 **Wall and Slabs >12" Thick**

10 DIRECTION VERT ANGLE

LOCATION CONTROL ROOM

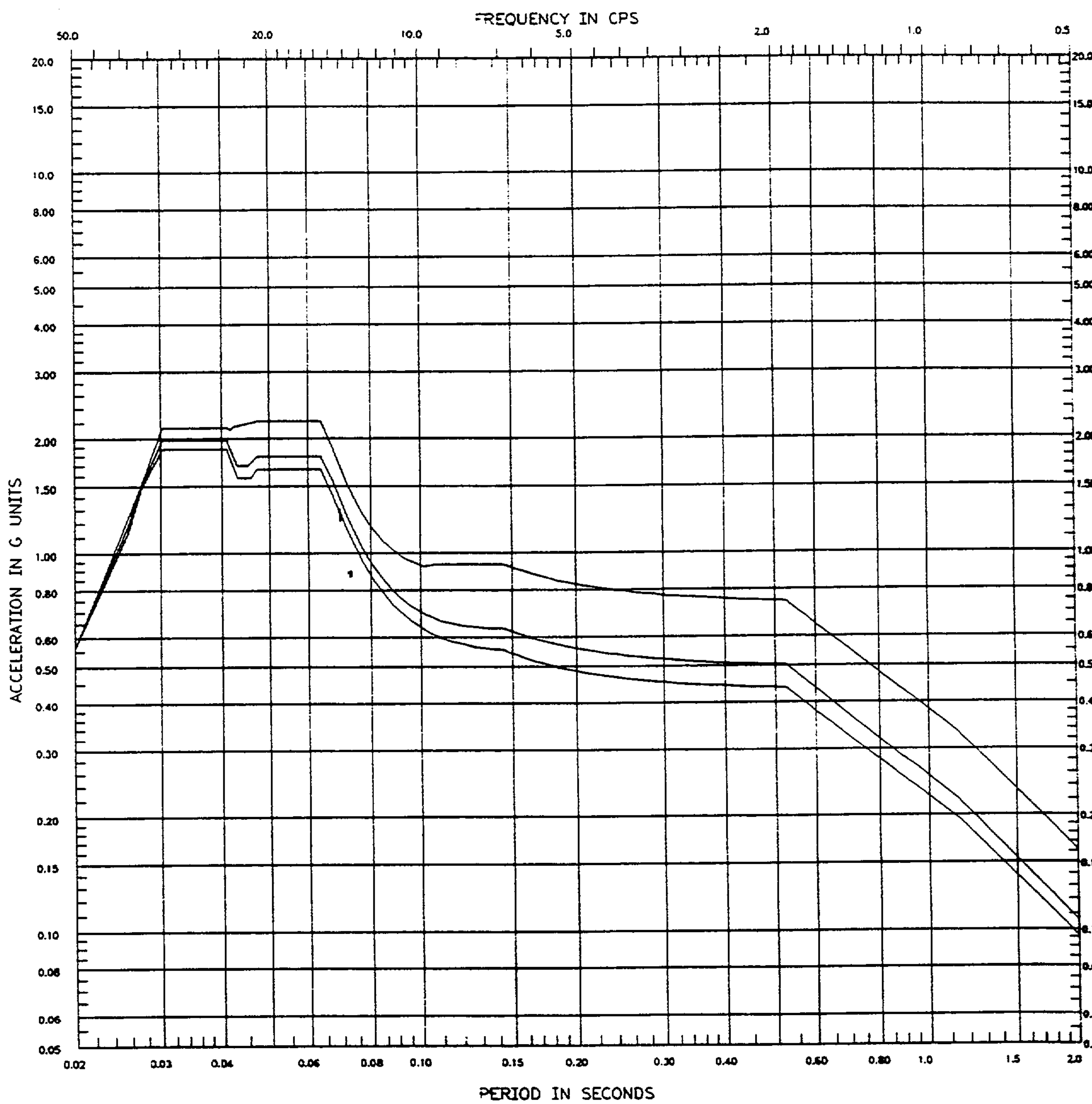
CALC NO. 8.11.6-2
SAFETY RELATED
PROJECT 9630-16 REV 0

SARGENT & LUNDY ENGINEERS

05/11/95
SME SPECTRA - Widened SLAB
NODE 53

PEAKS WIDENED ON EACH SIDE BY 15 %
DAMPING 0.005 0.03 0.05

Dresden Units 2&3



SPECTRA NO. SME
NODE 53 ELEVATION 549 **Slab ≤ 12" Thick**
DIRECTION VERT ANGLE LOCATION CONTROL ROOM

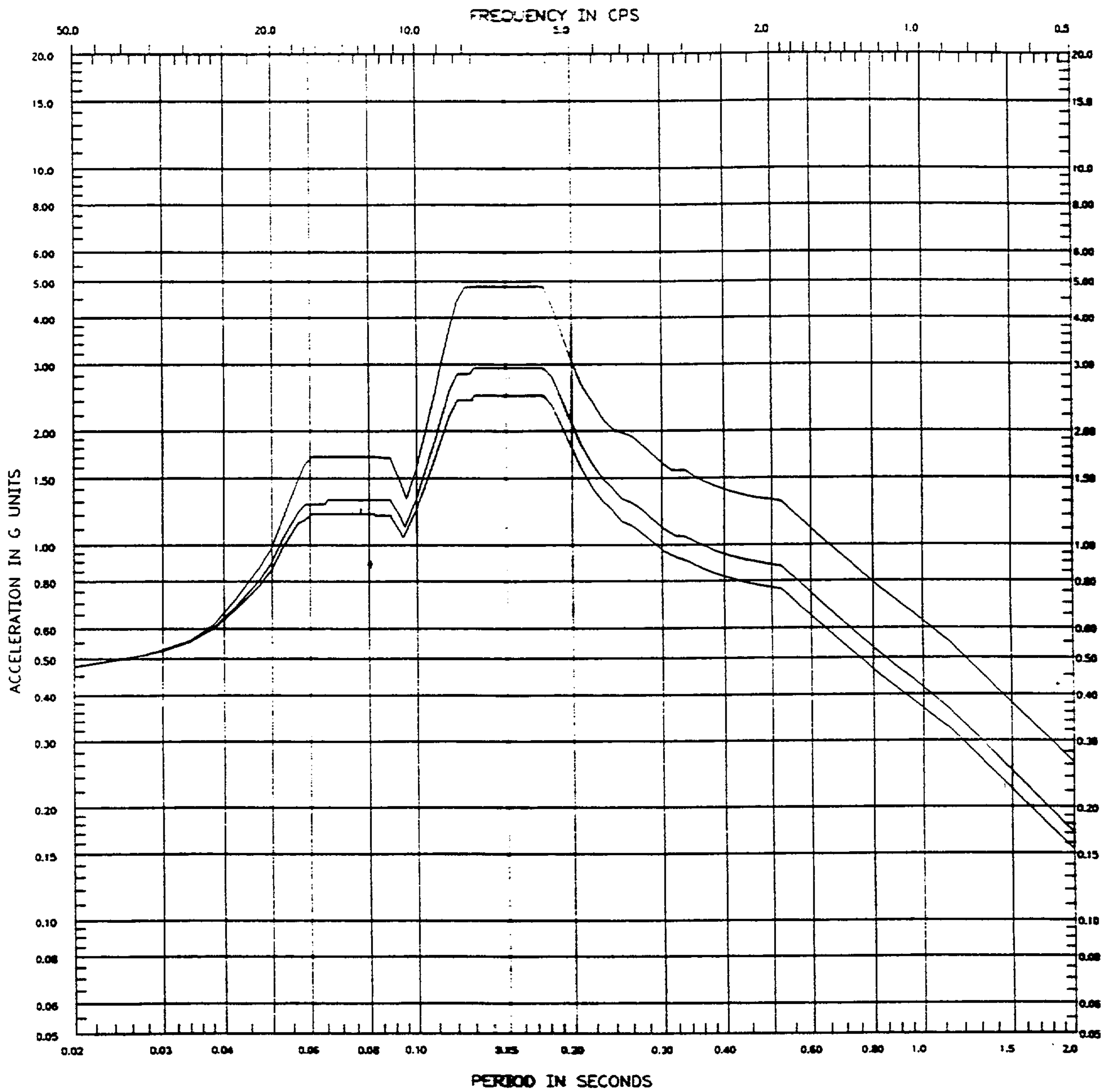
SARGENT & LUNDY
ENGINEERS

CALC NO. 8.11.6-2
SAFETY RELATED
PROJECT 9630-16 REV 0

05/04/95
SME SPECTRA - Widened
NODE 65

PEAKS WIDENED ON EACH SIDE BY 15 %
NS DAMPING 0.005 0.03 0.05

Dresden Units 2&3



		SPECTRA NO.	SME
NODE	65	ELEVATION	561
DIRECTION	NS Envelope For Entire Floor	LOCATION	CONTROL BUILDING

7

CALC NO. 8.11.6-2

SAFETY RELATED

SARGENT & LUNDY ENGINEERS

PROJECT 9630-16 REV 0

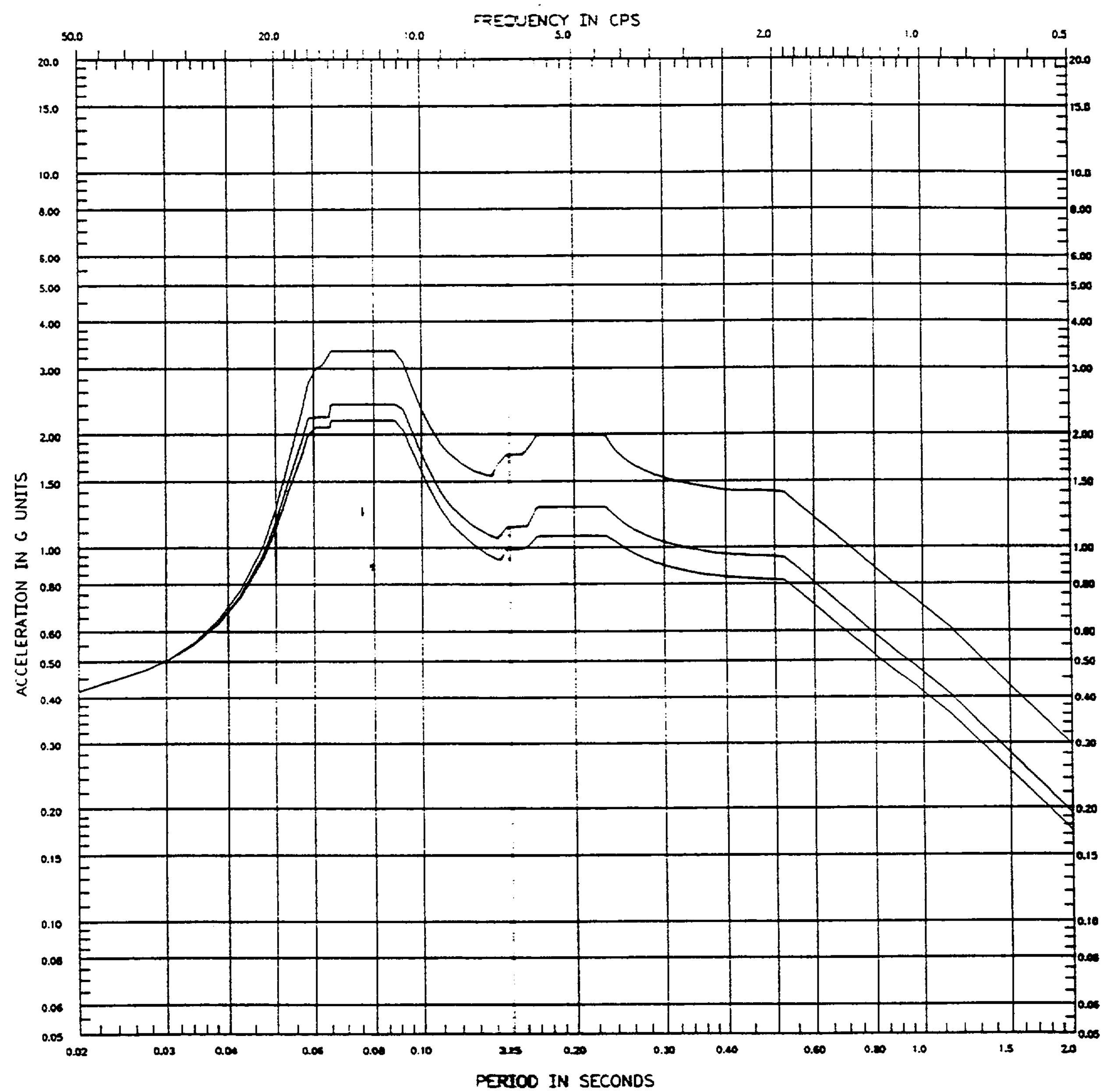
05/04/95

PEAKS WIDENED ON EACH SIDE BY 15 % -

SME SPECTRA - **Widened** EW DI DAMPING 0.005 0.03 0.05

NODE 65

Dresden Units 2&3



SPECTRA NO. SME

NODE 65 ELEVATION 561

7 DIRECTION EW **Envelope For Entire Floor** LOCATION CONTROL BUILDING

CALC NO. 8.11.6-2

SAFETY RELATED

PROJECT 9630-16 REV 0

SARGENT & LUNDY
ENGINEERS

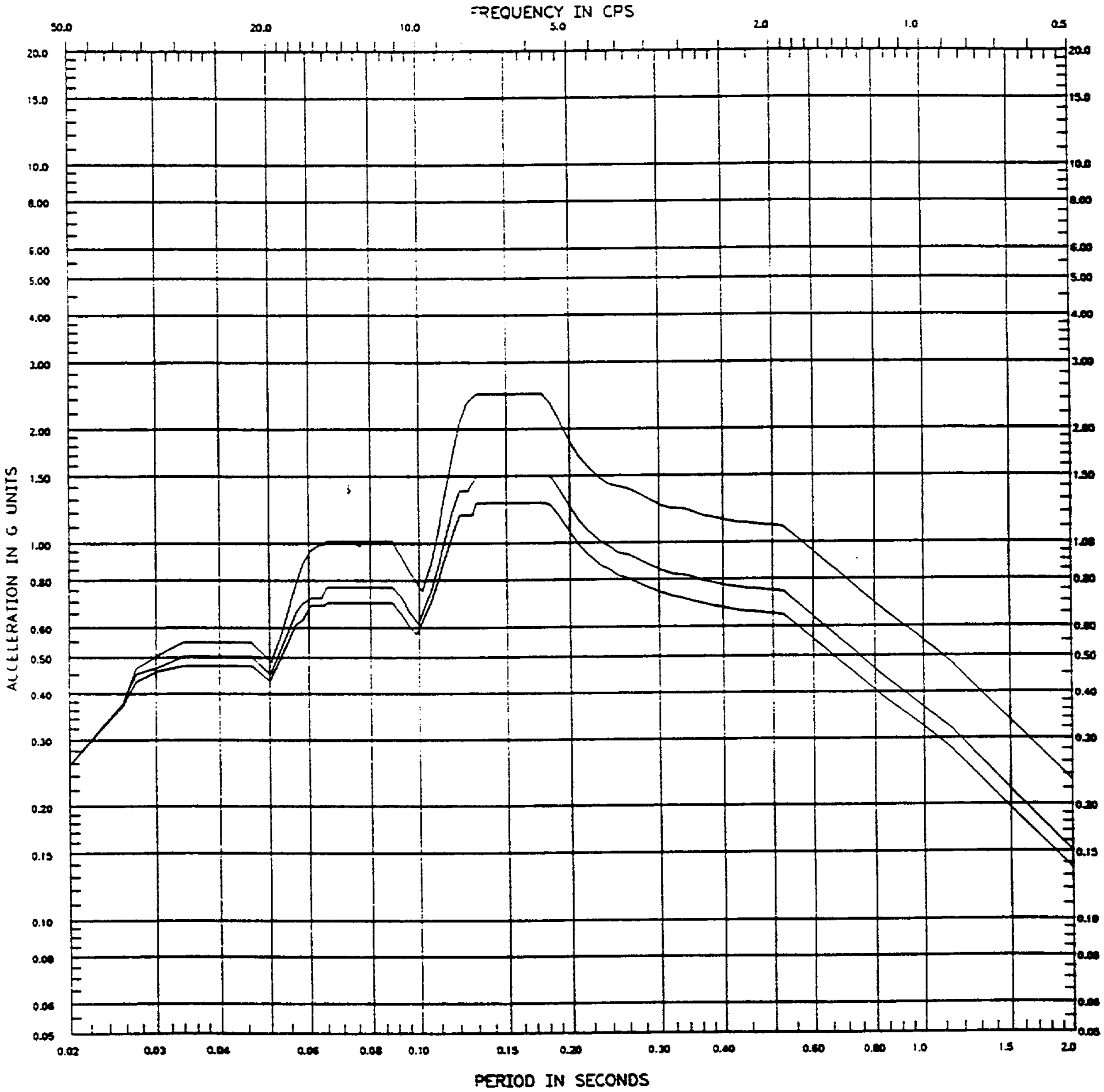
05/04/95

PEAKS WIDENED ON EACH SIDE BY 15 %

SME SPECTRA - **Widened** NS DI DAMPING 0.005 0.03 0.05

NODE 61

Dresden Units 2&3



SPECTRA NO. SME

NODE 61

ELEVATION 534

8 DIRECTION NS **Envelope For Entire Floor**

LOCATION TURBINE BUILDING

A2-47

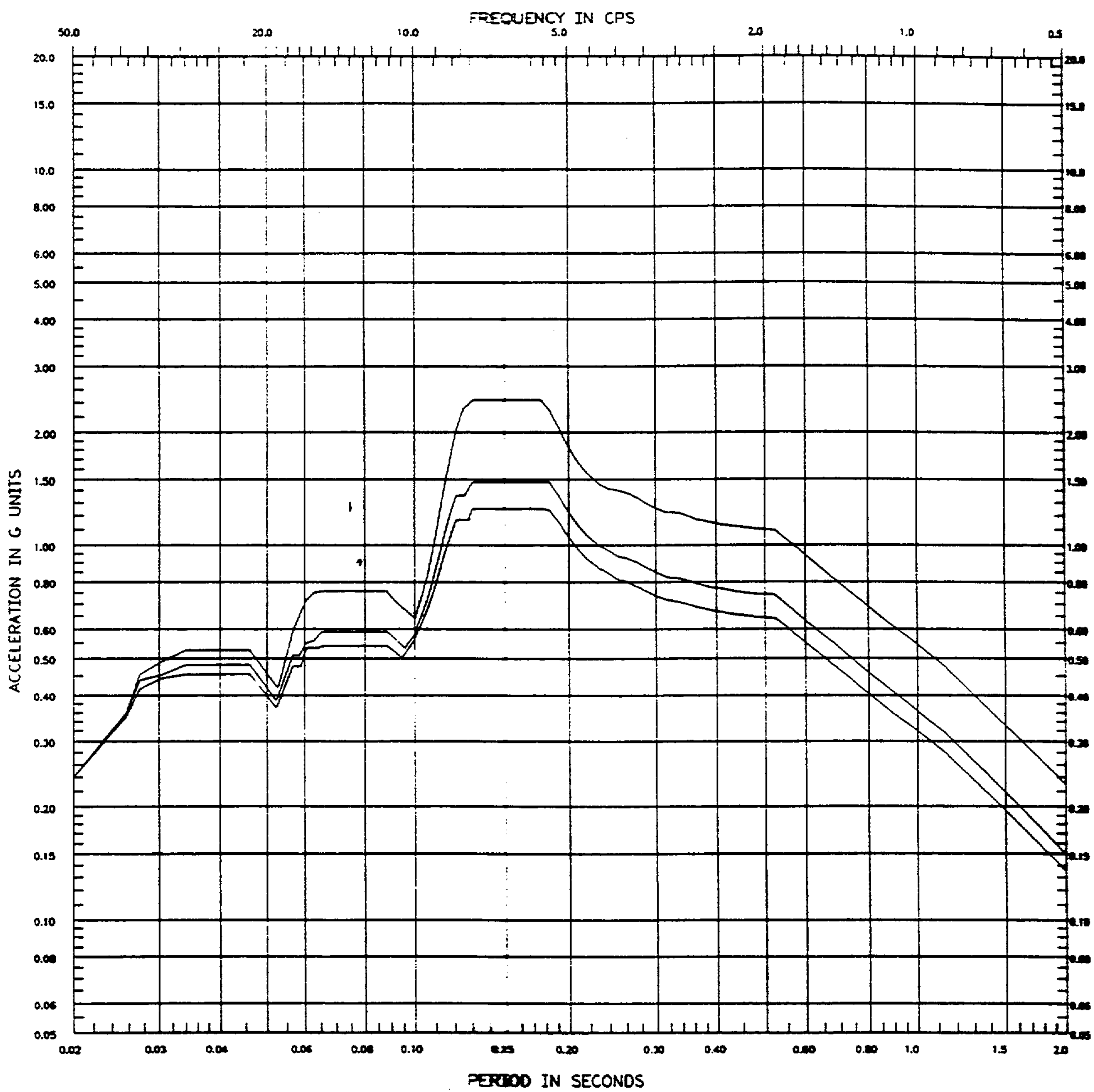
SARGENT & LUNDY
ENGINEERS

CALC NO. 8.11.6-2
SAFETY RELATED
PROJECT 9630-16 REV 0

05/04/95
SME SPECTRA **Widened** NS DIRECTION
NODE 16

PEAKS WIDENED ON EACH SIDE BY 15 %
DAMPING 0.005 0.03 0.05

Dresden Units 2&3



		SPECTRA NO.	SME
NODE	16	ELEVATION	534
DIRECTION	NS **Mass Center of Floor**	LOCATION	TURBINE BUILDING

9

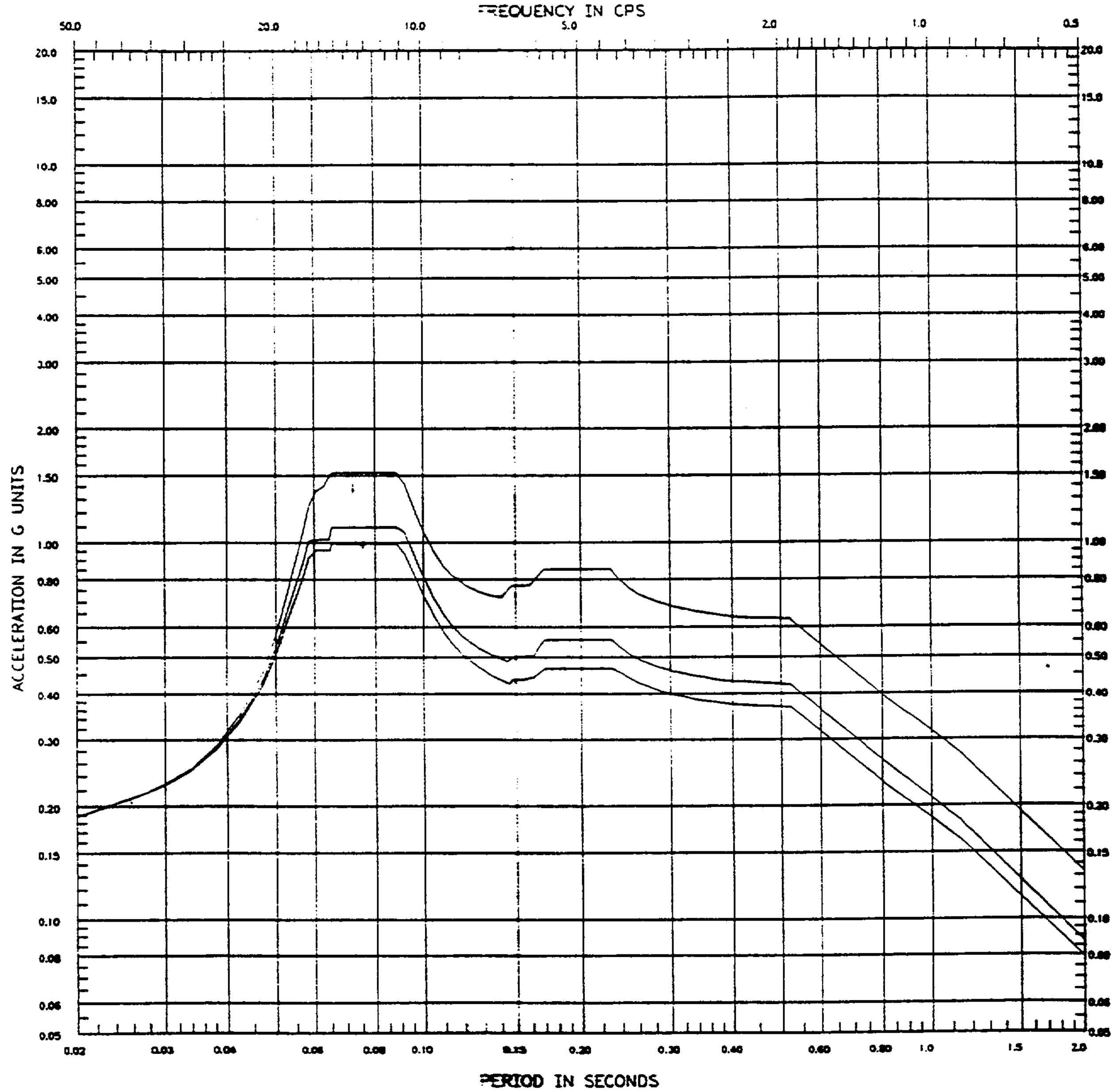
CALC NO. 8.11.6-2
SAFETY RELATED
PROJECT 9630-16 REV 0

SARGENT & LUNDY ENGINEERS

05/04/95
SME SPECTRA - Widened
NODE 61

PEAKS WIDENED ON EACH SIDE BY 15 %
EW DI DAMPING 0.005 0.03 0.05

Dresden Units 2&3



		SPECTRA NO.	SME
NODE	61	ELEVATION	534
DIRECTION	EW Envelope For Entire Floor	LOCATION	TURBINE BUILDING

CALC NO. 8.11.6-2 A2-49

SAFETY RELATED

SARGENT & LUNDY ENGINEERS

PROJECT 9630-16 REV 0

05/04/95

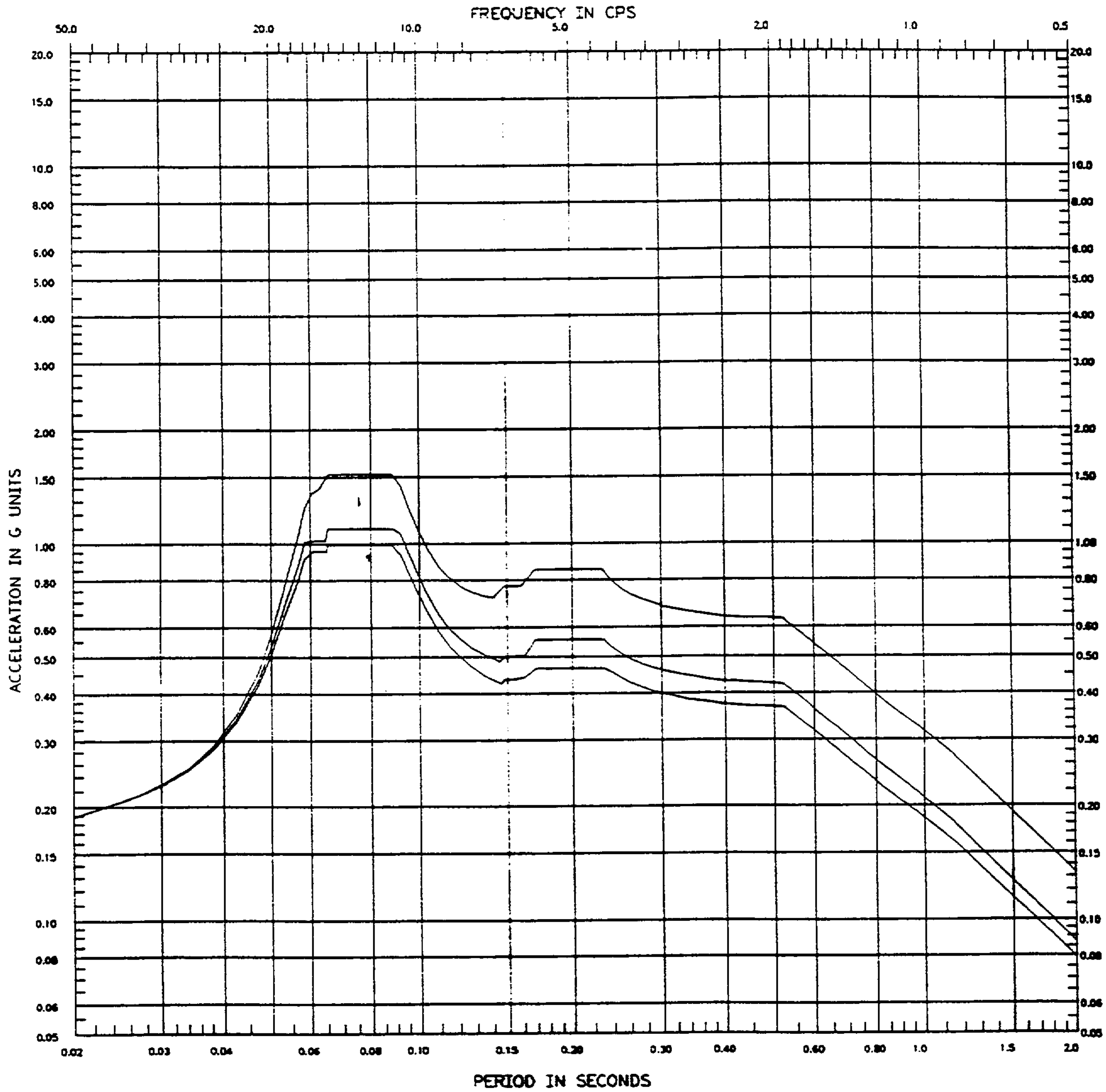
PEAKS WIDENED ON EACH SIDE BY 15 %

SME SPECTRA - **Widened** EW DIRECTION

DAMPING 0.005 0.03 0.05

NODE 16

Dresden Units 2&3



			SPECTRA NO.	SME
NODE	16		ELEVATION	534
DIRECTION	EW	**Mass Center of Floor**	LOCATION	TURBINE BUILDING

9

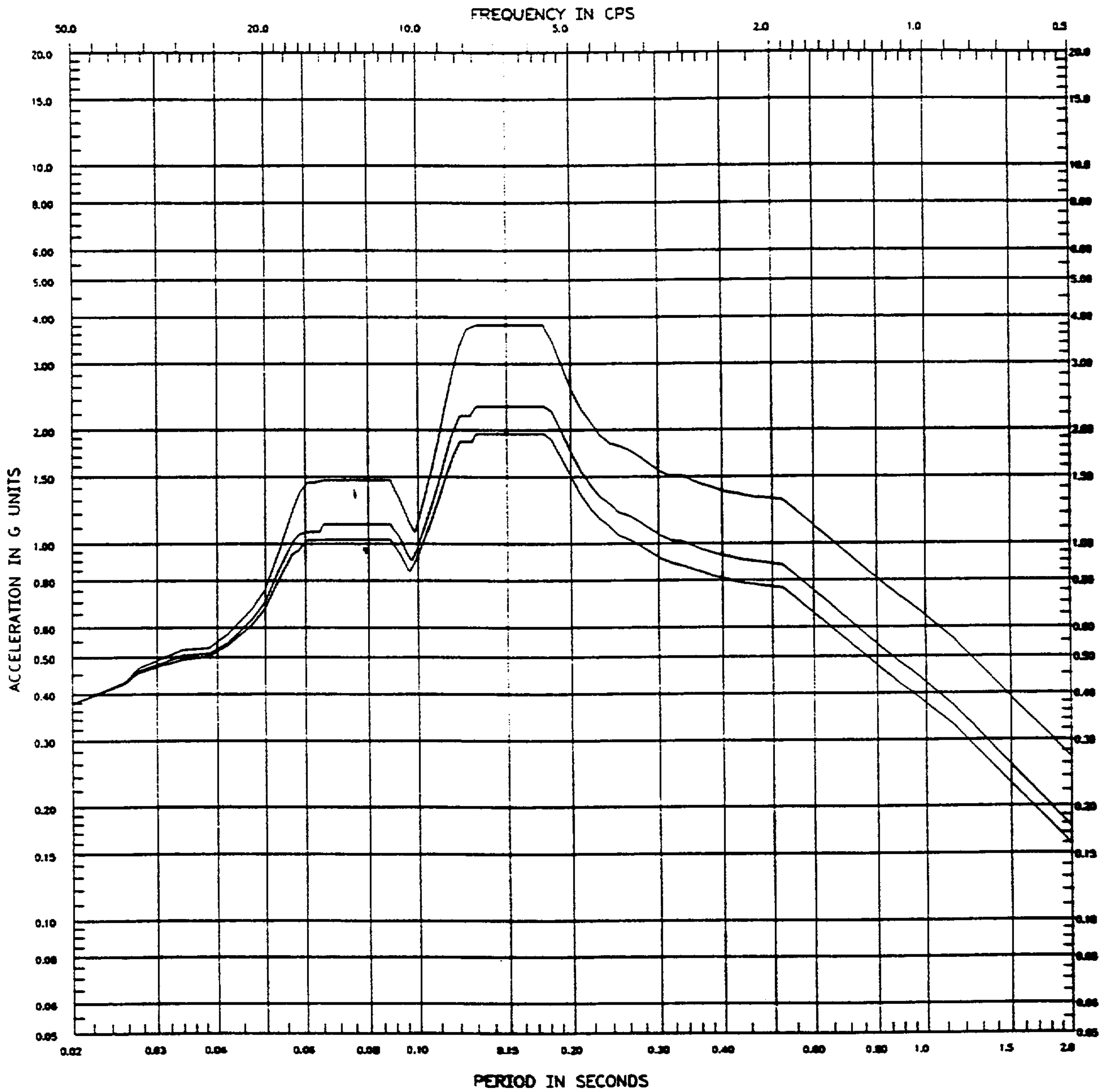
CALC NO. 8.11.6-2
SAFETY RELATED
PROJECT 9630-16 REV 0

SARGENT & LUNDY
ENGINEERS

05/04/95
SME SPECTRA - Widened
NODE 63

PEAKS WIDENED ON EACH SIDE BY 15 %
NS DI DAMPING 0.005 0.03 0.05

Dresden Units 2&3



		SPECTRA NO.	SME
NODE	63	ELEVATION	549
DIRECTION	NS Envelope For Entire Floor	LOCATION	TURBINE BUILDING

CALC NO. 8.11.6-2

SAFETY RELATED

PROJECT 9630-16 REV 0

SARGENT & LUNDY
ENGINEERS

05/04/95

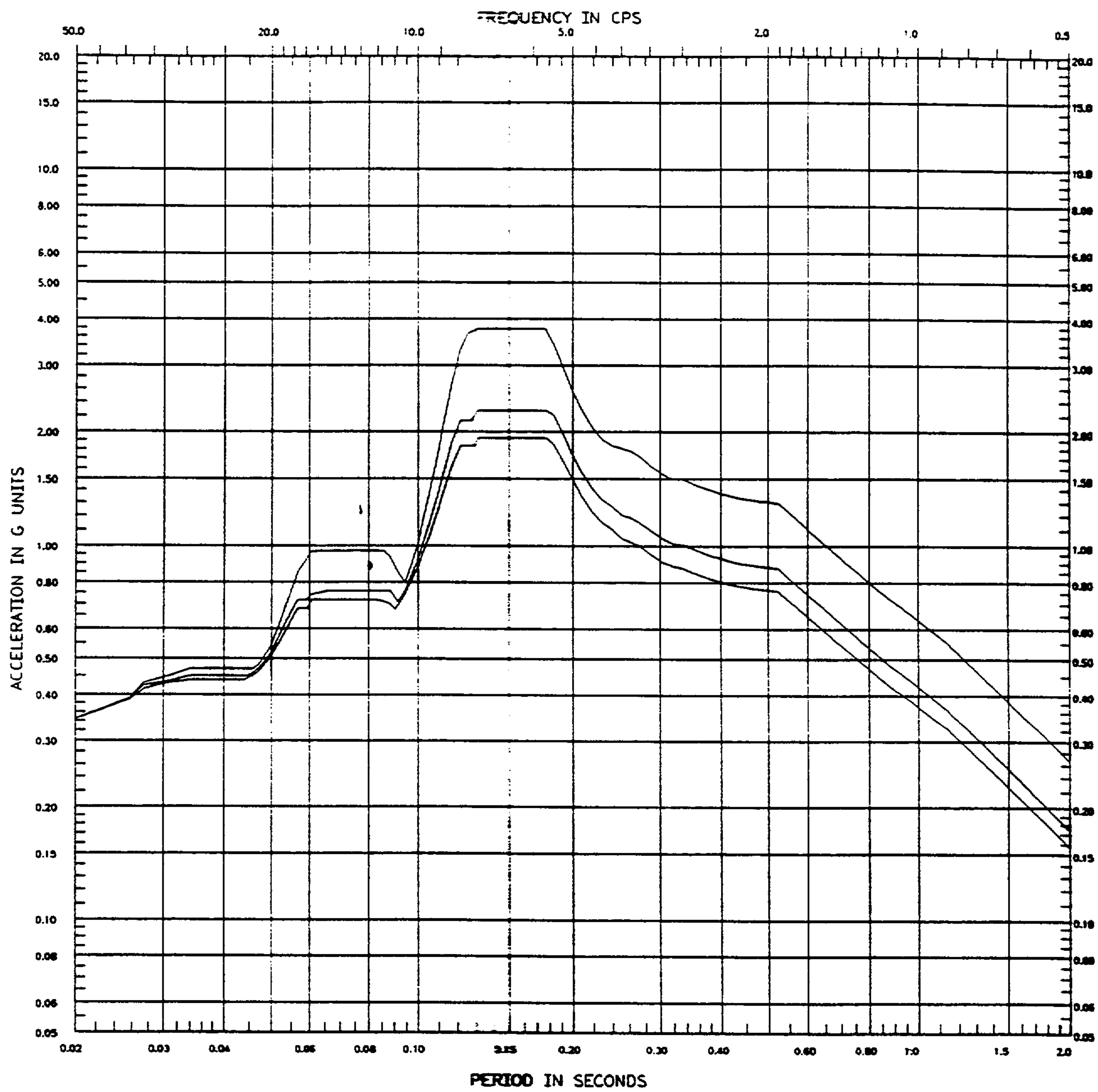
PEAKS WIDENED ON EACH SIDE BY 15 %

SME SPECTRA **Widened** NS DIRECTION

DAMPING 0.005 0.03 0.05

NODE 17

Dresden Units 2&3



SPECTRA NO. SME

NODE 17

ELEVATION 549

10 DIRECTION NS **Mass Center of Floor**

LOCATION TURBINE BUILDING

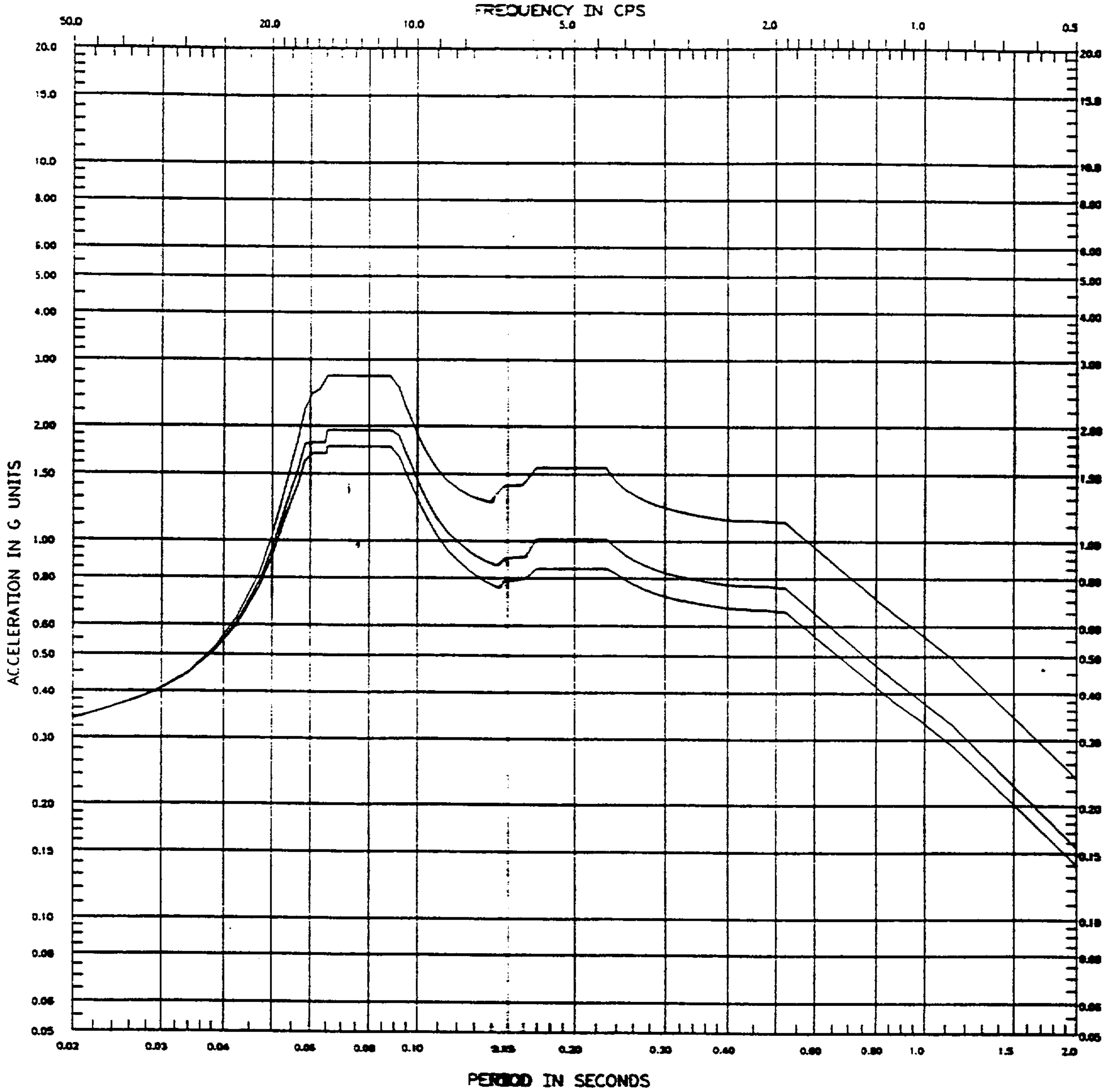
CALC NO. 8.11.6-2
SAFETY RELATED
PROJECT 9630-16 REV 0

SARGENT & LUNDY ENGINEERS

05/04/95
SME SPECTRA - **Widened**
NODE 63

PEAKS WIDENED ON EACH SIDE BY 15 %
EW DI DAMPING 0.005 0.03 0.05

Dresden Units 2&3



SPECTRA NO. SME
NODE 63
ELEVATION 549
9 DIRECTION EW **Envelope For Entire Floor**
LOCATION TURBINE BUILDING

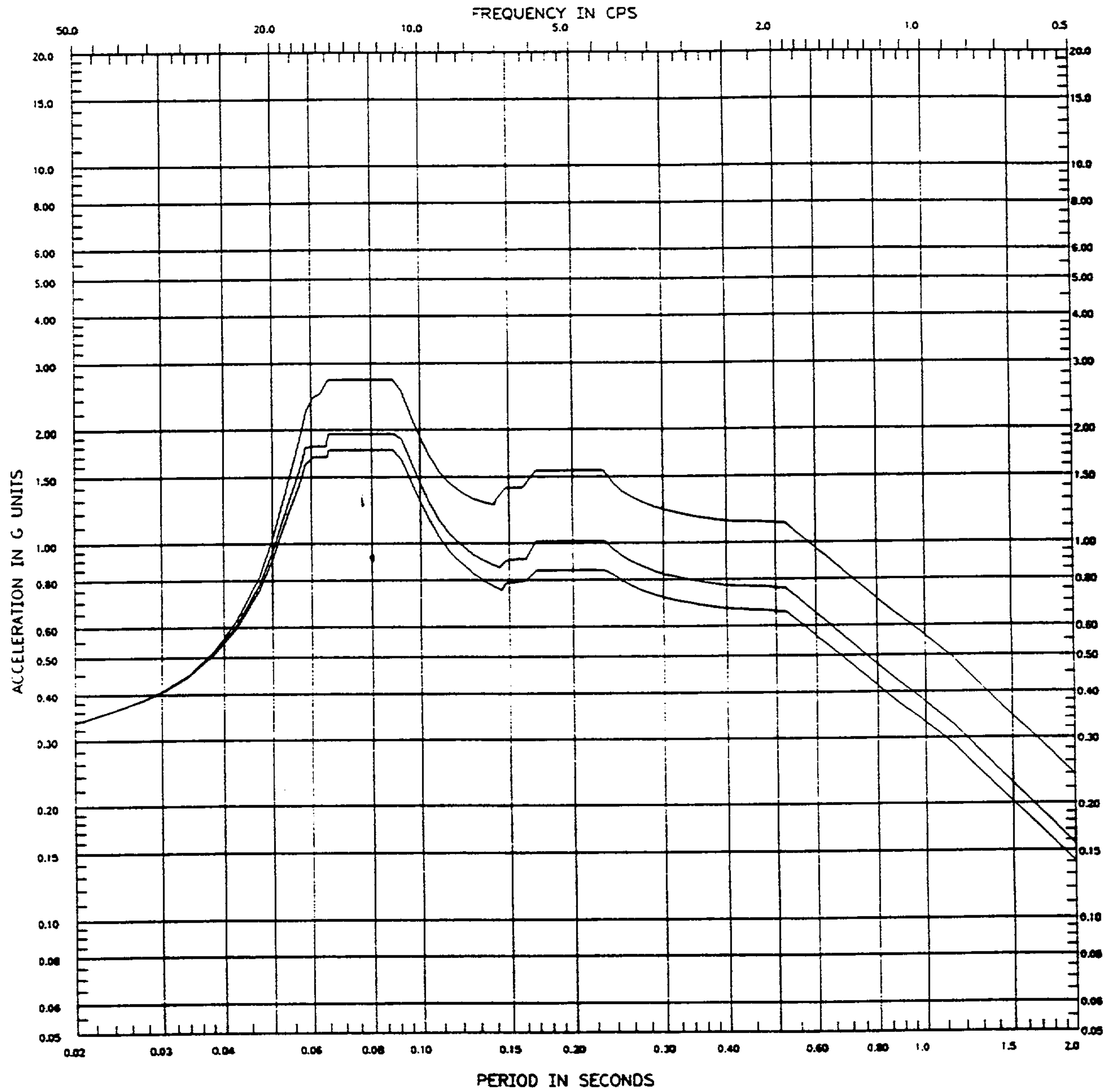
CALC NO. 8.11 6-2
SAFETY RELATED
PROJECT 9630-16 REV 0

SARGENT & LUNDY
ENGINEERS

05/04/95
SME SPECTRA - **Widened** EW DIRECTION
NODE 17

PEAKS WIDENED ON EACH SIDE BY 15 %
DAMPING 0.005 0.03 0.05

Dresden Units 2&3



			SPECTRA NO.	SME
NODE	17		ELEVATION	549
DIRECTION	EW	**Mass Center of Floor**	LOCATION	TURBINE BUILDING

10

CALC NO. 8.11.6-2

SAFETY RELATED

SARGENT & LUNDY ENGINEERS

PROJECT 9630-16 REV 0

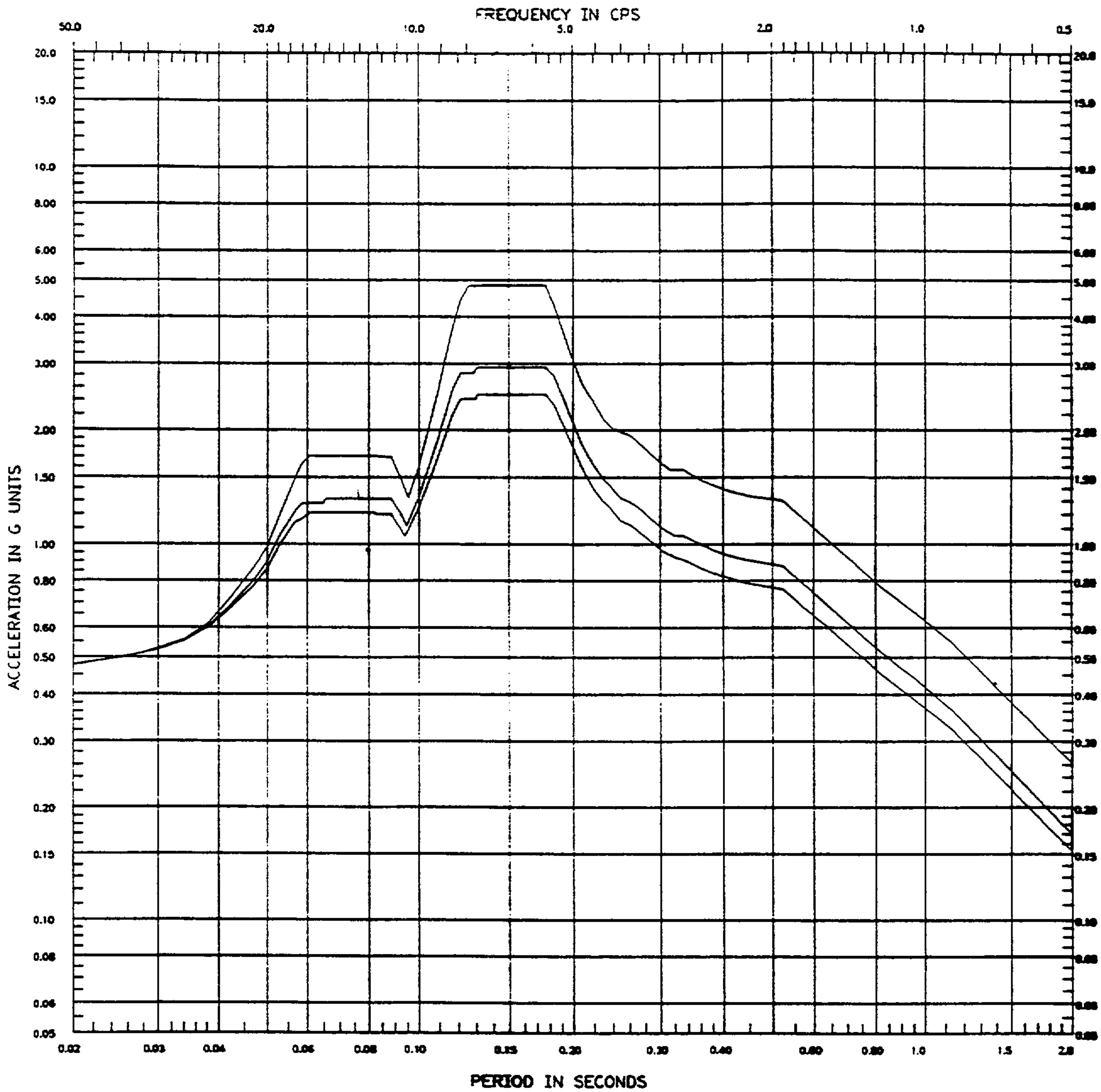
05/04/95

PEAKS WIDENED ON EACH SIDE BY 15 %

SME SPECTRA - **Widened** NS DI DAMPING 0.005 0.03 0.05

NODE 65

Dresden Units 2&3



		SPECTRA NO.	SME
NODE	65	ELEVATION	561
DIRECTION	NS **Envelope For Entire Floor**	LOCATION	TURBINE BUILDING

7

CALC NO. 8.11.6-2 **A2-55**

SAFETY RELATED

SARGENT & LUNDY ENGINEERS

PROJECT 9630-16 REV 0

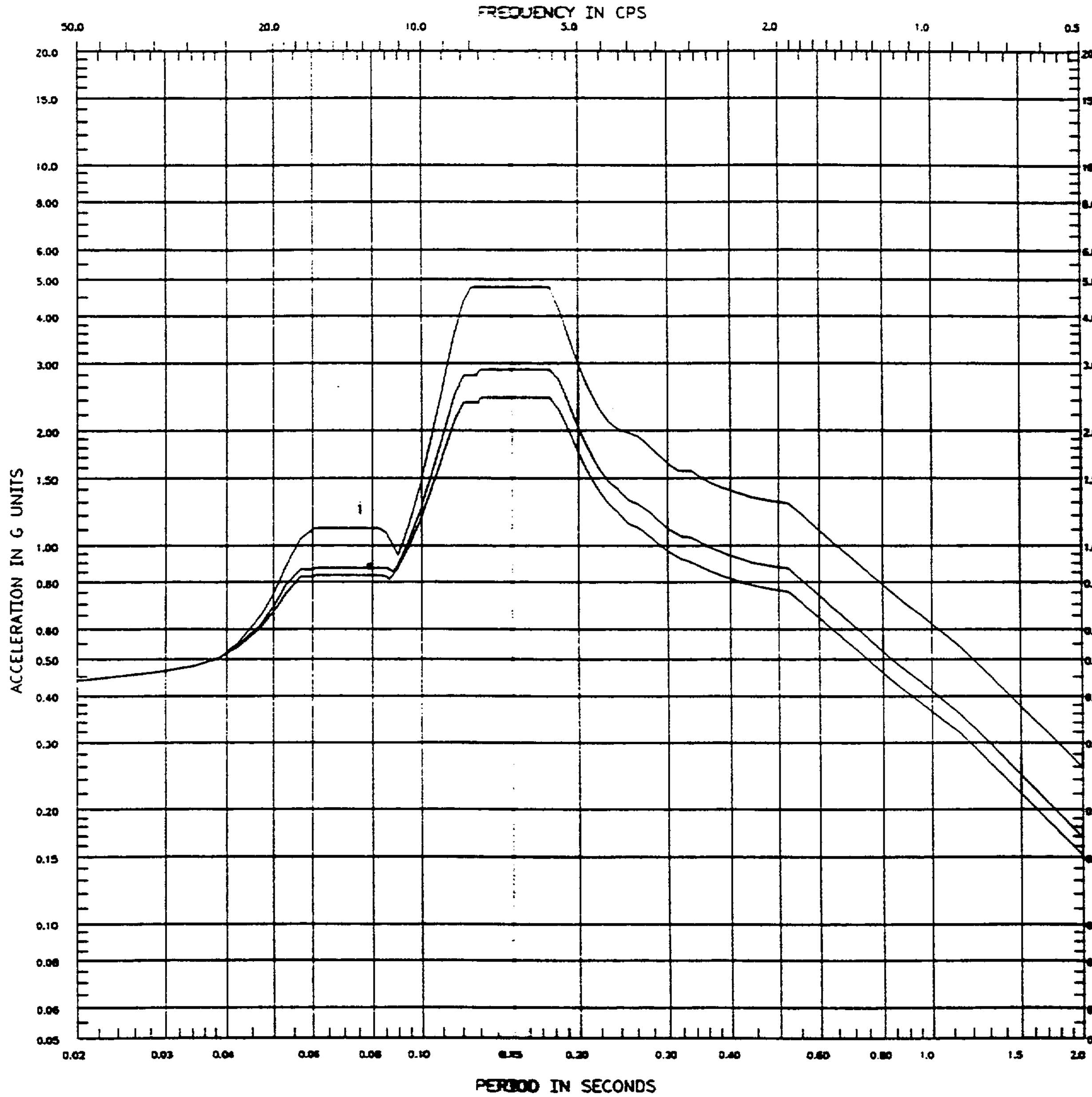
05/04/95

PEAKS WIDENED ON EACH SIDE BY 15 %

SME SPECTRA **Widened** NS DIRECTION DAMPING 0.005 0.03 0.05

NODE 13

Dresden Units 2&3



			SPECTRA NO.	SME
NODE	13		ELEVATION	561
DIRECTION	NS	**Mass Center of Floor**	LOCATION	TURBINE BUILDING

8

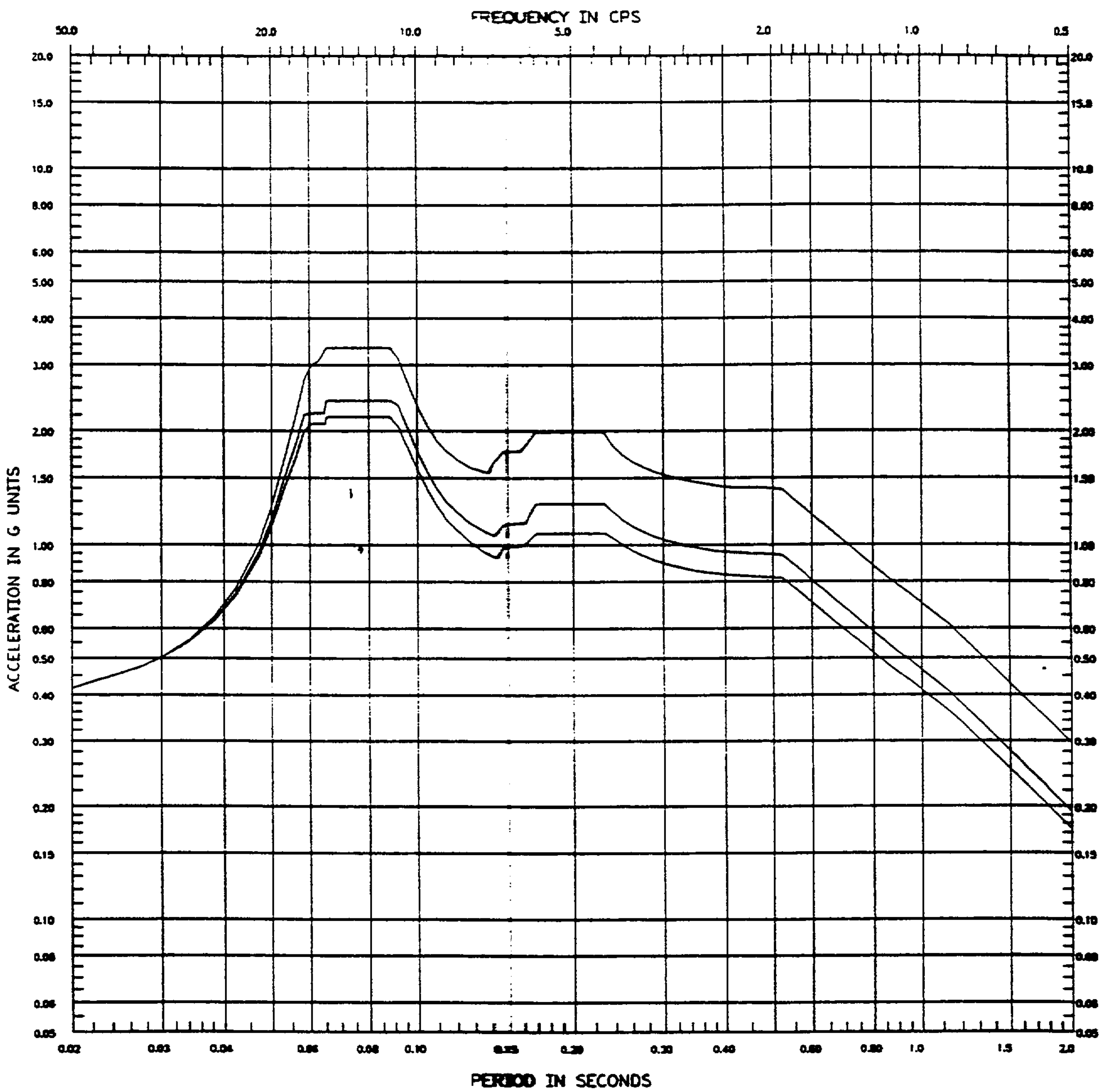
CALC NO. 8.11.6-2
SAFETY RELATED
PROJECT 9630-16 REV 0

SARGENT & LUNDY
ENGINEERS

05/04/95
SME SPECTRA - **Widened** EW DI
NODE 65

PEAKS WIDENED ON EACH SIDE BY 15 %
DAMPING 0.005 0.03 0.05

Dresden Units 2&3



		SPECTRA NO.	SME
NODE	65	ELEVATION	561
DIRECTION	EW **Envelope For Entire Floor**	LOCATION	TURBINE BUILDING

7

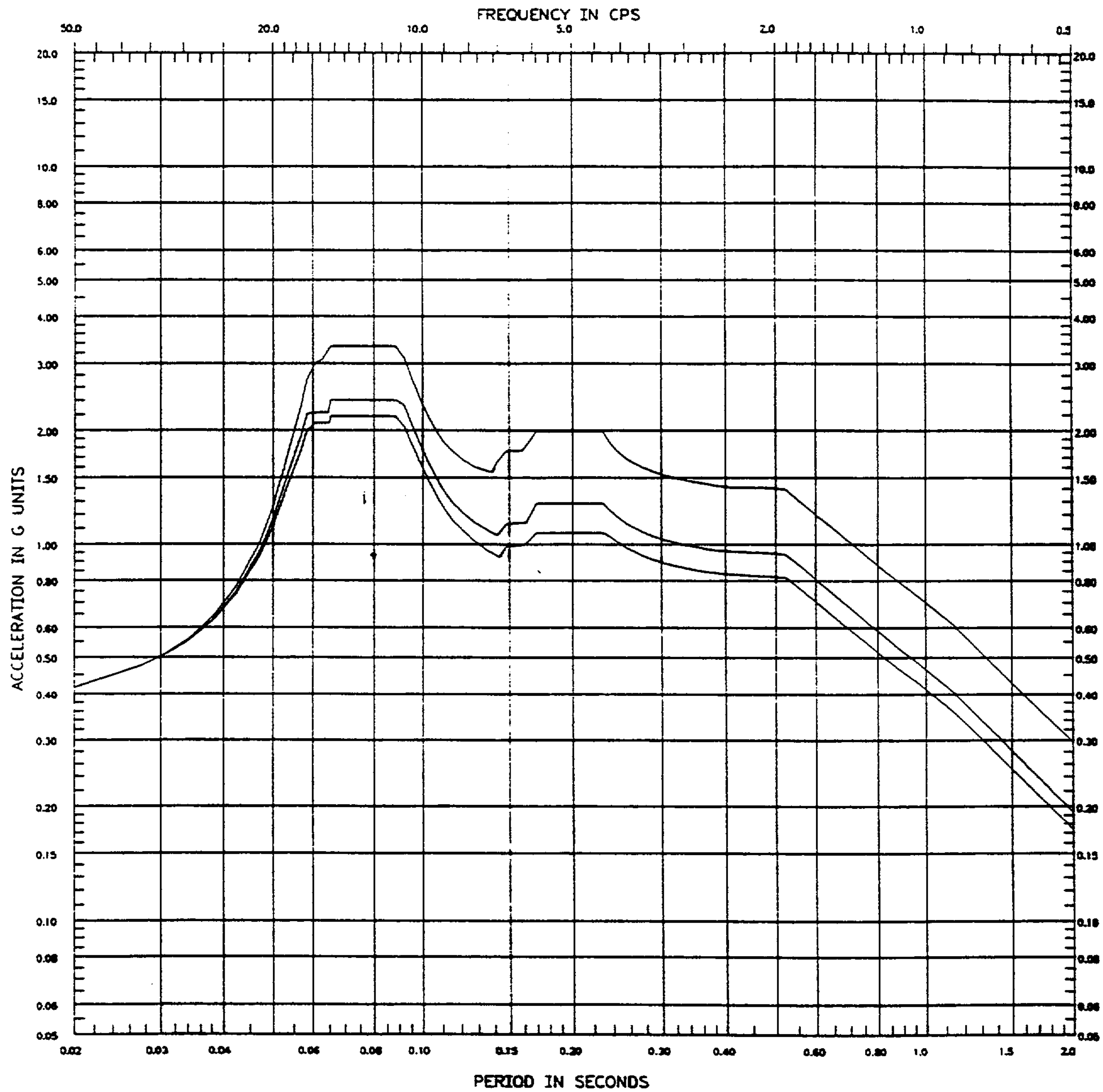
CALC NO. 8.11.6-2
SAFETY RELATED
PROJECT 9630-16 REV 0

SARGENT & LUNDY ENGINEERS

05/04/95
SME SPECTRA · **Widened** EW DIRECTION
NODE 13

PEAKS WIDENED ON EACH SIDE BY 15 %
DAMPING 0.005 0.03 0.05

Dresden Units 2&3



			SPECTRA NO.	SME
NODE	13		ELEVATION	561
DIRECTION	EW	**Mass Center of Floor**	LOCATION	TURBINE BUILDING

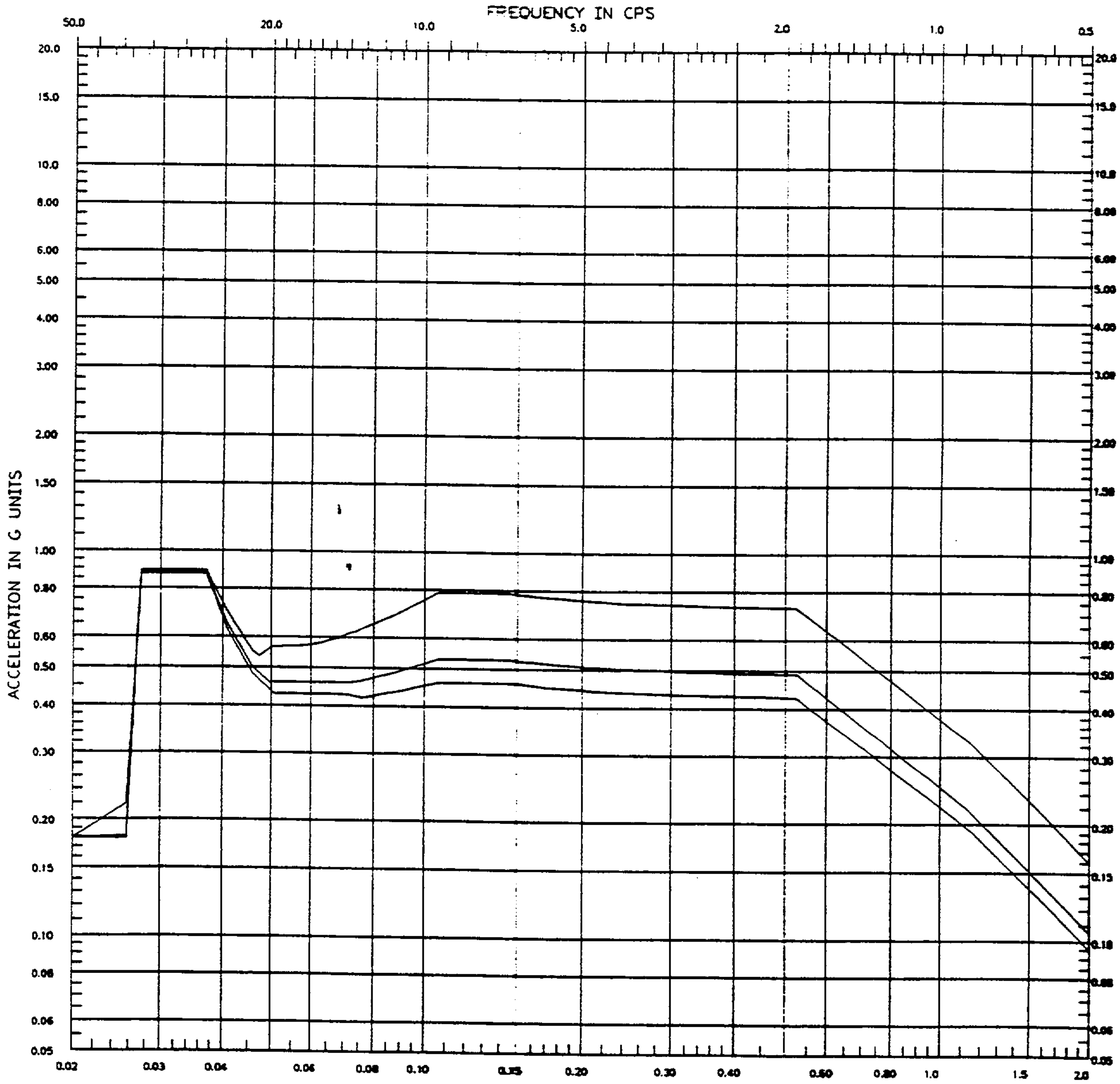
SARGENT & LUNDY
ENGINEERS

05/12/95
SME SPECTRA - **Widened**
NODE 13

CALC NO. 8.11.6-2
SAFETY RELATED
PROJECT 9630-16 REV 0

PEAKS WIDENED ON EACH SIDE BY 15 %
DAMPING 0.005 0.03 0.05

Dresden Units 2&3



NODE	13		SPECTRA NO.	SME
			ELEVATION	561 **Wall and Slabs >12" Thick**
DIRECTION	VERT	ANGLE	LOCATION	TURBINE BUILDING

SARGENT & LUNDY
ENGINEERS

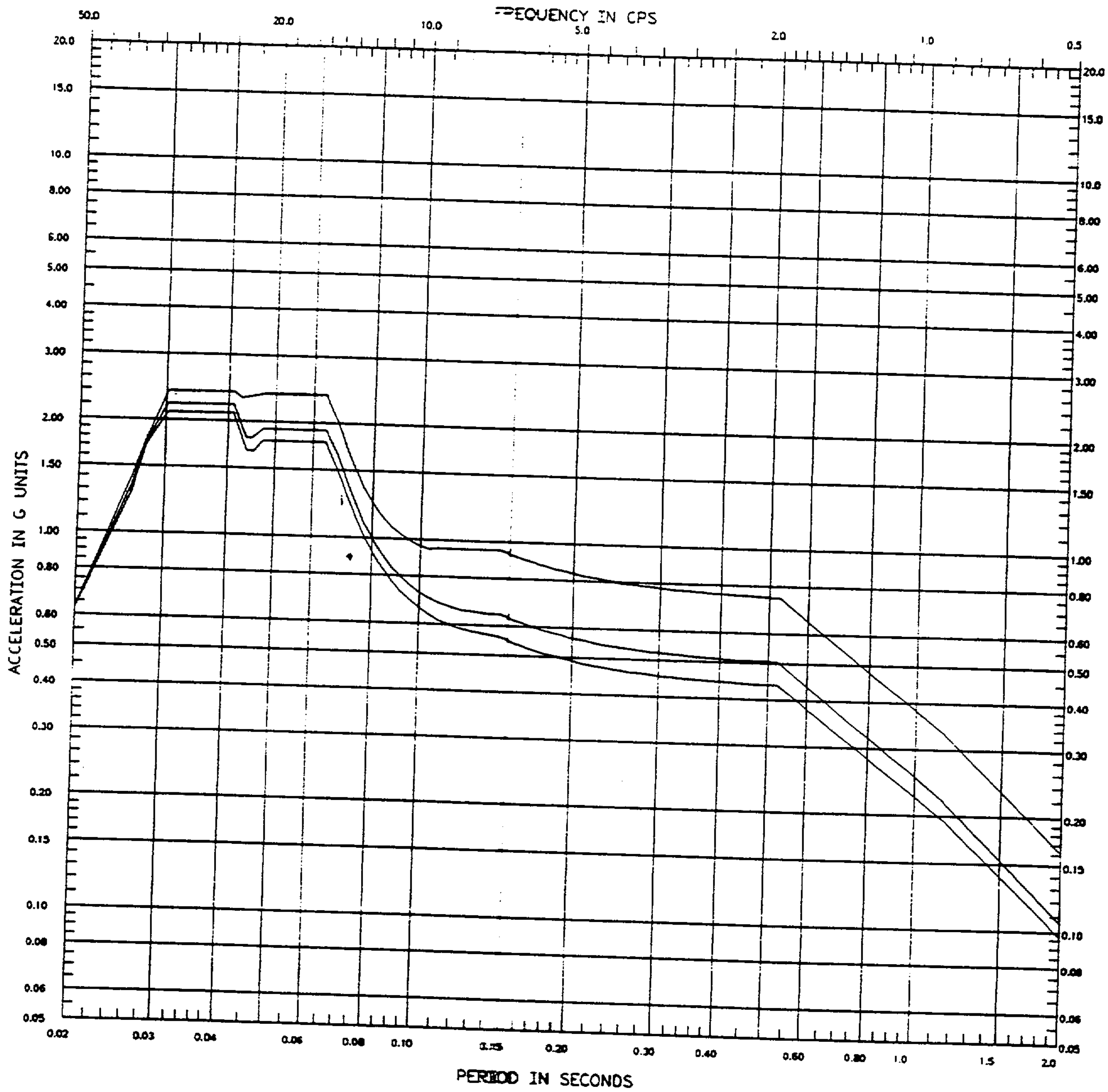
CALC NO. 8.11.6-2
SAFETY RELATED
PROJECT 9630-16 REV 0

A2-59
(Final)

05/11/95
SME SPECTRA - Widened LAB
NODE 49

PEAKS WIDENED ON EACH SIDE BY 15 %
DAMPING 0.005 0.03 0.05

Dresden Units 2&3



NODE	49	SPECTRA NO.	SME
DIRECTION	VERT	ELEVATION	561 Slab ≤ 12" Thick
ANGLE		LOCATION	TURBINE BUILDING

Attachment 6

Safe Shutdown Earthquake (SSE) Design Basis Ground and In-structure Response Spectra for Dresden Nuclear Power Station Units 2 and 3

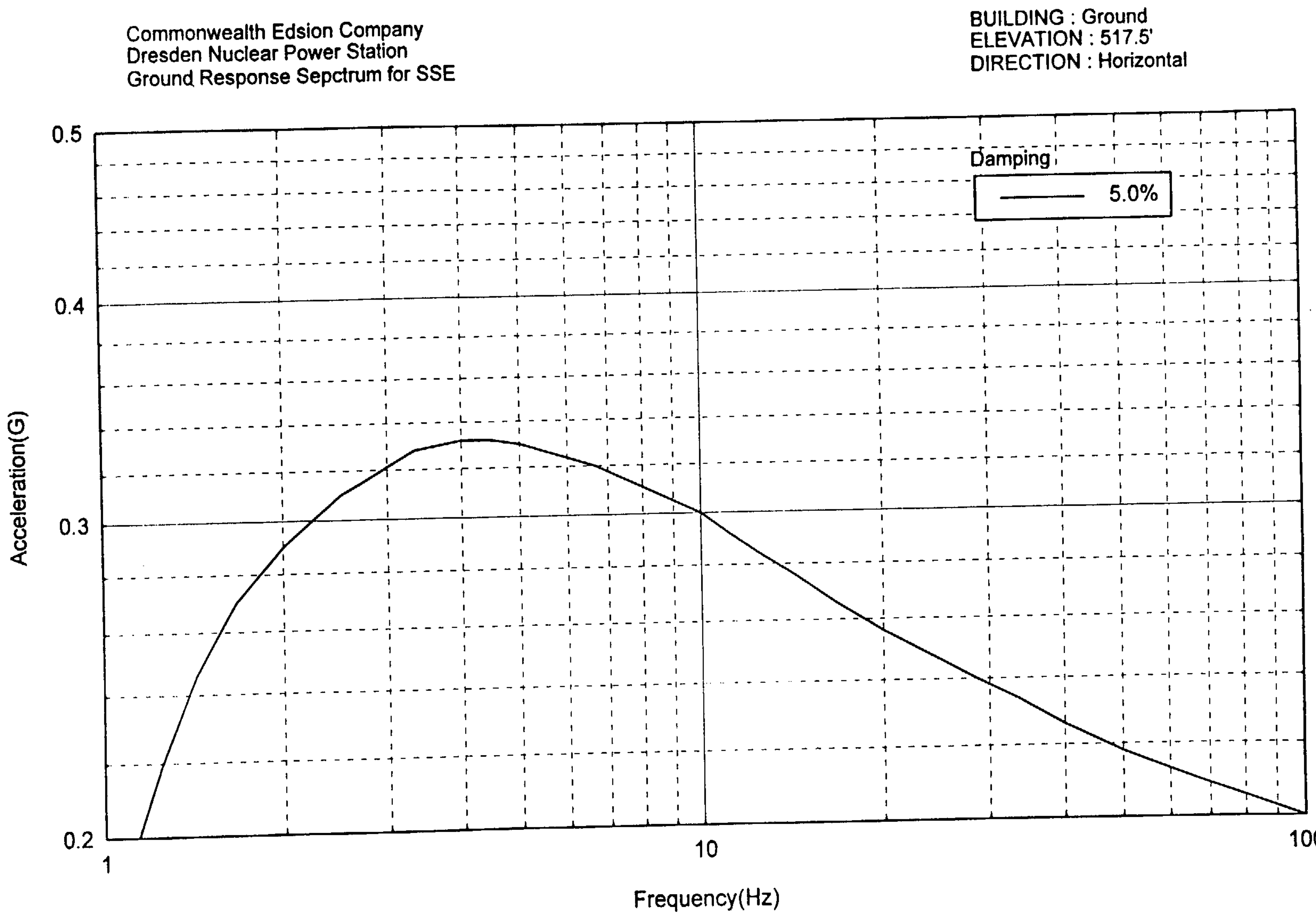

Commonwealth Edsion Company
Dresden Nuclear Power Station
Ground Response Sepctrum for SSE
BUILDING : Ground
ELEVATION : 517.5'
DIRECTION : Horizontal
Damping
5.0%
0.5
0.4
0.3
0.2
Acceleration(G)
1
10
100
Frequency(Hz)

Figure B-1

Commonwealth Edsion Company
Dresden Nuclear Power Station
In-Structure Response Sepctrum for SSE

BUILDING : Reactor
ELEVATION : 517.5'
DIRECTION : East-West

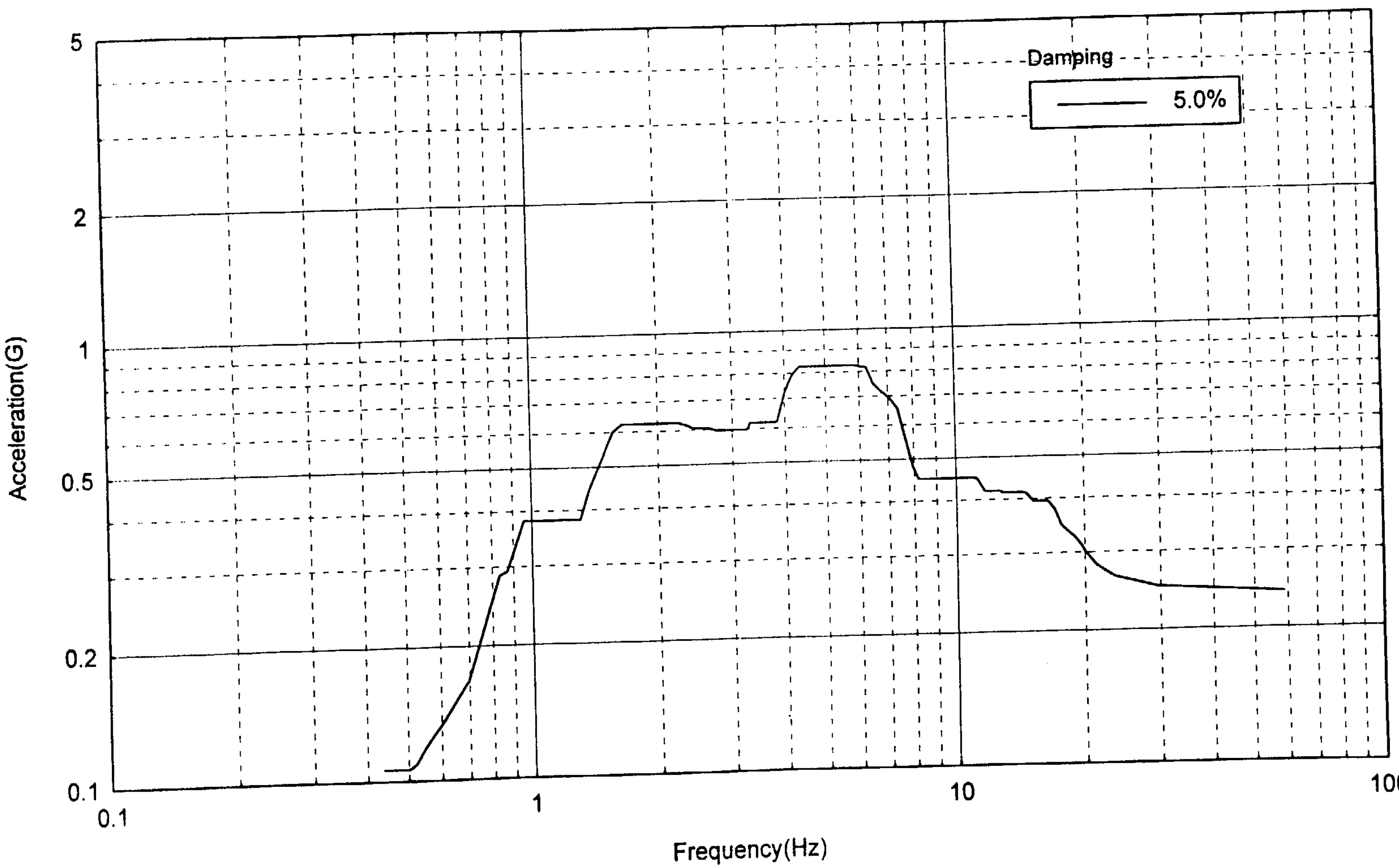
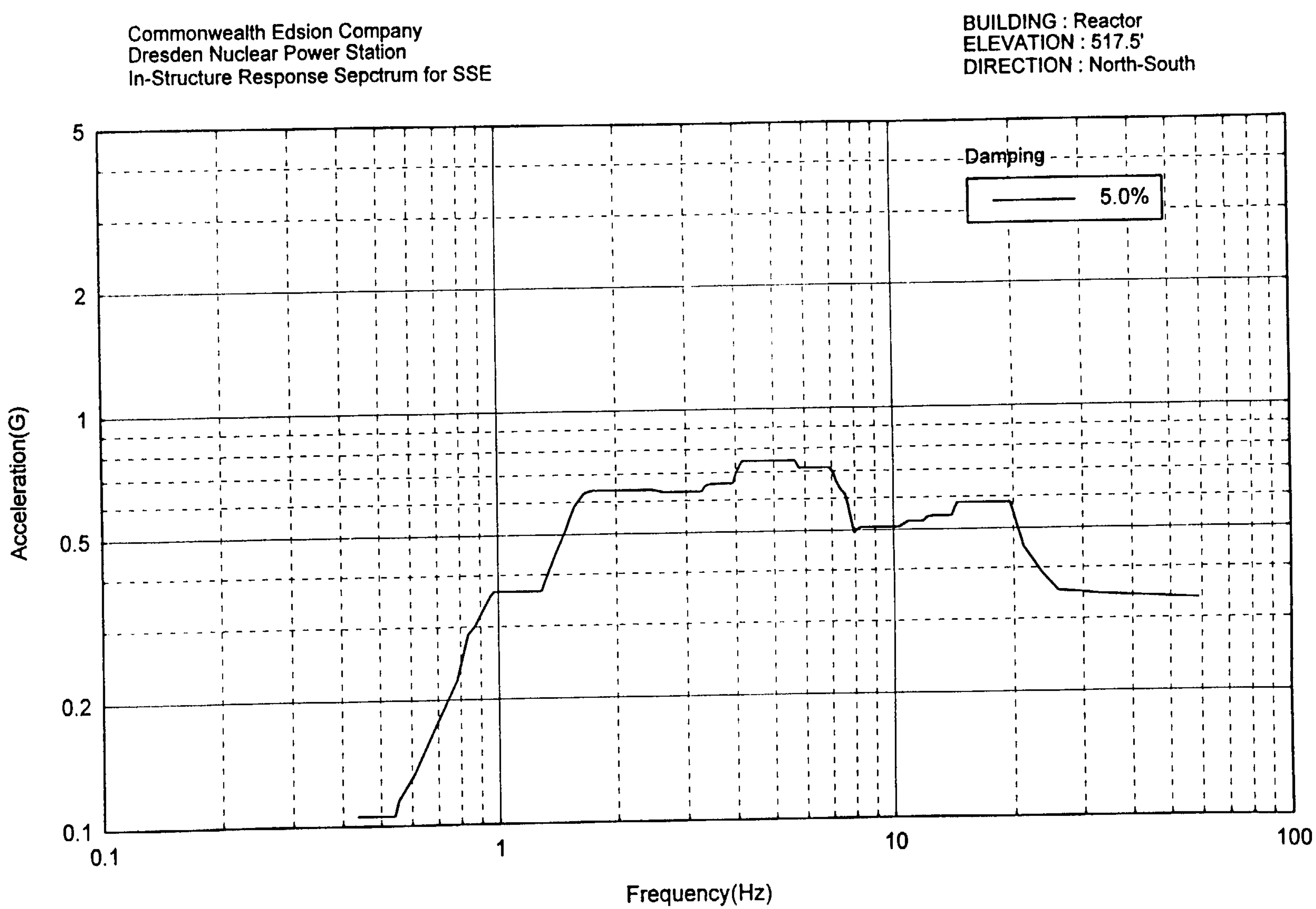


Figure B-2



Commonwealth Edsion Company
Dresden Nuclear Power Station
In-Structure Response Sepctrum for SSE
BUILDING : Reactor
ELEVATION : 517.5'
DIRECTION : North-South
Damping
5.0%
Acceleration(G)
5
2
1
0.5
0.2
0.1
0.1
1
10
100
Frequency(Hz)

Figure B-3

Commonwealth Edsion Company
Dresden Nuclear Power Station
In-Structure Response Sepctrum for SSE

BUILDING : Reactor
ELEVATION : 545.5'
DIRECTION : East-West

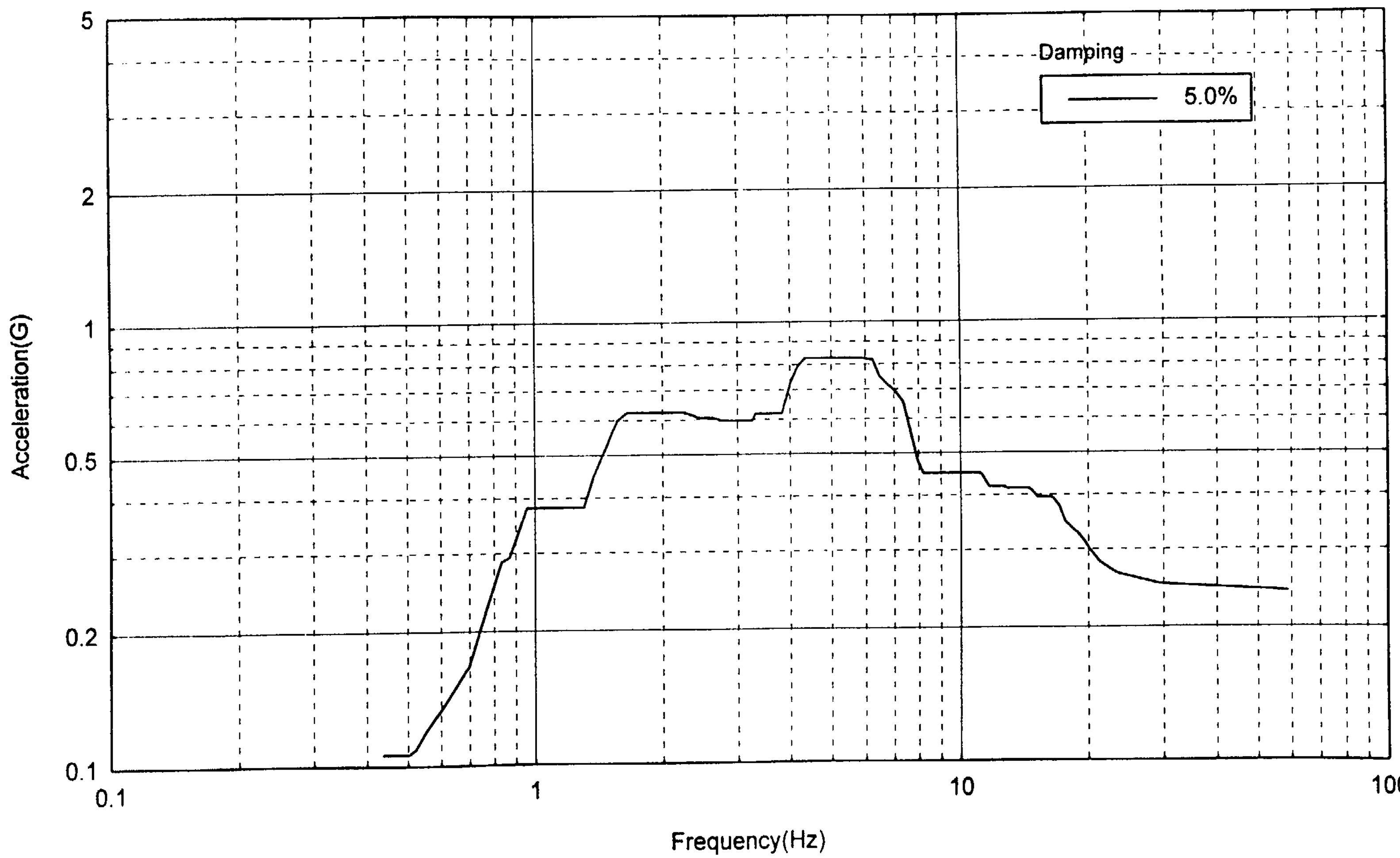


Figure B-4

Commonwealth Edsion Company
Dresden Nuclear Power Station
In-Structure Response Sepctrum for SSE

BUILDING : Reactor
ELEVATION : 545.5'
DIRECTION : North-South

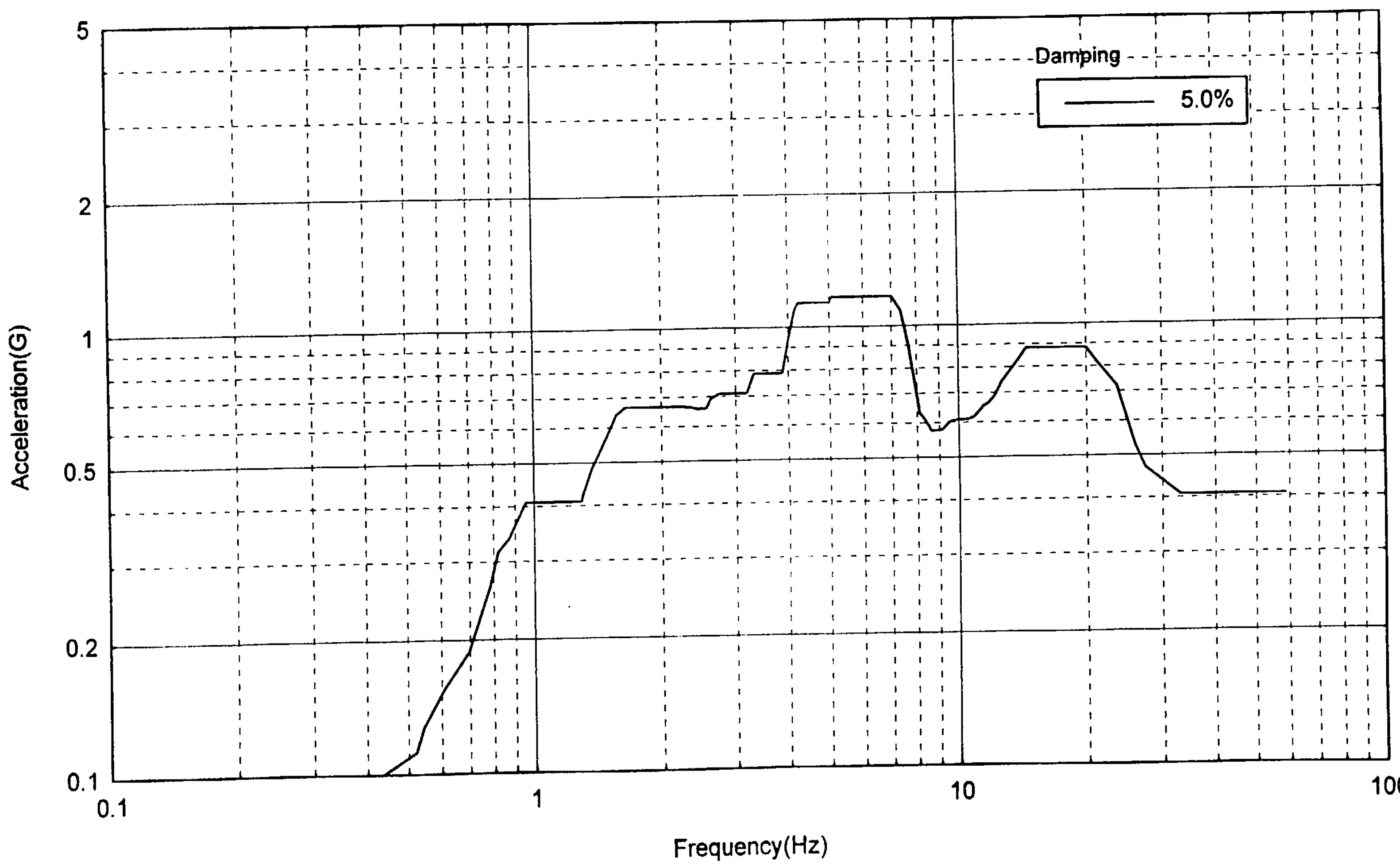


Figure B-5

Commonwealth Edsion Company
Dresden Nuclear Power Station
In-Structure Response Sepctrum for SSE

BUILDING : Reactor
ELEVATION : 570.0'
DIRECTION : East-West

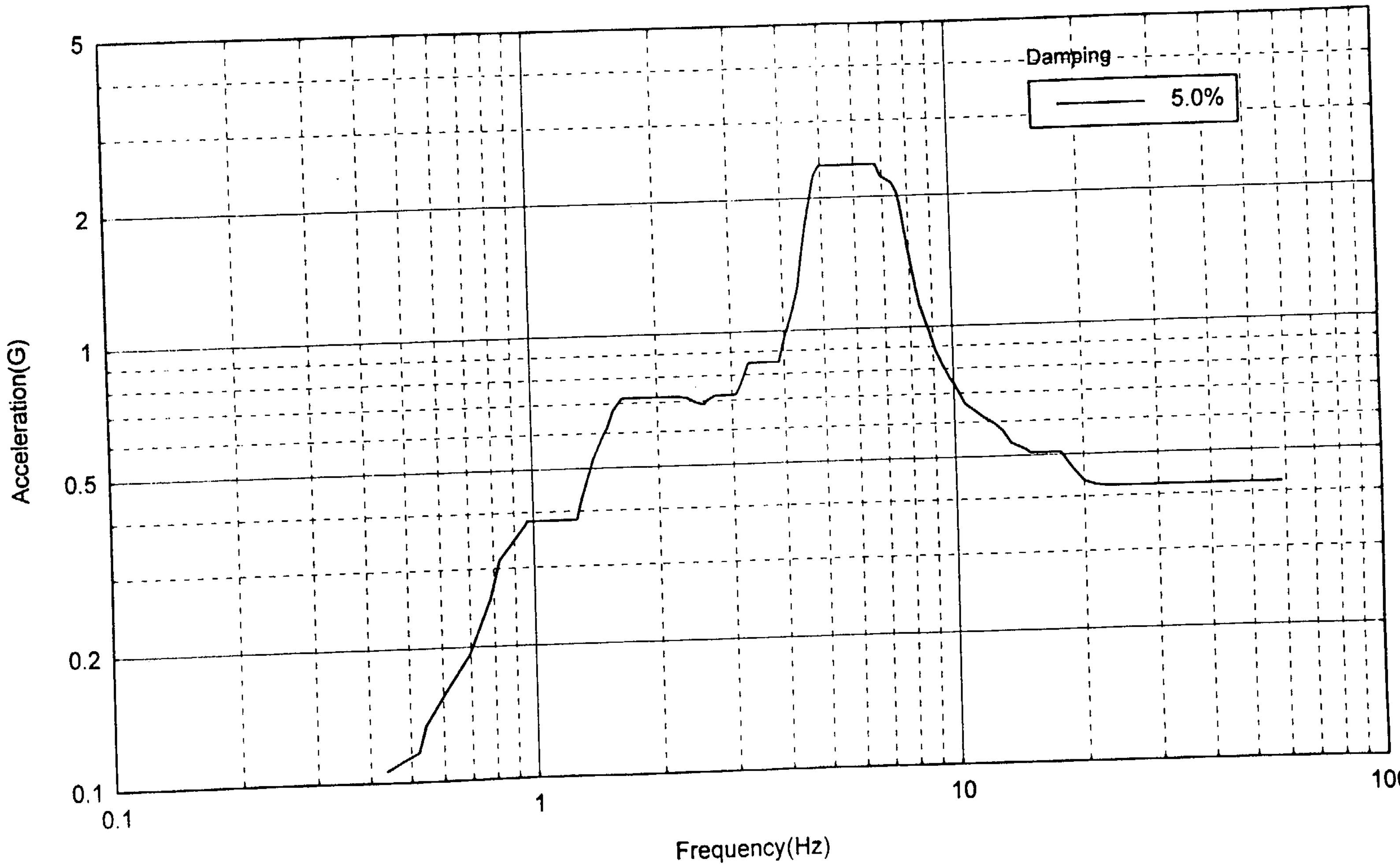


Figure B-6

Commonwealth Edsion Company
Dresden Nuclear Power Station
In-Structure Response Sepctrum for SSE

BUILDING : Reactor
ELEVATION : 570.0'
DIRECTION : North-South

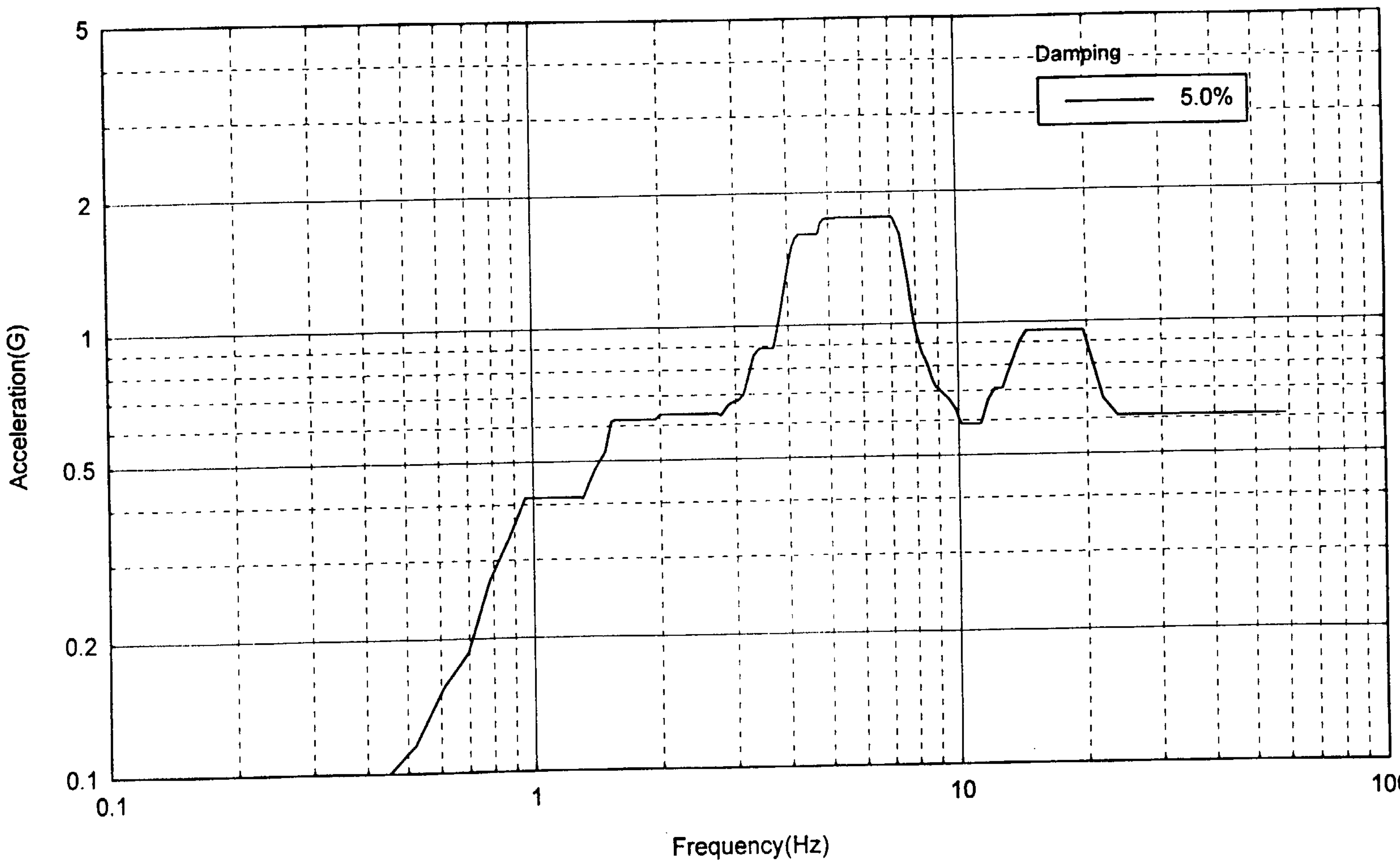
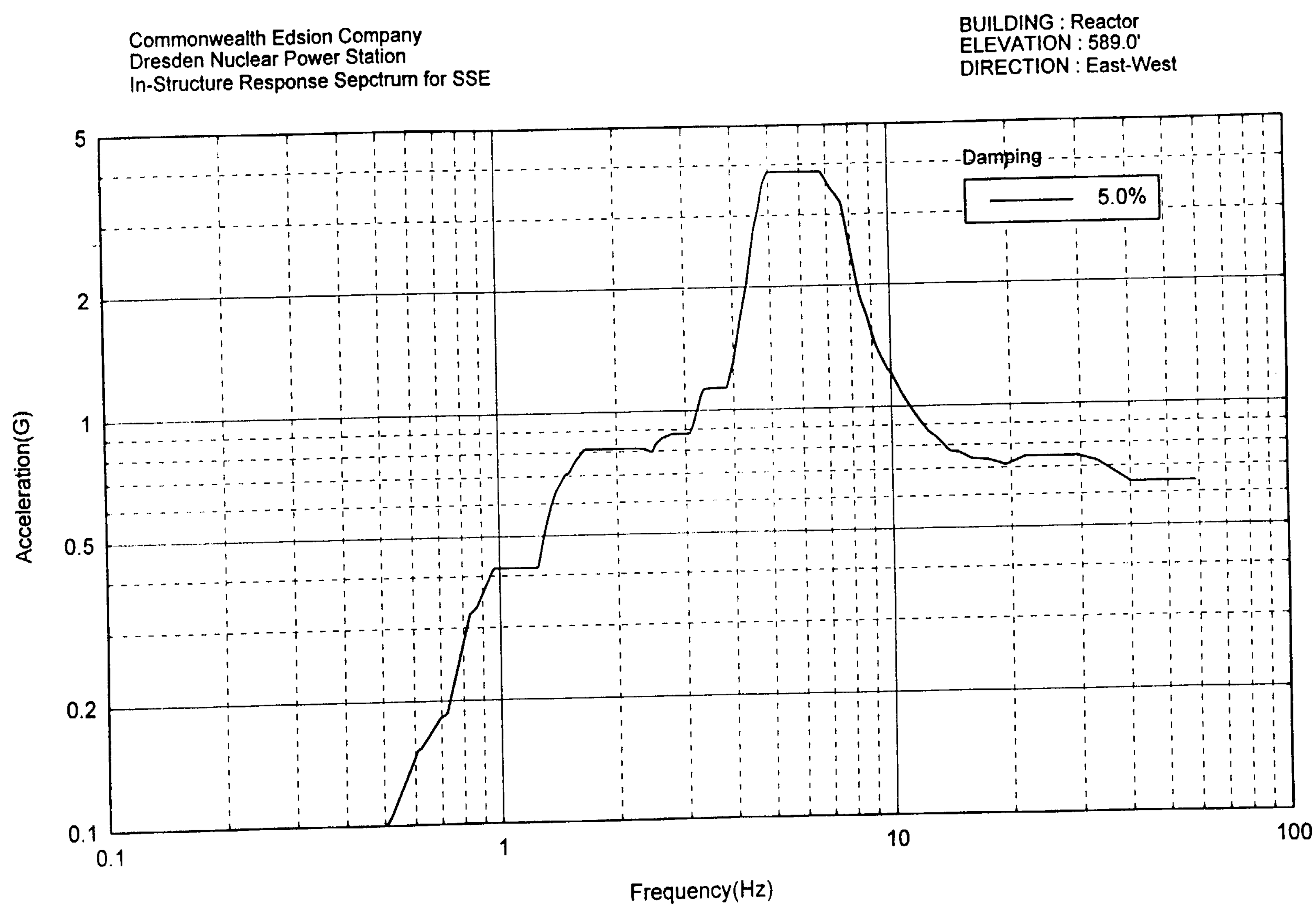


Figure B-7



Commonwealth Edsion Company
Dresden Nuclear Power Station
In-Structure Response Sepctrum for SSE
BUILDING : Reactor
ELEVATION : 589.0'
DIRECTION : East-West
Damping
5.0%
Acceleration(G)
5
2
1
0.5
0.2
0.1
0.1
1
10
100
Frequency(Hz)

Figure B-8

Commonwealth Edsion Company
Dresden Nuclear Power Station
In-Structure Response Sepctrum for SSE

BUILDING : Reactor
ELEVATION : 589.0'
DIRECTION : North-South

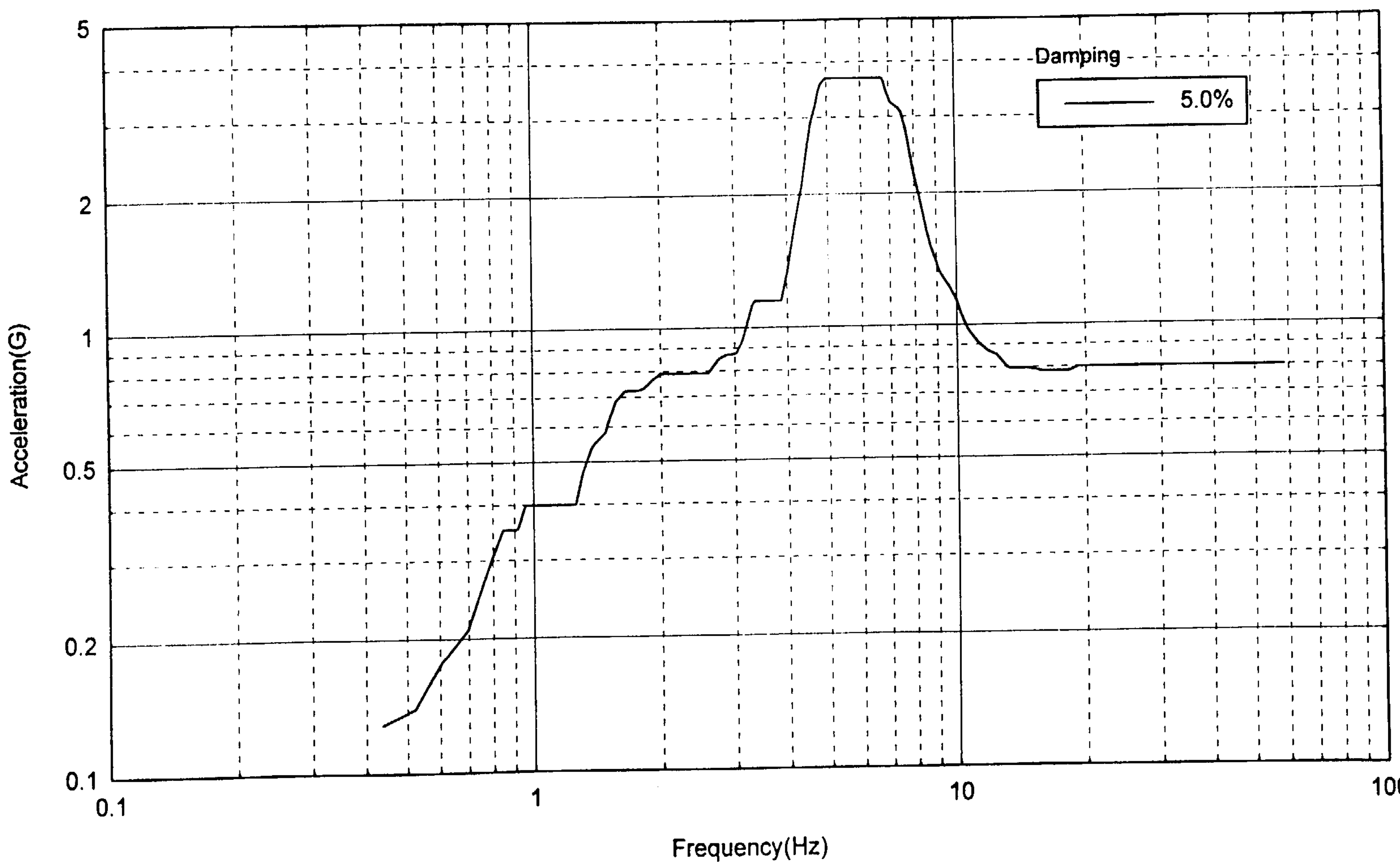


Figure B-9

Commonwealth Edsion Company
Dresden Nuclear Power Station
In-Structure Response Sepctrum for SSE

BUILDING : Turbine
ELEVATION : 538.0'
DIRECTION : East-West

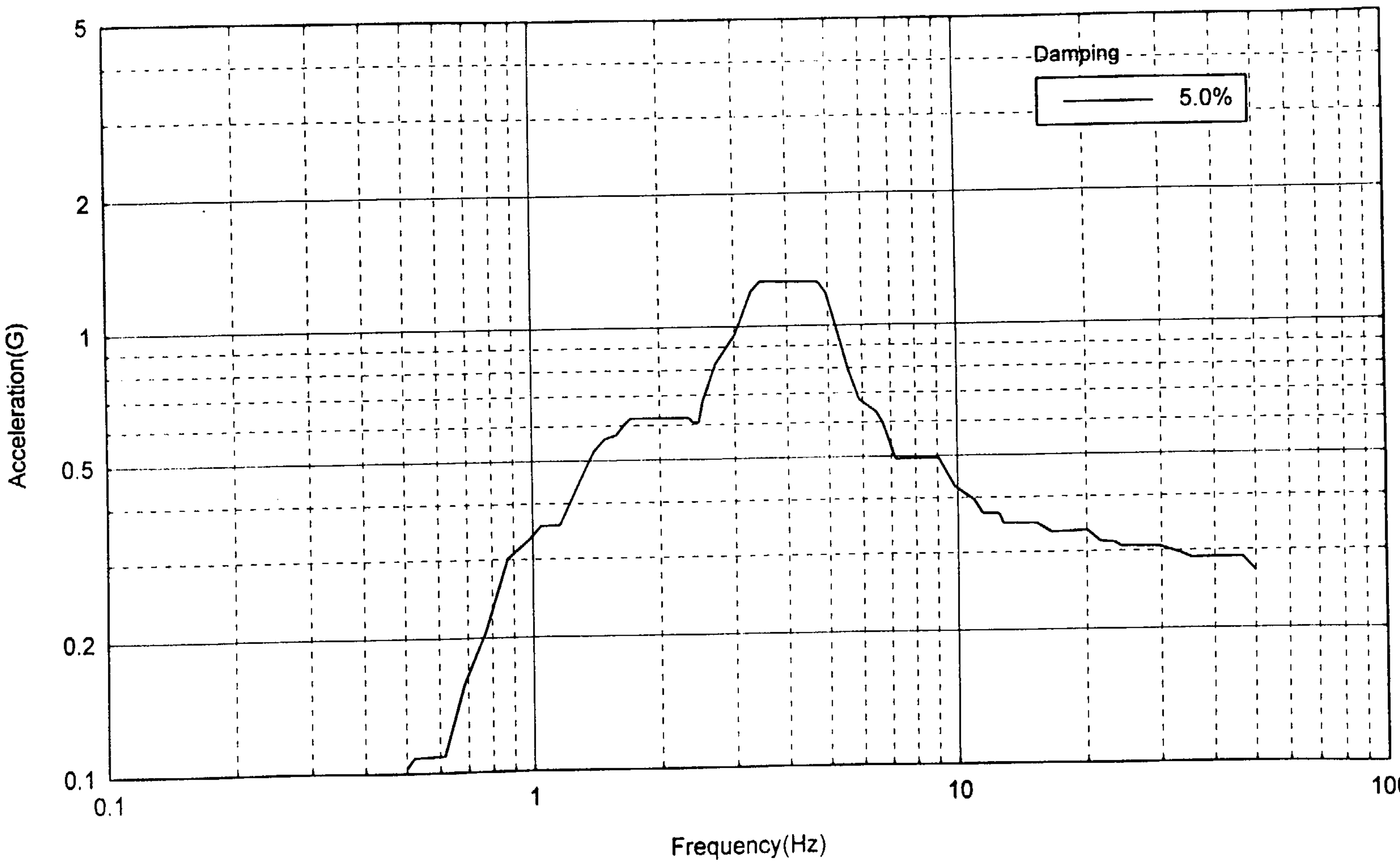


Figure B-10

Commonwealth Edsion Company
Dresden Nuclear Power Station
In-Structure Response Sepctrum for SSE

BUILDING : Turbine
ELEVATION : 538.0'
DIRECTION : North-South

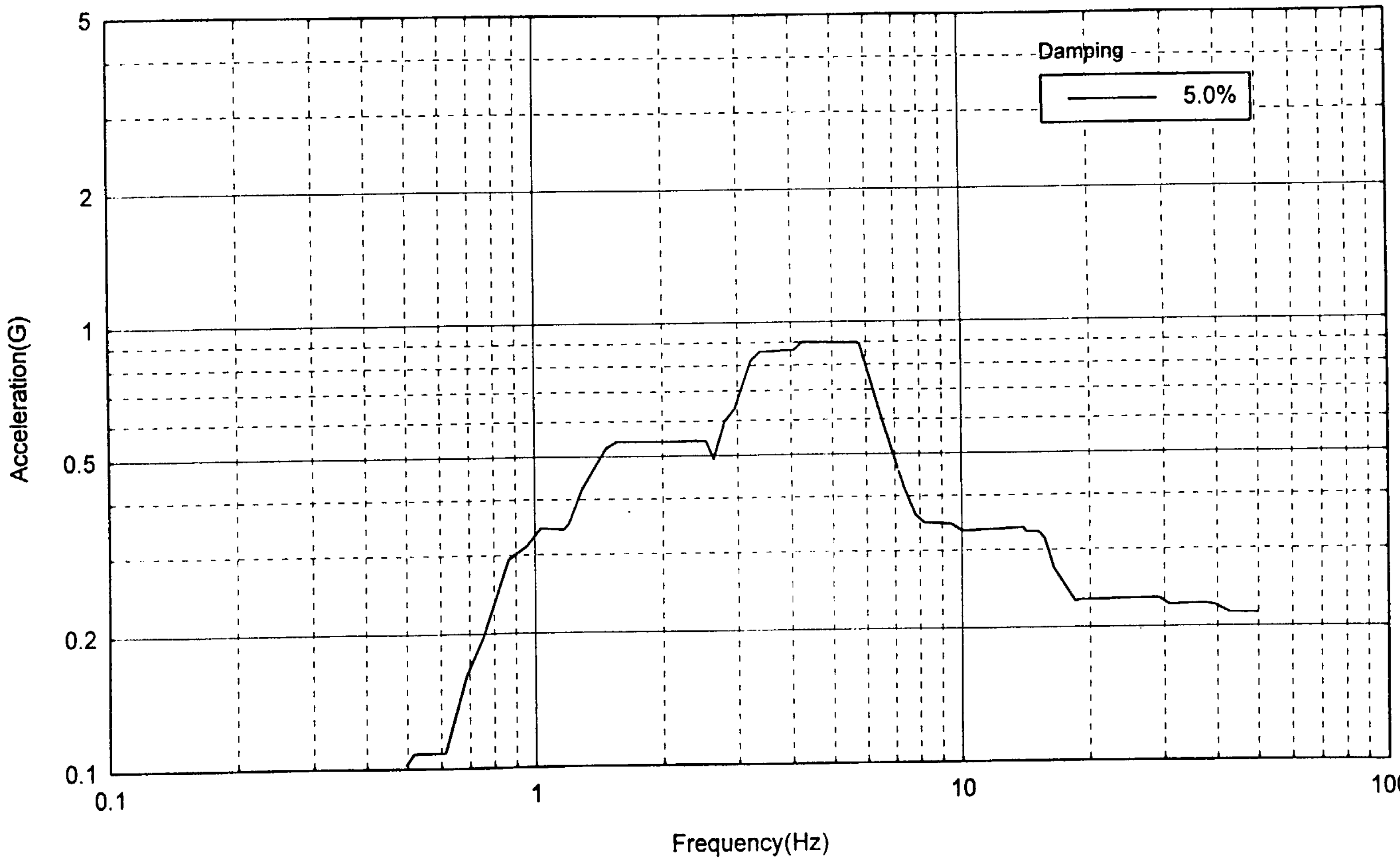


Figure B-11

Commonwealth Edsion Company
Dresden Nuclear Power Station
In-Structure Response Sepctrum for SSE

BUILDING : Turbine
ELEVATION : 561.5'
DIRECTION : East-West

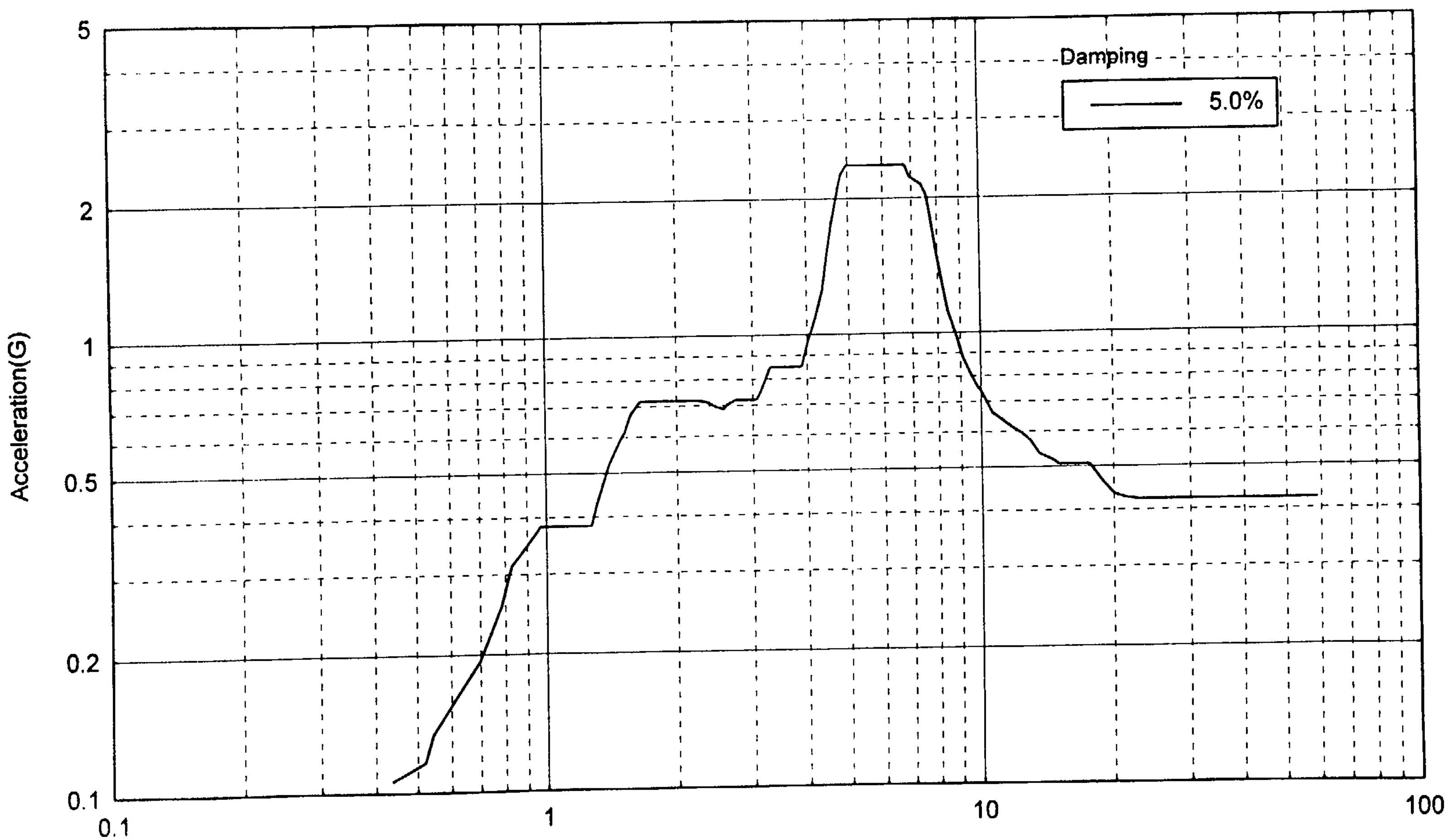


Figure B-12

Commonwealth Edsion Company
Dresden Nuclear Power Station
In-Structure Response Sepctrum for SSE

BUILDING : Turbine
ELEVATION : 561.5'
DIRECTION : North-South

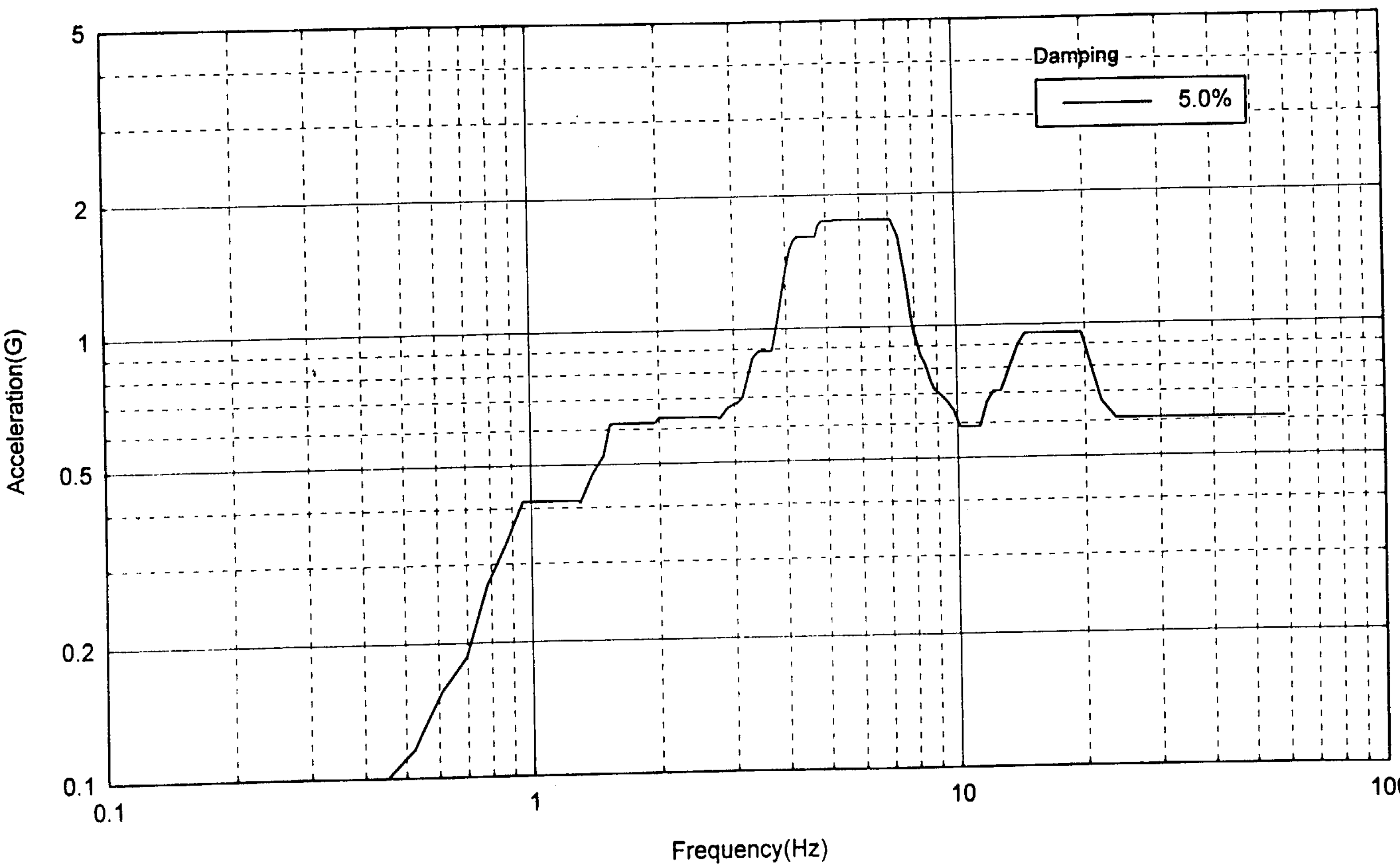


Figure B-13

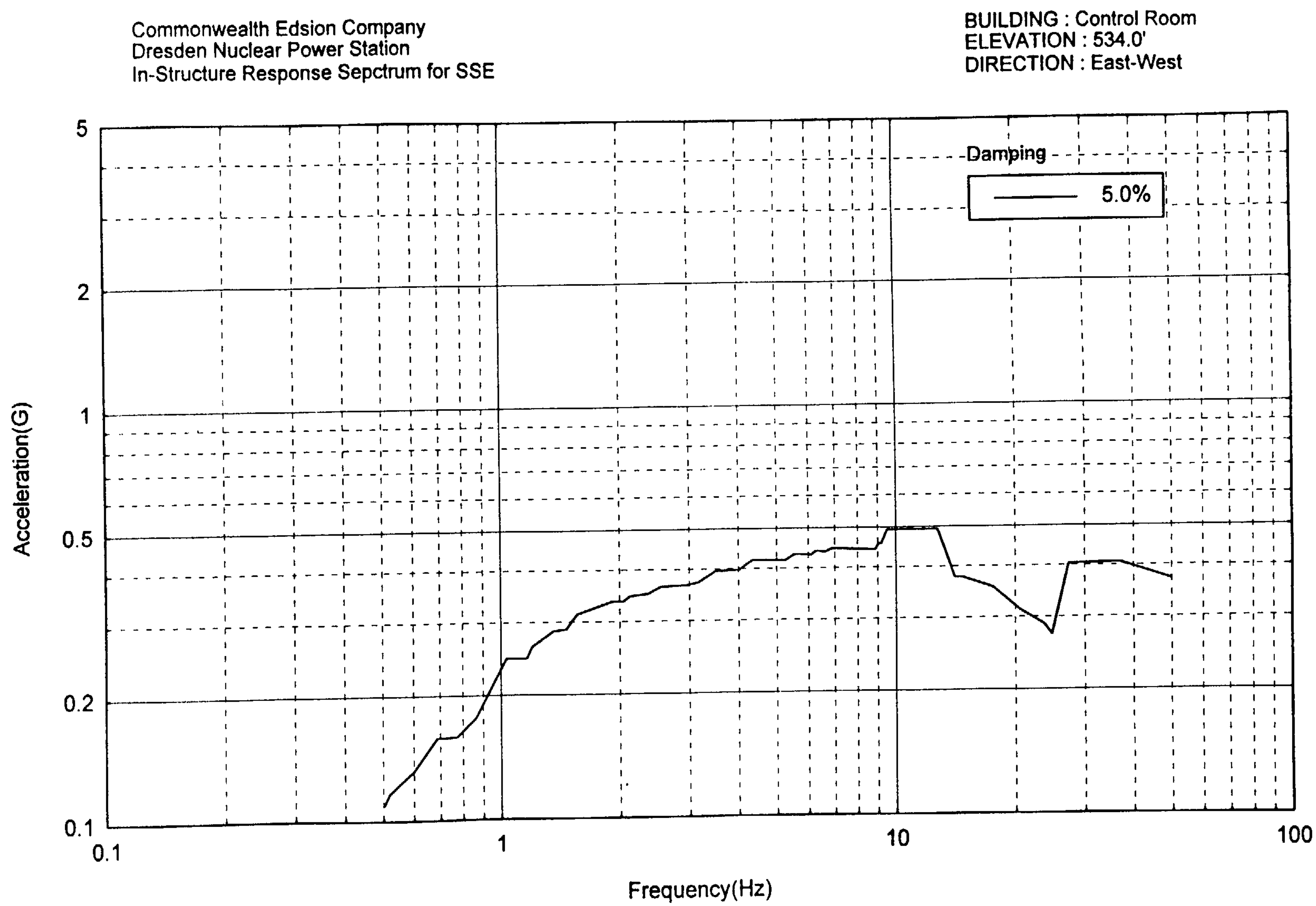

Commonwealth Edsion Company
Dresden Nuclear Power Station
In-Structure Response Sepctrum for SSE
BUILDING : Control Room
ELEVATION : 534.0'
DIRECTION : East-West
Damping
5.0%
Acceleration(G)
5
2
1
0.5
0.2
0.1
0.1
1
10
100
Frequency(Hz)

Figure B-14

Commonwealth Edsion Company
Dresden Nuclear Power Station
In-Structure Response Sepctrum for SSE

BUILDING : Control Room
ELEVATION : 534.0'
DIRECTION : North-South

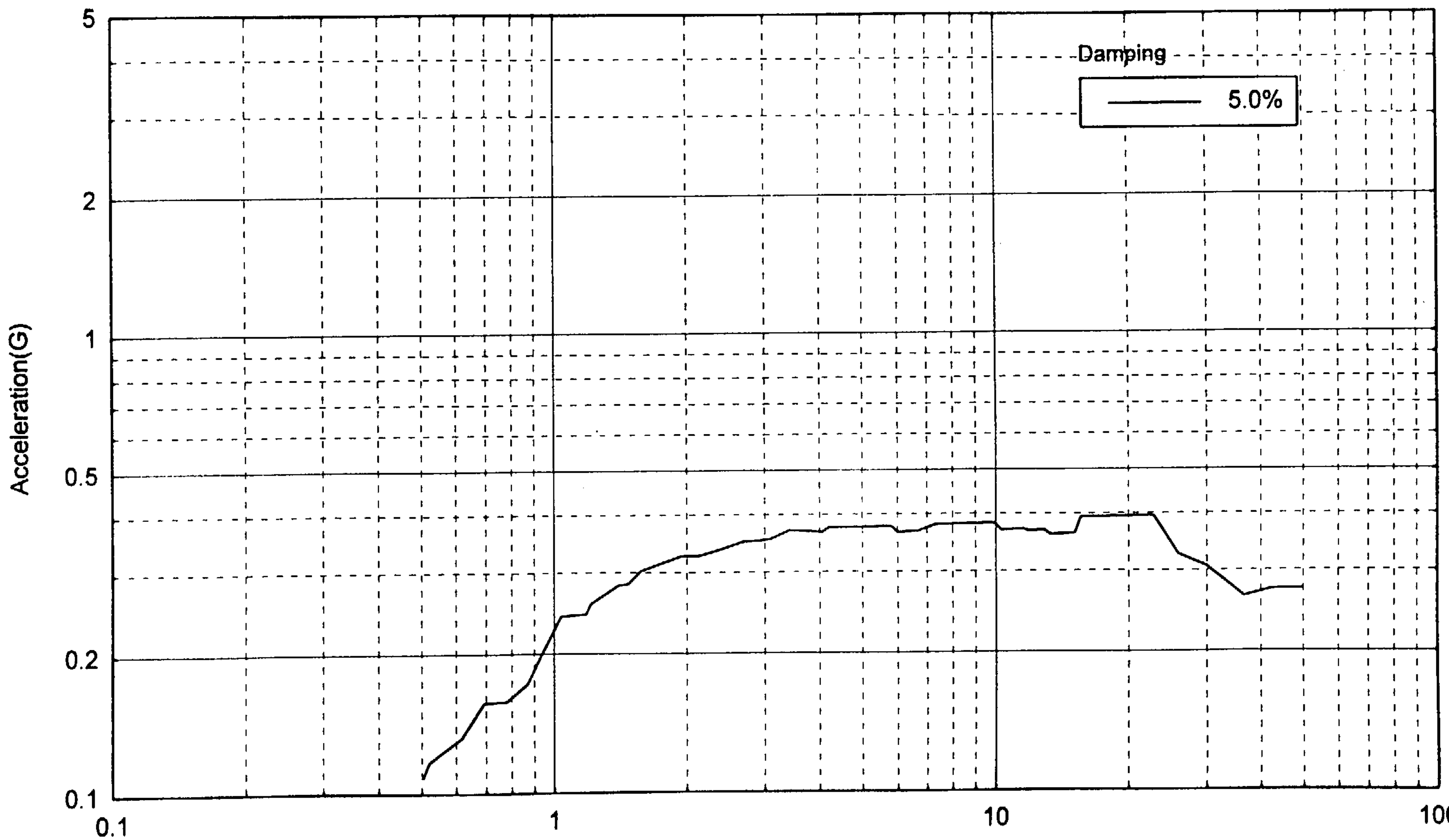


Figure B-15

Commonwealth Edsion Company
Dresden Nuclear Power Station
In-Structure Response Sepctrum for SSE

BUILDING : Battery Room
ELEVATION : 549.0'
DIRECTION : East-West

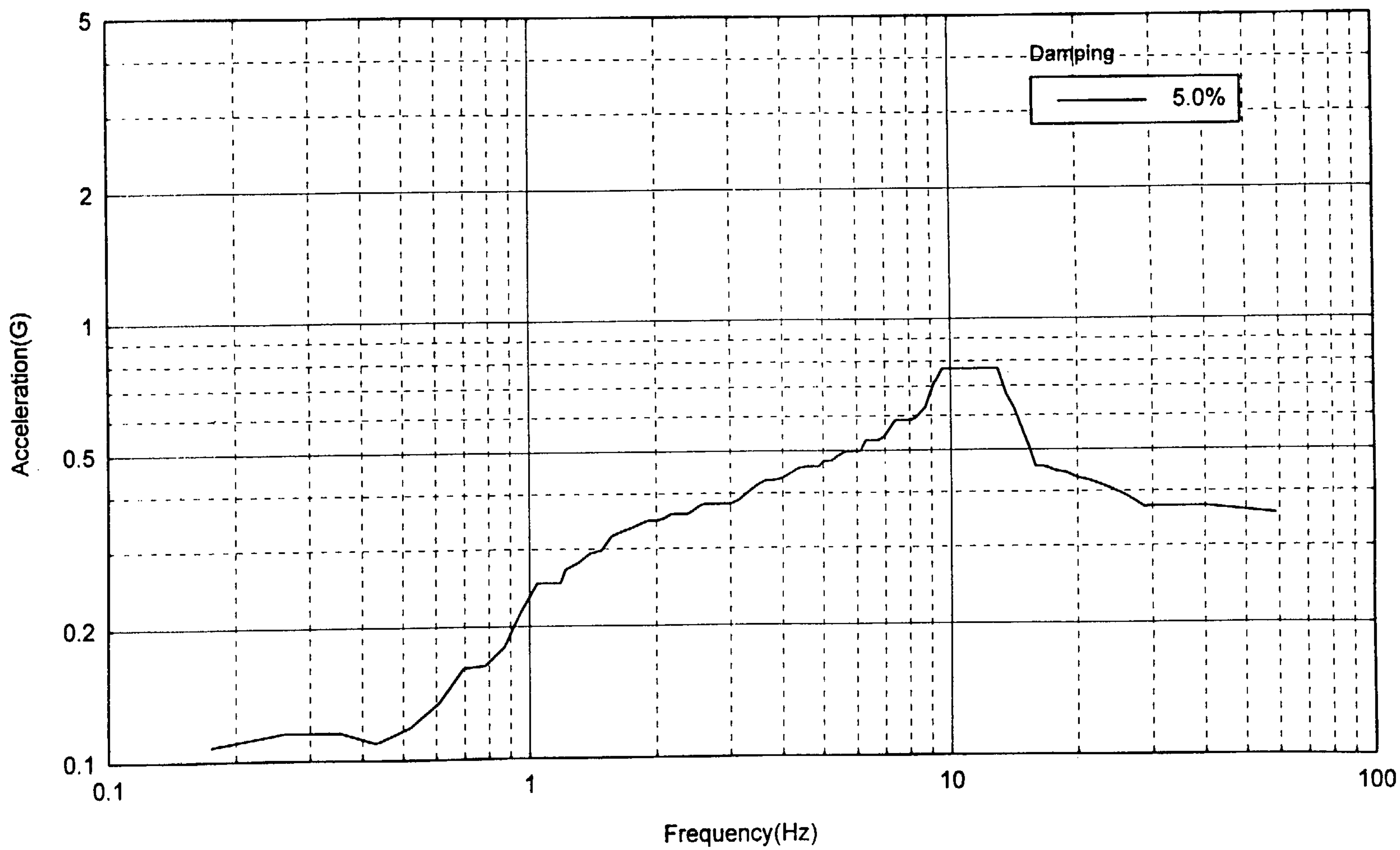


Figure B-16

Commonwealth Edsion Company
Dresden Nuclear Power Station
In-Structure Response Sepctrum for SSE

BUILDING : Battery Room
ELEVATION : 549.0'
DIRECTION : North-South

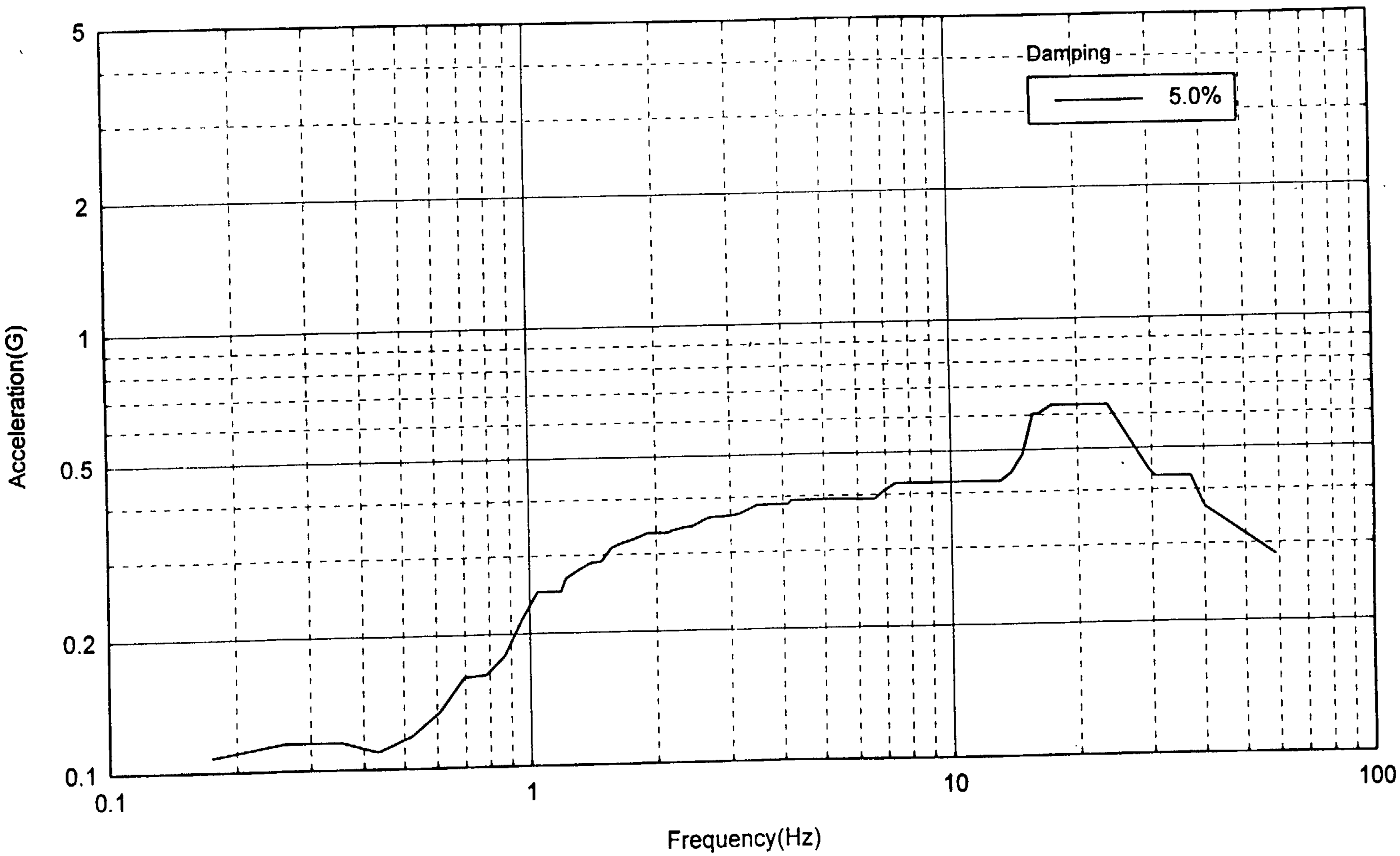


Figure B-17