GPU NUCLEAR

GPU Nuclear, Inc.
U.S. Route #9 South
Post Office Box 388
Forked River, NJ 08731-0388
Tel 609-971-4000

June 29, 2000
1940-00-20148

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

Dear Sir:

Subject: Oyster Creek Nuclear Generating Station
Docket No. 50-219
Reply to a Response for Additional Information
Individual Plant Examination of External Events

By letter dated March 3, 2000, the USNRC requested that GPU Nuclear, Inc. provide additional information relating to the Individual Plant Examination of External Events for the Oyster Creek Nuclear Generating Station (TAC No. M83652). This request was further discussed per telecon with the USNRC staff. The reply to that request is enclosed.

If any additional information or assistance is required, please contact Mr. John Rogers of my staff at 609.971.4893.

Very truly yours,

Sander Levin, Acting Site Director
Oyster Creek Nuclear Generating Station

SL/JJR
cc: Administrator, Region I
NRC Project Manager
Senior Resident Inspector

A011

RESPONSE TO THE SUPPLEMENTAL REQUEST FOR ADDITIONAL INFORMATION
OYSTER CREEK NUCLEAR GENERATING STATION
SEISMIC PORTION OF THE OYSTER CREEK
INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE)

The following provides the GPU Nuclear Corporation responses to the supplemental request for additional information. The NRC questions are provided in *Italics* and the GPU response underneath.

1) *The discussion in the submittal and RAI responses regarding the use of non-safety systems in the probabilistic risk assessment (PRA) model lacks sufficient detail. The concern is that these systems were not designed for the seismic motion of interest, or they may be housed in non-safety structures. In addition, there is a question of whether all the failure modes and all the components of such systems were adequately considered. While these systems already fail with a high probability in seismic ground acceleration ranges SEIS3 and SEIS4, centered at 0.54g and 0.72g, respectively (See Table 3-1 of the submittal), significant failure probabilities of such systems in seismic ground acceleration range SEIS2 (centered at 0.36g) might lead to substantial changes in the results of the seismic analysis. Therefore, please provide the following information regarding the firewater makeup to the isolation condenser, and the use of the combustion turbines and turbine building equipment for power recovery after a seismic event:*

The Oyster Creek Seismic Individual Plant Examination for External Events (IPEEE) models both the fire protection system and the station blackout combustion turbines. Both of these systems are non-safety related and non-seismic. The fire protection system is modeled as providing makeup to the isolation condensers following failure of the condensate transfer system. The station blackout combustion turbines are modeled as providing electric power following a station blackout event.

From a review of the results of industry seismic studies and the Oyster Creek Probabilistic Risk Assessment (Level 1), it was recognized early in the seismic IPEEE process that the loss of offsite power would contribute significantly to the total seismic risk. These systems can provide significant mitigation potential for loss of offsite power and station blackout events. In the interest of providing a realistic estimate of the risks associated with seismic events the fire protection system and the combustion turbines were included in the seismic model.

Given that the systems are both non-safety related and non-seismic, considerable effort was expended to ensure that these systems were appropriately addressed in the seismic model. These efforts included extensive plant walkdowns, a review of all potential seismic failure modes and calculation of equipment specific fragilities where necessary. The failure modes evaluated include seismic failures of the structures, seismic interaction failures, seismic failure of soils (i.e., liquefaction), and the seismic failure of the components within these systems.

(a) *What types of firewater pumps are used (e.g., electric, diesel driven, etc.)? Are any parts of the firewater makeup system, from the pumps to the isolation condenser, seismically qualified? If so, which ones?*

Oyster Creek employs a single electrical fire pump and two diesel driven firewater pumps for the purpose of suppression of fires. The electrical firewater pump takes suction from the redundant firewater tank. The electrically driven redundant firewater pump and the redundant fire water tank are not modeled in the Oyster Creek seismic PRA. (See page 3-87 of the submittal).

The diesel driven fire protection water pumps are modeled for the purpose of providing makeup to the isolation condensers. No portion of the fire protection system is seismically qualified. The isolation condenser system and the isolation condenser makeup valves (V-11-35 and V-11-36) are seismically qualified.

(b) *Were structural, support and other failure modes of all the components of the firewater makeup system evaluated? Please list which components were evaluated, what the failure modes and the calculated capacities were, and which failure modes and components were omitted, and Why?*

All failure modes and components of the fire protection water system that could impact the ability of the fire protection system to perform its seismic IPEEE function were investigated. The seismic IPEEE function of the fire protection system is to provide shell side makeup water to isolation condensers.

The structural failure modes of the fire protection system evaluated include the fire pond pump house, the fire pond dam and fire protection underground piping. A list of the failure modes and calculated capacities are provided on Table 1. The majority of the fire protection piping used to provide isolation condenser shell side makeup is located underground. Therefore, extensive evaluation of supports is not necessary. Other portions of the fire protection system were evaluated for the potential for seismically induced actuation. Seismic actuation or failure of the fire protection system can result in the potential for flow diversion and could impact of the ability of the fire protection system to perform its modeled function. See the response to question 1c for additional information on the treatment of the seismic actuation of the fire protection system. Section 4.8 of the IPEEE submittal report, "Treatment of Sandia Fire Risk Scoping Study Issues" provides details on the potential for seismic / fire interactions of the fire protection system.

In addition, Section 4.8 of the IPEEE submittal includes a complete description of the fire protection system (including several photographs of the diesel / pump arrangement and anchorage) as well as an evaluation of the potential for the seismically induced actuation of the fire suppression system. It should be noted that the all seven of the drop-weight actuation valves have been replaced (as a result of IPEEE findings) with newer and more reliable actuation valves. See IPEEE Submittal Section 7, Fire Analysis, Recommendation #6.

Table 1 – Fire Protection System Seismic Failure Modes Evaluated

No.	Description of Component, Structure or Failure Mode	$\hat{A}_g$	ϑ_R	ϑ_U	Reference No.
1	Fire Pond Pump House	1.21	0.25	0.44	1
2	Fire Pond Dam	high	n/a	n/a	1
3	Fire Pond Underground Piping	1.57	0.14	0.10	2
4	Diesel Driven Fire Protection Pump P-9-102A	high	n/a	n/a	3,4
5	Diesel Driven Fire Protection Pump P-9-102B	high	n/a	n/a	3.4
6	Relief Valve V-9-57	high	n/a	n/a	3
7	Relief Valve V-9-67	high	n/a	n/a	3
8	Diesel Driven Pump Fuel Tank No. 1	high	n/a	n/a	3
9	Diesel Driven Pump Fuel Tank No. 2	high	n/a	n/a	3
10	Diesel Fire Pump Batteries No. 1	high	n/a	n/a	3
11	Diesel Fire Pump Batteries No. 2	high	n/a	n/a	3
12	Isolation Condenser Makeup Valve V-11-35	high	n/a	n/a	3
13	Isolation Condenser Makeup Valve V-11-36	high	n/a	n/a	3

References:

1. Page 3-54, IPEEE Submittal Report (Reference 3-2 (provided with previous RAI response))
2. Page 3-55, IPEEE Submittal Report (Reference 3-2 (provided with previous RAI response))
3. Page 3-78, IPEEE Submittal Report, Table 3-6
4. Section 4.8, IPEEE Submittal Report, Figures 9,10, 11 and 12 (pictures of FP diesels)

(c) *Are there any special considerations regarding operator actions related to the use of this system following a seismic event, which do not exist for such actions modeled in the internal PRA (e.g., impeded access for actions outside the control room, or timing of such actions)? If so, what were these considerations and how were they taken into account?*

There were no special considerations modeled in the seismic IPEEE regarding operator actions related to the use of the fire protection system for isolation condenser makeup. The actions which take place outside the control room occur on the east side of the reactor building on the 23 foot elevation. The equipment located in the reactor building is seismically designed and access to this area of the plant is not expected to be impeded. The timing of the operator action associated with the fire protection makeup to the isolation condenser is not expected to change as a result of a seismic event as opposed to an internally initiated event.

However, a failure mode not previously considered involves the failure of portions of the fire protection system in non-safety related structures which could discharge (i.e., actuate) during a seismic event. The resulting failure of fire protection piping could result in the diversion of flow such that makeup to the isolation condenser was not sufficient.

An additional operator action to isolate the failed portion of the fire protection system was not modeled in the seismic IPEEE. Isolation of a failed portion of the fire protection system is not anticipated to require complex actions. Diagnosis of the failure is simple due to the availability of alarms which indicate actuation of the system. Access to perform the actions is not difficult since the fire protection system can be isolated from various locations including outside all buildings and at various locations along the ring header located in the yard.

A sensitivity study is performed to evaluate the seismic core damage frequency impact of an increase in the human error probability associated with makeup to the isolation condenser using the fire protection system. Details of this sensitivity case are provided in Attachment A.4. The human error probability increase assumes that a break in the fire protection system occurs in a non-safety related structure which must be isolated in order for the fire protection system to provide sufficient flow to the isolation condenser.

The current human error probability used in the seismic IPEEE for operators providing makeup to the isolation condensers using fire protection is 4E-04. In this sensitivity case, this human error probability is changed to a screening value of 0.1. This change in the human error probability is to account for the potential additional action of isolation of the fire protection header or termination of fire protection system actuation which may occur following a seismic event.

The core damage frequency, given the change to the human error probability for isolation condenser makeup, is 3.8E-06 per year. This increase represents an increase of approximately 5.6% over the seismic IPEEE core damage frequency of 3.6E-06 per year. This increase is expected given the importance of the makeup to the isolation condenser system in loss of offsite power and station blackout events.

(d) *For power recovery using the non-seismic combustion turbines, which failure modes and components were considered? Was failure of the combustion turbines' batteries considered (e.g., due to failure of the battery spacers)? If so, what were the results of the evaluation? If not, please provide the justification for not doing so?*

All failure modes and components of the station blackout combustion turbines that could impact the ability of the combustion turbines to provide power following a station blackout were evaluated. Three separate walkdowns of the station blackout combustion turbine were performed.

The first walkdown was performed to assess the capacity of the station blackout combustion turbines fuel oil tank and gas pipeline. Walkdown participants included:

Ken Canavan	–	GPU Nuclear, PRA Engineer
David Nakaki	–	EQE International, Seismic Capability Engineer
Maurice Power	–	Geomatrix Consultants, Soils and Liquefaction Expert
Chuck Ramdeen	–	GPU Nuclear, Seismic Capability/SQUG Engineer
Ken Whitmore	–	GPU Nuclear, Civil/Structural Engr., Seismic Capability Engr.

The results of these walkdowns are documented in Section 3.1.3 of the IPEEE Submittal Report. Additional references are 3-1, 3-2, 3-4 and 3-6. These references were provided in response to the previous request for additional information.

The second walkdown was performed on the station blackout combustion turbines themselves. This walkdown included all aspects of the combustion turbines including: combustion turbine battery compartments, local switchgear, local control panels, combustion chambers, exhaust stacks, building and supports, as well as the cooling systems (fin fan coolers and coolant pumps). Walkdown participants included:

Ken Canavan	–	GPU Nuclear, PRA Engineer
Thomas Kipp	–	EQE International, Seismic Capability Engineer
Chuck Ramdeen	–	GPU Nuclear, Seismic Capability/SQUG Engineer

Section 7 of the IPEEE Submittal Report, provides two recommendations based on these extensive walkdowns. These recommendations were:

"1) Ensure all bolts on the Forked River Combustion Turbine fin-fan coolers are installed and torqued properly. Although adequate capacity for withstanding the range of potential ground motions exist, additional margin could be obtained by ensuring all bolts are installed and torqued properly.

2) Consider the addition of battery spacers in the Combustion Turbine battery compartments. Although the current battery spacing is sufficient to prevent battery failure due to interactions, additional margin could result with the use of battery spacers."

These recommendations were based on improving the existing capacity of the station blackout combustion turbines. The station blackout combustion turbine seismic capacity used in the IPEEE was based on the as-found condition and the combustion turbine batteries were considered in the calculation of the fragility.

A third walkdown, performed in support of these recommendations, verified that little improvement would be achieved as a result of modifications to the battery compartments and the recommendation was closed. Fin-fan cooler bolts were added and torqued. Therefore, use of the existing station blackout combustion turbine seismic capacity is conservative. Participants on this walkdown included:

Ken Canavan	–	GPU Nuclear, PRA Engineer
Mike Godknecht	–	GPU Nuclear, System Engineer (SBO Combustion Turbines)
T. H. Chang	–	GPU Nuclear, Civil/Structural Engineer

The seismic capacity of the station blackout combustion turbines is provided in Table 3-6 of the IPEEE Submittal Report. Seismic capacities associated with the combustion turbine fuel oil tank and gas pipeline are contained in Section 3.1.3 of the IPEEE Submittal Report.

It should be noted that the seismic capacity of the gas pipeline is not used in the seismic analysis (i.e., the gas supply to the combustion turbines is not modeled). The fuel for the station blackout combustion turbines is modeled from the fuel oil tank only. This is due to concerns associated with other portions of the gas pipeline system and the uncertainty of gas supplies following a seismic event.

The following table provides a summary of the failure modes and components of the combustion turbines which were evaluated.

Table 2 – Combustion Turbine Seismic Failure Modes Evaluated

No.	Description of Component, Structure or Failure Mode	$\hat{A}_g$	ϑ_R	ϑ_U	Reference No.
1	Combustion Turbine Fuel Oil Tank	0.66	0.37	0.39	1
2	Combustion Turbine Gas Supply Piping	0.75	0.14	0.21	2
3	SBO Transformer	0.53	0.28	0.23	3
4	Liquefaction of Combustion Turbine Slab	high	n/a	n/a	4
5	Combustion Turbine No. 1 (includes building, stack, batteries, fin-fan coolers, control panels and all support components located at Forked River site)	0.60	0.37	0.39	3
6	Combustion Turbine No. 2 (includes building, stack, batteries, fin-fan coolers, control panels and all support components located at Forked River site)	0.60	0.37	0.39	3

References:
1. Page 3-56, IPEEE Submittal Report (Reference 3-2 (provided with previous RAI response))
2. Page 3-57, IPEEE Submittal Report (Reference 3-2 (provided with previous RAI response))
3. Page 3-78, IPEEE Submittal Report, Table 3-6
4. Geomatrix, "Assessment of Potential for Liquefaction and Permanent Ground Displacements at Designated Facilities, OCNGS", page 11.

(e) *There are potential failure modes of the emergency switchgear located in the turbine building, as well as the underground power cables to the combustion turbines, related to block wall failures, soil liquefaction or turbine building structural failure. How were these failure modes incorporated into the analysis of power recovery following a seismic event?*

The emergency switchgear located in the turbine building were evaluated for their ability to perform their seismic IPEEE function. All modes of seismically induced failure were investigated. These failure modes include:

- Seismically induced structural failure of the turbine building is assessed and documented in Section 3.1.3.2 of the IPEEE Submittal Report and Reference 3-6. Structural failure is not included in the seismic model since the seismic capacity calculated is for the onset of structural deformation and permanent displacement, not collapse. Since deformation and permanent displacement is not expected to fail the switchgear, the seismic capacity of the turbine building is not included. Cables associated with the station blackout transformer enter the building above grade level. Sufficient slack exists in the cables to prevent

failure following building deformation or permanent displacement (this aspect of seismically induced failure was addressed during the walkdown performed as part of the fragility evaluation for the SBO Transformer). Therefore, the turbine building seismic failure is not included in the seismic model.

Attachment A.3 to this report details a seismic model re-evaluation which includes the turbine building structural failure. The impacts of turbine building structural failure include the failure of all equipment housed in the turbine building.

- Seismically induced failure of the switchgear is assessed in Table 3-6: List of Components for Seismic Evaluation with Associated Fragilities, as well as References 3-3 and 3-8.

- Relay chatter failure of the switchgear. Relay chatter was conservatively assumed to fail the associated switchgear without credit for operator recovery. Results of the evaluation are contained in Table 3-7: Electrical Panel Fragilities Based on Relay Function. Details of the evaluation are contained in References 3-3 and 3-8.

- Seismic fragilities associated with required ventilation components are assessed in Table 3-6: List of Components for Seismic Evaluation with Associated Fragilities, and References 3-3 and 3-8.

- The impact of block walls is assessed in Section 3.1.4.3, Evaluation of Masonry Block Walls, and Reference 3-7. No other block walls impact the transmission of power from the station blackout combustion turbines and the switchgear.

- Liquefaction potential for the station blackout combustion turbine slab is investigated in reference 3-6.

  The potential for liquefaction failure of the soil supporting the combustion turbine fuel oil tank and gas supply piping is also investigated in reference 3-6. Although a seismic capacity of the combustion turbine gas supply line was calculated, it was not modeled in the Oyster Creek seismic IPEEE since offsite supply of gas may not be available following a seismic event.

  The liquefaction failure of the soil supporting the combustion turbine cables is not modeled. The combustion turbine power cables are sufficiently ductile that failure of the soil and permanent displacement of soil is not expected to fail the cables.

2) *It is not clear from the RAI response to the original seismic question 3, sent previously, whether the recirculation pump support failure was considered in developing the Nuclear Steam Supply System (NSSS) seismic capacity numbers. Failure of such supports have been shown to be an important contributor to seismically induced loss-of-coolant accidents (LOCAs) and to the seismic core damage frequency (CDF) in some boiling water reactors (BWRs).*

(a) *Please provide the seismic capacity estimated for the recirculation pump supports and discuss how this capacity was obtained.*

(b) *Was the failure of the recirculation pump supports considered as one of the failure modes in developing the NSSS piping seismic capacity? If it was not, what is the justification for omitting the support failure?*

In conducting seismic PRAs, the screening guidelines in EPRI NP-6041 for seismic margins assessment are utilized along with generic fragility calculations to determine the types of components and conditions under which components can be screened out. In the OCNGS seismic

PRA, a target fragility for screening was defined as having a HCLPF of 0.3g pga and a median capacity of 1.0g pga. The following discussion documents the rationale for screening out the recirculation pumps and their supports as having a capacity meeting or exceeding the target fragility screening level. Also presented are results of detailed licensing basis recirculation piping re-analyses recently conducted by GPU Nuclear that reinforce the prior decision to screen out the recirculation loops including the pumps and supports.

In EPRI NP-6041, Table 2-4, primary coolant systems of PWRs and BWRs can be screened at 1.2g 5% damped ground motion spectral acceleration (equivalent to 0.5g peak ground acceleration, pga). The only caveat on this is if there is suspected intergranular stress corrosion cracking (IGSCC) in BWR piping. The IGSCC issue in recirculation lines in BWRs was previously resolved, and with this resolution in mind we could state that the ground motion HCLPF of the recirculation system, including piping, pumps and pump supports, is 1.2g spectral acceleration (0.5g pga).

There is, however, a discussion in EPRI NP-6041, Appendix A, page A-9, that notes that some BWR recirculation pumps have estimated HCLPF capacities below 0.5g pga. In some of the early seismic PRAs, the recirculation pump supports were found to have large median capacities, well in excess of 1.0g pga, but the uncertainty was also high which resulted in a HCLPF in one instance that approached 0.3g pga. Thus the conclusions in Appendix A suggest that BWR reactor vessel supports and recirculation pump supports should be reviewed for SME levels between 0.3g and 0.5g pga. This recommendation was not, however, reflected in the Table 2-4 screening guidelines. Even so, we could conservatively conclude that the HCLPF for BWR recirculation loops, including the recirculation pump supports, is at least 0.3g pga. The 0.3g to 0.5g screening threshold implied by EPRI NP-6041 meets or exceeds the 0.3g target HCLPF for screening of OCNGS components.

The screening criteria in EPRI NP-6041 for primary coolant systems are primarily based on early PRAs of BWRs and on the results of extensive studies in the USNRC sponsored Load Combination Program. Studies by Lawrence Livermore National Laboratories and their contractors examined the probability of a large LOCA occurring due to direct and indirect failures of the primary system piping. The direct failures were assessed using probabilistic fracture mechanics models that examined the growth of cracks due to normal operating pressure and thermal loads and loads induced by earthquakes. The indirectly induced failures were examined by convolving seismic fragilities of primary system piping and component supports and a generic seismic hazard curve typical for an Eastern US site. Initially, all existing PWRs were studied and results were reported in NUREG/CR-3660 for Westinghouse PWRs, NUREG/CR-3663 for Combustion Engineering PWRs and NUREG/CR-4290 for Babcock and Wilcox PWRs. These studies were extended to BWRs, and results of a pilot study were reported in NUREG/CR-4792. In this BWR study, the conclusions were similar to those for PWRs except for the case where IGSCC was present in weld joints. In the absence of IGSCC, the probability of large LOCA from all sources was very low. The indirectly induced LOCA frequency due to failure of recirculation piping, primarily from failure of the recirculation pump supports, was calculated to be about 7.5E-8/year. The probability of LOCA due to failure of the reactor vessel supports was about 2E-8/year. Oyster creek was not specifically included in this BWR study, but the BWR pilot study, as well as the extensive studies of PWRs and PRA results for other BWRs, led the USNRC to conclude that the likelihood of double ended guillotine break caused by earthquakes was extremely low. As a result the Code of Federal Regulations was changed to allow the decoupling of LOCA and SSE.

In the fragility report for OCNGS, a generic calculation was provided in Section 5.1 to derive the median capacity of flexible systems in the reactor building. This calculation demonstrated that a component that meets the design basis seismic criteria would have a median capacity of about 1.35g pga. This is well above the threshold screening level of 1.0g pga used to determine if a component required a plant specific fragility calculation. This large generic capacity is primarily the result of the large degree of conservatism in the design basis in-structure response spectra in

the reactor building relative to median probabilistic spectra developed for IPEEE. The peak spectral accelerations of the median spectra resulting from IPEEE ground motion spectra anchored to an input pga equal to three (3) times the design basis pga are about equal to or less than the peaks of the design basis in-structure spectra. This comparison demonstrates a factor of conservatism of about three (3) in the design basis loading specified for components in the reactor building compared to loading that would result from an IPEEE spectrum. Appendix B of the component fragility report shows these comparisons of median probabilistic IPEEE spectra resulting from three (3) times the design basis pga input to the design basis spectra for several node points in the reactor building.

Based upon the above arguments and logic, it was concluded that a fragility that represents a large LOCA has a HCLPF value of between 0.3g and 0.5g pga and a median value of about 1.35g. Both of these values equal or exceed the target fragility parameters used to determine whether a component required a detailed fragility calculation or whether a generic screening level could be used.

A recent licensing basis reanalysis of the recirculation loops at OCNGS reinforces the above rationale used for the initial screening. GPU Nuclear Calculations C-1302-223-E540-034, 035 and 036 document the analyses conducted for the recirculation lines, including the return line from the emergency condenser in loop A. The calculations were conservatively conducted using envelope response spectra developed for the SSE ground motion. The recirculation pump is supported vertically by spring hangers and horizontally by two snubbers on the pump support and two snubbers on the pump motor. The snubbers on the pump support are the most highly loaded, the most critical having a demand to allowable load ratio of 0.92. However, there is sufficient conservatism in the analysis input motion and analysis method to conclude that the HCLPF and median capacities would exceed the target fragility threshold for screening. The following discussion of the design basis re-analyses demonstrates a capacity higher than the originally estimated screening level.

In the fragility report, Appendix B, it was shown that the median reactor building IPEEE response spectra resulting from a peak ground acceleration input motion of three times the design basis SSE were approximately equal to the design response spectra, implying a factor of three (3) conservatism. Further, comparing the actual envelope spectra used for the recirculation loop piping analysis to the median IPEEE response spectra for an input motion equal to three times the design basis SSE pga shows greater conservatism in the frequency range of 2.5 to 7.5 Hz. The attached Figures A-4, A-5 and A-6 from the recent GPU analysis show the envelope spectra used in the analysis compared to the spectra at different connection points of the recirculation loops (See Attachment B). Node 58, at an upper drywell penetration, defines the peak of the envelope response spectra used for the analyses. Spectral peaks on the reactor vessel are lower. The attached page B-12 from the EQE fragility report compares probabilistic spectra at Node 56 to design basis spectra. Node 56 is also on the drywell, about 16 feet below Node 58 as shown in the attached reactor building model. The envelope spectra used for the recirculation piping analyses are for CC N-411 damping. N-411 damping is a constant 5% up to 10 Hz, which is well beyond the peak of the horizontal spectra so comparison of 5% damped probabilistic to 5% damped design basis spectra at node 56 provides a close approximation to the ratio of 5% damped probabilistic spectra to CC N-411 damped design basis spectra at node 58 (IPEEE spectra were not available for Node 58).

The first mode for the "A" recirculation loop is approximately 4.7 Hz. Comparing the spectral accelerations of the IPEEE horizontal spectra resulting from three (3) times the SSE pga to the 5% damped SSE horizontal spectra used for the analysis, the minimum amount of conservatism is seen to be in the E-W direction and is about:

Conservatism in input = [Design Sa (SSE) / Probabilistic Sa (3xSSE)] [3]

$$= (0.50/0.20)\ (3) = 7.5$$

However, the pump supports will also see reaction forces from higher modes with less conservatism. Therefore, the factor of conservatism in the input should not be based solely on the first mode. Conservatively assuming only 50% of the support reaction force is from the first mode and the remainder is from the higher modes, the resulting factor of conservatism in the input can be estimated as:

$$(7.5 + 3.0) / 2 = 5.25$$

The maximum demand to allowable ratio for the pump supports from the reanalysis was 0.92. The governing element was a snubber. The allowable load in the snubber for the SSE load combination was taken as 1.2 times the Level A allowable. Snubbers are load rated by test and have a norminal ultimate strength of three (3) times the Level A rating; thus, the estimate of the median capacity, without considering conservatism in the piping response calculation, would be:

$$\text{Median} = [(5.25 / 0.92)\ (3.0 / 1.2)]\ [0.184g] = 2.62g \text{ pga.}$$

Studies documented in NUREG/CR-3996 and NUREG/CR-5073 quantify the conservatism in piping response when the envelope response spectrum method is used compared to using multiple support time history analysis. Examining the results of these two references, we can derive a lower bound factor on the conservatism in response. From NUREG/CR-3996, the Coefficients of Variation (COV) are reported for comparisons of the Standard Review Plan method and Best Estimate method of piping analysis. For logarithmic standard deviations less than about 0.5, the COV and logarithmic standard deviations are about equal. For the three piping models evaluated in the references, the COVs for support forces ranged from 0.20 to 0.61. For the same models, NUREG/CR-5073 reports the non-exceedance probabilities for comparisons of envelope response spectrum analysis (ERSA) using CC N-411 damping and multiple support time history analysis (MSTHA) using median damping. The minimum non-exceedance probability is stated to be greater than 92 percentile. This corresponds to about 1.4 standard deviations. Using a lognormal model and conservatively using the minimum COV of 0.2 for support forces, the median ratio of response between ERSA and MSTHA using median damping is:

$$\text{Ratio} = \exp\ (1.4)\ (0.2) = 1.32$$

A conservative estimate of median capacity for the pump supports can then be derived to be:

$$\text{Median} = (2.62g)\ (1.32) = 3.46g \text{ pga.}$$

The above estimate of median capacity for the recirculation pump supports exceeds the target median capacity screening level of 1.0g. Obviously, the HCLPF capacity is also well above the screening level of 0.3g. A minimum estimate of the HCLPF could be calculated by multiplying the conservatism in the input motion (minimum value) by the percent below allowable for the most highly loaded component.

$$\text{Minimum HCLPF} = (3 / 0.92)\ (0.184g) = 0.6\ g$$

The above discussion documents the logic used in the initial screening of the primary system and provides additional reinforcement using the results of a more recent licensing basis reanalysis of the OCNGS recirculation loops, including the supports for the recirculation pumps.

Since the recirculation pumps were initially screened, a screening threshold fragility for the recirculation pump supports has been incorporated into the Oyster Creek seismic model. The failure of the recirculation pump support is modeled as a large below core loss of coolant accident inside the drywell. A summary of the sensitivity study is provided below.

The table below provides the split fraction values and generic fragility used to determine the seismic failure probability of the recirculation pump supports.

Table 3 – Recirculation Pump Support Failure Split Fraction Values and Fragilities

Top Event	Split Fraction	Probability	Split Fraction Description		Component Fragilities
RX	RX1	1.63E-05	Recirc Pump Support Failure during SEIS1		FRAG00
	RX2	1.64E-02	Recirc Pump Support Failure during SEIS2		
	RX3	1.05E-01	Recirc Pump Support Failure during SEIS3		
	RX4	2.46E-01	Recirc Pump Support Failure during SEIS4		
Fragility	Fragility Description		A(g)	B(r)	B(u)
FRAG00	Generic Fragility		1.00g	0.40	0.32

The failure of the recirculation pump supports is modeled as a large below core loss of coolant accident. For successful mitigation of this event, the core spray system and the containment spray system are required for mitigation. The resulting seismic core damage frequency increases from 3.6E-06 per year to 3.7E-06 per year. This increase corresponds to approximately 1E-07 per year increase or a 3% increase in the seismic core damage frequency.

This small increase is expected due to low probability of seismic failure of the recirculation pump supports. That is, the convolved hazard curve and the generic fragility, produces an annual frequency of seismically induced recirculation pump LOCAs of 1.1E-06. A small potential for mitigation of this event (e.g., the combined probability of the core spray and containment spray systems) results in a small increase in seismic core damage frequency. See Attachment A.5 for additional details.

(3) *The RAI response to the original seismic question No. 4, regarding the basis for the screening threshold fragility, states that the threshold fragility, when combined with the mean seismic hazard curve, results in an annual probability of failure of 8.8E-6 per year. The threshold fragility has a median ground acceleration capacity of 1.0g and a high-confidence-low-probability of failure (HCLPF) capacity of 0.3g. Our interpretation of the application of this threshold for screening fragilities is that if a component had an estimated median ground acceleration capacity of greater than 1.0g and a HCLPF capacity of at least 0.3g, then the component was omitted from the seismic logic model, unless all components in a system were screened out, in which case the threshold fragility curve was used for a surrogate component to represent the system. Moreover, from Table 3-8 of the original submittal, it seems that the only case where such a surrogate component was used was for the fire protection system.*

As stated in the GPU IPEEE submittal, the total seismic CDF is 3.6E-6 per year. Since components were not included in the seismic logic model if their median ground acceleration capacity and HCLPF capacity exceeded the corresponding capacities of the threshold fragility, and the threshold fragility leads to an annual seismic failure frequency of 8.8E-6 per year, it appears that an appreciable underestimation of the seismic core damage frequency can occur. If the combination of the seismic event, and the failure of a screened out component (or two completely correlated similar components) leads to core damage, then such a sequence would contribute about 9E-6 per year to the core damage frequency. This exceeds the stated estimate of 3.6E-6 per year for the seismic CDF. With the low screening threshold fragility value used, the seismic PRA analysis may fail to identify important seismic sequences, and important insights into seismic system response may be overlooked. In light of the above, please:

The convolved hazard curve and threshold fragility were mistakenly reported, in both the IPEEE Submittal report and the subsequent request for additional information as resulting in a combined seismic failure frequency of 8.8E-6 per year. The 8.8E-6 per year corresponds to a seismic fragility of 0.68g with a high confidence low probability (HCLPF) of failure of 0.32g.

The correct frequency for the convolution of the hazard curve and the threshold fragility of 1.0g and a HCLPF of 0.3g is approximately 1.1E-6 per year.

Very few seismic PRAs have extensively incorporated screening threshold fragilities. Traditionally the PRAs which have extensively modeled generic or screening threshold fragilities are those which have seismically modeled only a few components. The modeling of a screening threshold fragility for all systems or components is not standard practice. If specific fragilities are developed for all single or multiple completely correlated components then the seismic core damage frequency will not change significantly when a screening threshold fragility is incorporated.

The Oyster Creek model contains 23 plant specific fragilities and the seismic analysis evaluated many more. This is commensurate with the level of detail in other seismic IPEEEs and PRAs. There are many issues associated with the extensive incorporation of the threshold fragility including fidelity and accuracy of the results. Therefore, this evaluation is presented as a sensitivity case and not as a final change to the seismic model.

It is felt that the current seismic IPEEE in conjunction with the attached sensitivity study, at a minimum, satisfies the intent of the Generic Letter 88-20, to:

(1) Develop an understanding of severe accident behavior
(2) Develop an understanding of the most likely severe accident sequences that could occur at Oyster Creek and
(3) Gain a more quantitative understanding of the overall probabilities of core damage.

In addition, it is also felt that the objective of Supplement 4 of Generic Letter 88-20 has been met. This supplement requests each licensee to "...perform an Individual Plant Examination of External Events to identify vulnerabilities, if any, to severe accidents and report the results together with any Licensee determined improvements and corrective actions to the Commission".

(a) *Increase the threshold for component screening, such that the annual failure probability of an individual component screened out with this threshold would represent only a small fraction of the total seismic CDF. (The screening threshold should be established such that the convolution of the threshold mean fragility curve with the mean site hazard curve leads to an annual component failure probability much less than the currently calculated mean seismic core damage frequency of 4E-6 per year. Then the insights as to dominant seismic sequences would not be affected by the screened out components.)*

It is not practical to reduce the screening threshold fragility to a level that "...the annual failure probability of an individual component screened out with this threshold would represent only a small fraction of the total seismic CDF." Using a screening threshold of approximately 1.5g results in a combined failure frequency of approximately 3E-7 per year which still represents a large fraction (almost 10%) of the total seismic CDF.

It is also not practical or reasonable to reproduce the current analysis with a new screening threshold value for the following reasons:

- Reproducing the current analysis with a new screening threshold value would require a re-evaluation of all components in the seismic analysis. The re-evaluation would require that walkdowns and additional fragility calculations be performed with little or no

additional insights. That is, low seismic margin components have been adequately identified using the existing fragility analyses.

- By inspection, there are no single components, excluding structures, or two or more completely correlated components whose failure results in core damage. The seismic evaluation centers on those components and structures which have the potential to significantly impact the core damage frequency. These include single and multiple completely correlated components and structures.

- Given the correct threshold fragility failure probability, even a single component or two or more completely correlated components, would only result in a maximum core damage frequency contribution of 1.1E-6 per year. Since all single components or multiple correlated components are already evaluated, then the expected core damage frequency increase should be substantially lower. In any case, a seismic CDF of 1.1E-6 could theoretically be considered a vulnerability. However, given the uncertainty of a seismic analysis and screening nature of a threshold fragility, it is likely that the issue would not warrant significant expenditure of resources.

Given that it is not practical to re-evaluate the analysis with a lower screening threshold, it is proposed that the consistent application of the current threshold value could be used to improve the analysis. That is, a consistent application of the threshold value could provide a reasonable method to determine area(s) of the seismic study which could benefit from additional fragility analysis or a lower threshold value. It is therefore proposed that the existing threshold value be applied to the systems that do not currently contain a detailed fragility as a sensitivity case to determine the areas of the seismic analysis which could benefit from additional fragility analyses or application of the threshold fragility.

The changes to the seismic IPEEE model and the new seismic results are presented in the following paragraphs and in Attachment A.6.

Several other changes to the Oyster Creek seismic model have been identified which could improve the overall study. These additional changes include:

- Extrapolation of the existing hazard curve from 0.82g to 1.5g as recommended in NUREG-1407.

- Correction of the fire protection system seismic split fraction values.

- The addition of the turbine building and reactor building fragilities. These detailed fragilities were not included in the original IPEEE submittal based on the fact that the fragility of the structure was the onset of deformation and permanent displacement, not structural failure. It was originally assumed that equipment within the structure would survive this mode of structural failure with a high likelihood of success. The impact of the failure of either the reactor building or the turbine building is modeled as core damage.

- A revised human action value for makeup to the isolation condenser to reflect the potential that flow diversion of the fire protection system could occur when flowpaths not associated with the makeup to the condenser fail. In these cases, manual action to isolate the flow diversion of fire protection would be required before isolation condenser makeup could be successful.

- Incorporation of the failure of recirculation pumps supports. Failure of the recirculation pump support is modeled as a large below core loss of coolant accident. This event utilizes the generic fragility since the recirculation pump supports screened.

The results of the re-evaluation of the seismic model changes above are presented in Attachment A.7. In this attachment, the changes are made to the base seismic IPEEE model and the resulting core damage frequency is reported on an individual change basis. This is to illustrate the change in seismic core damage frequency as well as the seismic sequences due to each individual change. In addition, all the model changes are integrated into the final Oyster Creek seismic model and the results presented.

b. *Evaluate the effect of this threshold increase on the existing seismic sequences.*

The following paragraphs detail the changes to the seismic IPEEE to evaluate the effect of the incorporation of the screening threshold fragility. The results of the incorporation of the threshold fragility are presented in the response to c. below.

Top events are generated for each of independent hardware top events in which all hardware was screened during the fragility evaluation. Each of the new seismic top events is assigned the generic fragility. The model is adjusted to incorporate the seismic top events. The top events contained in the table below are those seismic events which are added to the seismic IPEEE.

Table 4 – Top Events Added Using the Generic Fragility

Top Event	Split Fraction Description
DY	Seismic Failure of 125 VDC
EX	4160 VAC Bus 1D
XY	Seismic Failure of Long Term 125 VDC C
PX	Seismic Failure of Logic Actuation
RY	Seismic Failure of Reactor Scram System
BY	Seismic Failure of the TBVs Close
MX	Seismic Failure of the MSIVs to Close
CY	Seismic Failure of Condensate Transfer
VX	Seismic Failure of EMRVs to Open
SY	Seismic Failure of Safety Valves to Open
VY	Seismic Failure of EMRVs to Reclose
SZ	Seismic Failure of Safety Valve to Reclose
CU	Seismic Failure of the CRD System
OY	Seismic Failure of the Containment Vent

The split fractions for the seismic top events in Table 4 are quantified using the generic fragility. The split fraction values for a typical split fraction are represented as "XXY" in Table 5 below.

Table 5 – Generic Split Fraction Values

Top Event	Split Fraction	Probability	Split Fraction Description	Component Fragilities
XX	XX1	1.63E-05	Generic Split Fraction Failure during SEIS1	FRAG00
	XX2	1.64E-02	Generic Split Fraction Failure during SEIS2	
	XX3	1.05E-01	Generic Split Fraction Failure during SEIS3	
	XX4	2.46E-01	Generic Split Fraction Failure during SEIS4	

Fragility	Fragility Description	A(g)	B(r)	B(u)
FRAG00	Generic Fragility	1.00g	0.40	0.32

In the linked event tree methodology, the addition of top events, results in a decrease in the individual sequence frequency due to the additional success terms added to the sequence. At the same time, "new" sequences are added, at much lower frequencies, which contain the failure term. The term used most often to describe this phenomenon is the "fracture" of a sequence.

If both the successful and failed terms are contained in the database (i.e., a sufficiently low truncation limit is used) the total core damage frequency increases. The reported core damage frequency can actually decrease if the failure branch of the fractured sequence falls below the truncation frequency. The higher the failure probability of the added events the larger the reduction in individual sequence frequency. This is an important factor in the seismic model since at the higher accelerations, the failure probabilities are very high and significantly impact the dominant sequences.

The effect on the sequence database of the incorporation of additional top events has been the fracture of the existing sequences. The result is lowering of the dominant individual sequence frequencies with the same sequences with the failure term(s) appearing much lower in the database.

c. *Determine whether new seismic sequences could be established as a result of the threshold increase.*

No "new" seismic sequences, in the classic definition of the term, are added to the sequence database. This is a result of the fact that no new failure modes or equipment are being added to the model. That is, the addition of seismic top events (based on the addition of a screening threshold fragility) only increases the probability of previously defined sequences. See Table 15 for a comparison of the seismic IPEEE and Threshold Fragility top 25 seismic sequences.

d. *Update the Seismic CDF value.*

The seismic core damage frequency for the evaluation of the IPEEE seismic model with generic fragilities is approximately 4.4E-06 per year. This corresponds to an increase of 8E-07 per year (or about 22%). This increase is less than 1E-06 per year which is below the generally accepted criteria for the determination of risk significance. The relatively small increase in the total seismic core damage frequency, when compared with the number of additional top events added, is expected since the lowest seismic margin components have already been determined and included in the seismic model.

e. *Report the results of your re-analysis.*

The dominant contributors in the sequence database remain relatively unchanged. However, the number of sequences increases dramatically. See Attachment A.6, "Screening Threshold Fragility Incorporation" for a complete description of the model changes and the results of the evaluation.

Table 14 compares the number of sequences for the seismic IPEEE and the Threshold Fragility sensitivity case. An additional 32614 sequences contribute approximately 8E-7 to the seismic core damage frequency. This fact in conjunction with the relatively minor changes to the dominant sequence list ensures that the original IPEEE seismic model is capable of determining vulnerabilities due to seismically initiating events.

REFERENCES

1. GPU Nuclear Corporation, "Oyster Creek Individual Plant Examination for External Events", December 1995.

2. United States Nuclear Regulatory Commission, "Request for Additional Information Regarding Oyster Creek Nuclear Generating IPEEE Submittal (TAC No. M83652)", December 10, 1997.

3. GPU Nuclear Corporation, "Oyster Creek Nuclear Generating Station (OCNGS), Docket No. 50-219, Facility Operating License No. DPR-16, Individual Plant Examination for External Events – Response to Request for Additional Information", 1940-98-20188, May 21,1998.

4. United States Nuclear Regulatory Commission, "Request for Additional Information (RAI) – Individual Plant Examination of External Events (IPEEE) Oyster Creek Nuclear Generating Station (TAC No. M83652)", March 3, 2000.

5. EQE International, "Development of Equipment Seismic Fragilities for OCNGS IPEEE", Job Number 42113.08, July 1996.

6. Geomatrix Consultants, "Assessments of Potential for Liquefaction and Permanent Ground Displacements at Designated Facilities, Oyster Creek Nuclear Generating Station", Project No. 2529, December 1994.

7. EQE International, "Probabilistic Seismic Response Analysis of the Oyster Creek Nuclear Generating Station in Support of the IPEEE Program", Report Number 50124-R-003, July 1994.

8. EQE International, "Seismic Fragilities of Civil Structures at Oyster Creek Nuclear Generating Station", Report Number 990-2316, October 1994.

9. EQE International Memorandum, "Oyster Creek Fragilities – Updated Fragilities", April 27, 1995.

10. EQE International, "Seismic Fire Interaction Walkdown Report", Report No. P;\42113-08\-OYCEQFIR.DOC, Final Report, August 8, 1995.

11. EQE International, "Transmittal of Results of the Evaluation of the Masonry Walls at Oyster Creek Nuclear Generating Station for the Seismic IPEEE", June 5, 1995.

12. Electric Power Research Institute (EPRI), "Probabilistic Seismic Hazard Evaluation at Nuclear Power Plant Sites in the Central and Eastern United States: Resolution of the Charleston Issue," NP-6395-D, 1989.

13. EQE International, "OCNPS IPEEE Recirc. Pump RAI Question", NE-00-049, 240031-O-027, June 7, 2000.

ATTACHMENT A:

SEISMIC MODEL CHANGES AND RESULTS

As a result of comments on the Oyster Creek IPEEE seismic risk model several changes were made to the base model. In order to understand the impact of individual changes the model was revised in steps. Each model change is described in the paragraphs below along with the resulting change in the seismic core damage frequency and the seismic core damage sequences. The following table summarizes the changes to seismic model and the seismic core damage frequency.

Table 6 – Summary of Model Changes and Core Damage Frequency Results

Case No.	Model Name	Change Description	Seismic CDF	Percent Change
Base	OCSEIS	Seismic IPEEE Model	3.6E-06	n/a
1	RAI-01	Fire Protection Split Fraction Correction	1.3E-06	-164%
2	RAI-02	Seismic Hazard Curve Extrapolation to 1.5g	4.1E-06	+14%
3	RAI-03	Addition of Reactor and Turbine Building Failure	5.4E-06	+50%
4	RAI-04	Revised Isolation Condenser Makeup HEP	3.8E-06	+6%
5	RAI-05	Seismic Recirc Pump Loss of Coolant Accident	3.7E-06	+3%
6	RAI-06	Generic Fragility Incorporation	4.4E-06	+22%
7	SEISMIC1	Revised Seismic Model	4.7E-06	+33%

A.1 FIRE PROTECTION SPLIT FRACTION CORRECTION

An error was discovered in the seismic model. This error was associated with the seismic failure of the fire protection system. In the text of the Oyster Creek Seismic IPEEE, the correct values of the seismic failure of the fire protection system were displayed (i.e., split fractions FX1, FX2, FX3 and FX4), however, the incorrect values were used in the quantification of the seismic model. The values used in the quantification of the seismic model were significantly higher than those reported in the text of the seismic IPEEE. The higher values were the result of earlier screening fragilities, input for testing of the model, that were not removed before the final quantification of the seismic IPEEE.

Table 7 – Fire Protection Split Fraction Correction

Split Fraction	Split Fraction Description	Quantification Value	Reported Value
FX1	Seismic Failure of Fire Protection during SEIS1	2.61E-04	1.63E-05
FX2	Seismic Failure of Fire Protection during SEIS2	2.58E-01	1.64E-02
FX3	Seismic Failure of Fire Protection during SEIS3	7.94E-01	1.05E-01
FX4	Seismic Failure of Fire Protection during SEIS4	9.59E-01	2.46E-01

With seismic failure of the fire protection system overstated in the quantification of the seismic model, the resulting core damage frequency was also overstated. Given the importance of the fire protection system, the core damage frequency overstatement due to the erroneous fire protection failure values is significant. This sensitivity case replaces the erroneous fire protection values with the correct values reported in the original seismic IPEEE. The seismic core damage frequency decreases by 164% to 1.3E-06 per year from a reported value of 3.6E-06 per year.

A.2 HAZARD CURVE EXTRAPOLATION

It is recommended in NUREG-1407 that the existing hazard curves be extrapolated from the current endpoint to approximately 1.5g. This change results in additional initiating event frequency in the SEIS4 initiating event. The table below provides the changes in the SEIS4 initiating event frequency.

Table 8 – Hazard Curve Extrapolation Initiating Event Frequency Changes					
Seismic IPEEE Values			**Hazard Curve Extrapolation**		
Initiating Event	Ground Acceleration Range	Initiating Event Frequency	Initiating Event	Ground Acceleration Range	Initiating Event Frequency
SEIS1 (0.13g) (~1 x SSE)	0.007g – 0.26g	7.44E-3	SEIS1 (0.13g) (~1 x SSE)	0.007g – 0.26g	7.44E-3
SEIS2 (0.36g) (~2 x SSE)	0.26g – 0.46g	2.73E-5	SEIS2 (0.36g) (~2 x SSE)	0.26g – 0.46g	2.73E-5
SEIS3 (0.54g) (~3 x SSE)	0.46g – 0.62g	2.72E-6	SEIS3 (0.54g) (~3 x SSE)	0.46g – 0.62g	2.72E-6
SEIS4 (0.72g) (~4 x SSE)	**0.62g – 0.82g**	**9.83E-7**	**SEIS4 (1.1g) (> 4x SSE)**	**0.62g – 1.5g**	**1.40E-6**

The extrapolation of the hazard curve has two impacts on the Oyster Creek IPEEE seismic model. The first impact is apparent from the table above and is the change in the initiating event frequency of initiator SEIS4. The initiating event frequency for SEIS4 changes from 9.83E-06 to 1.40E-06 per year. The second impact is the change in the seismic split fractions associated with SEIS4 initiating event. These split fractions are those with a "4" in the final character of the split fraction designator (i.e., ZZ4). The table below compares the changes in the split fractions affected.

Table 9 – Hazard Curve Extrapolation Split Fraction Changes

Split Fraction Designator	Split Fraction Description	Seismic IPEEE Value	Hazard Curve Extrapolation Value
BX4	Standby Liquid Control during SEIS4	2.82E-03	1.05E-02
CX4	Containment Spray/ESW during SEIS4	6.48E-01	7.98E-01
CZ4	Core Spray during SEIS4	5.88E-01	7.00E-01
DU4	DG Building Liquefaction during SEIS4	9.59E-01	9.72E-01
DV4	DG Building after Liquefaction during SEIS4	8.77E-01	9.31E-01
EU4	Startup Transformer during SEIS4	7.74E-01	8.33E-01
EW4	4160 VAC Bus 1C during SEIS4	7.35E-01	7.91E-01
FX4*	Fire Protection during SEIS4	2.46E-01	3.39E-01
IY4	Isolation Condenser during SEIS4	2.66E-01	3.57E-01
LX4	Offsite Power Recovery during SEIS4	9.12E-01	9.47E-01
OX4	Offsite Power during SEIS4	9.34E-01	9.50E-01
SX4	Condensate Storage Tank during SEIS4	9.35E-01	9.52E-01
XX4	Long Term 125VDC Bus B during SEIS4	9.43E-01	9.57E-01
DW4	Diesel Generators during SEIS4	6.45E-01	7.85E-01
DX4	Short Term 125VDC Power during SEIS4	4.19E-01	5.21E-01

* Fire protection system seismic failure values are not changed in this model evaluation. The fire protection values are taken from Table 2, Quantified Value Column.

The core damage frequency increases from 3.6E-06 per year to 4.1E-06 per year. This change corresponds to approximately a 14% increase from the base seismic IPEEE core damage frequency. The minimal increase in seismic core damage frequency is the result of two factors. The first is the small change in the initiating event frequency for the SEIS4 from 9.83E-07 to 1.40E-06 per year. The second factor is that the conditional core damage frequency in this range of seismic motion is already almost unity (0.96).

A.3 ADDITION OF THE TURBINE AND REACTOR BUILDING SEISMIC FAILURE

The seismically induced failure of the turbine and reactor buildings are not modeled in the seismic IPEEE. These detailed fragilities were not included in the original IPEEE submittal based on the fact that the fragility of the structure corresponded to the onset of deformation and permanent displacement, not structural failure. It was originally assumed that equipment within these structures would survive this mode of failure with a high likelihood of success.

This model change assumes that the buildings fail at the calculated building fragility. This assumption is made since the response of equipment in the buildings and the consequences of permanent building displacements are not known. Building failure is assumed to result in core damage due to the large number of impacted systems.

Table 10 – Reactor and Turbine Building Split Fractions and Fragilities

Top Event	Split Fraction	Probability	Split Fraction Description	Component Fragilities
TZ	TZ1	2.16E-05	Turbine Building Failure during SEIS1	FRAG38
	TZ2	3.38E-02	Turbine Building Failure during SEIS2	FRAG39
	TZ3	2.57E-01	Turbine Building Failure during SEIS3	FRAG40
	TZ4	5.87E-01	Turbine Building Failure during SEIS4	
RZ	RZ1	6.22E-06	Reactor Building Failure during SEIS1	FRAG36
	RZ2	1.01E-02	Reactor Building Failure during SEIS2	FRAG37
	RZ3	8.46E-02	Reactor Building Failure during SEIS3	
	RZ4	2.26E-01	Reactor Building Failure during SEIS4	

Fragility	Fragility Description	A(g)	B(r)	B(u)
FRAG36	Reactor Building – Wall	2.96g	0.29	0.25
FRAG37	Reactor Building – Column	1.00g	0.38	0.27
FRAG38	Turbine Building – Diaphragm Shear	0.88g	0.20	0.26
FRAG39	Turbine Building – Column Anchor Bolt	0.74g	0.35	0.24
FRAG40	Turbine Building – Pedestal	2.17g	0.24	0.28

The core damage frequency increases from 3.6E-06 per year to 5.4E-06 per year. This corresponds to approximately a 50% increase. This substantial increase is expected due to the guaranteed core damage as a result of the failure of either the turbine or reactor buildings.

Within the top 16 sequences are the eight (8) sequences associated with the combinations of the initiating events and the failure of either the turbine or reactor buildings.

A.4 REVISED HUMAN ERROR PROBABILITY FOR FIRE PROTECTION MAKEUP TO THE ISOLATION CONDENSER

This model change incorporates a revised human error probability for makeup to the isolation condenser. This revised probability is designed to reflect the potential for a flow diversion of the fire protection system when flowpaths not associated with the makeup to the condenser fail. That is, sprinkler heads could actuate as a result of the seismic event and result in a significant diversion of fire water. In these cases, manual action to isolate the flow diversion of fire protection would be required before isolation condenser makeup could be successful. This failure mode of isolation condenser makeup was not modeled in the seismic IPEEE. However, it should be noted that it is possible to supply core spray from the torus to the shell side of the isolation condenser. This mode of isolation condenser makeup is not modeled since loss of offsite

power and station blackout dominates the cases where isolation condenser makeup via fire protection is required.

The current human action value for the successful makeup to the isolation condenser is 4E-4. This failure probability is essentially equal to the human error rate or HEP. Adjusting this value to a screening human error probability of 0.1 is appropriate for this type of action. The guidance to provide fire protection water to the isolation condenser is clear. In addition, ample time is available before the onset of core damage and alarms are available to indicate that the fire protection system has actuated. Multiple locations are available to isolate the fire protection header or fire protection flow to the affected portion of the system.

The core damage frequency given these changes made to the seismic IPEEE model increases from 3.6E-06 per year to 3.8E-06 per year. This increase represents an increase of approximately 5.6%. This increase is expected given the importance of the makeup to the isolation in loss of offsite power and station blackout events.

A.5 SEISMIC RECIRCULATION PUMP LOSS OF COOLANT ACCIDENT

In the seismic IPEEE the recirculation pump was screened from consideration during seismic walkdowns. However, a screening fragility was not modeled in the seismic IPEEE. As such, a plant specific estimation of the capacity of the recirculation pump supports has been performed and incorporated into the Oyster Creek seismic model. The plant specific fragility supports the use of the generic fragility originally estimated for the recirculation pump supports. Therefore, the generic fragility is incorporated into the model as resulting in a recirculation pump loss of coolant accident.

The failure of the recirculation pump supports are modeled as a large below core loss of coolant accident inside the drywell. The core damage frequency increase as a result of the modeling of the recirculation pump loss of coolant accident is from 3.6E-06 per year to 3.7E-06 per year. This increase corresponds to approximately 1E-07 per year or a 3% increase in the seismic core damage frequency.

The basis for the small increase in core damage frequency is the small frequency of a seismic failure of the recirculation pump support (i.e., approximately 1.1E-06 per year). The failure of the recirculation pump support is modeled as a large below core loss of coolant accident which can be mitigated by a single train of core spray and containment spray systems.

Table 11 provides a simplified estimation of the impact of the incorporation of a recirculation pump support failure. This simplified estimate is used to verify the results of the full model evaluation.

The Table 12 below provides the split fraction values for the failure of the recirculation pump support, core spray system, and containment spray system.

Table 11 – Simplified Estimate of Recirculation Pump Support Failure Core Damage Frequency

Initiating Event Name	Initiating Event Frequency	Recirc Pump Support Failure Probability	Core Spray System Failure Probability	Containment Spray/ESW Failure Probability	Estimation of Core Damage Frequency Increase
SEIS1	7.44E-03	1.63E-05	1.11E-04	4.74E-05	6.38E-16
SEIS2	2.73E-05	1.64E-02	7.20E-02	4.43E-02	1.43E-09
SEIS3	2.72E-06	1.05E-01	3.26E-01	2.96E-01	2.76E-08
SEIS4	9.83E-07	2.46E-01	5.88E-01	6.48E-01	9.21E-08
Total Estimate Recirc Pump Support Failure CDF					**1.21E-07**

Table 12 – Recirculation Pump Support Failure Split Fraction Values and Fragilities

Top Event	Split Fraction	Probability	Split Fraction Description	Component Fragilities
RX	RX1	1.63E-05	Recirc Pump Support Failure during SEIS1	FRAG00
	RX2	1.64E-02	Recirc Pump Support Failure during SEIS2	
	RX3	1.05E-01	Recirc Pump Support Failure during SEIS3	
	RX4	2.46E-01	Recirc Pump Support Failure during SEIS4	
CX	CX1	4.74E-05	Containment Spray/ESW during SEIS1	FRAG15
	CX2	4.43E-02	Containment Spray/ESW during SEIS2	FRAG16
	CX3	2.96E-01	Containment Spray/ESW during SEIS3	FRAG33
	CX4	6.48E-01	Containment Spray/ESW during SEIS4	FRAG35
CZ	CZ1	1.11E-04	Core Spray during SEIS1	FRAG17
	CZ2	7.20E-02	Core Spray during SEIS2	FRAG32
	CZ3	3.26E-01	Core Spray during SEIS3	
	CZ4	5.88E-01	Core Spray during SEIS4	

Fragility	Fragility Description	A(g)	B(r)	B(u)
FRAG00	Generic Fragility	1.00g	0.40	0.32
FRAG15	Containment Spray Pump	1.48g	0.40	0.35
FRAG16	Emergency Service Water Pump	1.18g	0.40	0.35
FRAG33	Relay Panel 8A and 8B	0.88g	0.28	0.43
FRAG35	Intake Structure	0.82g	0.18	0.26
FRAG17	Core Spray Pump	0.82g	0.40	0.35
FRAG32	Relay Panel 18A and 18B	0.88g	0.28	0.43

A.6 SCREENING THRESHOLD FRAGILITY INCORPORATION

In the seismic IPEEE the screening threshold fragility was not consistently used in the evaluation of the seismic core damage frequency. The screening threshold fragility was not consistently applied in the seismic IPEEE since detailed fragilities were performed for all single or multiple correlated components and these were already included in the seismic model. Therefore, the impact of the inclusion of the screening threshold fragility would not produce any significant changes in the seismic core damage frequency or add any new significant seismic sequences.

Very few seismic PRAs have extensively incorporated a screening threshold fragility. Traditionally the PRAs which have extensively modeled screening threshold fragilities are those which have modeled only a few components. The Oyster Creek model contains 23 plant specific fragilities and the seismic analysis evaluated many more. This is commensurate with the level of detail and modeling in other seismic IPEEEs and PRAs. There are many issues associated with the extensive incorporation of a screening threshold fragility including fidelity and accuracy of the results. Therefore, the following evaluation is presented as a sensitivity case and not as a final change to seismic model.

In the linked event tree methodology, the addition of top events, results in a decrease in the individual sequence frequency due to the additional success terms added to the sequence. At the same time, "new" sequences are added, at much lower frequencies, which contain the failure term. The term used most often to describe this phenomenon is the "fracture" of a sequence.

If both the successful and failed terms are contained in the database (i.e., a sufficiently low truncation limit is used) the total core damage frequency increases. The reported core damage frequency can actually decrease if the failure branch of the fractured sequence falls below the truncation frequency. The higher the failure probability of the added events the larger the reduction in individual sequence frequency. This is an important factor in the seismic model since at the higher accelerations, the failure probabilities are very high, in some cases approaching unity, and can significantly impact the dominant sequences.

In order to reduce the number of sequences which are fractured as a result of the additional top events, seismic top events are incorporated into the independent failure split fractions, in specific cases. The specific cases where this treatment is applied is where the independent split fractions are not conditional on previous events and few independent split fractions are developed. For example, the failure of 125 VDC power Bus B is modeled in a single split fraction, DB1. This split fraction is not conditional on any other events. With the addition of the seismic failure, several new split fractions are added which are conditional on the seismic event. Rather than model a separate seismic top event, the independent top event and new split fractions which are now conditional on the seismic event are used to evaluate the combination of the seismic failure and the independent failure. This is illustrated below:

DX1	INIT=SEIS1	Seismic Failure of 125 VDC Power Bus B which is conditional on initiating event only.
DX2	INIT=SEIS2	
DX3	INIT=SEIS3	
DX4	INIT=SEIS4	
DB1	DX=S	Independent failure which is conditional on seismic failure only.

The new model changes the above two top events and five split fractions to a single top event with four split fractions.

DB1	INIT=SEIS1	These split fraction values are determined by manually adding the seismic failure rate which is conditional on the initiating event only with the independent failure rate which is not conditional.
DB2	INIT=SEIS2	
DB3	INIT=SEIS3	
DB4	INIT=SEIS4	

This modeling treatment is applied to the following top events: DB, DC, XB, XC, FP, OX and ST. The result is a substantial decrease in the number of top events and therefore the number of resulting sequences.

This evaluation incorporates the seismic generic fragility for hardware in the seismic model. Simplifications are made such as the correlation of components that are actually not seismically correlated. This simplification is extremely conservative but reduces the burden of sensitivity model development as well as the number of seismic sequences that result when the sensitivity case is quantified. Table 13, below, provides the seismic top events that are added to the seismic IPEEE model.

Table 13 – Top Event and Split Fractions Added Using the Generic Fragility

Top Event	Split Fraction	Probability	Split Fraction Description	Component Fragilities
DY (DC)	DY1	1.63E-05	Failure of 125 VDC C during SEIS1	FRAG00
	DY2	1.64E-02	Failure of 125 VDC C during SEIS2	
	DY3	1.05E-01	Failure of 125 VDC C during SEIS3	
	DY4	2.46E-01	Failure of 125 VDC C during SEIS4	
EX	EX1	1.63E-05	4160 VAC Bus 1D during SEIS1	FRAG00
	EX2	1.64E-02	4160 VAC Bus 1D during SEIS2	
	EX3	1.05E-01	4160 VAC Bus 1D during SEIS3	
	EX4	2.46E-01	4160 VAC Bus 1D during SEIS4	
XY (XC)	XY1	1.63E-05	Long Term 125 VDC C during SEIS1	FRAG00
	XY2	1.64E-02	Long Term 125 VDC C during SEIS2	
	XY3	1.05E-01	Long Term 125 VDC C during SEIS3	
	XY4	2.46E-01	Long Term 125 VDC C during SEIS4	
PX	PX1	1.63E-05	Logic Fails (auto actuation) during SEIS1	FRAG00
	PX2	1.64E-02	Logic Fails (auto actuation) during SEIS2	
	PX3	1.05E-01	Logic Fails (auto actuation) during SEIS3	
	PX4	2.46E-01	Logic Fails (auto actuation) during SEIS4	

Table 13 – Top Event and Split Fractions Added Using the Generic Fragility				
Top Event	Split Fraction	Probability	Split Fraction Description	Component Fragilities
RY	RY1	1.63E-05	Reactor Scram Failure during SEIS1	FRAG00
	RY2	1.64E-02	Reactor Scram Failure during SEIS2	
	RY3	1.05E-01	Reactor Scram Failure during SEIS3	
	RY4	2.46E-01	Reactor Scram Failure during SEIS4	
BY	BY1	1.63E-05	TBVs Close during SEIS1	FRAG00
	BY2	1.64E-02	TBVs Close during SEIS2	
	BY3	1.05E-01	TBVs Close during SEIS3	
	BY4	2.46E-01	TBVs Close during SEIS4	
MX	MX1	1.63E-05	MSIVs Close during SEIS1	FRAG00
	MX2	1.64E-02	MSIVs Close during SEIS2	
	MX3	1.05E-01	MSIVs Close during SEIS3	
	MX4	2.46E-01	MSIVs Close during SEIS4	
CY	CY1	1.63E-05	Condensate Transfer Failure during SEIS1	FRAG00
	CY2	1.64E-02	Condensate Transfer Failure during SEIS2	
	CY3	1.05E-01	Condensate Transfer Failure during SEIS3	
	CY4	2.46E-01	Condensate Transfer Failure during SEIS4	
VX	VX1	1.63E-05	EMRVs Fail to Open during SEIS1	FRAG00
	VX2	1.64E-02	EMRVs Fail to Open during SEIS2	
	VX3	1.05E-01	EMRVs Fail to Open during SEIS3	
	VX4	2.46E-01	EMRVs Fail to Open during SEIS4	
SY	SY1	1.63E-05	Safety Valves Fail to Open during SEIS1	FRAG00
	SY2	1.64E-02	Safety Valves Fail to Open during SEIS2	
	SY3	1.05E-01	Safety Valves Fail to Open during SEIS3	
	SY4	2.46E-01	Safety Valves Fail to Open during SEIS4	
VY	VY1	1.63E-05	EMRVs Fail to Reclose during SEIS1	FRAG00
	VY2	1.64E-02	EMRVs Fail to Reclose during SEIS2	
	VY3	1.05E-01	EMRVs Fail to Reclose during SEIS3	
	VY4	2.46E-01	EMRVs Fail to Reclose during SEIS4	
SZ	SZ1	1.63E-05	Safety Valve Fail to Reclose during SEIS1	FRAG00
	SZ2	1.64E-02	Safety Valve Fail to Reclose during SEIS2	
	SZ3	1.05E-01	Safety Valve Fail to Reclose during SEIS3	
	SZ4	2.46E-01	Safety Valve Fail to Reclose during SEIS4	
CU	CU1	1.63E-05	CRD Fails during SEIS1	FRAG00
	CU2	1.64E-02	CRD Fails during SEIS2	
	CU3	1.05E-01	CRD Fails during SEIS3	
	CU4	2.46E-01	CRD Fails during SEIS4	
OY	OY1	1.63E-05	Containment Vent Fails during SEIS1	FRAG00
	OY2	1.64E-02	Containment Vent Fails during SEIS2	
	OY3	1.05E-01	Containment Vent Fails during SEIS3	
	OY4	2.46E-01	Containment Vent Fails during SEIS4	

Fragility	Fragility Description	A(g)	B(r)	B(u)
FRAG00	Generic Fragility	1.00g	0.40	0.32

The core damage frequency following the incorporation of the screening threshold fragility is 4.4E-06 per year. This corresponds to an increase of 8E-07 per year (or about 22%). This is a relatively small increase in seismic core damage frequency when considering the number of seismic top events added to the model. The relatively small increase is expected since the lowest seismic margin components have already been determined and included in the seismic model.

Table 15 compares the top 25 sequences in the IPEEE with the top 25 sequences generated following the incorporation of the threshold fragility. The table provides the top 25 seismic IPEEE sequences listed on the left-hand side of the table with the corresponding sequence number and frequency from the threshold fragility sensitivity analysis. New sequences are indicated by a "new" in the leftmost column. Sequences that appear in the top 25 sequences in the threshold sensitivity analysis are listed following the top 25 sequences from the seismic IPEEE. In these cases, the leftmost column contains the seismic IPEEE sequence number and corresponding frequency.

As can be seen from Table 15, six "new" sequences appear in the top 25 sequences. Four of the six sequences are related to a seismic failure to scram following a SEIS2 initiating event. The remaining two sequences are related to EMRV failures to open or close following a SEIS3 initiating event. Although a significant re-ordering of the seismic sequences occur with SEIS3 and SEIS4 initiated sequences dropping dramatically very few new sequences are added. The significant drop in the SEIS3 and SEIS4 initiated sequences is due to the multiplication of the existing seismic sequences with very high success terms.

The dominant contributors in the sequence database remain relatively unchanged. However, the number of sequences increases dramatically. The table below compares the number of sequences for the seismic IPEEE and the Threshold Fragility sensitivity case.

Table 14 – Comparison of the Number of Accident Sequences Using the Same Cutoff

Initiating Event	Number of Accident Sequences		
	IPEEE Model	Threshold Model	Difference
SEIS1	213	268	55
SEIS2	2159	7684	5525
SEIS3	2111	17889	15778
SEIS4	801	12148	11347
TOTAL	5284	37898	32614

From the Table 14 above, it can be seen that the 32614 additional sequences contribute approximately 8E-7 to the seismic core damage frequency (i.e., sensitivity case CDF minus seismic IPEEE CSF). This fact in conjunction with the relatively minor changes in the dominant sequence list ensures that the original IPEEE seismic model is capable of determining vulnerabilities due to seismically initiating events.

Table 15 – Comparison of Sequences

No.	Initiating Event	Frequency	Percent	Failed Events	Split Fraction Descriptions	New No.	New Frequency
1	SEIS3	2.76E-07	7.61%	OX3	Seismic Failure of Offsite Power	4	6.54E-08
				DU3	Diesel Generator Building Liquefaction		
				DV3	Diesel Generator Building Failure		
				FX3	Seismic Fire Protection Failure		
				SX3	Condensate Storage Tank Failure		
				LX3	Seismic Offsite Power Recovery Failure		
2	SEIS4	2.42E-07	6.66%	OX4	Seismic Failure of Offsite Power	50	6.16E-09
				DU4	Diesel Generator Building Liquefaction		
				DV4	Diesel Generator Building Failure		
				FX4	Seismic Fire Protection Failure		
				SX4	Condensate Storage Tank Failure		
				LX4	Seismic Offsite Power Recovery Failure		

Table 15 – Comparison of Sequences							
No.	Initiating Event	Frequency	Percent	Failed Events	Split Fraction Descriptions	New No.	New Frequency
3	SEIS4	1.71E-07	4.71%	DX4	Short Term 125 VDC B (Wall No 53)	89	4.36E-09
				OX4	Seismic Failure of Offsite Power		
				DU4	Diesel Generator Building Liquefaction		
				DV4	Diesel Generator Building Failure		
				FX4	Seismic Fire Protection Failure		
				SX4	Condensate Storage Tank Failure		
				LX4	Seismic Offsite Power Recovery Failure		
4	SEIS2	1.22E-07	3.37%	EW2	4160 VAC Bus 1C	1	9.88E-08
				XX2	Long Term 125 VDC Bus B		
				FX2	Seismic Fire Protection Failure		
				SX2	Condensate Storage Tank Failure		
5	SEIS2	1.08E-07	2.98%	OX2	Seismic Failure of Offsite Power	2	8.76E-08
				EW2	4160 VAC Bus 1C		
				XX2	Long Term 125 VDC Bus B		
				FX2	Seismic Fire Protection Failure		
				SX2	Condensate Storage Tank Failure		
6	SEIS4	9.61E-08	2.65%	OX4	Seismic Failure of Offsite Power	134	2.45E-09
				DU4	Diesel Generator Building Liquefaction		
				DV4	Diesel Generator Building Failure		
				FX4	Seismic Fire Protection Failure		
				SX4	Condensate Storage Tank Failure		
				IY4	Isolation Condenser		
7	SEIS1	7.43E-08	2.05%	VR2	EMRVs Fail to Reclose	3	7.47E-08
				CS1	Core Sprays Failure		
8	SEIS3	7.39E-08	2.04%	OX3	Seismic Failure of Offsite Power	8	1.75E-08
				DU3	Diesel Generator Building Liquefaction		
				DW3	Diesel Generators		
				FX3	Seismic Fire Protection Failure		
				SX3	Condensate Storage Tank Failure		
				LX3	Seismic Offsite Power Recovery Failure		
9	SEIS4	6.81E-08	1.88%	DX4	Short Term 125 VDC B (Wall No 53)	206	1.74E-09
				OX4	Seismic Failure of Offsite Power		
				DU4	Diesel Generator Building Liquefaction		
				DV4	Diesel Generator Building Failure		
				FX4	Seismic Fire Protection Failure		
				SX4	Condensate Storage Tank Failure		
				IY4	Isolation Condenser		
10	SEIS3	6.77E-08	1.87%	DX3	Short Term 125 VDC B (Wall No 53)	10	1.61E-08
				OX3	Seismic Failure of Offsite Power		
				DU3	Diesel Generator Building Liquefaction		
				DV3	Diesel Generator Building Failure		
				FX3	Seismic Fire Protection Failure		
				SX3	Condensate Storage Tank Failure		
				LX3	Seismic Offsite Power Recovery Failure		
11	SEIS3	5.48E-08	1.51%	OX3	Seismic Failure of Offsite Power	15	1.30E-08
				DU3	Diesel Generator Building Liquefaction		
				DV3	Diesel Generator Building Failure		
				FX3	Seismic Fire Protection Failure		
				LX3	Seismic Offsite Power Recovery Failure		

Table 15 – Comparison of Sequences							
No.	Initiating Event	Frequency	Percent	Failed Events	Split Fraction Descriptions	New No.	New Frequency
12	SEIS3	5.47E-08	1.51%	OX3	Seismic Failure of Offsite Power	16	1.29E-08
				DU3	Diesel Generator Building Liquefaction		
				DV3	Diesel Generator Building Failure		
				FX3	Seismic Fire Protection Failure		
				SX3	Condensate Storage Tank Failure		
				IY3	Isolation Condenser		
13	SEIS3	5.06E-08	1.40%	OX3	Seismic Failure of Offsite Power	19	1.10E-08
				DU3	Diesel Generator Building Liquefaction		
				EW3	4160 VAC Bus 1C		
				XX3	Long Term 125 VDC Bus B		
				FX3	Seismic Fire Protection Failure		
				SX3	Condensate Storage Tank Failure		
14	SEIS3	4.09E-08	1.13%	OX3	Seismic Failure of Offsite Power	21	9.68E-09
				DW3	Diesel Generators		
				FX3	Seismic Fire Protection Failure		
				SX3	Condensate Storage Tank Failure		
				LX3	Seismic Offsite Power Recovery Failure		
15	SEIS2	3.48E-08	0.96%	OX2	Seismic Failure of Offsite Power	5	2.81E-08
				DU2	Diesel Generator Building Liquefaction		
				EW2	4160 VAC Bus 1C		
				XX2	Long Term 125 VDC Bus B		
				FX2	Seismic Fire Protection Failure		
				SX2	Condensate Storage Tank Failure		
16	SEIS3	2.92E-08	0.80%	EU3	Startup Transformers	48	6.32E-09
				DU3	Diesel Generator Building Liquefaction		
				DV3	Diesel Generator Building Failure		
				FX3	Seismic Fire Protection Failure		
				SX3	Condensate Storage Tank Failure		
17	SEIS3	2.80E-08	0.77%	OX3	Seismic Failure of Offsite Power	52	6.09E-09
				EW3	4160 VAC Bus 1C		
				XX3	Long Term 125 VDC Bus B		
				FX3	Seismic Fire Protection Failure		
				SX3	Condensate Storage Tank Failure		
18	SEIS3	2.45E-08	0.68%	OX3	Seismic Failure of Offsite Power	64	5.33E-09
				DU3	Diesel Generator Building Liquefaction		
				EW3	4160 VAC Bus 1C		
				XX3	Long Term 125 VDC Bus B		
				FX3	Seismic Fire Protection Failure		
				SX3	Condensate Storage Tank Failure		
				CZ3	Core Spray Failure		
19	SEIS2	2.39E-08	0.66%	OX2	Seismic Failure of Offsite Power	6	1.93E-08
				EC2	Independent Failure of 4160 Bus 1C		
				XX2	Long Term 125 VDC Bus B		
				FX2	Seismic Fire Protection Failure		
				SX2	Condensate Storage Tank Failure		
20	SEIS3	2.35E-08	0.65%	OX3	Seismic Failure of Offsite Power	39	7.11E-09
				DU3	Diesel Generator Building Liquefaction		
				DV3	Diesel Generator Building Failure		
				FX3	Seismic Fire Protection Failure		
				SX3	Condensate Storage Tank Failure		
				CX3	Containment Spray/ESW		

Table 15 – Comparison of Sequences							
No.	Initiating Event	Frequency	Percent	Failed Events	Split Fraction Descriptions	New No.	New Frequency
21	SEIS3	2.21E-08	0.61%	DX3	Short Term 125 VDC B (Wall No 53)	67	5.25E-09
				OX3	Seismic Failure of Offsite Power		
				DU3	Diesel Generator Building Liquefaction		
				EW3	4160 VAC Bus 1C		
				FX3	Seismic Fire Protection Failure		
				SX3	Condensate Storage Tank Failure		
				LX3	Seismic Offsite Power Recovery Failure		
22	SEIS4	2.19E-08	0.60%	OX4	Seismic Failure of Offsite Power	170	2.01E-09
				DU4	Diesel Generator Building Liquefaction		
				DV4	Diesel Generator Building Failure		
				FX4	Seismic Fire Protection Failure		
				SX4	Condensate Storage Tank Failure		
				LX4	Seismic Offsite Power Recovery Failure		
23	SEIS3	2.11E-08	0.58%	EW3	4160 VAC Bus 1C	82	4.58E-09
				XX3	Long Term 125 VDC Bus B		
				FX3	Seismic Fire Protection Failure		
				SX3	Condensate Storage Tank Failure		
24	SEIS2	2.03E-08	0.56%	XX2	Long Term 125 VDC Bus B	7	1.75E-08
				FX2	Seismic Fire Protection Failure		
				SX2	Condensate Storage Tank Failure		
				CX2	Containment Spray/ESW		
25	SEIS3	1.97E-08	0.54%	OX3	Seismic Failure of Offsite Power	80	4.66E-09
				DU3	Diesel Generator Building Liquefaction		
				DV3	Diesel Generator Building Failure		
				FX3	Seismic Fire Protection Failure		
				SX3	Condensate Storage Tank Failure		
				LY2	Independent Failure of Offsite Pwr Rec		
new	SEIS2			XX2	Long Term 125 VDC Bus B	9	1.66E-08
				RY2	Seismic Reactor Scram Failure		
				SX2	Condensate Storage Tank Failure		
28	SEIS2	1.71E-08		OX2	Seismic Failure of Offsite Power	11	1.48E-08
				XX2	Long Term 125 VDC Bus B		
				FX2	Seismic Fire Protection Failure		
				SX2	Condensate Storage Tank Failure		
				CX2	Containment Spray/ESW		
28	SEIS2	1.68E-08		FX2	Seismic Fire Protection Failure	12	1.46E-08
				SX2	Condensate Storage Tank Failure		
				CX2	Containment Spray/ESW		
new	SEIS2			OX2	Seismic Failure of Offsite Power	13	1.40E-08
				XX2	Long Term 125 VDC Bus B		
				RY2	Seismic Reactor Scram Failure		
				SX2	Condensate Storage Tank Failure		
new	SEIS2			RY2	Seismic Reactor Scram Failure	14	1.38E-08
				SX2	Condensate Storage Tank Failure		
36	SEIS2	1.418E-08		OX2	Seismic Failure of Offsite Power	17	1.23E-08
				FX2	Seismic Fire Protection Failure		
				SX2	Condensate Storage Tank Failure		
				CX2	Containment Spray/ESW		
new	SEIS2			OX2	Seismic Failure of Offsite Power	18	1.17E-08
				RY2	Seismic Reactor Scram Failure		
				SX2	Condensate Storage Tank Failure		

Table 15 – Comparison of Sequences							
No.	Initiating Event	Frequency	Percent	Failed Events	Split Fraction Descriptions	New No.	New Frequency
49	SEIS1	9.87E-09		VR2 CC5 OV1	Independent EMRVs Fail to Reclose Independent Containment Spray Failure Independent Containment Vent Failure	20	9.93E-09
46	SEIS2	1.07E-08		OX2 DW2 FX2 SX2 LX2	Seismic Failure of Offsite Power Diesel Generators Seismic Fire Protection Failure Condensate Storage Tank Failure Seismic Offsite Power Recovery Failure	22	8.69E-09
new	SEIS3			OX3 DU3 DV3 FX3 SX3 VY3 LX3	Seismic Failure of Offsite Power Diesel Generator Building Liquefaction Diesel Generator Building Failure Seismic Fire Protection Failure Condensate Storage Tank Failure Seismic EMRV Failure to Reclose Seismic Offsite Power Recovery Failure	23	8.18E-09
new	SEIS3			OX3 DU3 DV3 FX3 SX3 VX3 LX3	Seismic Failure of Offsite Power Diesel Generator Building Liquefaction Diesel Generator Building Failure Seismic Fire Protection Failure Condensate Storage Tank Failure Seismic Failure of EMRVs to Open Seismic Offsite Power Recovery Failure	24	8.05E-09
50	SEIS2	9.72E-09		OX2 DW2 FX2 LX2	Seismic Failure of Offsite Power Diesel Generators Seismic Fire Protection Failure Seismic Offsite Power Recovery Failure	25	7.88E-09

A.7 FINAL SEISMIC MODEL

This model incorporates the change detailed above with the exception of the generic fragility sensitivity case. In summary the model changes include:

- Extrapolation of the existing hazard curve from 0.82g to 1.5g as recommended in NUREG-1407.
- Correction of the fire protection system seismic split fraction values.
- The addition of the turbine building and reactor building fragilities. These detailed fragilities were not included in the original IPEEE submittal based on the fact that the fragility of the structure was the onset of deformation and permanent displacement, not structural failure. It was originally assumed that equipment within the structure would survive this mode of failure with a high likelihood of success. The impact of the failure of either the reactor building or the turbine building is now modeled as core damage.
- A revised human error probability for makeup to the isolation condenser to reflect the potential that flow diversion of the fire protection system could occur when flowpaths not associated with the makeup to the condenser fail. In these cases, manual action to isolate the flow diversion of fire protection would be required before isolation condenser makeup could be successful. The new human error probability for operators providing makeup to isolation condenser using the fire protection system is 0.1.

- Incorporation of the failure of recirculation pumps supports. This event uses a screening threshold or generic fragility and is modeled as a large below core loss of coolant accident.

The following tables and paragraphs provide a summary of the results. The total mean seismic core damage frequency is 4.74E-06 per year. This represents an increase of approximately 33% from the reported seismic IPEEE.

Table 16 – Final Seismic Model Initiating Event Results

	SEIS1	SEIS2	SEIS3	SEIS4
Ground Acceleration Range	0.007 – 0.26g	0.26 – 0.46g	0.46 – 0.62g	0.62 – 1.5g
Initiating Event Frequency	7.44E-03	2.73E-05	2.72E-06	1.40E-06
Core Damage Frequency	3.42E-07	1.66E-06	1.40E-06	1.34E-06
Percent of Total CDF	7.2%	35.0%	29.5%	28.3%
Conditional CDF	4.60E-05	0.061	0.515	0.957
Unaccounted Frequency	2.23E-09	1.28E-09	2.21E-10	7.41E-10
Percent Unaccounted	0.65%	0.08%	0.02%	0.06%

A.7.1 Initiating Event Contributions

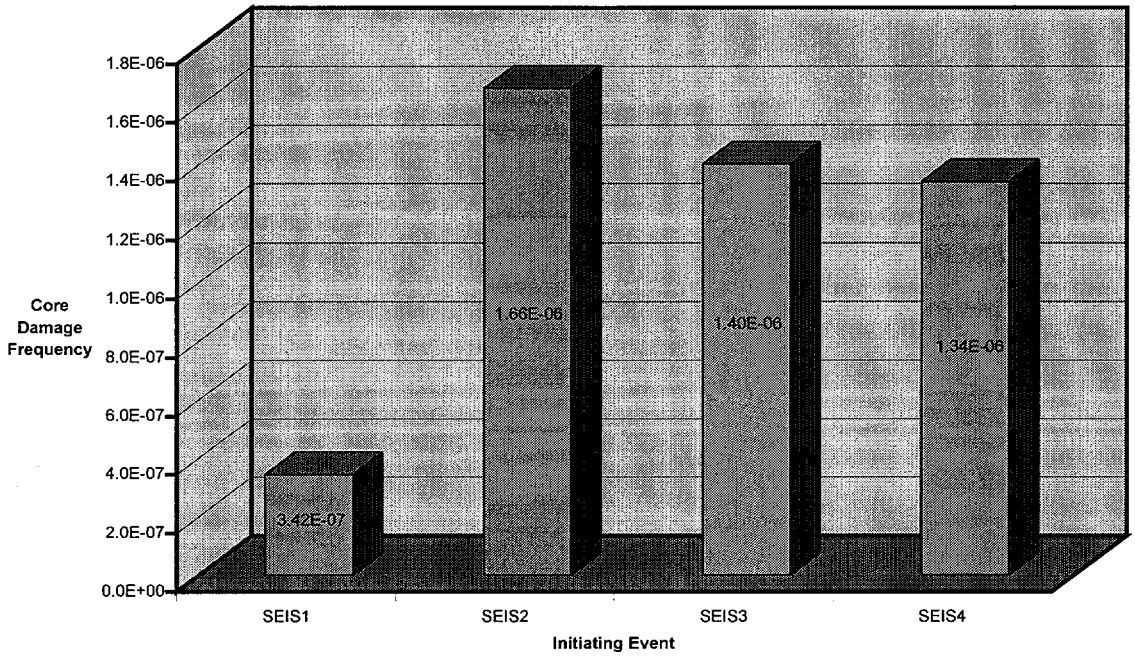
As in the case with the seismic IPEEE the lowest acceleration level accounts for 99% of the total seismic initiating event frequency while contributing only 7% to the seismic core damage frequency. This primarily is a result of the relatively low likelihood of seismic failure of equipment at these acceleration levels. This re-enforces the fact that Oyster Creek is relatively tolerant of ground accelerations up to and including the safe shutdown earthquake.

The SEIS2 initiating event (0.26 – 0.46g) accounts for 35% of the total seismic core damage frequency. This initiating event is the largest initiating event contributor to the seismic core damage frequency. This initiating event contribution has risen from a contribution of 26.6% in the seismic IPEEE model. The contributions in the SEIS2 initiating event have increased due to significant contributions of the reactor turbine building failures which have mean capacities of approximately 1.0g and 0.74 respectively and are assumed to directly result in core damage.

The SEIS3 initiating event (0.46 – 0.62g) accounts for 29.5% of the total seismic core damage frequency. This initiating event was the largest initiating event contributor in the seismic IPEEE. However, its contribution has dropped significantly in the new seismic model. This is primarily a result of increases in the SEIS2 initiating event contributions.

The SEIS4 initiating event contribution increases slightly. This increase is primarily a result of the hazard curve extrapolation which adds initiating event frequency.

Figure 1 - Initiator Contribution to Seismic Core Damage Frequency



A.7.2 System Importance

As in the case with the seismic IPEEE, systems that support or backup the function of the isolation condenser remain significant contributors. However, two significant additions are the seismic failures of the reactor and turbine buildings which are the top two Fussel-Vesely importance contributors.

Another addition to the list, when compared to the seismic IPEEE top event importance, is isolation condenser makeup using the fire protection system (top event MU). This addition is due to the change in the human error probability from a 4E-04 to a screening value of 0.1. The change in human error probability is based on the potential need to isolate the fire protection header following inadvertent seismically initiated fire protection system actuation.

Table 17 – Final Seismic Model Top Event Importance

No.	Top Event	Fussel-Vesely	Risk Achievement	Risk Reduction	Top Event Description
1	TZ	3.54E-01	1.57E+03	6.46E-01	Seismic Turbine Building Failure
2	RZ	9.69E-02	1.57E+03	9.03E-01	Seismic Reactor Building Failure
3	EW	8.61E-02	1.63E+00	9.14E-01	Seismic 4160 VAC 1C Failure
4	MU	6.56E-02	1.63E+00	9.34E-01	Independent Iso Cond Makeup Failure
5	SX	5.48E-02	1.34E+00	9.45E-01	Seismic Condensate Tank Failure
6	XX	5.34E-02	1.05E+00	9.47E-01	Seismic Long Term 125 VDC B Failure
7	IY	5.12E-02	4.48E+00	9.49E-01	Seismic Failure of Offsite Pwr Recovery
8	VR	4.40E-02	1.66E+00	9.56E-01	Indep Failure of EMRVs to Reclose
9	FX	3.55E-02	1.89E+00	9.65E-01	Seismic Fire Protection Failure
10	OX	3.42E-02	1.32E+00	9.66E-01	Seismic Offsite Power Failure

A significant addition to the seismic IPEEE fragility importance are the seismic failures of the reactor and turbine buildings. Another addition is the failure of recirculation pump support (Top Event RX) which is modeled as large below core loss of coolant accident. It should be noted that the seismic failure of the turbine building is listed twice in the table. This is due to the fact that both the column failure mode and shear failure mode are significant.

Table 18 – Seismic Fragility Fussel-Vesely Importance

Fragility ID	Top Event	Fussel-Vesely	Risk Achievement Worth	Risk Reduction Worth	10% Median Acceleration Decrease	Component Description
FRAG39	TZ	2.97E-01	1.57E+03	7.03E-01	1.14E+00	Turbine building - column
FRAG37	RZ	9.64E-02	1.57E+03	9.04E-01	1.05E+00	Reactor building - column
FRAG05	EW	8.61E-02	1.63E+00	9.14E-01	1.02E+00	Switchgear room fan
FRAG10	SX	5.48E-02	1.34E+00	9.45E-01	1.01E+00	Condensate storage tank
FRAG04	XX	5.34E-02	1.05E+00	9.47E-01	1.01E+00	Battery room fans
FRAG18	IY	5.12E-02	4.48E+00	9.49E-01	1.02E+00	Isolation condenser
FRAG38	TZ	4.11E-02	1.57E+03	9.59E-01	1.04E+00	Turbine building - shear
FRAG00	FX	3.55E-02	1.89E+00	9.65E-01	1.01E+00	Generic fragility for others
FRAG01	OX	3.42E-02	1.32E+00	9.66E-01	1.00E+00	Offsite power
FRAG00	RX	3.27E-02	2.80E+00	9.67E-01	1.01E+00	Generic fragility for others

A.7.3 Sequence Importance

The top 25 sequences in the final seismic model contribute approximately 80% to the total seismic core damage frequency. Table 19 provides a listing of the sequences. The top 15 sequences (75% of the total seismic core damage frequency) are described in the paragraphs below.

It should be noted that the sequences that are characterized by the failure of the turbine building failure and reactor building failure are non-minimal with respect to the failure of the recirculation pump supports. That is, failure of either the reactor or turbine building will result in core damage, however, these scenarios are fractured with the failure of the recirculation pump support failure. All reactor and turbine building failure scenarios appear in the database and the top event importance measures in Table 17 are accurate. References to the fractured sequence is provided in cases where the fractured sequence contributes over 0.5% to the total seismic core damage frequency.

- Sequence 1 is defined by the seismic failure of the turbine building following a SEIS2 initiating event. This sequence contributes 19%. (See also sequence number 22).

- Sequence 2 is defined by the seismic failure of the turbine building following a SEIS4 initiating event. This sequence contributes 14%. (See also sequence number 4).

- Sequence 3 is defined by the seismic failure of the turbine building following a SEIS3 initiating event. This sequence contributes 13%. (See also sequence number 10).

- Sequence 4 is defined by the seismic failure of the turbine building following a SEIS4 initiating event and the failure of the recirculation pump supports. This sequence is non-minimal since the failure of the turbine building is assumed to result in core damage. This sequence contributes 7%. (See also sequence number 2).

- Sequence 5 is defined by the seismic failure of the reactor building following a SEIS2 initiating event. This sequence contributes 6%.

- Sequence 6 is defined by the seismic failure of the turbine building following a SEIS1 initiating event. This sequence contributes 3%.

- Sequence 7 is defined by the seismic failure of the reactor building following a SEIS3 initiating event. This sequence contributes 6%. (See also sequence number 21).

- Sequence 8 is defined by the seismic failure of the reactor building following a SEIS4 initiating event. This sequence contributes 6%. (See also sequence number 14).

- Sequence 9 is defined by the independent failure of the EMRVs to reclose following initial post trip pressure relief followed by the independent failure of the core spray system following a SEIS1 initiating event. This sequence is the first independent failure scenario. Since main feedwater is assumed failed in seismic events and fire protection is not modeled as providing vessel inventory in seismic events, core damage results. This sequence contributes 1.6% to the seismic core damage frequency.

- Sequence 10 is defined by the seismic failure of the turbine building and the recirculation pump supports following a SEIS3 initiating event. This sequence is non-minimal since core damage is assumed following the failure of the turbine building. This sequence contributes 1.5% to the seismic core damage frequency. (See also sequence number 3).

- Sequence 11 is defined by the seismic loss of offsite power with the seismic failure of the diesel generator building (top events DU and DV) and a recirculation pump support failure following a SEIS3 initiating event. The failure of the recirculation pump support is assumed to result in a large loss of coolant accident. With a large loss of coolant event, the recovery of offsite power is not considered. The result is a large loss of coolant accident with a station blackout event. This sequence contributes 1% to the seismic core damage frequency.

- Sequence 12 is defined by the seismic failure of the reactor building following a SEIS1 initiating event. Failure of the reactor building is assumed to result in core damage. This sequence contributes 1% to the seismic core damage frequency.

- Sequence 13 is defined by the seismic failure of the 4160 VAC Bus 1C following a SEIS2 initiating event. Heat removal is initially provided by isolation condensers. The failure of makeup to the isolation condenser results in a loss of heat removal. Combined with the eventual long term loss of all DC power, the EMRVs become inoperable and prevent depressurization (or allow repressurization). Without high pressure sources of inventory makeup, core damage occurs. This sequence contributes 1% to the seismic core damage frequency.

- Sequence 14 is defined by the seismic failure of the reactor building and the seismic failure of the recirculation pump supports following a SEIS4 initiating event. This sequence is non-minimal since the failure of the reactor building is assumed to result in core damage. This sequence contributes 0.9% to the total seismic core damage frequency. (See also sequence number 8).

- Sequence 15 is defined as similar to sequence 13 except that a seismic failure of offsite power occurs. This sequence contributes 0.9% to the total seismic core damage frequency.

A.7.3 Sequence Importance

The top 25 sequences in the final seismic model contribute approximately 80% to the total seismic core damage frequency. Table 19 provides a listing of the sequences. The top 15 sequences (75% of the total seismic core damage frequency) are described in the paragraphs below.

It should be noted that the sequences that are characterized by the failure of the turbine building and reactor building are non-minimal with respect to failure of the recirculation pump supports. That is, failure of either the reactor or turbine building will result in core damage, however, these scenarios are fractured with the failure of the recirculation pump support failure. All reactor and turbine building failure scenarios appear in the database and the top event importance measures in Table 17 are accurate. References to the fractured sequence is provided in cases where the fractured sequence contributes over 0.5% to the total seismic core damage frequency.

- Sequence 1 is defined by the seismic failure of the turbine building following a SEIS2 initiating event. This sequence contributes 19%. (See also sequence number 22).

- Sequence 2 is defined by the seismic failure of the turbine building following a SEIS4 initiating event. This sequence contributes 14%. (See also sequence number 4).

- Sequence 3 is defined by the seismic failure of the turbine building following a SEIS3 initiating event. This sequence contributes 13%. (See also sequence number 10).

- Sequence 4 is defined by the seismic failure of the turbine building following a SEIS4 initiating event and the failure of the recirculation pump supports. This sequence is non-minimal since the failure of the turbine building is assumed to result in core damage. This sequence contributes 7%. (See also sequence number 2).

- Sequence 5 is defined by the seismic failure of the reactor building following a SEIS2 initiating event. This sequence contributes 6%.

- Sequence 6 is defined by the seismic failure of the turbine building following a SEIS1 initiating event. This sequence contributes 3%.

- Sequence 7 is defined by the seismic failure of the reactor building following a SEIS3 initiating event. This sequence contributes 6%. (See also sequence number 21).

- Sequence 8 is defined by the seismic failure of the reactor building following a SEIS4 initiating event. This sequence contributes 6%. (See also sequence number 14).

- Sequence 9 is defined by the independent failure of the EMRVs to reclose following initial post trip pressure relief followed by the independent failure of the core spray system following a SEIS1 initiating event. This sequence is the first independent failure scenario. Since main feedwater is assumed failed in seismic events and fire protection is not modeled as providing vessel inventory in seismic events, core damage results. This sequence contributes 1.6% to the seismic core damage frequency.

- Sequence 10 is defined by the seismic failure of the turbine building and the recirculation pump supports following a SEIS3 initiating event. This sequence is non-minimal since core damage is assumed following the failure of the turbine building. This sequence contributes 1.5% to the seismic core damage frequency. (See also sequence number 3).

Table 19 – Final Seismic Model Top 25 Seismic Sequences						
No.	Initiating Event	Frequency	Percent	Cum. Percent	Split Fraction	Split Fraction Description
1	SEIS2	9.08E-07	19.1%	19.1%	TZ2	Seismic Turbine Building Failure
2	SEIS4	6.76E-07	14.2%	33.4%	TZ4	Seismic Turbine Building Failure
3	SEIS3	6.26E-07	13.2%	46.6%	TZ3	Seismic Turbine Building Failure
4	SEIS4	3.46E-07	7.3%	53.9%	TZ4	Seismic Turbine Building Failure
					RX4	Seismic Recirc Support Failure
5	SEIS2	2.62E-07	5.5%	59.4%	RZ2	Seismic Reactor Building Failure
6	SEIS1	1.61E-07	3.4%	62.8%	TZ1	Seismic Turbine Building Failure
7	SEIS3	1.53E-07	3.2%	66.0%	RZ3	Seismic Reactor Building Failure
8	SEIS4	8.20E-08	1.7%	67.7%	RZ4	Seismic Reactor Building Failure
9	SEIS1	7.56E-08	1.6%	69.3%	VR2	Indep EMRV Failure to Reclose
					CS1	Indep Core Spray Failure
10	SEIS3	7.34E-08	1.5%	70.9%	TZ3	Seismic Turbine Building Failure
					RX3	Seismic Recirc Pump Support Failure
11	SEIS3	4.77E-08	1.0%	71.9%	OX3	Seismic Loss of Offsite Power
					DU3	Seismic DG Building Liquefaction
					DV3	Seismic DG Building After Liquefaction
					RX3	Seismic Recirc Pump Support Failure
12	SEIS1	4.63E-08	1.0%	72.8%	RZ1	Seismic Reactor Building Failure
13	SEIS2	4.59E-08	1.0%	73.8%	EW2	Seismic 4160 VAC 1C Failure
					XX2	Seismic Long Term 125 VDC B Failure
					SX2	Seismic Condensate Tank Failure
					MU2	Indep Iso Condenser Makeup Failure
14	SEIS4	4.20E-08	0.9%	74.7%	RZ4	Seismic Reactor Building Failure
					RX4	Seismic Recirc Pump Support Failure
15	SEIS2	4.07E-08	0.9%	75.5%	OX2	Seismic Loss of Offsite Power
					EW2	Seismic 4160 VAC 1C Failure
					XX2	Seismic Long Term 125 VDC B Failure
					SX2	Seismic Condensate Tank Failure
					MU2	Indep Iso Condenser Makeup Failure
16	SEIS3	3.79E-08	0.8%	76.3%	OX3	Seismic Loss of Offsite Power
					DU3	Seismic DG Building Liquefaction
					DV3	Seismic DG Building After Liquefaction
					SX3	Seismic Condensate Tank Failure
					IY3	Seismic Isolation Condenser Failure
17	SEIS4	2.43E-08	0.5%	76.9%	DX4	Seismic 125 VDC B Failure
					OX4	Seismic Loss of Offsite Power
					DU4	Seismic DG Building Liquefaction
					DV4	Seismic DG Building After Liquefaction
					RX4	Seismic Recirc Pump Support Failure
18	SEIS4	2.28E-08	0.5%	77.3%	OX4	Seismic Loss of Offsite Power
					DU4	Seismic DG Building Liquefaction
					DV4	Seismic DG Building After Liquefaction
					RX4	Seismic Recirc Pump Support Failure
19	SEIS3	2.26E-08	0.5%	77.8%	OX3	Seismic Loss of Offsite Power
					DU3	Seismic DG Building Liquefaction
					DV3	Seismic DG Building After Liquefaction
					FX3	Seismic Fire Protection Failure
					SX3	Seismic Condensate Tank Failure
					LX3	Seismic Offsite Pwr Recovery Failure

Table 19 – Final Seismic Model Top 25 Seismic Sequences						
No.	Initiating Event	Frequency	Percent	Cum. Percent	Split Fraction	Split Fraction Description
20	SEIS3	1.99E-08	0.4%	78.2%	OX3	Seismic Loss of Offsite Power
					DU3	Seismic DG Building Liquefaction
					DV3	Seismic DG Building After Liquefaction
					SX3	Seismic Condensate Tank Failure
					MU2	Indep Iso Condenser Makeup Failure
					LX3	Seismic Offsite Pwr Recovery Failure
21	SEIS3	1.80E-08	0.4%	78.6%	RZ3	Seismic Reactor Building Failure
					RX3	Seismic Recirc Pump Support Failure
22	SEIS2	1.51E-08	0.3%	78.9%	TZ2	Seismic Turbine Building Failure
					RX2	Seismic Recirc Pump Support Failure
23	SEIS4	1.50E-08	0.3%	79.3%	DX4	Seismic 125 VDC B Failure
					OX4	Seismic Loss of Offsite Power
					DU4	Seismic DG Building Liquefaction
					DV4	Seismic DG Building After Liquefaction
					SX4	Seismic Condensate Tank Failure
					IY4	Seismic Isolation Condenser Failure
24	SEIS4	1.41E-08	0.3%	79.5%	OX4	Seismic Loss of Offsite Power
					DU4	Seismic DG Building Liquefaction
					DV4	Seismic DG Building After Liquefaction
					SX4	Seismic Condensate Tank Failure
					IY4	Seismic Isolation Condenser Failure
25	SEIS4	1.32E-08	0.3%	79.8%	DX4	Seismic 125 VDC B Failure
					OX4	Seismic Loss of Offsite Power
					DU4	Seismic DG Building Liquefaction
					DV4	Seismic DG Building After Liquefaction
					FX4	Seismic Fire Protection Failure
					SX4	Seismic Condensate Tank Failure
					LX4	Seismic Offsite Pwr Recovery Failure

ATTACHMENT B:

COMPARISON OF RESPONSE SPECTRA

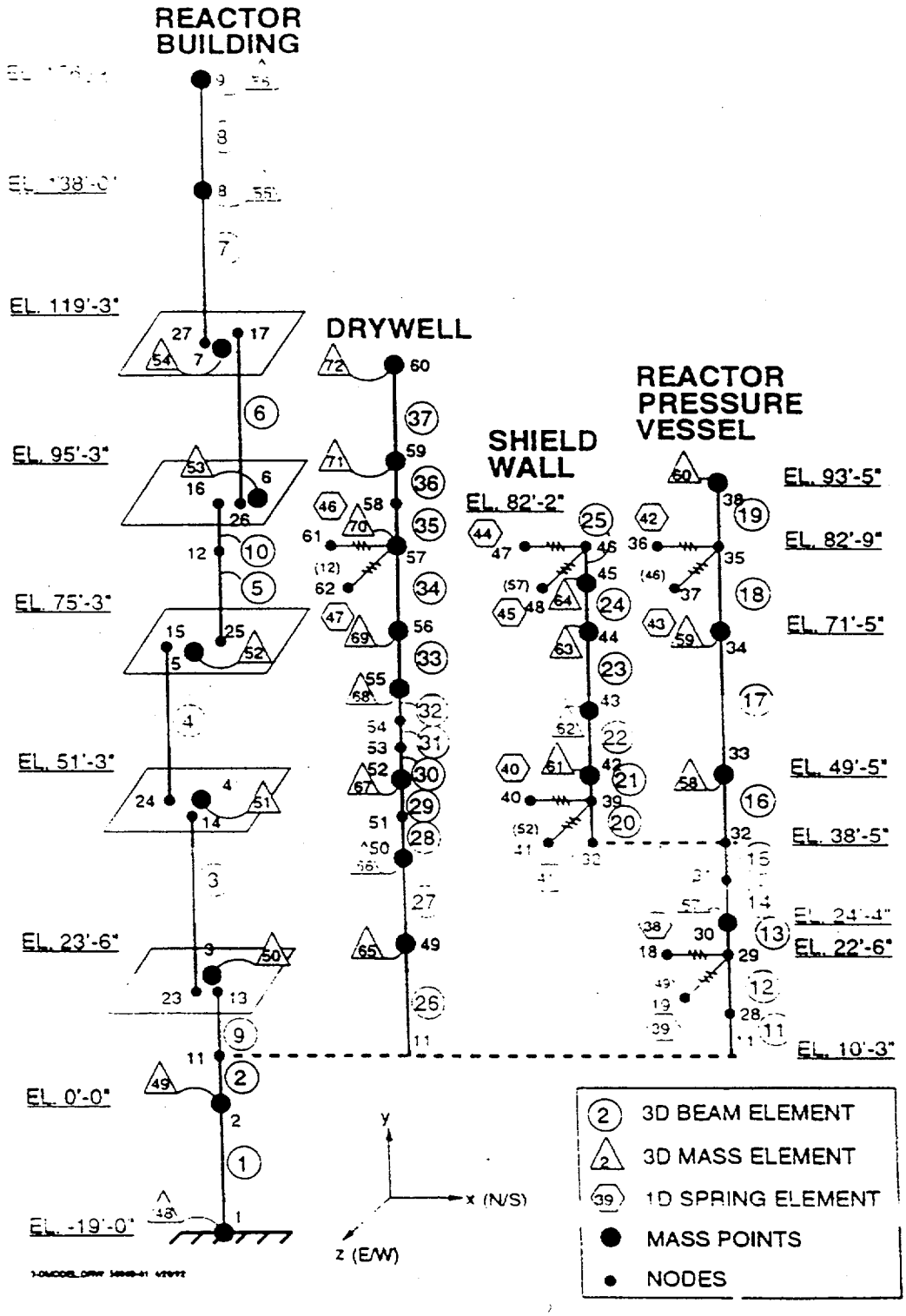


Figure 4-1: 3-D coupled model. Oyster Creek Reactor Building.

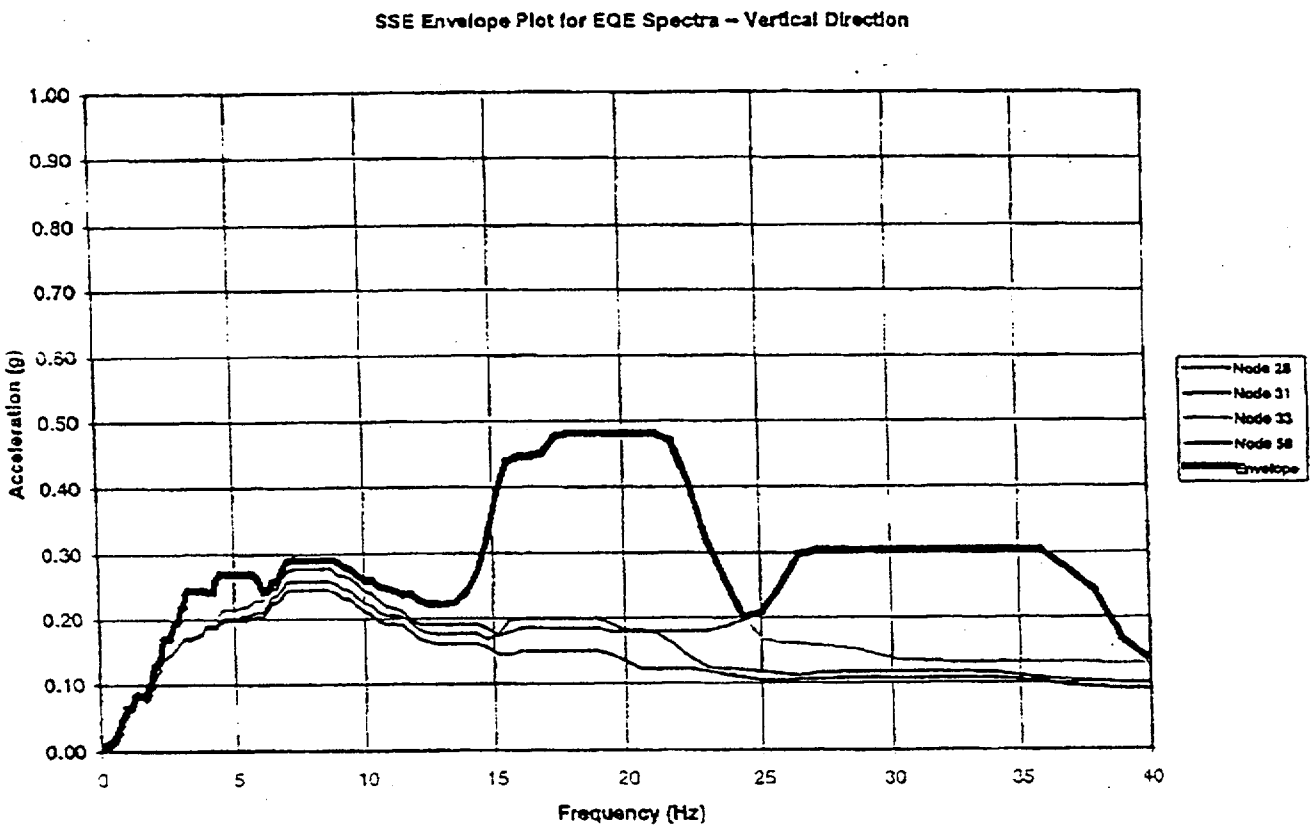


Figure A-4. Envelope Plot for Vertical EQE Spectra (SSE)

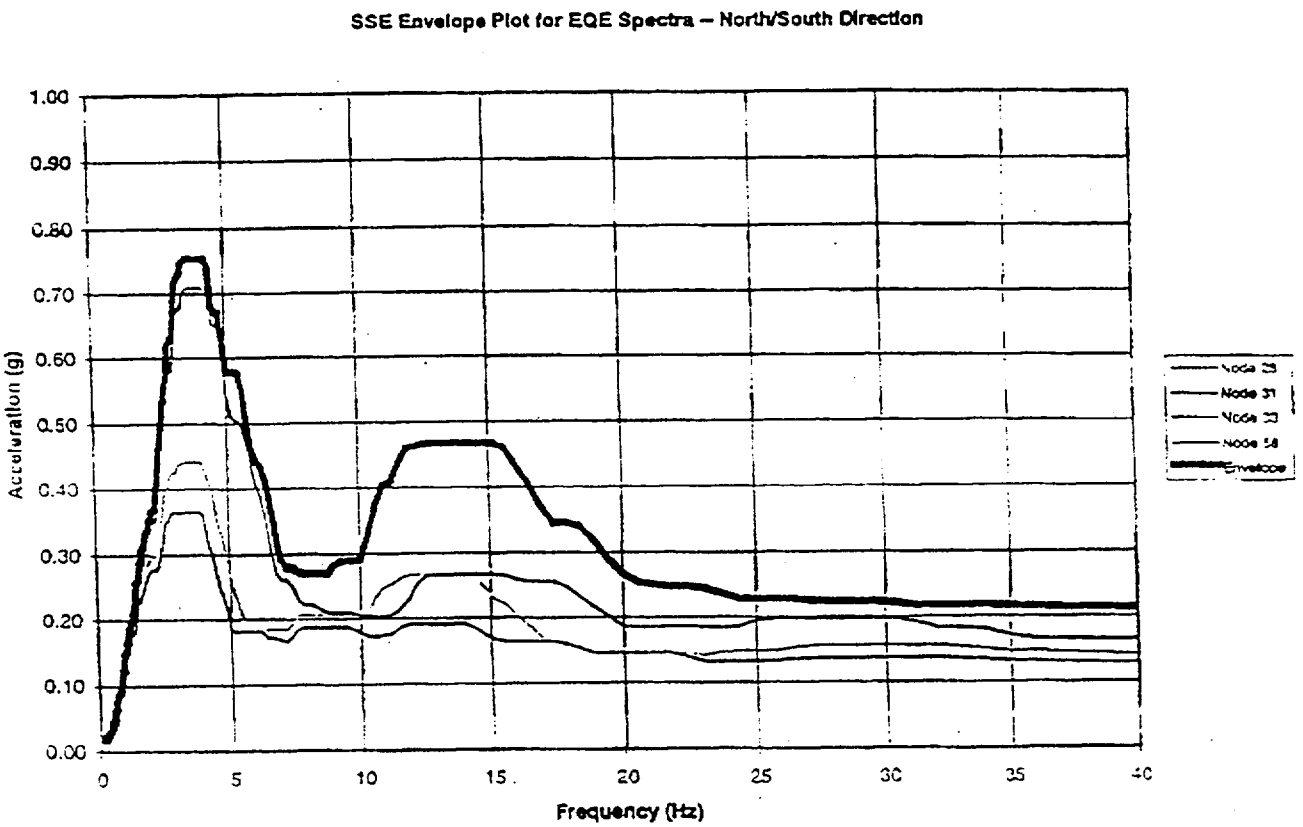


Figure A-5. Envelope Plot for North/South EQE Spectra (SSE)

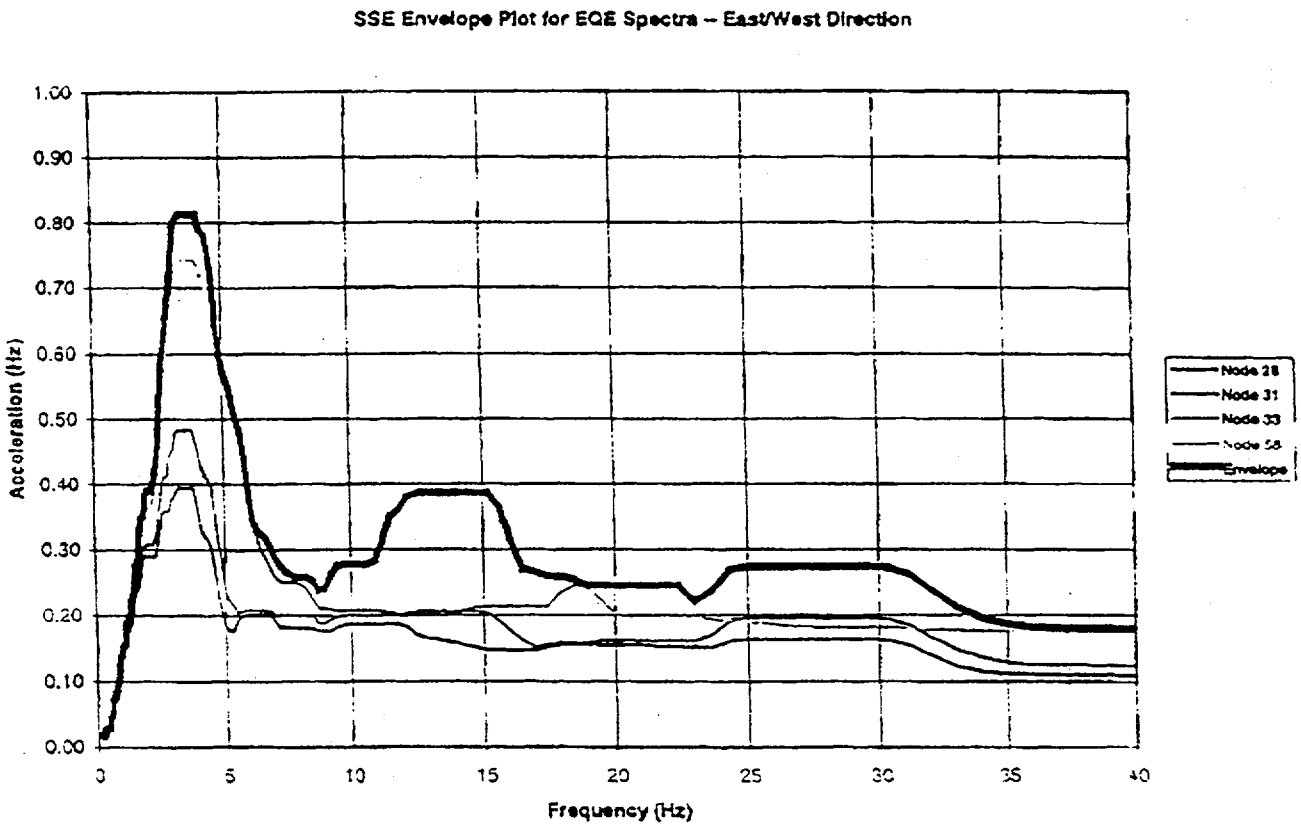
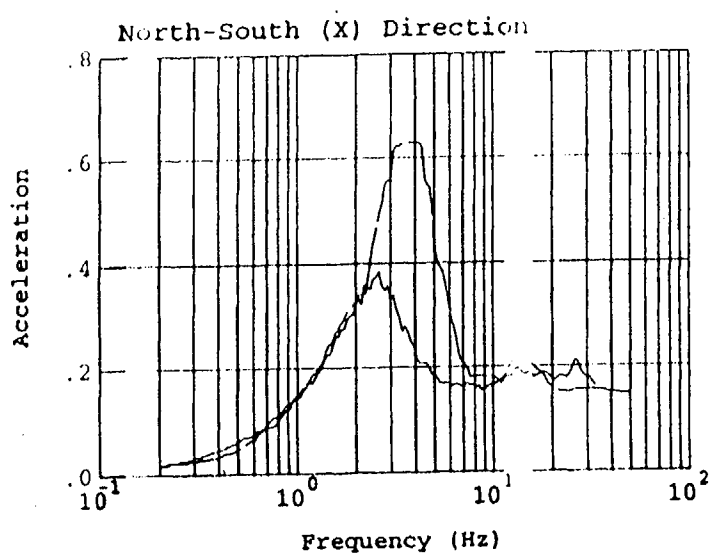
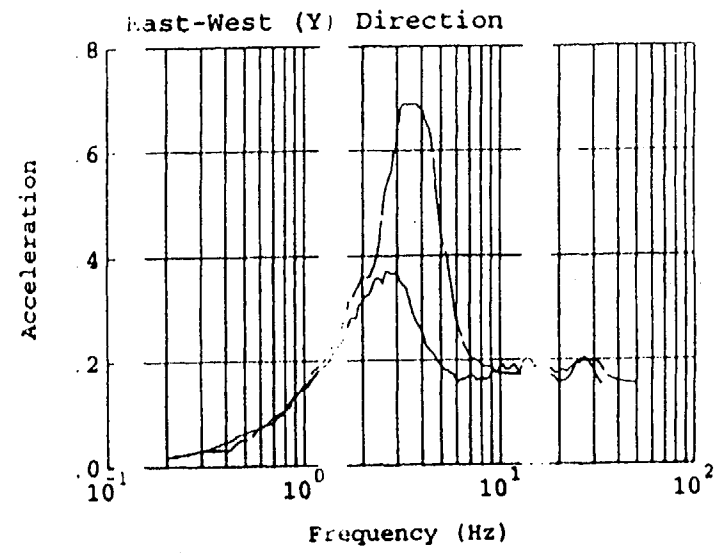
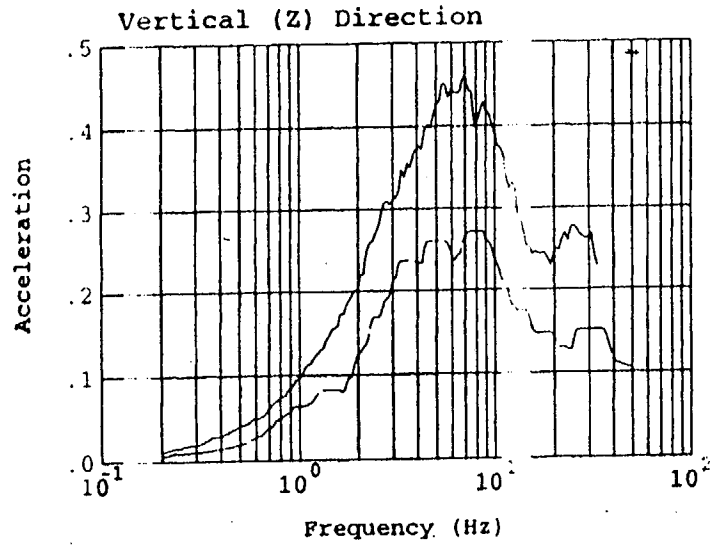


Figure A-6. Envelope Plot for East/West EQE Spectra (SSE)







Legend:

Median (3xSSE)

A-46

Notes:

Accelerations in g's

5% Damped Spectra

OCNGS RB: Comparison of A-46 and IPEEE Response Spectra

Drywell, Elevation 71'-6", Node 56