4.0 DESIGN CRITERIA AND COMPLIANCE

4.1 INTRODUCTION

A general description of GE-MO and a summary of operational functions are contained in Section 1. Original design criteria for GE-MO facilities were developed and established as part of the design for a fuel reprocessing plant - the Midwest Fuel Recovery Plant (MFRP). Criteria herein are those applicable to the use of those facilities for spent fuel storage.

4.1.1 Material Stored

The material stored at GE-MO is irradiated light water UO_2 fuel with initial enrichment of 5% U-235 or less, stainless steel, or Zircaloy cladding and in a "bundle of rods" geometry. The calculated fission product activity contents of fuel irradiated at 40 kW/kgU, exposed at 24,000 MWd/TeU and 44,000 MWd/TeU, and cooled 1 year are presented in Table 4-1.

Fuel stored at GE-MO has exposures from 177.9 MWd/TeU to 36,712.9 MWd/TeU. The average burnup of the fuel bundles is 17,740.1 MWd/TeU and the median burnup is 19327.8 MWd/TeU. The cooling periods range from 17 to 34 years with an average cooling time of about 27 years as of April 2004.

Included in the fuel stored, GE-MO currently stores four fuel bundles from San Onofre Unit 1 that exhibited high radionuclide transfer rates, and 753 bundles from Dresden Unit 2 that are warranty returns.

The four San Onofre fuel bundles (numbers C-21, C-28, C-46 and C-47) are stored in basket P-117, located in Fuel Basin II, grid B-13. These fuel bundles exhibited higher than normal radionuclide transfer rates during sipping testing at San Onofre in May 1976. Further testing provided evidence that the fuel bundles were within regulatory limits for shipping. The radionuclide transfer rate decreased with time in storage, and storage of these fuel bundles has not had an unacceptable effect on the Morris fuel basin. The bundles were received during February and March of 1978. The bundles are 14 x14 Stainless Steel clad PWR bundles discharged on June 2, 1973. Burnup on these bundles range from 30,946 to 32,804 MWd/TeU.

The 753 fuel bundles stored at Morris from Dresden Unit 2 are GE 7 x 7 BWR bundles with Zircaloy cladding and discharge dates from July 15, 1970 to April 13, 1972. The burnup on the Dresden fuel bundles ranges from 178 to 5,708 MWd/TeU. Analysis on several fuel pins from this batch showed evidence of cladding hydriding.

Sipping and visual inspections of fuel were performed in the early 1980s to verify the radionuclide transfer rates and the physical integrity of the fuel stored at Morris Operation.

The quality of the water in the GE-MO basin is strictly maintained to inhibit corrosion of Zircaloy and Stainless Steel cladding and components. Perforations in fuel cladding expose the Uranium oxide pellets to water. Uranium oxide pellets have been observed to be highly stable when in contact with pool water (IAEA 1012 pg. 55). Basin radiochemistry is routinely monitored, an appreciable change in the radionuclide transfer rate of fuel bundles in storage would be evident.

Realistic exposures based on fuel in storage have been used in some analyses, as appropriate. Table 4-2 contains a list of analyses, fuel exposures and cooling times on which each is based.

Heat load calculations for basin water temperature and evaporation rates, basin water cooler design, and ventilation air cooling design are based on heat loads from fuel currently in storage and that expected to be stored.

4.1.2 Storage Conditions

Normal storage conditions at GE-MO impose much less stress on fuel than does the normal operational environment within a reactor. Maintaining basin chemistry and the integrity of fuel rods provides protection against uncontrolled release of radioactive material from fuel in storage. Instrumentation and other equipment are provided to warn of unsafe conditions or the approach of unsafe conditions. However, the approach of unsafe conditions is relatively slow in all cases, so rapid response and prompt, automatic initiation of corrective action - as in a reprocessing plant or reactor in non-storage conditions - is not required.

Table 4-1
SPENT FUEL FISSION PRODUCT ACTIVITY
(Ci/TeU)
Specific Power = 40 kW/kgU
Cooling Time = 1 Year

CLASS	ISOTOPE	HALF LIFE	24,000 MWd/TeU	44,000 MWd/TeU
Noble Gases	Kr-85	10.701y	7,620	12,000
Halogens	I-129	1.57×10^7y	.021	.044
Tritium	H-3	12.346y	416	766
Transuranics	Am-241	432y	99	250
	Am-243	7370y	2.6	32
	Cm-242	162.76d	1,350	9,160
	Cm-244	18.099y	169	5,090
Total			1,621	14,532
All Remaining Fission Products	Rb-86	18.82d	.000693	-
	Sr-89	50.55d	9,410	7,140
	Sr-90	28.82y	64,700	103,000
	Y-90	64.06h	64,800	103,000

CLASS	ISOTOPE	HALF LIFE	24,000 MWd/TeU	44,000 MWd/TeU
	Y-91	58.51d	20,800	16,500
	Zr-93	1.53×10^6y	2.3	3.9
	Zr-95	63.98d	41,500	38,300
	Nb-95m	86.6hd	527	487
	Nb-95	34.97d	87,800	81,800
	Tc-99	2.14×10^5y	10.8	18.9
	Ru-103	39.35d	2,680	3,280
	Rh-103m	56.116m	2,680	3,290
	Ru-106	366.5d	172,000	344,000
	Rh-106	29.8s	172,000	344,000
	Ag-110m	26.42d	12,300	51,700
	Ag-110	24s	160	672
	Cd-113m	14.6y	15	42.8
	Cd-115m	44.8d	3.5	4.9
	Sn-119m	250d	26.4	40.2
	Sn-123	129d	6113	801
	Sb-124	60.2d	3.2	8.1
	Sb-125	2.71y	4,840	10,100
	Te-125m	58d	1,180	2,470
All Remaining Fission Products	Sn-119m	250d	26.4	40.2
	Sn-123	129d	6113	801
	Sb-124	60.2d	3.2	8.1
	Sb-125	2.71y	4,840	10,100
	Te-125m	58d	1,180	2,470
	Te-127m	109d	1,320	1,870
	Te-127	9.35h	1,300	1,830
	Te-129m	33.52d	43.1	52.7
	Te-129	69.5m	27.4	33.5
	Cs-134	2.062y	88,900	283,000
	Cs-137	30.174y	77.900	142,000
	Ba-137m	2.5513m	73,700	134,000
	Ce-141	32.55d	800	772
	Ce-144	284.5d	530,000	594,000
	Pr-144	17.3m	530,000	594,000
	Pr-144m	7.2m	6,360	7,130
	Pm-147	2.62344y	104,000	91,400
	Pm-148m	41.29d	94.5	88.0
	Pm-148	5.37d	6.5	6.07
	Sm-151	87y	936	1,350
	Eu-152	13.2y	6.9	8.0
	Eu-154	8.5y	4,390	16,000

CLASS	ISOTOPE	HALF LIFE	24,000 MWd/TeU	44,000 MWd/TeU
	Eu-155	4.96y	1,020	3,100
	Gd-153	241.6d	3.9	21.0
	Tb-160	72.1y	16.6	63.3
Total of All Remaining Fission Products			2.08×10^6	2.98×10^6

Table 4-2
ANALYSES, FUEL EXPOSURES, AND COOLING TIMES USED

Section	Type of Analysis	Exposure and Cooling Time Used MWd/TeU	Months
5.4.4.3	Storage Basket Heat Transfer	44,000	4
7.3.1	Radiation Sources	24,000	12
7.3.2	Fission Gases Released	24,000	12
7.4.2	Direct Radiation from Fuel	24,000	12
7.7.2	Maximum Off-site Exposures	24,000	12
8.6	Fuel Drop Accidents	44,000	12
8.7	Missile Impact Accidents	24,000	12

4.2 STRUCTURAL AND MECHANICAL SAFETY CRITERIA

Structures, systems and components (SSCs) contributing to prevention of accidents (or to mitigation of consequences of accidents) which could affect public health and safety have been designed, fabricated, erected, operated, and maintained in compliance with established performance and quality standards. Under these standards, GE-MO will withstand, without loss of important protection capability, all credible operating and accident stresses, including forces that might be imposed by natural phenomena such as earthquakes, tornadoes, or flooding conditions.

Standards for ensuring SSCs will adequately perform required safety functions for their intended service life with a low probability of failure have been based on temperatures, corrosion rates and other stress conditions derived from comprehensive analyses, including consideration of:

a. accessibility for in-service surveillance, monitoring and repair (or replacement);

b. potential for short-term exposure to abnormal operating or accident conditions;

c. consequences of component failure - no single component failure or multiple failures caused by a single initiating event shall result in significant radiation exposure to the public;

d. accessibility for emergency services, including ambulance attendants, fire and police services, and other emergency activity.

4.2.1 Wind and Tornado Loadings

4.2.1.1 Criteria

Final structures and components essential for safety shall be designed to withstand effects of short-term wind velocities of 300 mph with pressure differentials of up to 3 psi without damage to fuel in storage to an extent endangering public health and safety. The site is located in USNRC Tornado Intensity Region I, as defined in Regulatory Guide 1.76.

4.2.1.2 Compliance

The fuel basin structure (enclosure) was analyzed with calculated wind loads applied as uniform static loads on vertical or horizontal projected areas of the walls and roof. Only dead load was considered as resisting uplift. Horizontal wind loads are distributed by the walls to the floor and roof systems, which transfer loads to the lateral load-carrying elements of the structures.

Plant structures and components were designed to withstand sustained wind velocities of 110 mph without loss of functions. At higher velocities, enclosure covering may fail or blow away.

These analyses included consideration of a drop in atmospheric pressure of 3 psi in 3 seconds. This condition would damage the basin enclosure, probably damage or even remove much of the roof and wall sheathing from the basin enclosure, but would cause no off-site radiological effect.

4.2.2 Tornado Missile Protection

4.2.2.1 Criteria

Plant SSCs essential for safety shall be designed to withstand effects of windborne missiles without damage to fuel in storage to an extent endangering public health and safety.

4.2.2.2 Compliance

Analyses in Appendix A.15 indicate the public health and safety would not be endangered as a result of tornado missiles impacting fuel storage structures or components.

4.2.3 Water Level (Flood) Design

4.2.3.1 Criteria

Structural integrity of fuel storage buildings and components shall not be endangered by flooding.

4.2.3.2 Compliance

Analysis has shown the maximum water level of a hypothetical flood greater than the maximum recorded flood at the site is below the site elevation (Appendix A.6).

4.2.4 Seismic Design

4.2.4.1 Criteria

Fuel storage structures and components essential to integrity of stored fuel, or fuel in the process of being transferred from shipping cask to the storage basin, shall be constructed to withstand a seismic event which, based on studies of area seismic history and geology, has a predicted recurrence of once per 1,000 years.

4.2.4.2 Compliance

The main building, including all portions of the structure now used for irradiated fuel storage, was originally constructed to seismic criteria based on a design earthquake and a maximum earthquake. The design earthquake was defined as a seismic event that has a reasonable probability of occurrence during the life of the facility, based on studies of historical seismically and structural geology. The design earthquake has a horizontal ground acceleration of 0.1 G.

The maximum earthquake is rated at twice the acceleration of the design earthquake, or 0.2-G. The design basis earthquake (DBE) can be sustained by these structures without exceeding allowable stresses. The maximum earthquake (ME) can be sustained without exceeding yield stress limits of the structure.

The 1940 El Centro, California earthquake has been thoroughly studied and well documented and provided most of the seismic data for time-history analyses available at the time of MFRP design. Illinois is not noted for earthquakes and no equally well studied seismic data base was available for Illinois.

Comparisons have been made between the El Centro earthquake spectrum and the spectrum in Regulatory Guide (RG) 1.60 for both Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) conditions. Results are shown in Figures 4-1 and 4-2. In generating spectra for the El Centro earthquake, damping values of 2% for DBE and 5% for ME were used. These damping values are consistent with those used in design of the basin structure. Sampling values for the RG 1.60 spectrum are 4% for OBE and 7% for SSE conditions, per RG 1.61. Differences between these spectra are insignificant.

A new fuel storage system was completed in 1976 to replace the original MFRP storage system. Since the new system is fabricated and installed as a separate entity in relation to the civil structures, it was designed to criteria in accordance with 10 CFR 100, Appendix A, and Regulatory Guide 1.60.

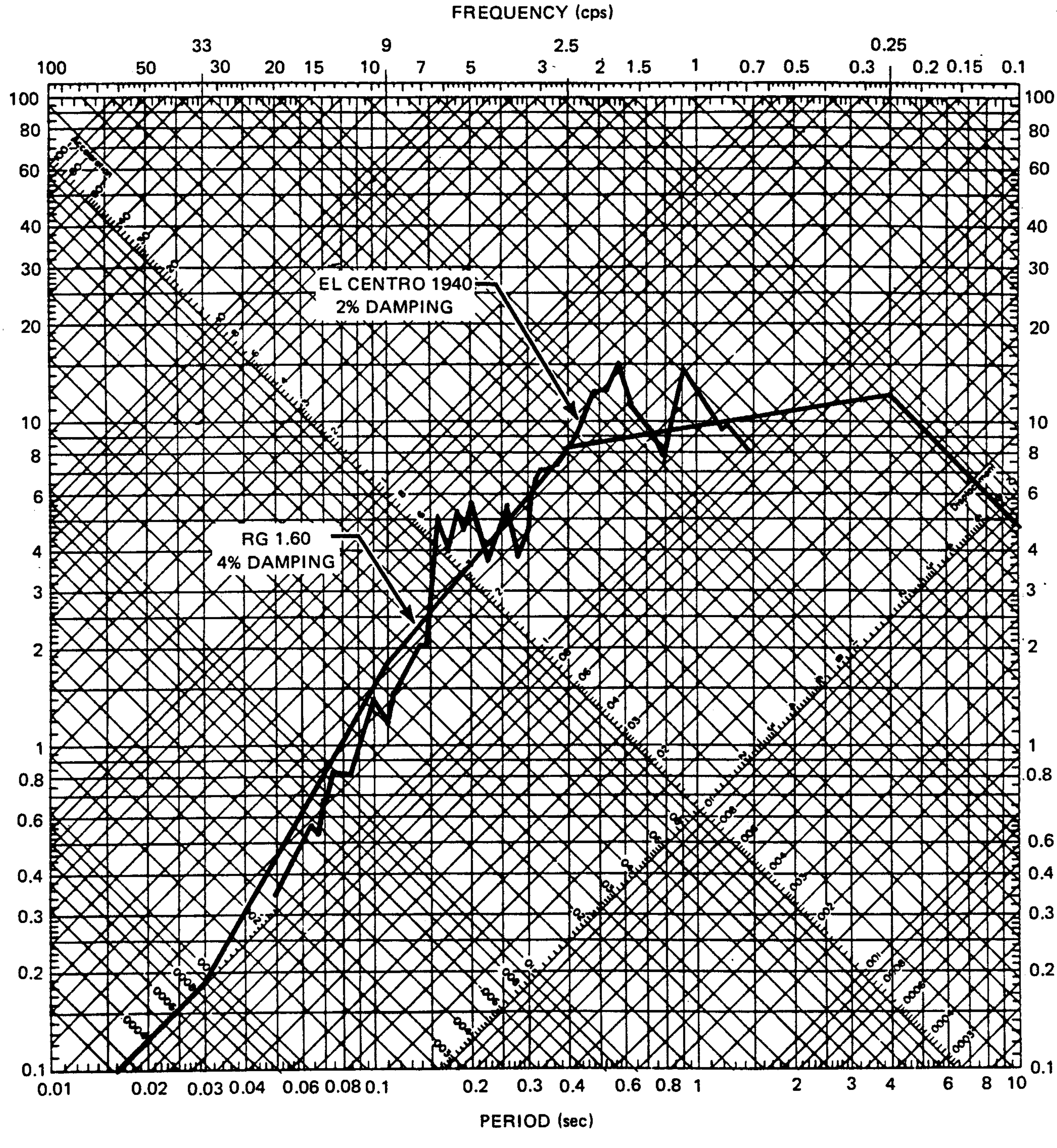


Figure 4-1. Spectra Comparison – 0,10G Ground Acceleration – RG 1.60 vs. El Centro 1940 N-S

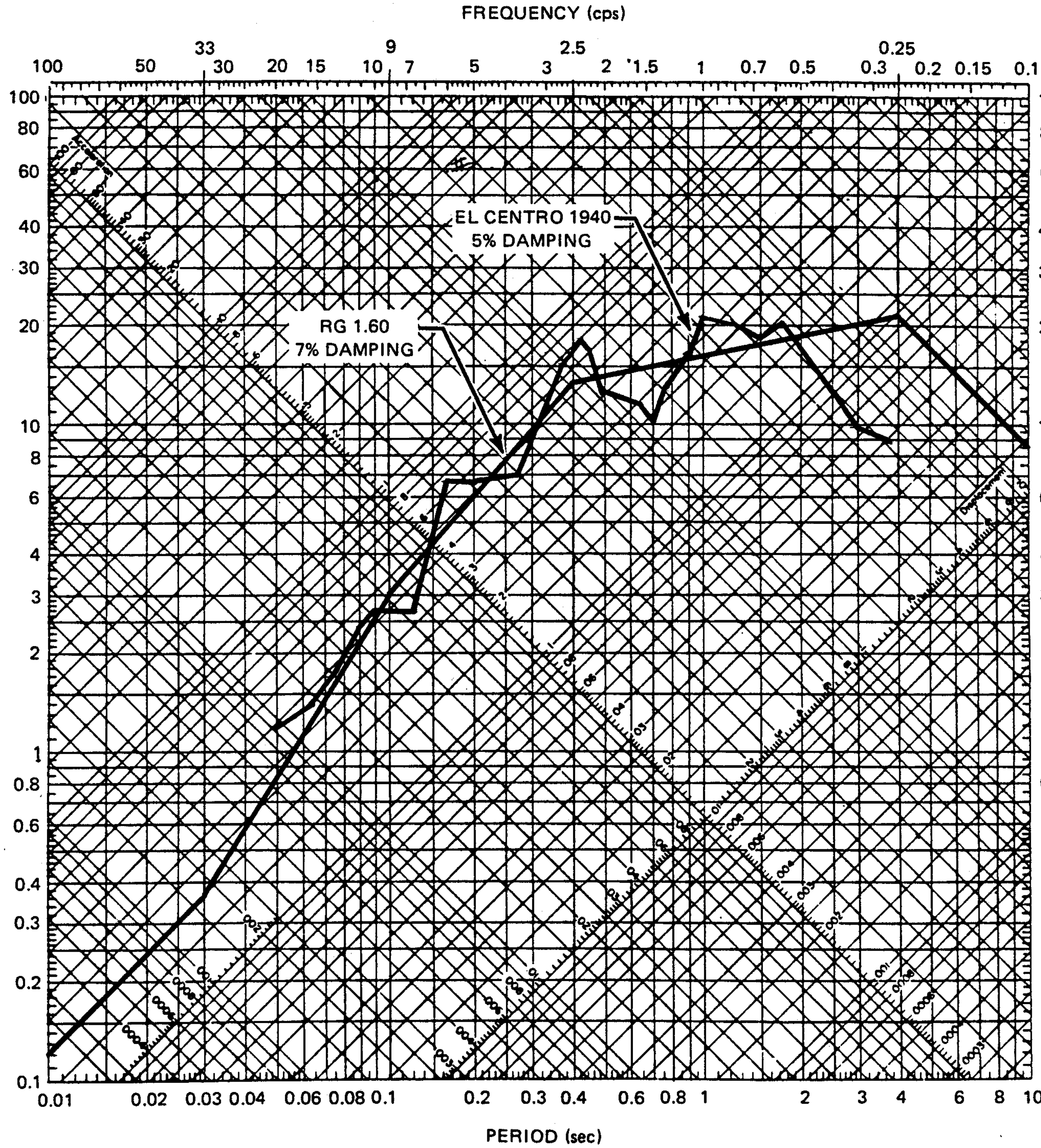


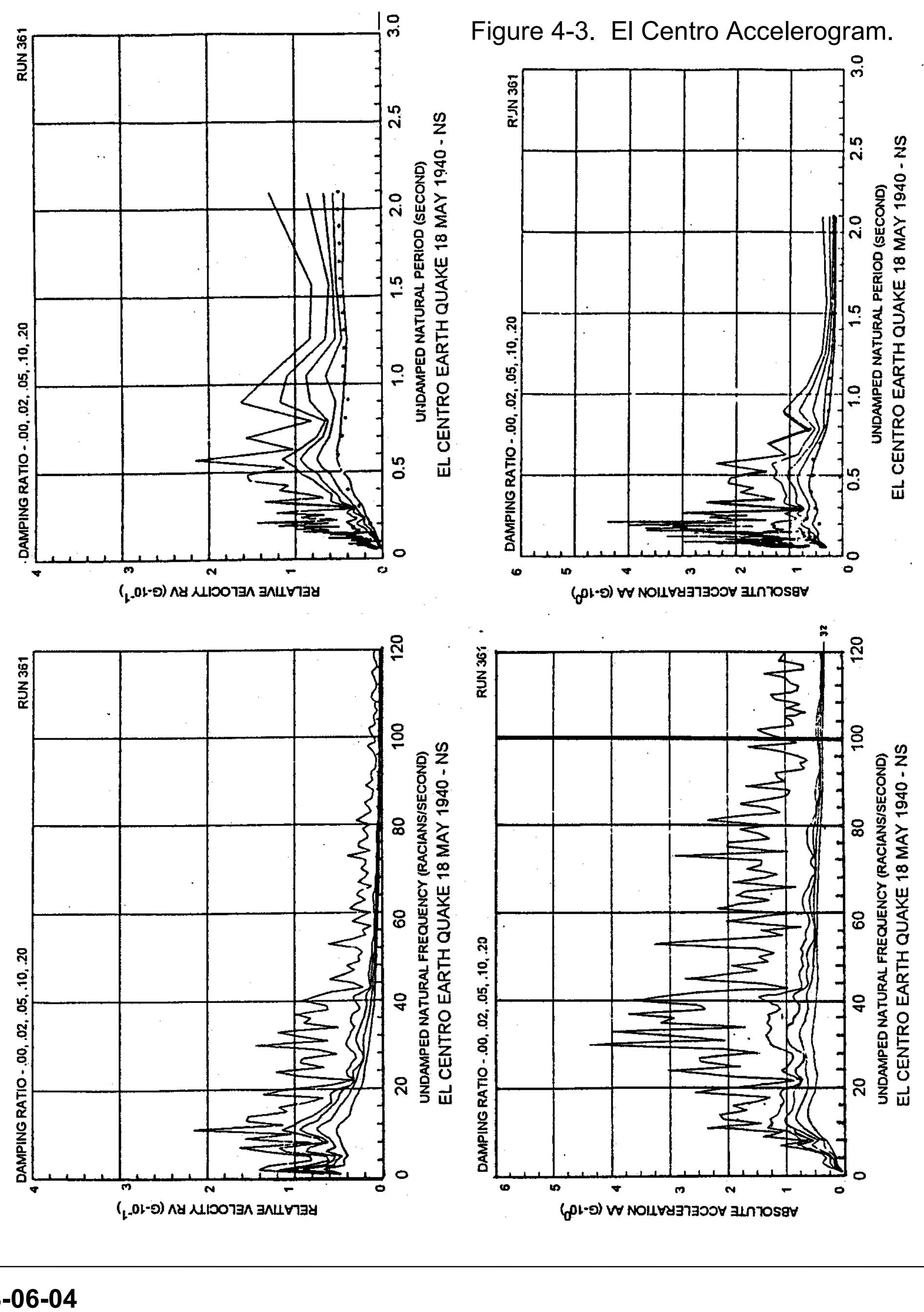
Figure 4-2. Spectra Comparison – 0.20G Ground Acceleration – RG 1.60 vs. El Centro 1940 N-S

4.2.4.2.1 Seismic Accelerations - Basins and Related Structures

a. Design Response Spectra

Structural (and equipment supported at grade) accelerations resulting from the DBE are defined by design response spectra. Design of fuel unloading and storage basins and underground vaults was based on north-south components of the 1940 El Centro earthquake normalized to 0.1G and 0.2G for the maximum earthquake case. The El Centro accelerogram is shown in Figure 4-3. The time used for the floor-level (main building) spectra was 6 seconds. Comparison of ground motion spectra for the 30 second period shows no measurable differences in the range provided.

Figure 4-3. El Centro Accelerogram.



b. Design Response Spectra Derivation

Absolute acceleration response spectra for ground motion are shown in Figures 4-4, 4-5 and 4-6 for damping ratio values of 0.005, 0.010, and 0.020, respectively. These spectra result from a time-history analysis of the 1940 El Centro earthquake.

c. Damping values used for both design and maximum earthquake dynamic analyses of basin and vault structures, excluding basket and grid system, are:

ITEM	% CRITICAL DAMPING
Reinforced concrete structures	5.0
Steel frame structures	2.0
Welded assemblies	1.0
Bolted and riveted assemblies	2.0
Piping systems containing radioactive material	0.5
Underground vaults and basins containing radioactive material	0.5

d. Bases for Site-Dependent Analysis

A site-dependent analysis was not used. Section 3 describes the basis for specifying vibratory ground motion for design use.

e. Soil-supported Structures

Structures important for safety are founded on existing rock material exposed by excavation. The foundation support materials will withstand pressures imposed by appropriate loading combinations without failure (Appendix B.2).

4.2.4.2.2 Seismic System Analysis - Basins and Related Structures

Seismic system analyses applicable to basins, vaults, and related structures are discussed in the following paragraphs and Appendix B.4.

a. Seismic Analysis Methods

Hydrodynamic effects were a main consideration in analysis of vaults and tanks; specifically, cladding vault, fuel unloading basin, and fuel storage basins. Because the mathematically precise procedure for analysis is very complex, a simplified approach based on References 5 through 8 was used.

When a tank containing fluid of weight W is accelerated in a horizontal direction, a certain portion of the fluid behaves similarly to a solid mass in rigid contact with the wall. This mass exerts a maximum horizontal force directly proportional to the maximum acceleration of the tank bottom. Acceleration also causes another portion of the fluid to respond as

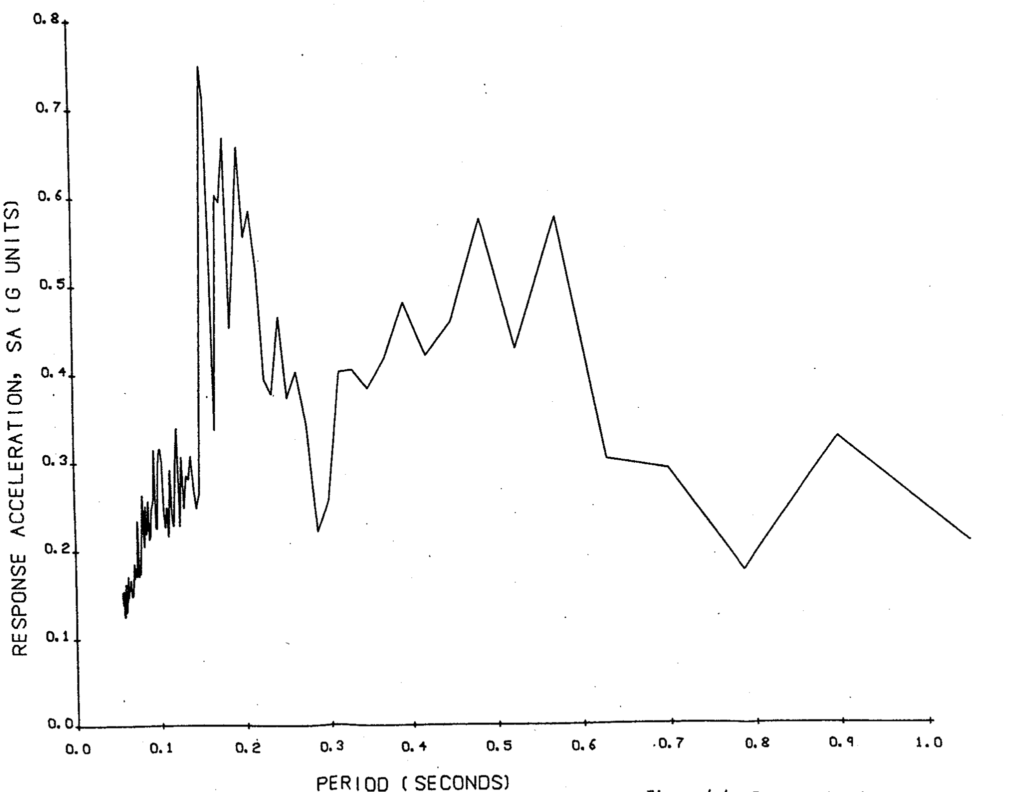


Figure 4-4. Response Acceleration Spectrum - Morris Operation - Main Building, Ground Motion, Damping Ratio - 0.005

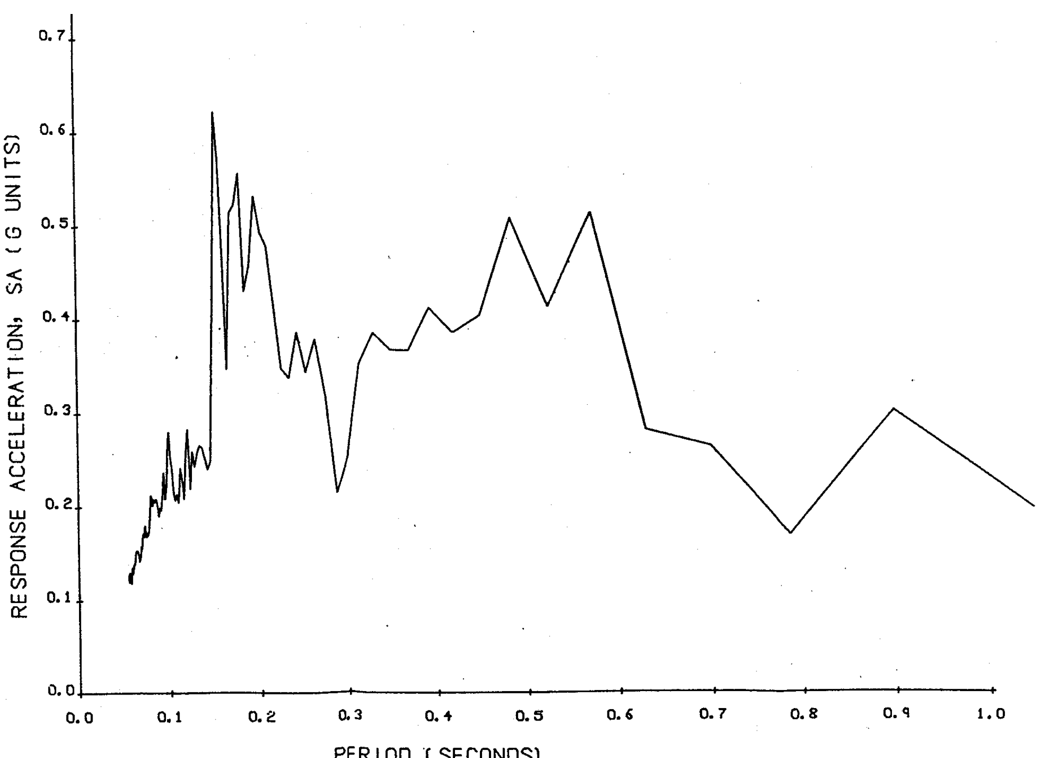


Figure 4-5. Response Acceleration Spectrum - Morris Operation - Main Building, Ground Motion, Damping Ratio - 0.010

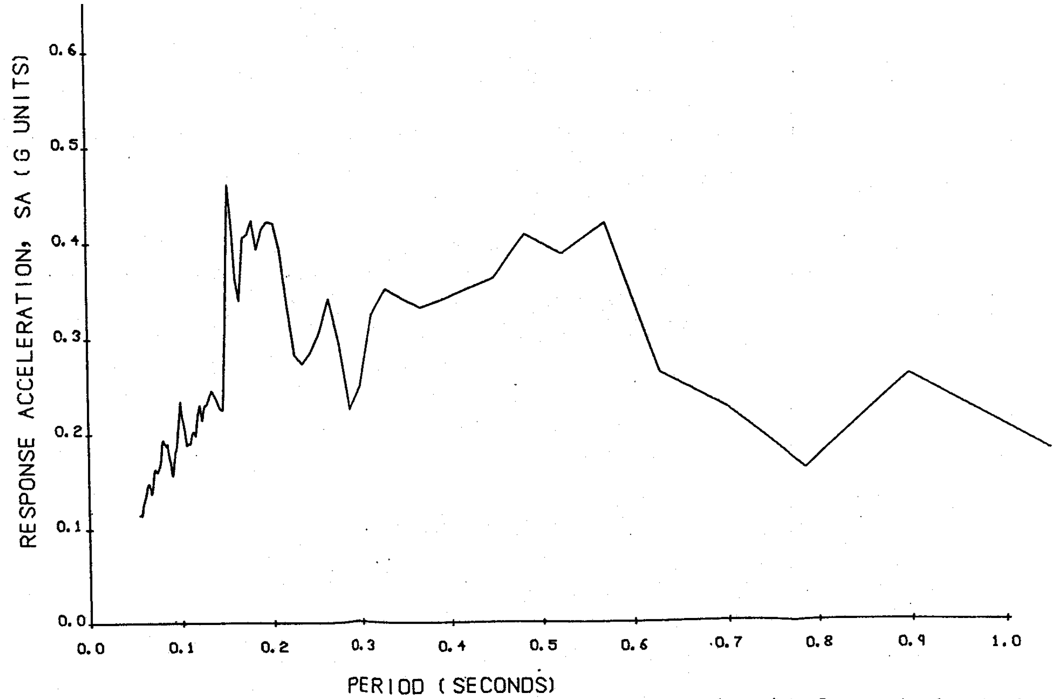


Figure 4-6. Response Acceleration Spectrum - Morris Operation - Main Building, Ground Motion, Damping Ratio = 0.02

though it were a solid oscillating mass flexibly connected to the walls. The maximum amplitude of the mass relative to the walls determines both maximum vertical displacement of the water surface (slosh height) and horizontal force exerted on the walls.

Figure 4-7 provides dynamic constants (aspect ratios) used in determining period and magnitude of sloshing. In this figure, alpha is the ratio of twice the height to average width of the tank.

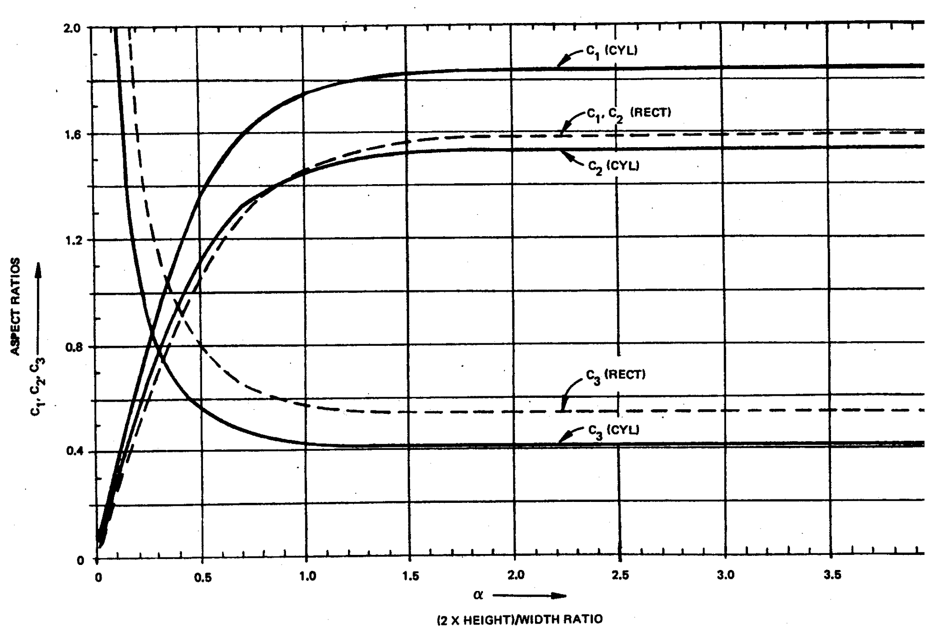


Figure 4-7. Hydrodynamic Constants for Rectangular and Cylindrical Tanks

b. Rocking and Translational Response Summary

Because underground vaults and tanks are embedded in sound rock, lateral soil pressures on these elements are negligible. An evaluation of vaults and tanks (section a. above) was made on the basis of a shearing stress of 330 psi in the rock. Resulting deformations in the rock and concrete were used to calculate stresses. Special attention was given to points of stress concentration caused by cavities behind the concrete and to localized deformations at corners. Distortion was considered, caused by the discontinuity of rock at cavity sides and bottom of the cavities, and stresses in the vaults were calculated on the basis of resulting deformations.

Stresses were most severe at corners of thick walls of short span and where interior walls are formed into outer walls. Stresses in concrete walls were found to be less than allowable stresses in concrete or steel.

Periods of sloshing for vessels and tanks are given below.

Element	Period of Sloshing (Seconds)
Cladding Vault	3.7
Fuel Unloading Pit	2.2
Fuel Storage Basin I	3.5
Fuel Storage Basin II	3.9

Rocking and translational loads in the basket and grid system are transferred through the grid to walls of the fuel storage basin. An analysis was performed to determine if basin walls and liner can safely sustain maximum load combinations of the basket and grid system and water mass in the basin. The following stresses in the basin walls were found to be less than allowable stresses of concrete or steel:

(1) Bearing stresses at the base of the wall due to the support mechanism of the fuel storage system.

(2) Peripheral or punching shear at the base of the wall due to the support mechanism of the fuel storage system.

(3) Shear-friction of concrete in the wall; a crack is assumed to occur along the shear path. Relative displacement can be resisted by friction maintained by shear-friction reinforcement available across the potential crack.

(4) Stress due to skin-friction of the bearing plate (wedge) on the basin liner.

c. Methods Used to Couple Soil with Seismic System Structures

Cladding vault, cask unloading basin, and fuel storage basins are deeply embedded in rock. Consequently, they are assumed to be rigid and move with the rock.

d. Development of Floor Response Spectra

Floor response spectra are the same as those discussed in Section 4.2.4.2.1.

e. Differential Seismic Movement of Interconnected Components

Allowable stresses for extreme loads are 90% of yield strength. (In design of the fuel storage system, allowable stresses of 1.5 times AISC allowable stresses were used.)

f. Use of Constant Vertical Load Factors

No constant vertical load factors are used for structures, systems and components. The method of analysis used for both vertical and horizontal directions is the response spectrum method. Induced forces, moments and stresses due to motions in vertical and two horizontal directions are combined by the square root of the sum of the squares technique.

g. Seismic Restraint of Overhead Cranes

Overhead cranes that could potentially fall into the fuel unloading basin or fuel storage basins have seismic retainer attachments, or are designed otherwise to prevent dislodging during a seismic event.

4.2.4.2.3 Seismic Acceleration and Response Spectra - Fuel Storage System

a. Response spectra for the fuel storage basket and grid system were derived as follows:

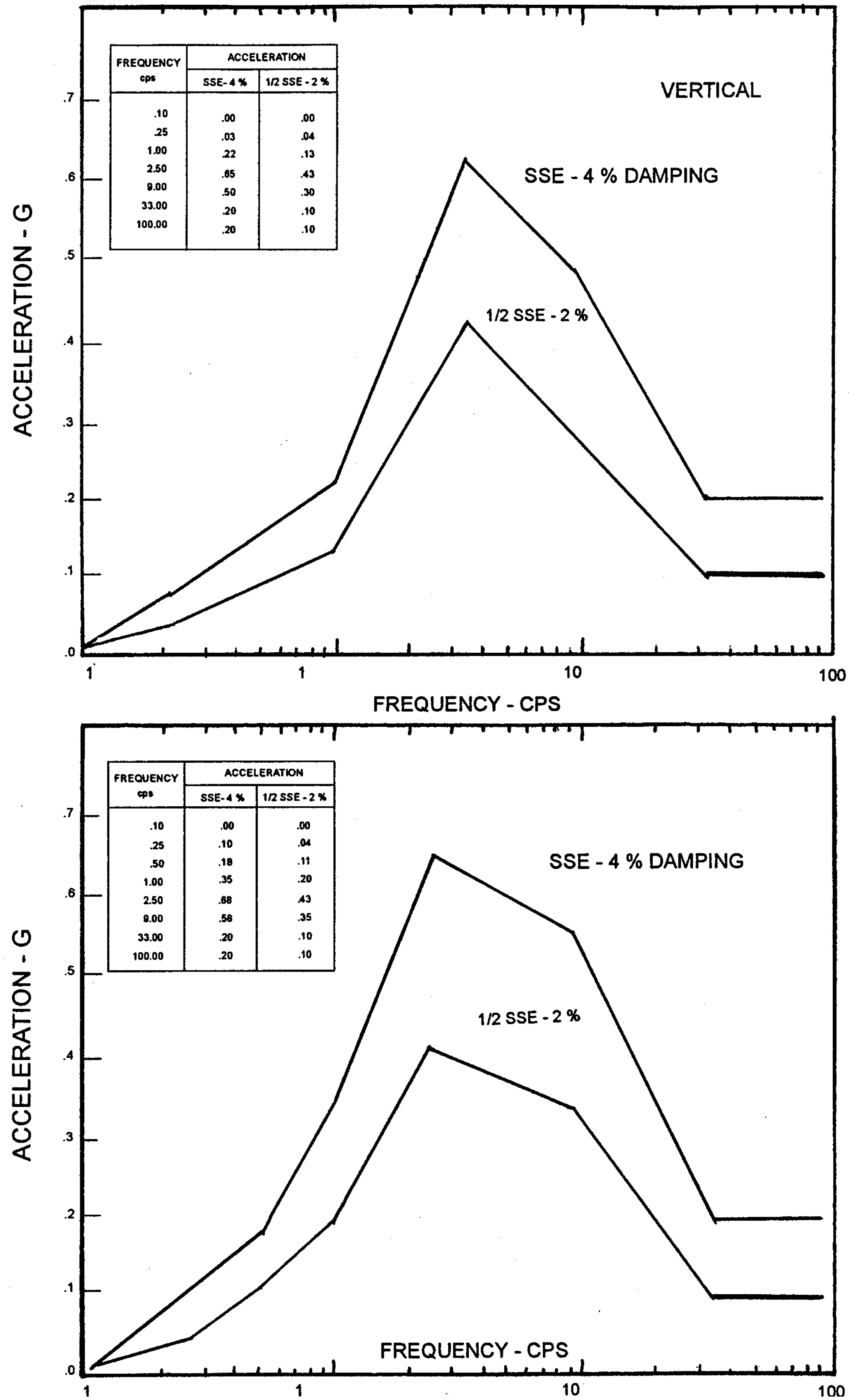
(1) Horizontal and vertical component design response spectra are scaled to a maximum horizontal ground acceleration of 0.20 G for SSE at 4% damping as specified in Regulatory Guides 1.60 and 1.61.

(2) Horizontal and vertical component design response spectra are scaled to a maximum horizontal ground acceleration of 0.10 G for 1/2 SSE at 2% damping as specified in Regulatory Guides 1.60 and 1.61.

A plot of these spectra is shown in Figure 4-8.

b. Peak vertical acceleration of the response spectra for the basket and grid system occurs at a frequency of 3.5 cps. The fundamental frequency is 0.68 cps.

c. Damping values used for design and maximum earthquake dynamic analyses of the basket and grid design shall be (from Regulatory Guide 1.61) 2% (1/2 SSE) and 4% (SSE) for welded steel structures.



FREQUENCY cps	ACCELERATION SSE-4 %	ACCELERATION 1/2 SSE - 2 %
.10	.00	.00
.25	.03	.04
1.00	.22	.13
2.50	.65	.43
9.00	.50	.30
33.00	.20	.10
100.00	.20	.10

FREQUENCY cps	ACCELERATION SSE-4 %	ACCELERATION 1/2 SSE - 2 %
.10	.00	.00
.25	.10	.04
.50	.18	.11
1.00	.35	.20
2.50	.68	.43
9.00	.58	.35
33.00	.20	.10
100.00	.20	.10

Figure 4-8. Vertical and Horizontal Design Response Spectra for Nuclear Power Plants

4.2.4.2.4 Seismic System Analysis - Fuel Storage System

a. Seismic Analysis Methods

In the seismic analysis a detailed mathematical model of the fuel storage baskets and support grid was subjected to horizontal and vertical design response spectra by the use of a computer system (SAP IV). The same mathematical model was used for both static and dynamic analyses.

The analysis used to obtain seismic response of the mathematical model is based on standard equations of motion for damped linear systems. Matrix equations were used to find the lowest natural frequencies, corresponding mode shapes of the system and response spectrum.

The SAP IV program calculates maximum responses in each of the lowest modes based on the spectra (accelerations) in the x, y and z directions. Total response for displacements and stress resultants is calculated as the square root of the sum of the squares of the modal maximum responses.

Seismic responses were obtained for N-S, E-W and vertical directions of storage baskets and grid. Degrees-of-freedom at the tops of the basket were "slaved" to six "master" baskets by partitioning 270 baskets into six groups. Seismic response of the lowest six modes was considered. Primary participation was derived from the first two modes.

Analyses used to obtain vertical dead load stresses and displacements were based on the same model as described above, except static loads were applied. The model was also subjected to two sets of static loads at 1.0G corresponding to N-S and E-W directions. A fourth load condition approximated a static equivalent analysis of the response spectra by applying horizontal loads at 0.6G and vertical loads at 0.2G.

b. Natural Frequencies and Response Loads

Frequencies and periods of vibration of basket modules and bottom holding grid are listed below for the first six modes.

MODE	FREQUENCY (HERTZ)	PERIOD (SEC)
1	5.99	0.167
2	7.59	0.132
3	8.71	0.115
4	9.97	0.100
5	10.05	0.100
6	10.32	0.097

c. Procedures Used to Lump Masses

Spent fuel storage baskets and grid were idealized as a finite element model consisting of over 1,300 nodal points and over 4,000 flexural beam-column elements. The grid was assumed to be on rollers on the basin floor and in axial contact with the wall at two adjacent sides of the basin. Basket modules were modeled as an equivalent cantilever beam connected to the grid by four artificial beam-type elements representing the holddown device. A segment of mathematical model used in the analysis is shown in Appendix B.

Material and section properties for 12 basic elements were determined. In most elements these properties were extracted directly from the American Institute of Steel Construction (AISC) tables on steel sections. In other cases these properties were derived from combined shapes, built-up sections or castings. (See Appendix B.)

4.2.4.2.5 Seismic Subsystem Analysis

a. Determination of Number of Earthquake Cycles

Structures and equipment are designed on the basis of ground motion response spectra defined previously. Design of such structures and equipment is not controlled by fatigue because most stresses and strains occur only a small number of times. Full design strains from earthquakes and accidents occur too infrequently and with too few cycles to require a fatigue design basis for these structures.

b. Root Mean Square Basis

The total maximum value of any response quantity Q (shear, moment, deflection stress and acceleration) is based on the absolute sum, or on probability considerations, by the square root of the sum of the squares procedure according to the following equation:

$$Qmax = [(Q_1\ max)^2 + (Q_2\ max)^2 + (Q_3\ max)^2 + ... + (Q_n\ max)^2]^{1/2}$$

4.2.5 Combined Loads

4.2.5.1 Criteria

Stress levels for structures and equipment shall be limited to allowable stresses set forth in applicable codes, without allowance for short-term loading. Stresses arising from seismic motion in both vertical and horizontal directions shall be added to stresses arising from other applicable loadings. No significant concrete cracking shall occur as a result of design loading conditions. For maximum seismic ground motion or tornado wind conditions, combined stresses may approach but shall not exceed yield stresses.

4.2.5.2 Compliance

In general, concrete sections are designed so that failure would occur by yielding of the reinforcement rather than by crushing of the concrete. Where calculations indicated that a structure or component would be stressed beyond the yield point an analysis was made to determine its energy absorption capacity to ensure it exceeds the energy input from the initiating condition. In addition, such designs were reviewed to ensure any resulting deflections or distortions would not prevent performance of functions essential to continued confinement of radioactive materials and would not impair proper functioning of other structures and components from a safety point of view.

4.2.5.2.1 Loads - Definitions of Terms and Nomenclature

a. Normal Loads

Normal loads are those encountered during normal facility operation. They include:

D = Deadloads, or related internal moments and forces, including any permanent equipment loads.

L = Live loads, or related internal moments and forces, including any movable equipment loads and other loads which vary with intensity and occurrence, such as soil pressure.

T_o = Thermal effects and loads during normal operating conditions based on the most critical transient or steady state condition.

b. Severe Environmental Loads

Severe environmental loads are those that could be encountered infrequently during the life of the facility. Included in this category are:

E = loads resulting from the design earthquake

W = loads resulting from the specified design wind.

c. Extreme Environmental Load

Extreme environmental load is the load that is credible but highly improbable. It is:

W_t = loads resulting from design tornado, including wind velocity pressures, pressure differential and tornado-generated missiles, where applicable.

d. Abnormal Loads

Abnormal loads are those generated by a postulated accident, e.g., cask drop. They include:

T_a = Thermal loads resulting from an accident condition; specifically, this shall include design of fuel storage basins for thermal loads resulting from boiling basin water (212 °F), which could occur under certain conditions due to loss of basin cooling.

P_a = Pressure loadings resulting from an accident condition.

e. Other Definitions

u = section strength for concrete structures that is required to resist design loads and based on methods described by the American Concrete Institute in ACI 318.

s = section strength for structural steel based on elastic design methods, and allowable stresses against which calculated actual stresses are compared, are to be taken as 35/36 times allowable stresses defined by AISC Steel Construction Manual, Seventh Edition, Appendix A for 36,000 psi yield strength steel.

The yield strength for 304 stainless steel is used as 35,000 psi at 0.2% offset and a modulus of elasticity of 2.9×10^7. Allowable stresses for elements directly in the lifting load train are based on a safety factor of 5/1 on yield.

Y = section strength for structural steel required to resist design loads taken as 90% of yield strength. Allowable stresses of 1.5 times AISC allowable stresses are used, which are equal to or less than 90% of yield strength.

4.2.5.2.2 Load Combination and Acceptance Criteria for Concrete Structures

a. Load combinations used for normal operating conditions are:

(1) $u = 0.9D + 1.9E$

(2) $u = 0.75\ (1.4D + 1.7L + 1.7T_o)$

(3) $u = 0.75\ (1.4D + 1.7L + 1.9E + 1.7T_o)$

(4) $u = 0.9D + 0.75\ (1.9E + 1.7T_o)$

(5) $u = 1.4D + 1.7L + 1.9E$

b. Load combinations used for factored load conditions are:

(1) $u = D + L + T_o + W_t$

(2) $u = D + L + T_a$

4.2.5.2.3 Load Combinations and Acceptance Criteria for Steel Structures

a. Load combinations used for normal operating conditions are:

(1) $s = D + L$

(2) $s = D + L + 0.5E$

(3) $s = D + L + W$

(4) $1.5s = D + L + T_o$

(5) $1.5s = D + L + T_o + 0.5E$

(6) $1.5s = D + L + T_o + W$

b. Load combinations used for factored load conditions are:

(1) $Y = D + L + T_o + E$

(2) $Y = D + L + T_o + Wt$

(3) $Y = D + L + T_a$

(4) $Y + D + L + T_o + 1.5 P_a$

c. Local yielding or buckling due to tornado winds and missile loadings is allowed unless this results in excessive release of radioactive materials to the environs.

4.2.6 Subsurface Hydrostatic Loadings

4.2.6.1 Criteria

Subsurface hydrostatic loading shall be considered in analysis of below-grade structures.

4.2.6.2 Compliance

Subsurface water is present at the interface between below-grade structures and surrounding rock, at least at the points of intersection with identified perched water zones. Lateral flow rates through rock are rather slow but are sufficient for hydraulic pressure head to accumulate outside below-grade structures. Magnitude of the pressure head varies with time and seasonal changes but only within the range of upper perched water zone level variations. This hydrostatic load is combined with other loads described in Section 4.2.5.

4.2.7 Basin Water Cooling

4.2.7.1 Criteria

Means shall be provided to maintain water temperature less than 200 °F (93.3 °C).

4.2.7.2 Compliance

Basin water is cooled by a system described in Section 5.5.3.

4.3 SAFETY PROTECTION SYSTEMS

4.3.1 General

There are no site-related factors sufficiently unusual to require protection systems or special design considerations beyond those normally required for a facility of this type. Operations take into account DNPS proximity to ensure cumulative effects of these operations do not constitute an unreasonable risk to public health and safety.

4.3.2 Protection by Multiple Confinement Barriers and Systems

The total confinement system consists of one or more individual confinement barriers and systems that successively minimize potential for release of radioactive material to the environment. These features also protect fuel in storage by protecting the fuel from damage and providing a favorable environment.

4.3.2.1 Criteria

Equipment and systems containing radioactive or potentially contaminated materials shall provide a continuous boundary against escape of such material and be designed to have a low probability of gross failure or significant uncontrolled leakage during the design lifetime.

Secondary confinement barriers such as vaults, ventilation system, etc., shall be designed and constructed to contain results of primary system failure, under conditions that may have initiated such failure, without loss of required integrity and to continue operation for the maximum anticipated period of stress.

Storage vaults and basins shall be designed and constructed for low probability of gross failure or uncontrolled leakage, with means provided to monitor leakage and preclude transport of radioactive materials to underlying aquifers. For lined structures containing radioactive or potentially contaminated liquids, leak detection and empty-out means shall be provided between liner and structure so that release of radioactive material to the environs can be avoided by pumping leakage back into storage, effecting repairs where leaks can be located and are accessible, or installing additional facilities in the event repair is not feasible. Water systems shall be designed to prevent accidental removal of water from basins by any means to less than a safe level. Basin water level shall be indicated and alarmed (low water alarm) in the CAS/SAS.

4.3.2.2 Compliance

All criteria described above have been satisfied; refer to Section 5.

4.3.3 Building Ventilation

4.3.3.1 Criteria

Radioactive material in building ventilation exhaust shall be reduced to levels **A**s **L**ow **A**s **R**easonably **A**chievable (**ALARA**) before being released to the environs. Special venting lines and enclosures shall be employed when necessary, to confine airborne radioactive particulate materials.

4.3.3.2 Compliance

Principal methods used to meet these criteria include:

a. Generation: Airborne radioactive material may originate from; preparation of contaminated equipment for disposal; and from operation of low-activity liquid waste treatment systems. Other than these principal sources and minor H-3 and Kr-85 leakage from fuel in storage, no other significant source exists[1]. These activities (other than fuel storage) can be suspended on short notice whenever higher than prescribed levels of radioactive materials are detected in the ventilation air exhaust stream. The waste evaporator system is designed to limit radioactive material in its effluent.

b. Confinement: The building ventilation system utilizes pressure differentials to maintain air flow paths to exhaust all ventilation air through the filter system and discharge stack. Special venting systems and special enclosures may be employed to confine airborne particulates from cask venting, decontamination activities, or similar sources to the filter - discharge stack system. The ventilation system is designed for all credible normal or anticipated off-normal conditions.

c. Release: Most of ventilation air is passed through a sand filter of demonstrated capability for removing particulate matter, and released through a 300 foot high discharge stack. Two streams are filtered through HEPA filters before release to the stack.

4.3.4 Protection by Equipment and Instrumentation

4.3.4.1 Criteria

Equipment and instrumentation shall be provided to monitor radiation and other parameters of operation, and to perform related control functions in accordance with the following:

a. Equipment and systems shall be set and adjusted to alarm and/or initiate action such that specified limits are not exceeded as a result of normal or abnormal occurrences.

b. Redundancy and independence shall be provided to a degree sufficient to ensure that no single failure of an instrument or equipment item can result in loss of control functions.

c. Equipment shall be designed to permit inspection, testing, and maintenance.

d. Monitoring of important systems and functions during normal operations and under anticipated off-normal or accident conditions is performed.

4.3.4.2 Compliance

Equipment is designed to permit inspection, maintenance, and periodic testing of functions to specified parameters. Temporary removal of single items of equipment from service has no safety significance.

Instrumentation is provided to ensure proper operation or notification of the failure of systems. Instrumentation is designed or specified to standards of known reliability.

Alarms that indicate a set point has been exceeded are annunciated in the CAS/SAS. Alarms with safety significance sound locally as well as in the CAS/SAS.

4.3.5 Nuclear Criticality Safety

4.3.5.1 Criteria

Every reasonable precaution is taken to ensure a criticality incident does not occur. Design controls are utilized and complemented by administrative control.

4.3.5.2 Compliance

The design of the spent fuel storage system includes the following controls to preclude a criticality incident:

a. Initial analyses were made in sufficient detail to demonstrate that criticality control concepts considered (e.g., control of geometry) were feasible under all credible conditions. Additional detailed nuclear criticality safety evaluations of the final design were made by qualified experts in the field to ensure final dimensions and other factors safety margins were adequate to prevent a criticality incident. Additional detailed analyses required to confirm the final design are included in Appendices A.10, B.5 and B.15.

b. In the derivation of subcritical limits, the k_{eff} permitted for the most reactive credible conditions was specified as 0.95 at a 95 percent confidence level[2].

Operation of the spent fuel storage facility includes the following administrative controls to preclude a criticality incident:

a. Safety evaluation, review and approval of operating procedures related to design control parameters.

b. Verification of nuclear fuel parameters for fuel scheduled to be stored at GE-MO.

c. Verification of fuel identity for fuel received at GE-MO for storage.

d. Maintenance of fuel storage location records.

e. Specific fuel and cask handling procedures when these tasks are performed.

f. Personnel training.

Independent reviews and audits are utilized to determine adequacy of nuclear safety control provisions and effectiveness of implementing activities.

4.3.6 Radiological Protection

4.3.6.1 Criteria

Radiation and radioactive contamination conditions at GE-MO are controlled to provide protection of personnel health and safety at all times. Emphasis is placed on minimizing both individual exposures and total exposure (man-Rem) to **A**s **L**ow **A**s **R**easonably **A**chievable (**ALARA**).

During normal operations, including anticipated occurrences, the annual dose equivalent to any person located beyond the OCA boundary does not exceed 25 mRem to the whole body, 75 mRem to the thyroid and 25 mRem to any other organ as a result of either planned discharges or direct radiation from the facility.

Any person located at or beyond the nearest boundary of the OCA will not receive a dose greater than 5 Rem to the whole body or any organ from a design basis accident.

4.3.6.2 Compliance

Criteria are satisfied through the following design features and operational practices:

a. Confining radioactive materials to prescribed locations.

b. Clearly defining areas in which significant radiation or contamination levels exist.

c. Applying special provisions and appropriate procedures to assure personnel safety.

d. Applying rigorous surveillance, housekeeping, and clean-up practices.

e. Providing comprehensive personnel training in radiological safety.

Dosimeters are provided for ensuring accurate detection and assessment of personnel exposure to ionizing radiation in accordance with applicable procedures. Thermoluminescent dosimeters (TLDs) are positioned throughout the site to assess trends in background dose rates so that increases may be detected and corrective plans initiated.

4.3.6.2.1 Access Control (Controlled Areas)

Provisions have been established for controlling personnel access to areas in which radioactive material is present and are maintained to keep potential for contamination spread and exposure to radiation **ALARA**. This is accomplished by maintaining a series of access control barriers with increasingly restrictive occupancy constraints and access authorization requirements. These access controls were designed as follows:

a. General Electric Tract: Agricultural fencing with appropriate posting encloses the tract. Routine surveillance by operating and security personnel is utilized to ensure that unauthorized occupancy for significant periods of time is prevented.

b. OCA: An double 8 ft. high chain link fence topped with razor wire surrounds the OCA in which GE-MO storage facilities are located. Personnel and vehicle access gates are locked or guarded by security personnel at all times. Vehicles, materials and equipment are checked into and out of the area following procedures that require potentially contaminated or radioactive items to be monitored and cleared before entry or exit is authorized.

c. Radiologically Controlled Area (RCA): Personnel access to RCAs in which radioactive material is stored is controlled by limiting entrance such that occupancy authorization requirements can be strictly enforced. Access to various areas is controlled by structural compartmentalization and by authorization procedures commensurate with conditions existing in the particular areas. Access to all potentially contaminated areas is limited to specific routes in accordance with prescribed procedures and clothing and monitoring requirements which are varied according to conditions. Exit from RCAs, except under emergency conditions, is by the same controlled routes through necessary clothing change stations and monitoring facilities. Routine radiation surveys of the area are performed and TLDs are posted. Equipment requiring access (e.g., basin coolers) can be decontaminated to permit maintenance.

   Materials and equipment required for operation and maintenance will be checked into the areas and will be monitored before leaving the areas in accordance with prescribed control procedures. Access for transfer of such items is limited to specific points which are provided with means for precluding unauthorized usage.

Additional requirements are utilized to limit access into areas of known or potential of high radiation levels or contamination levels. High Radiation Areas will be locked or guarded continuously.

4.3.6.2.2 Shielding

Radiation shielding is provided to control personnel exposure to **ALARA** levels.

4.3.6.2.3 Radiation Alarm Systems

Sampling and detection systems are provided that have sufficient sensitivity and scope of coverage to ensure any radiation or contamination condition of potential safety significance is accurately and promptly assessed.

Area radiation monitors (ARMs) meet the following requirements:

a. Monitors will detect gamma radiation within the range of 0.1 to 1,000 mRem/hr.

b. The high level alarm is audible locally.

c. The criticality accident alarm system meets the following requirements:

    (1) The system has gamma-sensitive monitors that meet sensitivity requirements of 10 CFR 70.24(a)(1).

    (2) The system produces a unique audible alarm.

    (3) Two detectors are provided in the storage basin area, but are not underwater.

    (4) The system is continuously functional.

    (5) The high level alarm circuits for the system are arranged in parallel so that either alarm will energize all criticality alarms.

    (6) The alarm circuit that energizes the criticality horns is designed to stay on until a manual reset in the SAS is employed to silence the horns (assuming radiation level is below trip point).

4.3.6.2.4 Effluent Monitoring

Sampling and monitoring systems and associated procedures are provided to measure radionuclides in ventilation effluent and in sample wells. Documentation and procedures for assessment of dose to the public from GE-MO effluents is contained in the GE-MO Off-site Dose Calculation Manual (ODCM).

4.3.7 Fire and Explosion Protection

4.3.7.1 Criterion

Structures, systems and components directly involved in storage of fuel shall be protected so that performance of their functions is not impaired when exposed to credible fire and explosion conditions.

4.3.7.2 Compliance

This criterion is met by using noncombustible and heat-resistant materials whenever practical throughout the facility, particularly in locations vital to functioning of confinement barriers and systems such as the basin areas and pump room. Fire detection, alarm, and suppression systems are installed in warehouse areas, and certain areas of the main building where deemed necessary. Fire extinguishers are strategically located throughout the facility. Fire training is furnished to all personnel. Fire alarms are audible in the CAS/SAS.

4.3.8 Fuel Handling and Storage

4.3.8.1 Criterion

Cask and fuel handling systems shall provide safe, reliable and efficient handling of casks and fuel.

4.3.8.2 Compliance

GE Morris Operation (GE-MO) is capable of receiving irradiated fuel bundles in shielded casks mounted on trucks or railroad cars. All major equipment such as cranes located above basin areas containing fuel are designed to ensure that components will not fall into the basin. The cask handling system has been designed to preclude a cask from being moved over fuel storage basins. Means are provided to preclude lifting a fuel bundle or a fuel storage basket to an elevation within a basin such that the shield provided by basin water is reduced to less than the prescribed depth.

Cask drop analyses have determined that energy absorption provisions in the fuel unloading basin are adequate.

Treatment of the storage basin water is adequate to minimize corrosion and prevent undue exposure of personnel.

4.3.9 Radioactive Waste Treatment

4.3.9.1 Criteria

Radioactive waste shall be stored in a manner that does not preclude retrieval and transfer off-site. Provisions shall be made for inspection and sampling of the material. No liquid radioactive waste shall be discharged from the site to the environs. Solid radioactive waste shall be disposed of in accordance with current regulations.

4.3.9.2 Compliance

Radioactive liquid waste is processed using the GE-MO or vendor radwaste system and is periodically concentrated by evaporation to reduce volume. Solid waste is disposed of via a licensed contractor.

4.3.10 Utility Systems

4.3.10.1 Criterion

Utility systems shall maintain the capability to perform safety related functions assuming a single failure.

4.3.10.2 Compliance

See Section 5.7.1.

4.4 CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

The objective of GE-MO is to prevent conditions that could result in undue risk to public health and safety by providing quality structures and reliable systems and components.

The degree of reliability that must be provided for various structures, components, and systems is determined primarily by consequences of failure of that unit. Failure of some structures, systems, or components could - if uncorrected - expose people to ionizing radiation (See Section 8). However, in a passive facility such as a fuel storage basin, repair or replacement of the failed structure, system or component can usually be accomplished long before consequences pose undue risk to public or employee health and safety. Failure of other structures, systems or components could result in an unacceptable loss of operating efficiency, but would pose no significant long or short-range risk to employees or the public.

Quality Assurance history and a list of safety related structures, systems and components are in Section 11. The quality assurance plan is NEDE-31559, "GE-MO Quality Assurance Plan”.

4.4.1 Intensity of Natural Phenomena

Monitoring of natural remarkable events is provided by local, state, and federal agencies. These events are self evident and appropriate response is documented in the GE-MO Emergency Plan.

4.5 DECOMMISSIONING

4.5.1 Criterion

The GE-MO facility shall effect decontamination and decommissioning activities to an extent consistent with existing regulatory requirements.

4.5.2 Compliance

GE-MO design provides a stainless-steel-lined basin that includes cleaning, volume-reducing waste management facilities and a ventilation sand filter that will facilitate decontamination and decommissioning operations.

Codes, guides, and standards applicable to the GE-MO facility, as noted in this report, are listed in Table 4-3.

4.6 REFERENCES

1. K. J. Eger, Operating Experience - Irradiated Fuel Storage at Morris Operation, General Electric Company, January 1972 through December 1982 (NEDO-209969B).

2. See ANSI N18,2A-1975, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants.

Table 4-3
CODES, GUIDES, AND STANDARDS

Item	Section Where Referenced
Uniform Building Code	5.3.1
ASTM C150 (Cement)	5.5.1.2
ASTM A15 (Rebar)	5.5.1.2
ASTM 262 (Stainless Steel Liner)	5.5.1.3
Regulatory Guide 1.76	4.2.1.1
Regulatory Guide 1.60	4.2.4.2
Regulatory Guide 1.61	4.2.4.2
AISC Steel Construction Manual 7th Edition, Appendix	4.2.4.2.4[a]
ACI 318	4.2.5.2.1
ANSI-N18.2A 1975	4.3.5.2
ASTM A514 (Stainless Steel)	Appendix A.8
ASTM A285 (Stainless Steel)	Appendix A.13
ASTM A240 (Stainless Steel)	Appendix A.13
AWS-ASTM (welding rod)	Appendix A.13

[a] Other references, also.