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August 5, 1980

Docket No. 50-213
B10051

Director of Nuclear Reactor Regulation
Attn: Mr. Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Reference: (1) W. G. Counsil letter to D. L. Ziemann dated January 17, 1980.

Gentlemen:

Haddam Neck Plant
Systematic Evaluation Program
Seismic Reevaluation

In Reference (1), Connecticut Yankee Atomic Power Company (CYAPCO) provided a summary of the seismic reevaluation program being conducted for the Haddam Neck Plant. Included was a breakdown of the five segments which form the program as well as a criteria document which delineated the ground rules for reassessment of the structural adequacy of safety-related structures and the reactor coolant system piping and components under SSE loads.

Recent CYAPCO efforts and telephone discussions and meetings with the Staff have resulted in additional information regarding the scope and status of the seismic reevaluation program. This information is enclosed in the following three segments:

Attachment 1 - A summary of the major portions of the program including status and schedular information.

Attachment 2 - An update of the criteria document originally provided in Reference (1).

Attachment 3 - A site-dependent response spectra for the Haddam Neck site prepared by Weston Geophysical.

Although a significant portion of the ongoing program will continue independent of NRC feedback, resolution of the site-spectra issue is a particularly high priority item.

It is our judgement that the spectra currently in use form a conservative basis upon which to conduct this reevaluation, and Staff concurrence in this regard would alleviate current concerns.

We remain available to amplify or clarify the attached information as you may require.

Very truly yours,

CONNECTICUT YANKEE ATOMIC POWER COMPANY

W. G. Counsil

W. G. Counsil
Senior Vice President

Attachment

DOCKET NO. 50-213

ATTACHMENT 1

HADDAM NECK PLANT

SYSTEMATIC EVALUATION PROGRAM

SEISMIC REEVALUATION

AUGUST, 1980

SUMMARY OF SEISMIC REEVALUATION PROGRAM

HADDAM NECK PLANT

1.0 SELECTION OF SITE SPECIFIC RESPONSE SPECTRA

A recommended site specific response spectra defining the maximum earthquake potential (safe shutdown earthquake) of the Haddam Neck site has been generated by our consultant, Weston Geophysical. The safe shutdown spectra (SSE) recommended by Weston Geophysical is shown in Figure 1. The spectra shown are based on the log-normal mean of 20 accelerograms determined to conservatively represent the seismic risk for the Haddam Neck site. The basis for this determination is detailed in Attachment 3, Site Dependent Response Spectra, Haddam Neck Site. Figure 2 shows the log-normal 84th percentile response spectra generated using the methodology documented in Attachment 3. Also shown in Figure 2 is the present licensing basis response spectra for the Haddam Neck site (0.17g Housner). CYAPCO is proceeding with SEP related seismic analyses based upon a conservative enveloping of these two spectra. The actual spectra in use for these analyses is also shown in Figure 2. The enveloping spectra chosen was developed in concert with our consultant for structural seismic analyses, URS/J. A. Blume, for use in analyses to conservatively bound what could reasonably be expected to result from on-going NRC Staff efforts in this regard. As such, the spectra in use is considered extremely conservative by CYAPCO, and is subject to revision in the less conservative direction pending the results of on-going structural seismic analyses. It is anticipated the NRC review of the enclosed site spectra will result in a more realistic assessment of the earthquake potential at the Haddam Neck site.

2.0 CRITERIA DOCUMENT

In Reference (1), CYAPCO transmitted criteria proposed for use in the analysis and evaluation of identified plant structures and the reactor coolant system piping, components, and component supports. Subsequent conversations with members of the NRC Staff resulted in a few agreed-upon changes to the criteria. These changes are reflected in the revised criteria document in Attachment 2. (Note that changes are indicated by a line in the margin.) Criteria for balance of plant piping analyses will be submitted on or about January 1981.

3.0 ANALYSES OF STRUCTURES HOUSING SAFETY RELATED EQUIPMENT

In Reference (1), CYAPCO informed the NRC of its intentions to analyze the containment, primary auxiliary building, screenwell and auxiliary feedwater pump structures. Detailed modeling of these structures is approximately 75% complete. Analyses and structural evaluations based on the enclosed response spectra should be complete by January 1981. CYAPCO expects to complete any required structural modifications prior to or during an early 1983 refueling outage. CYAPCO re-emphasizes the need for resolution of the site spectra issue prior to development of floor response spectra, modification of structures, or preparation of a final report.

4.0 REACTOR COOLANT SYSTEM ANALYSIS

In Reference (1), CYAPCO stated that the RCS piping, RCS components, and RCS component supports (including the pressurizer, pressurizer supports, and surge line piping) would be modeled and analyzed for the safe shutdown earthquake. Detailed modeling of the reactor coolant system is approximately 70% complete. Again, resolution of the site spectra issue prior to performing the final analyses or preparing a final report is appropriate.

5.0 BALANCE OF PLANT ANALYSES

Although CYAPCO has not specifically addressed semismic related evaluations beyond the scope discussed above, additional activities are planned to complete the seismic review of the Haddam Neck facility as follows.

5.1 Piping Analyses

(a) Unisolable portions of the reactor coolant system pressure boundary greater than two inch (2") nominal size.

(b) Piping required to safely shut down the plant assuming no accidents with loss of off-site power (>2" nominal size).

(c) ECCS piping; high pressure safety injection, low pressure safety injection (>2" nominal size)

CYAPCO has begun efforts aimed at accomplishing these analyses. As-built isometrics of these piping systems have been prepared and preliminary modeling of some systems has been done.

Piping analyses and any required modifications to piping or pipe supports are scheduled for completion prior to or during the early 1983 refueling outage.

5.2 Seismic Qualification of Safety Related Electrical Equipment

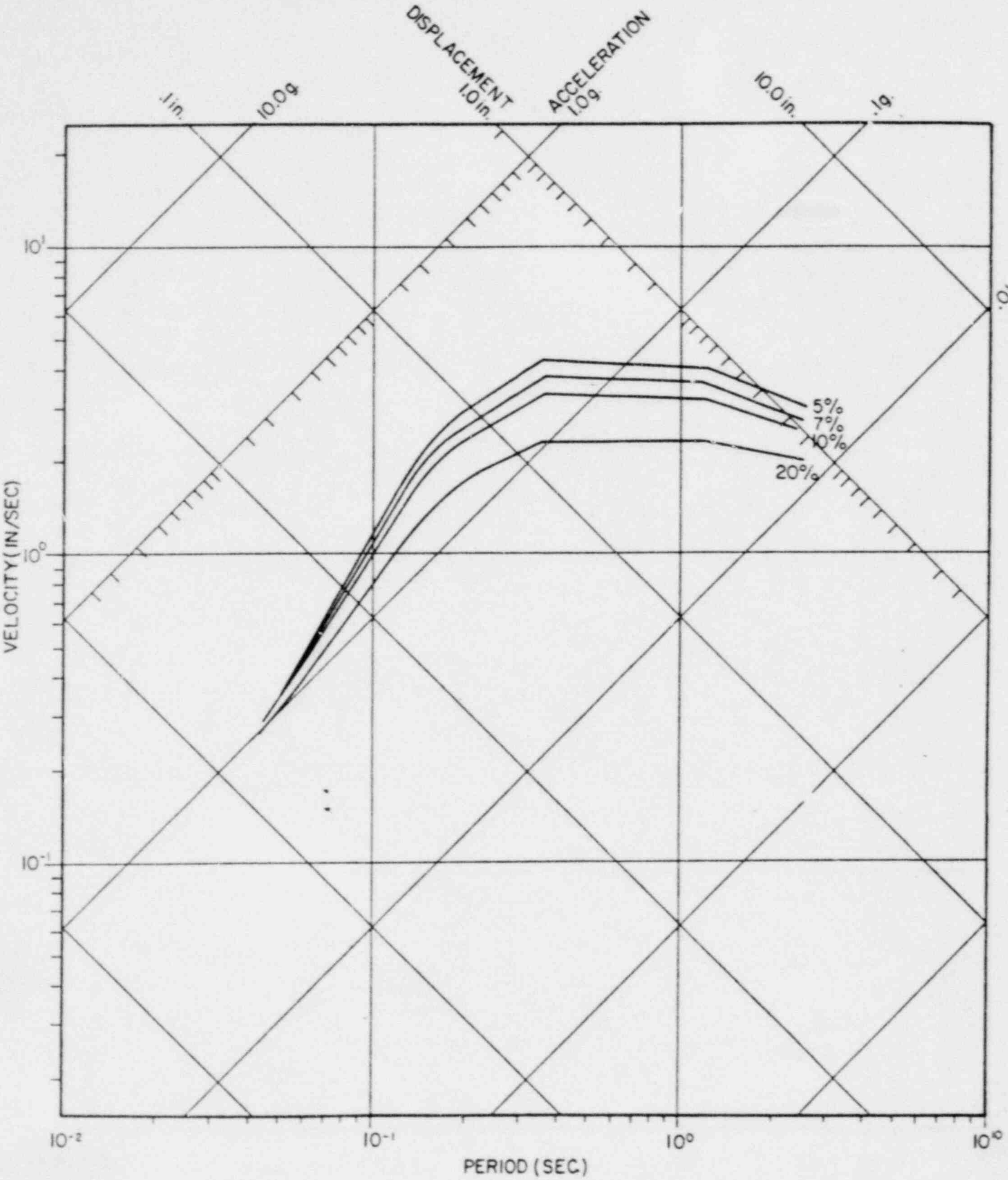
CYAPCO perceives this item as an activity which should be addressed generically, perhaps through an owners group effort. CYAPCO currently envisions working within the SEP Owners Group with interested utilities to develop a program aimed at verifying the seismic adequacy of this equipment.

CYAPCO further endorses certain Staff comments made in SECY-80-325, regarding new unresolved safety issues. Specifically, CYAPCO concurs in the Staff conclusion:

> *"That if sufficient anchoring is provided for equipment, it should function properly in the event of an earthquake, during the interim period until the overall seismic qualification issue can be resolved."*

CYAPCO has previously committed to resolve the anchorage issue on an expedited basis, and concurs that additional time is necessary to resclve the broader issue of seismic qualification.

1 3



HORIZONTAL SEISMIC DESIGN-RESPONSE SPECTRA
HADDAM NECK NUCLEAR POWER PLANT
(5,7,10, and 20% Critical Damping)

FIGURE 1

Weston Geophysical

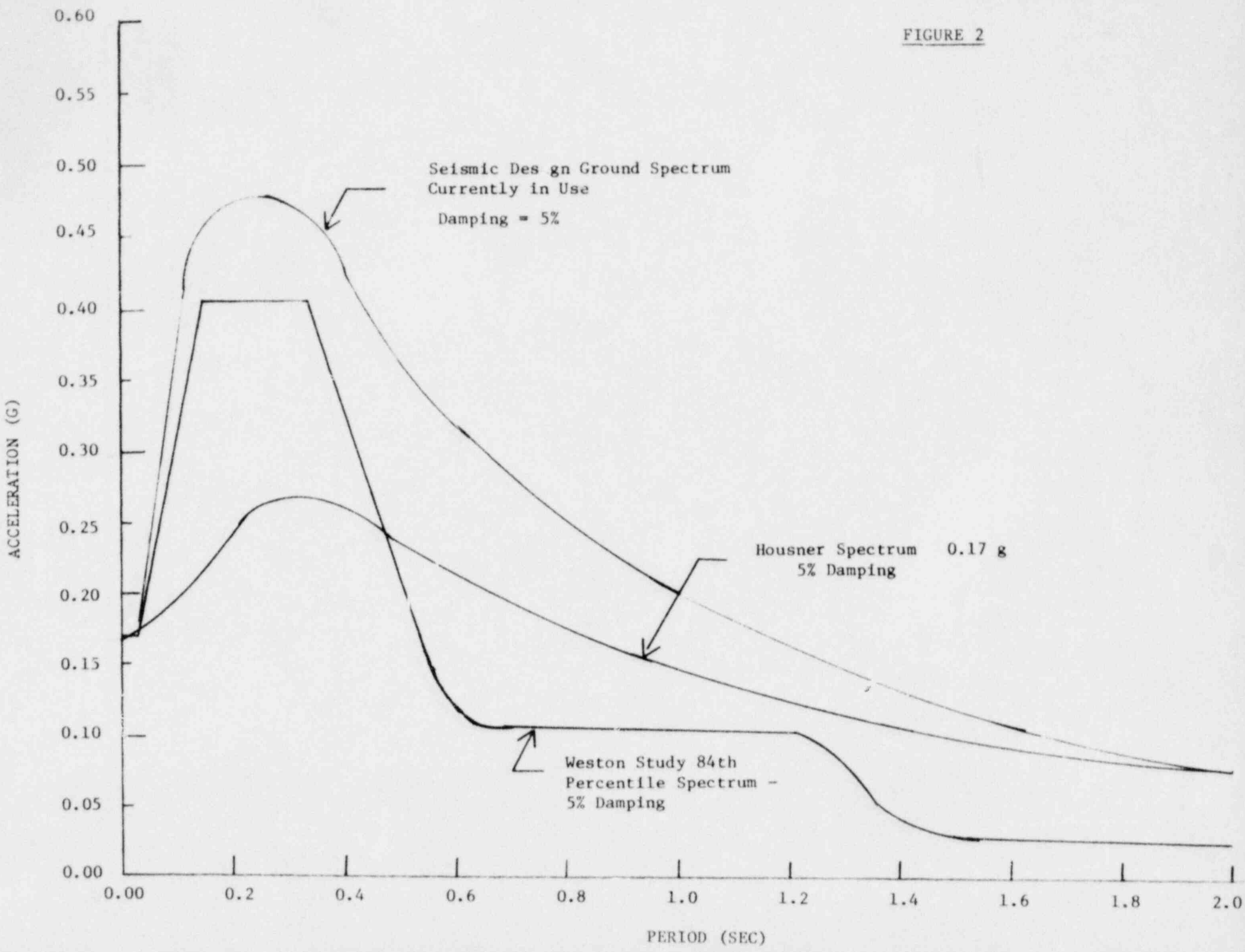


FIGURE 2
Seismic Des gn Ground Spectrum
Currently in Use
Damping = 5%
Housner Spectrum 0.17 g
5% Damping
Weston Study 84th
Percentile Spectrum -
5% Damping
ACCELERATION (G)
PERIOD (SEC)
0.60
0.55
0.50
0.45
0.40
0.35
0.30
0.25
0.20
0.15
0.10
0.05
0.00
0.00
0.2
0.4
0.6
0.8
1.0
1.2
1.4
1.6
1.8
2.0

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ATTACHMENT 2

HADDAM NECK PLANT

SEISMIC REEVALUATION PROGRAM

CRITERIA DOCUMENT

JANUARY, 1980
Revised: JULY 30, 1980

PRIMARY COOLANT LOOP SYSTEM ANALYSIS

A. SCOPE

The purpose of this document is to present the analytical methods and stress criteria which will be used for the Connecticut Yankee primary coolant loop system seismic qualification program. The program will include static analysis of the primary piping/support system for normal operating thermal, pressure, and deadweight loads along with dynamic system analysis for seismic loads. Stress criteria will be presented for the piping, supports, and primary equipment.

B. BACKGROUND

In the years since the Connecticut Yankee generating station was designed seismic analysis methods have become more rigorous and the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, has been published reflecting changes in analysis, design, and quality control techniques. The purpose of this criteria document is to establish requirements for performing the upgrading seismic analyses of the primary coolant loop system with current technology.

The original design criteria used for analysis of this plant's primary piping system is the ANSI B31.1 Code for Pressure Piping. The reactor pressure vessel, steam generator, and reactor coolant pump were designed and analyzed to the rules of the ASME Code Section VIII.

For the purposes of this document, the reactor coolant loop piping shall be considered to consist of the hot legs, cold legs, crossover legs, and pressurizer surge line. The primary equipment considered in this document consists of control rod drive mechanism, reactor vessel internals, reactor pressure vessel, steam generator, reactor coolant pump, and pressurizer. The supports covered by the criteria in this document are those for the reactor pressure vessel, the steam generator, reactor coolant pump, and pressurizer.

C. LOADING CONDITIONS

The reactor coolant loop piping, supports, and components will be analyzed for the following loading conditions:

1. Normal condition operating pressure, deadweight, and temperature.
2. SSE Condition Seismic - Safe Shutdown Earthquake (SSE) combined with operating pressure and deadweight.

D. STRESS CRITERIA

1. Piping

The piping analysis that will be performed for the Connecticut Yankee evaluation is based on the rules of the ANSI B31.1-1973 Code, the Summer 1973 Addenda.

The loading combinations and associated stress limits to be used for the piping systems which are part of the seismic qualification program are given in Table 1. The stress limits used for the SSE condition correspond to faulted condition allowables. The piping stresses are to be calculated using formulas given in ANSI B31.1-1973, 1973 Summer Addenda.

The maximum loads that the primary coolant loop piping is permitted to transmit to the pressurizer, steam generator, reactor coolant pump, and reactor pressure vessel nozzles are listed in Table 2.

Since the loop isolation valves are much thicker and stronger than the attached piping, and since valves of this design have no history of gross failure of their pressure boundaries (as long as the stresses of the piping attached to the valve remain within the limits defined in this document) the valve integrity is assured.

2. Supports

For linear type supports (i.e., reactor coolant pump hangers), the basis used for the stress criteria in this section is the AISC steel construction manual. The other supports in the primary coolant loop system can be classified as plate and shell types. The stress criteria for the plate and shell supports that is outlined in this document are based on the ASME Code, Section III, Subsection NF.

The load combinations and stress limits for both the linear and plate and shell primary equipment supports are presented in Table 3. The information presented in the table will provide allowables for normal operating and seismic conditions.

3. Components

The basis of the stress criteria outlined in this section for the primary equipment is the ASME Code, Section III, Subsection NB. The load combinations and stress limits to be used with those combinations are presented in Table 4.

E. ANALYSIS PROCEDURES

1. General Procedures

The reactor coolant loop piping/support system will be evaluated with three-dimensional static or dynamic models, depending on the load requirements, which include the effects of the equipment supports and equipment. Static analysis of the piping systems will be performed using displacement techniques with lumped parameters and stiffness matrix representations of supports. It will assume that all components and piping behave in a linear elastic manner. The methods to be used for dynamic analysis depend upon which of two techniques is chosen, response spectra or time history. Details of the two dynamic analysis procedures are presented in the following two sections.

The primary equipment that will be evaluated as part of this program shall have dynamic analyses performed in accordance with the same procedures as those presented below for piping systems. In addition to the detailed models that are developed for the evaluations of the individual components, reduced models will be produced for use in the reactor coolant loop system analysis.

Analytical representations of the primary equipment supports shall be produced for inclusion in the reactor coolant loop system model. The loads that are generated by the reactor coolant loop system model shall be used to qualify the component supports.

2. Response Spectrum Analysis Procedures

If a decision is made to perform a response spectrum seismic analysis, a three-dimensional linear dynamic analytic model of the primary coolant loop system will be developed. The model will include analytical representations of the components, component supports, and associated piping. The boundaries of the model will be defined as the component support to containment concrete interface.

The analysis will be performed assuming that the seismic event is initiated with the plant at normal full power condition. The damping values that will be used are four percent (4%) of critical for the SSE condition.[1] Since the components are supported at different

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[1]Reference: "Damping Values of Nuclear Plant Components", Westinghouse Electric Corporation, WCAP-7921-AR, May 1974.

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floor elevations within the containment building, the response spectrum in each direction shall be an envelope of the applicable floor spectra.

The analysis shall be performed with a simultaneous input of the two horizontal components and one vertical component of the earthquake. The modal response for each item of interest (e.g., force, displacement, stress) shall be obtained by the square root of the sum of the squares method.

$$R_T = \left[\sum_{i=1}^{3} R_i^{2} \right]^{1/2}$$

where: $R_i \quad \left[\sum_{j=1}^{N} R_{ij}^{2} \right]^{1/2}$

where: R_T = total combined response at a point

R_i = value of combined response of direction i

R_{ij} = absolute value of response for direction i, mode j

N = total number of modes considered.

For systems having modes with closely spaced frequencies, the above method shall be modified to include the possible effect of these modes. Combined total response for systems which have such closely spaced modal frequencies will be obtained in accordance with Regulatory Guide 1.92, or as an acceptable alternative, the following method. The groups of closely spaced modes shall be chosen such that the difference between the frequencies of the first mode and the last mode of the group does not exceed ten percent (10%) of the lower frequency. Frequency groups are formed starting from the lowest frequency and working toward successively higher frequencies. No frequency should be included in more than one group. The resultant unidirectional response for systems having such closely spaced modal frequencies shall be obtained by the square root of the sum of: (a) the sum of the squares of all modes, and (b) the product of the responses of the modes in various groups of closely spaced modes and associated coupling factors, ε. The mathematical expression for this method (with "R" as the item of interest) is:

$$R_i^2 = \sum_{j=1} R_{ij}^2 + 2 \sum_{j=1}^{S} \sum_{K=M_j}^{N_j-1} \sum_{\ell=K+1}^{N_j} R_{iK} R_{i\ell} \varepsilon_{K\ell}, \quad \text{for: } \ell \neq K$$

where: S = number of groups of closely spaced modes

M_j = lowest model number associated with group j of closely spaced modes

N_j = highest model number associated with group j of closely spaced modes

$\varepsilon_{K\ell}$ = coupling factor with

$$\varepsilon_{K\ell} = \left[1 + \left[\frac{\omega_K^1 - \omega_\ell^1}{(\beta_K^1 \omega_K + \beta_\ell^1 \omega_\ell)}\right]^2\right]^{-1}$$

and:

$$\omega_K^1 = \omega_K \left[1 - (\beta_K^1)^2\right]^{1/2}$$

$$\beta_K^1 = \beta_K + \frac{2}{(\omega_K t_d)}$$

ω_K = frequency of closely spaced mode K (rad/sec)

β_K = fraction of critical damping in closely spaced mode K

t_d = duration of the earthquake (seconds)

The analyses performed for piping and supports will not include stresses resulting from SSE induced differential motion. These stresses are secondary in nature, based on ASME Code rules for piping (NB-3653, NB-3656, F-1360) and component supports (NF-3231). The SSE being a very low probability single occurrence event, is treated as a faulted condition.

The analysis of the components subjected to seismic loading will involve several steps that are similar to those outlined above for the system analysis. A three-dimensional linear elastic analystic representation of the component is developed. The component supports and attached primary coolant loop piping shall be represented by stiffness matrices. The analysis shall be performed with the simultaneous input of three response spectra, two horizontal and one vertical. Damping values of four percent (4%) for SSE will be used. The model combination techniques outlined for the system analysis shall also be used for the component analysis.

3. Time History Seismic Analysis Procedures

In the event that time history seismic analysis is required, the following procedures shall be used. A three-dimensional elastic non-linear model of the reactor coolant loop system shall be used. The model shall include a simplified representation of the containment interior concrete structure, the components, the component supports and the attached piping. The effects of the large auxiliary piping systems (e.g., main steam, feedwater) shall be accounted for with stiffness elements in the form of linear springs or stiffness matrices. Damping for the system model shall be provided using the Rayleigh method based on a computed modal energy distribution.

The three time histories will be applied simultaneously using direct integration to determine the total response. 7/30/80

F. MODELING TECHNIQUES

The piping system components, and component supports are to be represented by an ordered set of data which numerically describes the physical system.

The spatial geometric description of the model is to be based upon the as-built isometric piping drawings and equipment drawings. Node point coordinates and incremental lengths of the members are determined from these drawings. Node point coordinates are input on network cards. Incremental member lengths are input on element cards. The geometrical properties along with the modulus of elasticity, E, the coefficient of thermal expansion, α, the average temperature changes from the ambient temperature, ΔT, and the weight per unit length, ω, are specified for each element. The supports are represented by stiffness matrices which define restraint characteristics of the supports.

A network model is to be made up of a number of sections, each having an overall transfer relationship formed from its group of elements. The linear elastic properties of the section are to be used to define the characteristic stiffness matrix for the section. Using the transfer relationship for a section, the loads required to suppress all deflections at the ends of the section arising from the thermal and boundary forces for the section are obtained. These loads are incorporated into the overall load vector.

After all the sections have been defined in this manner, the overall stiffness matrix (K) and associated load vector to suppress the deflection of all the network points is to be determined. The flexibility matrix is multiplied by the negative of the load vector to determine the network point deflections due to the thermal and boundary force effects. Using the general transfer relationship, the deflections and internal forces are then determined at all node points in the system. The support loads (F) are also computed by multiplying the stiffness matrix (K) by the displacement vector (δ) at the support point.

The models used in the static analyses are to be modified for use in the dynamic analyses by including the mass characteristics of the piping and equipment.

The lumping of the distributed mass of the piping systems is to be accomplished by locating the total mass at points in the system which will approximately represent the response of the distributed system. Effects of the equipment motion will be obtained by modeling the mass and the stiffness characteristics of the equipment in the overall system model when required. The supports are again represented by stiffness matrices in the system model for the dynamic analysis.

From the mathematical description of the system, the overall stiffness matrix (K) is to be developed from the individual element stiffness matrices using the transfer matrix (K_R) associated with mass degrees-of-freedom only. From the mass matrix and the reduced stiffness matrix, the natural frequencies and the normal modes are to be determined.

The effect of eccentric masses, such as valves and extended structures, are considered in the seismic piping analyses. These eccentric masses are modeled in the system analysis, and the torsional effects caused by them are evaluated and included in the total system response. The total response must meet the limits of the criteria applicable to the safety class of the piping.

TABLE 1

LOADING COMBINATIONS AND STRESS LIMITS FOR PIPING

LOADING COMBINATIONS	STRESS LIMITS
1. Normal:	
Design Pressure + Deadweight	$\leq S_h$
2. SSE:	
Operating Pressure + Deadweight + Maximum Potential Earthquake Loads (SSE)	$\leq 2.4\ S_h$

where: S_h = allowable stress from USAS B31.1 Code for Pressure Piping.

TABLE 2

STEAM GENERATOR LOADS INLET AND OUTLET NOZZLES

LOAD	FORCE (kips)			MOMENT (in-kips)		
	X	Y	Z	X	Y	Z
Thermal	160	100	25	1200	3000	15000
DW	20	- 20	10	70	100	150
Pressure	+1700	40	15	1500	1500	2000
SSE	250	200	120	6000	6200	7000

REACTOR COOLANT PUMP LOADS INLET NOZZLE

LOAD	FORCE (kips)			MOMENT (in-kips)		
	X	Y	Z	X	Y	Z
Thermal	100	30	30	8000	7000	3000
DW	+ 20	1	1	50	200	200
Pressure	+1700	30	20	1000	6000	3000
SSE	350	200	275	6000	13000	10000

REACTOR COOLANT PUMP LOADS OUTLET NOZZLE

LOAD	FORCE (kips)			MOMENT (in-kips)		
	X	Y	Z	X	Y	Z
Thermal	50	50	40	3000	3000	7000
DW	1	- 10	1	50	20	150
Pressure	+1400	10	10	1000	700	500
SSE	450	150	300	13000	1500	15000

NOTE: 1. All loads are ± unless noted.
2. Coordinate system.

X-Y Plane Vertical

Z By Right Hand Rule

TABLE 2 (Continued)

REACTOR PRESSURE VESSEL LOADS INLET NOZZLE

LOAD	FORCE (kips)			MOMENT (in-kips)		
	X	Y	Z	X	Y	Z
Thermal	50	100	30	5000	7000	5000
DW	1	- 20	1	200	60	800
Pressure	+1400	1	10	800	700	200
SSE	300	130	300	9000	13000	10300

REACTOR PRESSURE VESSEL LOADS OUTLET NOZZLE

LOAD	FORCE (kips)			MOMENT (in-kips)		
	X	Y	Z	X	Y	Z
Thermal	60	150	30	1000	4000	20000
DW	1	- 20	1	75	100	800
Pressure	1500	5	5	70	900	400
SSE	500	90	160	1600	14000	7000

PRESSURIZER SURGE NOZZLE

LOAD	FORCE (kips)			MOMENT (in-kips)		
	X	Y	Z	X	Y	Z
Thermal	3	7	7	1200	1000	400
DW	+ 30	1	1	15	10	35
SSE	3	5	5	250	350	350

NOTES: 1. All loads are ± unless noted.
2. Coordinate system.

X-Y Plane Vertical

Z By Right Hand Rule

TABLE 3

LOADING COMBINATIONS AND STRESS LIMITS FOR SUPPORTS

LOADING COMBINATION	LINEAR TYPE SUPPORTS LIMITS[3]	PLATE AND SHELL SUPPORTS LIMITS
$P + D + T_D$	Working Stress[4]	$P_m \leq 1.0\ S_m$ $P_m + P_b \leq 1.5\ S_m$
$P + D + T_O + E^1$	Within lesser of $\frac{1.2\ F_y}{F_t}$ or $\frac{0.7\ S_u}{F_t}$ times working limits[4]	$P_m \leq 1.2\ F_y$[1] $P_m + P_b \leq 1.8\ F_y$[2]

1 Not to exceed $0.7\ S_u$

2 Not to exceed $1.05\ S_u$

3 Compressive axial member loads should be kept to less than 0.9 times the critical buckling load.

4 Working stress allowables per Appendix XVII of ASME III.

NOTES:
- P = pressure
- D = deadweight
- T_D = thermal-design temperature
- T_O = thermal-operating temperature
- E^1 = SSE
- F_y = material yield strength
- F_t = allowable tensile stress per ASME Section III, Appendix XVII

TABLE 4

LOADING COMBINATIONS AND STRESS LIMITS FOR COMPONENTS

LOADING COMBINATION	STRESS LIMIT
Design Pressure + Deadweight	$P_m \leq S_m$
	$P_L\ (P_m) + P_B \leq 1.5\ S_m$
Operating Pressure + Deadweight + SSE	$P_m \leq 2.4\ S_m$[1]
	$P_L\ (P_m) + P_B \leq 3.6\ S_m$[2]

1 Not to exceed $0.7\ S_u$

2 Not to exceed $1.05\ S_u$

NOTES:
- P_m = general primary membrane stress
- P_L = primary local membrane stress
- P_B = primary bending stress
- S_m = allowable stress intensity per ASME, Section III
- S_u = ultimate stress at operating temperature

ANALYSIS AND EVALUATION OF PLANT STRUCTURES

1. Basic Approach

This section outlines criteria that form the basis for the reassessment of the structural adequacy of the safety-related structures to resist the SSE loads. The structures that will be included in the reevaluation are:

a) Containment Shell
b) Containment Internal Structure
c) Screenwall House
d) Primary Auxiliary Building
e) Service-Turbine Building Complex
f) Auxiliary Feedwater Building

All of these structures may be classified as seismic Category 1 structures except for some areas of the service-turbine building complex. The new diesel generator building, a recent addition to the plant, has been designed as a seismic Category 1 structure using currently accepted techniques and is not included in this reassessment.* All structures will be reevaluated using dynamic analyses. Where the preliminary evaluations indicate a considerable margin of safety with respect to the postulated seismic event, a simplified equivalent static procedure may be used.

The following documents establish acceptable methods, stresses and properties and are discussed in detail in the sections to follow:

USNRC Standard Review Plan - Sections 3.7.2, 3.8.3, 3.8.4

USNRC Regulatory Guides 1.60, 1.61, 1.92

ACI Codes 318-71, 349-76, 359-77

ASME B&PV Code, Section III Subsections NE, NF

AISC Specification for Design, Fabrication and Erection of Structural Steel for Buildings

Uniform Building Code - 79 Edition for Unreinforced Brick and Hollow Unit Masonry

Although the proposed criteria are essentially the current standard ones, they may subsequently be modified. If the proposed modification departs significantly from the present NRC positions, justifications will be documented to support any changes.

*The new diesel generator building was analyzed in 1974. A three dimensional space frame model was constructed and an evaluation made based on a Reg. Guide 160 input normalized to 0.17 G.

2. Time-History Motions

The seismic input has been described in terms of response spectra in Section III. The seismic input is also needed in terms of time-histories for the computation of floor response spectra as well as in the time-history analyses procedures for structural response computations.

Time-histories that will be developed for such purposes will match the design response spectra of Section III within the limits required by USNRC Standard Review Plan Section 3.7.1. The overall duration and the rise, strong motion and decay portions of the time-history will be consistent with the hypothesized SSE. If differing horizontal and vertical design response spectra are specified, the time histories will match them.

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3. Material Properties

For the determination of the strength and stiffness of the structures under the postulated seismic conditions, the material properties will be taken as *either* those specified on the contract drawings and documents *or* the average of actual material properties obtained from tests at the time of construction. In lieu of construction test data, tests on selected cores or samples from existing construction may be performed to obtain actual material properties.

Table 1 lists the specified material properties for concrete, reinforcement and structural steel in various structures.

Damping in reinforced concrete and structural steel shall be taken as 5 percent of critical except 7 percent of critical damping may be used when the stresses induced in the structure by seismic, gravity, and operating loads (see section 6.1) are high (close to allowables, see section 6.2).

TABLE 1

SPECIFIED MATERIAL PROPERTIES

I. Containment & Internals Structure

A. Reinforced Concrete

1. f'_c = 3000 psi @ 28 days (dome concrete - f'_c = 5000 psi) @ 28 days)

2. Reinforcing Steel

(a) #14 & #18 (ASTM A408)

(1) Typical fy = 50,000 psi min

(2) Foundation Mat and Exterior Wall Dowels:
fy = 40,000 psi min

(b) #11 (ASTM A-15 & A-305)

(1) Typical: fy = 40,000 psi min

(2) Exterior Wall above elev. 31'-6" and dome:
fy = 50,000 psi min

(c) #10 and smaller: ASTM A-15 and A-305, intermediate grade,
fy = 40,000 psi min

B. Structural Steel

ASTM A-36, Fy = 36,000 psi min

II. Primary Auxilary Building, Turbine-Service Building Complex and Screenwell House

A. Reinforced Concrete

1. F'_c = 3000 psi @ 28 days

2. Reinforcing Steel: ASTM A-15 & A-305, intermediate grade,
fy - 40,000 psi min.

B. Structural Steel

ASTM A-36, Fy - 36,000 psi min

C. Unreinforced Brick and Hollow Unit Masonry:

S'_m = 1500 psi

III. Auxiliary Feedwater Building

A. Structural Steel

ASTM A-36, Fy = 36,000 psi min

4. Analytical Procedures

Linear elastic dynamic analyses procedures are intended to be used for all structures. If nonlinear inelastic procedures are to be used for any structure a separate criterion will be developed for the non-linear analysis procedures and acceptance criterion.

The USNRC Standard Review Plan Section 3.7.2 shall be followed in those matters not explicitly covered by this document. The following dynamic analyses procedure may be used:

- o Response spectrum modal superposition
- o Time-history modal superposition
- o Time-history direct integration

Equivalent static procedure may also be used where justified.

4.1 Soil-Structure Interaction

Most of the structures at the CY plant are founded on rock (shear wave velocity, $V_s > 3500$ ft./sec.). Screenwell house is founded on lean concrete fill of 2 ft. to 20 ft. depth over rock. A small portion of Turbine Building is also founded on lean concrete fill of small depth over rock. A lightly loaded region of Service Building is founded on select compacted fill (soil) of about 10 ft. depth over rock.

Soil compliance effects in those portions of any structure founded on soil backfill will be considered. The seismic input for all structures will be as described in Section 2.

4.2 Structural Modeling

Dynamic structural models will be used to calculate the structural responses to the horizontal and vertical components of the ground motion. Material properties used in these models will be as defined in Section 4.3.

In general, the stiffness of reinforced concrete structural members will be calculated using gross cross-sections. Cracked sections will be used when necessary for a realistic assessment of the stiffness.

Mass calculations shall include the dead weight of the structures as well as the equipment. The mass of non-structural elements (e.g., partitioning) and small pieces of equipment (e.g., electrical cabinets) will be estimated as a uniform weight across the whole floor.

4.3 Coupling

Simplified models of the Nuclear Steam Supply System (NSSS) components will be coupled to the dynamic structural model of the Containment Internal structure. Responses at the equipment supports will be calculated for later use in NSSS qualification.

Structures which are physically connected by structural elements will be analyzed using coupled dynamic models except where it can be shown that coupling does not significantly influence relevant structural responses.

4.4 Torsion

Significant eccentricity between mass and stiffness will induce torsional response in a structure subjected to horizontal component of ground motion. Such eccentricity will be taken into account in the modeling of structures. In addition, to account for variation in location of mass and stiffness in the model and in the structure as well as possible torsional input into the structure, accidental eccentricity or equivalent will be considered.

For structures with rigid diaphragms or equivalent which are modeled by lumped mass models, accidental eccentricity shall be taken equal to 5% of the plan dimension normal to horizontal input component. Such accidental eccentricity will be additive to geometrical eccentricity that may exist at that level.

For other structures where accidental eccentricity cannot be accounted for in a simple manner, the responses to the horizontal input component shall be increased by 5% to account for the effects of accidental eccentricity.

Torsional responses shall be combined with translational responses on an absolute sum basis.

5. Floor Response Spectra

The peaks in the floor response spectra at structural frequencies are usually broadened to account for the uncertainties in these frequencies due to uncertainties in material properties, and approximations in modeling techniques and analyses procedures.

When minimum specified properties of structural materials are used in the model, the spectral peaks at structural frequencies will be broadened by 15% on each side of such frequencies.

If actual average structural material properties determined from test data are used in the models, a portion of this uncertainty is accounted for. The average material properties are usually higher than the minimum specified properties and leads to somewhat higher values for structural frequencies. In this case the spectral peaks at structural frequencies will be broadened by 5% on the high side and 15% on the low side of the structural frequencies.

Lesser peaks and valleys will be smoothed by free-hand enveloping.

6. Acceptance Criteria

6.1 Load Combination

The following load combination will be considered in evaluating the structure:

$$U = D + L' + O + E$$

where

U = total load to be resisted

L' = actual live load

O = operating temperature and pressure loads, if any

E = SSE load

D = dead weight

6.2 Allowable Stresses

The allowable stresses for reinforced concrete portions of structures will be per ACI Code 359-77 for the Containment Exterior and Internal structures and ACI Code 349-76 for other structures.

The stresses for steel portions of structures will be checked per Part 1 of AISC Specification, 1979 edition, except that the allowable stresses will be as delineated in NRC Standard Review Plan Section 3.8.3 and 3.8.4.

6.3 Structural Foundations

The structural foundations will have a factor of safety   1.1 against sliding and overturning for the following load combination:

$$U = D + L' + E$$

where U, D, L' and E are as defined in Section 6.1.

DOCKET NO. 50-213

ATTACHMENT 3

HADDAM NECK PLANT

SYSTEMATIC EVALUATION PROGRAM

SEISMIC REEVALUATION

AUGUST, 1980