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RECIP.NAME RECIPIENT AFFILIATION
EISENHUT,D.G. Division of Licensing

SUBJECT: Forwards "Seismic Qualification of Auxiliary Feedwater Sys," in response to 810210 Generic Ltr 81-14.Results of walkdown revealed existence of minor non-seismic Category 1 components.

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FLORIDA POWER & LIGHT COMPANY

September 18, 1981
L-81-405

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U.S. NUCLEAR REGULATORY COMMISSION

Office of Nuclear Reactor Regulation
Attention: Mr. Darrell G. Eisenhut, Director
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Eisenhut:

Re: Turkey Point Units 3 & 4
Docket Nos. 50-250 & 50-251
Generic Letter No. 81-14
Seismic Qualification of Auxiliary Feedwater System

Please find attached our report providing the information concerning auxiliary feedwater seismic design that was requested in Generic Letter No. 81-14, dated February 10, 1981. The Turkey Point Units 3 & 4 Auxiliary Feedwater System (AFWS) is a seismically designed system as described in the FSAR. It is designed, constructed, and maintained in a manner consistent with other safety grade systems in the plant. Our architect-engineer has verified the seismic qualification of each of the AFWS components and supporting systems.

A walkdown, as requested in your letter, was performed for those portions of the AFWS where sufficient information was not retrievable to verify its seismic qualification. The results of the walkdown, indicated that the AFWS as currently exists contains several minor non-seismic Category I components. These items, and the corrective actions taken to upgrade the system are summarized in Section IV of the attachment.

Very truly yours,

Robert E. Uhrig
Vice President
Advanced Systems & Technology

A048
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1/1

REU/JEM/ras

cc: Mr. J. P. O'Reilly, Region II
Harold F. Reis, Esquire

STATE OF FLORIDA)
) ss.
COUNTY OF DADE)

Robert E. Uhrig, being first duly sworn, deposes and says:

That he is a Vice President of Florida Power & Light Company, the Licensee herein;

That he has executed the foregoing document; that the statements made in this said document are true and correct to the best of his knowledge, information, and belief, and that he is authorized to execute the document on behalf of said Licensee.

Robert E. Uhrig

Subscribed and sworn to before me this

18 day of September, 1981

Cheryl L. Fredrick
NOTARY PUBLIC, in and for the county of Dade, State of Florida

My commission expires: Notary Public, State of Florida at Large My Commission Expires October 30, 1983 Bonded thru Maynard Bonding Agency

SEISMIC QUALIFICATION OF THE

AUXILIARY FEEDWATER SYSTEM

CONTENTS

I. INTRODUCTION

The Turkey Point Units 3 and 4 Auxiliary Feedwater System is classified as a seismic system and was included within the scope of NRC I. E. Bulletins 79-02, 79-04, 79-07, 79-14, and 80-11, and I. E. Information Notice 80-21. Although the system was not originally designed and classified as a seismic system, it was upgraded and reclassified prior to the Turkey Point Plant receiving an operating license to meet the seismic criteria imposed by the AEC at that time. This report contains the additional information requested by Generic Letter 81-14, Seismic Qualification of Auxiliary Feedwater Systems. It includes a brief system description, a discussion of the methodologies, and a list of non-seismic components.

II. SYSTEM DESCRIPTION

(Extracted verbatim from NRC Letter to FPL dated 10/16/79 and updated. Portions updated noted by a bar and asterisk in righthand margin.)

Configuration-Overall Design

The auxiliary feedwater system (AFWS) for the Turkey Point Plant (Units 3 and 4), as shown in Figure 1, consists of three steam turbine driven pumps, i.e., one pump normally aligned to each unit and the third pump is a shared standby for either unit. Each pump normally delivers 600 gpm (@ 2775 ft. head) feedwater to the three steam generators (SG) in each unit. Also, the control room operator can manually direct flow from any pump to all three steam generators of either unit. Under a design basis accident, only one pump would be required in order to cool the plant down to a condition where the RHR system can be put into operation to continue the safe plant shutdown process.

Primary water supply for the AFWS comes from the Seismic Category I condensate storage tanks (CST) of both units. Each CST has a capacity of 250,000 gallons with a minimum reserved storage capacity of 185,000 gallons of demineralized water. With this quantity of water, the licensee indicated that the unit can be kept at hot standby condition

for 15 hours and then cooled to 350°F, at which point the RHR system can be put in service, or the unit can be kept at hot standby condition for about 23 additional hours. All the manually operated valves associated with CST's are locked open. A secondary water supply comes from the non-seismic Category 1 water treatment system. An additional feedwater supply can be provided from the main feedwater system of the adjacent Units 1 & 2 (non-nuclear power plant).

Components - Design, Classification

The AFWS is designed according to seismic Category I requirements. The AFWS is classified as an engineered safety related system and its associated instrumentation and controls are designed accordingly.

Power Sources

The turbine driven pumps are supplied with steam from the main steam line of either or both units upstream of the MSIV. The operator normally selects the steam supply from the Unit which has lost its normal feedwater supply. The turbines have an atmosphere exhaust. Steam can also be supplied from the Unit having normal feedwater supply and from an auxiliary steam system connection to Units 1 & 2. The turbine driven pump steam supply line has a normally closed AC motor operated valve in series with a normally closed DC solenoid air operated valve. The pump discharge control valves are DC solenoid operated air valves.

Instrumentation and Control

Controls

The steam generator water level is manually controlled by the control room operator using either one of the DC solenoid operated air valves. A seismically installed nitrogen back-up system supplements the non-seismic instrument air supply to these valves. Local manual operation of these valves can also be performed. The AFW pump feedwater discharge rate is always greater than the turbine steam consumption when the steam *

pressure is higher than 120 psig. When the steam pressure is reduced to 120 psig, the RHR system is started and the AFW pumps are shut down.

Information Available to Operator

Low water level in the condensate storage tank will alarm and annunicate in the main control room. In addition, AFW flow indication, SG water level, and control valve position indication are provided in the control room.

Initiating Signals for Automatic Operation

All three AFW pumps will automatically start by any of the following signals from either Unit:

(a) safety injection

(b) low-low water level in any of the three steam generators

(c) loss of voltage on both 4160V buses

(d) loss of both main feedwater pumps.

Any one of these signals will also automatically open the normally closed motor operated and air operated valves in series which isolate the main steam line from the steam supply header of each AFW pump turbine. Air to operate the AFW control valves to the steam generators is supplied when the steam supply valves commence opening. The AFWS can also be started manually in the control room or from the local station.

In accordance with NUREG 0578 and 0737, the following AFW system modifications are in process:

(a) design and installation of a safety-related initiation and flow indication system

(b) design and installation of a qualified lube oil cooling system for the AFW pumps

(c) replacement of two AC operators on the AFW steam admission valves with DC operators for each unit

(d) addition of redundant steam supply lines to the AFW pump turbines

(e) addition of redundant discharge piping from the AFW pumps

(f) addition of redundant safety grade condensate storage tank level indication.

III. SEISMIC METHODOLOGY

Attachment A is a reproduction from the FSAR for Turkey Point Units 3 and 4, describing the seismic criteria and methodology used to qualify the majority of the Class I (seismic) structures, equipment and components. In addition, reproductions of applicable follow-up questions are included. Additional information on methods of qualification and scope not specifically described in the before mentioned excerpts from the FSAR is provided below.

Initially the pumps and drives were not procured to any specific criteria. They were later certified by the supplier to be capable of functioning under the imposed seismic loadings.

Class I structures were designed for an OBE, but later checked for the maximum earthquake (SSE).

Steam supply piping to the AFW turbines, suction piping from the condensate storage tanks to the AFW pumps, discharge piping from the pumps to a point downstream of the main feedwater isolation valves and AFW pump recirculation piping were considered to be within the scope of NRC Bulletins 79-02, 79-04, 79-14 and 80-11.

Branch lines to the first valve were also included.

Valves were analyzed with the piping taking into account the C. G. of valve operators when applicable.

Electrical equipment was purchased under specifications that included a description of the seismic design criteria for the plant.

Instrumentation, controls and panels supplied by the NSSS are covered under WCAP-7397-L and its supplements.

Conduit supports were installed in accordance with written procedures and based on conduit manufacturer's recommendations. Typical conduit supports have been evaluated and comply with the seismic requirements of the Turkey Point FSAR.

Transfer switches (120 VAC) mounted on the 120 VAC distribution panels were supplied by Airpax Electronics Corporation. These switches will withstand shocks of 100 G without tripping, while carrying full rated current when tested per M1L-STD-202C (Method 213) and will withstand vibrations of 10 G without tripping while carrying full rated current when tested per M1L-STD-202C (Method 201A).

IV. NON-SEISMIC CATEGORY I COMPONENTS

The following is a list of components which have been identified as presently being non-seismic Category I. The items are arranged in the order presented in Table 1 of Generic Letter 81-14 and those that are being upgraded are identified.

(1) Pump/Motors

Condensate Transfer Pump - The condensate transfer pump, located in a branch line from the AFW pump suction, acts as a pipe anchor. Piping to the pump is seismically

supported to maintain the nozzle loads on the pump nozzles to within good engineering limits. The pump is not required to function. This branch line will be disconnected from the AFW pump suction as a result of new demineralized water system modifications.

Condensate Recovery Transfer Pump - The condensate recovery transfer pump located in a branch line from the condensate transfer pump suction, acts as a pipe anchor. Piping to the pump is seismically supported to maintain nozzle loads on the pump nozzles to within good engineering limits. The pump is not required to function. This branch line will be disconnected from the AFW pump suction as a result of new demineralized water system modifications.

(2) Piping

Condenser Make-Up Line - The condenser make-up line, a branch line from the AFW pump suction does not have the required valving arrangement. Currently, a new demineralized water system is being installed which will include a new condenser make-up line. The branch line from the AFW suction will be cut and capped when the new system is placed in service.

(3) Valves and Actuators

Air Operated Vent Valve - Three-quarter (3/4) inch valves located downstream of the motor-operated steam admission valves to the AFW turbines are provided to prevent the AFW turbines from turning due to valve leakage. Failure of the valves in the open position will not prevent the AFW system from performing its required function.

(4) Power Supplies

No non-seismic power supplies were identified.

(5) Primary Water and Supply Paths

Condensate Recovery Tank - The condensate recovery tank supports the condensate recovery pump which acts as a seismic pipe anchor. The condensate recovery line will be disconnected from the AFW pump suction as a result of new demineralized water system modifications.

(6) Secondary Water and Supply

Not applicable to Turkey Point Units 3 and 4.

(7) Initiation and Control Systems

Condensate Storage Tank Level Transmitter - The condensate tank level transmitter was procured to control grade. Loss of function of the transmitter and indicator will not prevent the AFW system from performing its function. Redundant safety grade indication is currently being added.

Local Pressure Indicators - Local pressure indicators were procured to control grade (industrial grade standards). Loss of the pressure indicator (gauge) function will not prevent the AFW system from performing its required function.

Pressure Switches - Pressure switches located upstream of the AFW turbine trip and throttle valves are currently used to initiate the air supply to the normally closed turbine pressure reducing valve. The use of these pressure switches with the new high pressure AFW turbines has yet to be determined,

since the normally closed turbine pressure reducing valves will be replaced by normally open trip and throttle valves.

AFW Flow Control and Indication - AFW flow control and indication was originally procured to control grade standards. This is currently being upgraded to safety grade (Class IE).

N_2 Backup System - N_2 Backup system is provided to supplement the instrument air used for AFW control. The components of the system were procured to industrial standards and installed seismically.

(8) Structures Supporting or Housing AFW System Items

A. Turbine Building - The turbine building is not a Class I structure. However, the turbine building is a substantial steel and concrete structure with considerable inherent rigidity and resistance to the low OBE and SSE loads for the Turkey Point Plant. Portions of the AFW system piping is supported along the east side of the turbine building. These pipe supports have been analyzed for seismic loads and the portions of the turbine building to which they are attached have been analyzed for the seismic loads and are within allowable stresses. ⚠

B. Other Supporting Structures - The following structures were walked down to evaluate the current structural condition of concrete, steel and anchor bolts and to identify any readily recognized deficiencies in seismic resistance. Several minor maintenance action items

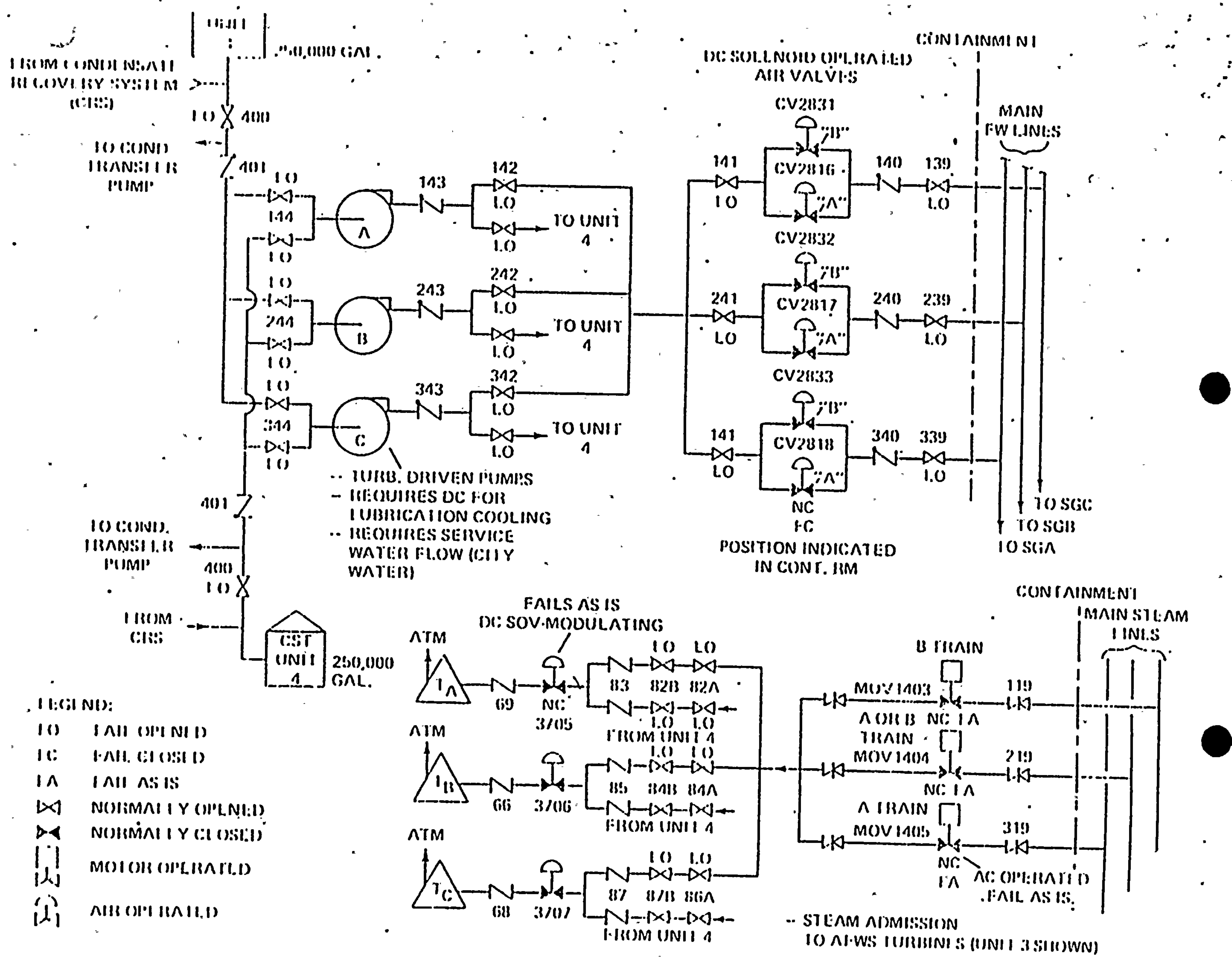
were identified and have been corrected. As granted by Generic Letter 81-14 and its enclosures, engineering judgement was used to determine the adequacy of the following structures to withstand the low OBE and SSE loads at the Turkey Point Plant.

a) Condensate Transfer Pump Foundation.

b) Condensate Recovery Pump Attachment to the Condensate Recovery Tank.

c) Condensate Recovery Tank Foundation.

d) Condensate Storage Tank Level Transmitter Attachment to Condensate Storage Tank.



Auxiliary Feedwater System

Turkey Point (Units 3 and 4)

Figure 1

<u>ATTACHMENT (A)</u>

Comprising:

FSAR Appendix 5A - Seismic Classification & Design Basis for Structures, Systems and Equipment for Turkey Point (27 pages)

FSAR Questions (5 pages)

APPENDIX 5A

SEISMIC CLASSIFICATION & DESIGN BASIS

FOR

STRUCTURES, SYSTEMS AND EQUIPMENT

FOR

TURKEY POINT

The design bases for structures at normal operating conditions are governed by the applicable building design codes. The design bases for specific systems and equipment are stated in the appropriate FSAR section. The design bases for the containment structure are contained in Appendix 5B. The basic design criterion for the maximum hypothetical accident and earthquake conditions is that there be no loss of function if that function is related to public safety.

I. Classes of Structures, Systems and Equipment

Class I structures, systems and equipment are those whose failure could cause uncontrolled release of radioactivity in excess of the established guidelines as prescribed in 10 CFR 100, those essential for immediate and long-term operation following a loss-of-coolant accident to either cool the core or reduce the containment pressure, those required to function after a loss of power occurrence or steam line break to permit a controlled NSSS cooldown, or those required for a safe shutdown. Associated with Class I structures, systems and equipment are their supports, enclosures, piping, wiring, controls, power sources and switchgear. They are designed to withstand the appropriate earthquake loads applied simultaneously with other applicable loads without

loss of function. When a system as a whole is referred to as Class I the portions not associated with the loss of function of the system may be designated as Class III as appropriate. There are no components or structures designated as being Class II. The following are Class I structures, systems and equipment:

1. Reactor Coolant System
   - Reactor vessel
   - Reactor vessel internals
   - RCC assemblies and drive mechanisms
   - Steam generators
   - Reactor Coolant pumps
   - Pressurizer and relief tank
   - All reactor coolant piping, plus any other lines carrying reactor coolant under pressure.
2. Containment System
   - Containment structure
   - Containment penetrations
   - All lines penetrating the containment, up to and including the first isolation valves.
3. Main Steam & Feedwater Lines within the Containment
4. Main Steam Safety, Isolation and Atmospheric Dump Valves
5. New Fuel Storage Facilities
6. Auxiliary Feedwater System
   - Auxiliary feedwater pumps and turbine drivers
   - Condensate storage tank
   - Steam, condensate and feedwater lines of auxiliary feedwater system

7. Emergency Diesel Generators, Day Tanks and Storage Tanks

8. Containment Polar Crane and Rail Support (Unloaded)

9. Refueling Water Storage Tanks

10. Emergency Containment Cooling and Filtering Units

11. Intake Cooling Water Systems

- Intake structure and crane supports
- Intake cooling water pumps and motors
- Intake cooling water piping, from pumps to component cooling water heat exchanger inlets.

12. Component Cooling System

- Component cooling heat exchangers
- Component cooling pumps and motors
- Residual heat removal pumps and motors (low-head safety injection pumps)
- Residual heat removal heat exchangers
- Component cooling surge tanks

13. Spent Fuel Storage Facilities

- Spent fuel pit and racks
- Spent fuel pit pump and motor
- Spent fuel pit heat exchanger

14. Safety Injection System

- Containment spray pumps and motors
- Low-head safety injection pumps and motors (residual heat removal pumps)
- High-head safety injection pumps and motors.
- Containment spray headers
- Boron injection tank

- Boron injection tank accumulator
- Accumulator tanks
- Containment recirculation sumps

15. Chemical and Volume Control System
   - Charging pumps
   - Volume control tank
   - Boric acid blender
   - Boric acid tanks
   - Boric acid transfer pumps
   - Boric acid filters

16. Fuel Transfer Tube

17. Post Accident Containment Venting System
   17 - Piping within containment and to the second valve outside containment

I. Design Bases

a) Class I Structure Design

1) Normal Operation - For loads to be encountered during normal operation, Class I structures are designed in accordance with design methods of accepted standards and codes insofar as they are applicable.

2) Hypothetical Accident, Wind and Earthquake Conditions - The Class I structures are proportioned to maintain elastic behavior when subjected to various combinations of dead loads, accident loads, thermal loads and wind or seismic loads. The upper limit of elastic behavior is considered to be the yield strength of the effective load-carrying structural materials. The yield strength for steel (including reinforcing steel) is considered to be the minimum as given in the appropriate ASTM Specification. Concrete

structures are designed for ductile behavior whenever possible; that is, with steel stress controlling the design. The values for concrete, as given in the ultimate strength design portion of the ACI 318-63 Code, are used in determining "Y", the required yield strength of the structure. Limited yielding is allowable provided the deflection is checked to ensure that the affected Class I systems and equipment (except reactor vessel internals under MHA loadings) are not stressed beyond the values given below.

The structure design loads are increased by load factors based on the probability and conservatism of the predicted normal design loads.

The Class I structures <u>outside</u> the containment structure satisfy the most severe of the following:

$$Y = 1/\phi\ (1.25D + 1.25E)$$

$$Y = 1/\phi\ (1.25D + 1.0R)$$

$$Y = 1/\phi(1.25D + 1.25H + 1.25E)$$

$$Y = 1/\phi\ (1.0D + 1.0E)$$

where Y = required yield strength of the structures.

D = dead load of structure and equipment plus any other permanent loads contributing stress, such as soil or hydrostatic loads. In addition, a portion of "live load" is added when such load is expected to be present when the unit is operating. An allowance is also made for future

permanent loads.

R = force or pressure on structure due to rupture of any one pipe.

H = force on structure due to restrained thermal expansion of pipes under operating conditions.

E = design earthquake load.

E' = maximum earthquake load.

W = wind load (to replace E in the above load equations whenever it produces higher stresses than E does).

ϕ = 0.90 for reinforced concrete in flexure.

ϕ = 0.85 for tension, shear, bond, and anchorage in reinforced concrete.

ϕ = 0.75 for spirally reinforced concrete compression members.

ϕ = 0.70 for tied compression members.

ϕ = 0.90 for fabricated structural steel.

b) Class I Systems and Equipment Design

All Class I systems and equipment are designed to the standards of the applicable Code. The loading combinations which are employed in the design of Class I systems and equipment are given in Table 5A-1.

Table 5A-1 also indicate the stress limits which are used in the design of the listed equipment for the various loading combinations.

To perform their function, i.e., allow core shutdown and cooling, the reactor vessel internals must satisfy deformation limits which

are more restrictive than the stress limits shown on Table 5A-1. For this reason the reactor vessel internals are treated separately.

Piping and Vessels

The reasoning for selection of the load combinations and stress limits given in Table 5A-1 is as follows: For the design earthquake, the nuclear steam supply system is designed to be capable of continued safe operation, i.e., for the combination of normal loads and design earthquake loading. Critical equipment needed for this purpose is required to operate within normal design limits.

In the case of the maximum hypothetical earthquake, it is only necessary to ensure that critical components do not lose their capability to perform their safety function, i.e., shut the unit down and maintain it in a safe condition. This capability is ensured by maintaining the stress limits as shown in Table 5A-1. No rupture of a Class I pipe is caused by the occurrence of the maximum hypothetical earthquake.

Careful design and thorough quality control during manufacture and construction and inspection during unit life, ensures that the independent occurrence of a reactor coolant pipe rupture is extremely remote. If it is assumed that a reactor coolant pipe ruptures, the stresses in the unbroken leg will be as noted in line 4 of Table 5A-1.

TABLE 5A-1

LOADING COMBINATIONS AND STRESS LIMITS

LOADING COMBINATIONS	VESSELS- REACTOR COOLANT SYSTEM	PIPING REACTOR COOLANT SYSTEM	PIPING OTHER CLASS I PIPING
Normal Loads	$P_m \leq S_m$ $P_L + P_B \leq 1.5\ S_m$	$P_m \leq S$ $P_L + P_B \leq S$	$\sigma_p + \sigma_g \leq S$
Normal + Design Earthquake Loads	$P_m \leq S_m$ $P_L + P_B \leq 1.5\ S_m$	$P_m \leq 1.2\ S$ $P_L + P_B \leq 1.2\ S$	$\sigma_p + \sigma_g + \sigma_{sd} \leq 1.2S$
Normal + Maximum Potential Earthquake Loads	$P_m \leq 1.2\ S_m$ $P_L + P_B \leq 1.2\ (1.5\ S_m)$	$P_m \leq 1.2\ S$ $P_L + P_B \leq 1.2\ S$	$\sigma_p + \sigma_g + \sigma_{sm} \leq Sy$ (1)
Normal + Pipe Rupture Loads	$P_m \leq 1.2\ S_m$ $P_L + P_B \leq 1.2\ (1.5\ S_m)$	$P_m \leq 1.2\ S$ $P_L + P_B \leq 1.2\ S$	Not applicable - See Pipe Restraint Criteria.

Where:

P_m = primary general membrane stress; or stress intensity

P_L = primary local membrane stress; or stress intensity

P_B = primary bending stress; or stress intensity

S_m = stress intensity value from ASME B & PV Code, Section III

S = allowable stress from USAS B31.1 Code for Pressure Piping

σ_p = longitudinal pressure stress

σ_g = gravity-caused stress

σ_{sd} = seismic stress due to design earthquake

σ_{sm} = seismic stress due to maximum potential earthquake

Sy = Minimum yield strength at operating temperature

Note (1) - This equation satisfies no loss of function criteria.

Reactor Vessel Internals

Design Criteria for Normal Operation

The internals and core are designed for normal operating conditions and subjected to load of mechanical, hydraulic, and thermal origin. The response of the structure under the design earthquake is included in this category.

The stress criteria established in the ASME Boiler and Pressure Vessel Code, Section III, Article 4, have been adopted as a guide for the design of the internals and core with the exception of those fabrication techniques and materials which are not covered by the Code. Earthquake stresses are combined in the most conservative way and are considered primary stresses.

The members are designed under the basic principles of: (1) maintaining distortions within acceptable limits, (2) keeping the stress levels within acceptable limits, and (3) prevention of fatigue failures.

Seismic Analysis of Reactor Internals

The maximum stresses are obtained by combining the contributions from the horizontal and vertical earthquakes in the most conservative manner. The following paragraphs describe the horizontal and vertical contributions.

The reactor building with the reactor vessel support, the reactor vessel, and the reactor internals are included in this analysis. The mathematical model of the building, attached to ground, is similar to that used to evaluate the building structure.

The reactor internals are mathematically modeled by beams, concentrated masses, and linear springs.

All masses, water, and metal are included in the mathematical model. All beam elements have the component weight or mass distributed uniformly, e.g., the fuel assembly mass and barrel mass. Additionally, wherever components are attached somewhat uniformly their mass is included as an additional uniform mass, e.g., baffles and formers acting on the core barrel. The water near and about the beam elements is included as a distributed mass.

Horizontal components are considered as a concentrated mass acting on the barrel. These concentrated masses also include components attached to the horizontal members since this is the media through which the reaction is transmitted. The water near and about these separated components is considered as being additive at these concentrated mass points.

The concentrated masses attached to the barrel represent the following: a) the upper core support structure, including the upper vessel head and one-half the upper internals; b) the upper core plate, including one-half the thermal shield and the other half of the upper internals; c) the lower core plate, including one-half of the lower core support columns; d) the lower one-half of the thermal shield; and e) the lower core support, including the lower instrumentation and the remaining half of the lower core support columns.

The modulus of elasticity is chosen at its hot value for the three major materials found in the vessel, internals, and fuel assemblies. In considering shear deformation, the appropriate cross-sectional areas are selected along with a value for Poisson's ratio. The fuel assembly moment of inertia is

derived from experimental results by static and dynamic tests performed on fuel assembly models. These tests provide stiffness values for use in this analysis.

The fuel assemblies are assumed to act together and are represented by a single beam. The following assumptions are made in regard to connection restraints. The vessel is pinned to the vessel support which is the surrounding concrete structure and part of the containment building. The barrel is clamped to the vessel at the barrel flange and spring connected to the vessel at the lower core barrel radial support. This spring corresponds to the radial support stiffness for two opposite supports acting together. The beam representing the fuel assemblies is pinned to the barrel at the locations of the upper and lower core plates.

Modal analysis, plus the response spectrum method[1] is used in this analysis. The modal analysis is studied by the use of a transfer matrix method.

The maximum deflection, acceleration, etc., is determined at each particular point by summing the absolute values obtained for all modes. With the shear forces and bending moments determined, the earthquake stresses are then calculated.

Figure 5A-3 shows the mathematical model studies.

The reactor internals are modeled as a single degree of freedom system for vertical eathquake analysis. The maximum acceleration at the vessel support is increased by the amplification due to the building soil interaction.

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(1) Shock and Vibration Handbook, edited by Harris and Crede, Volume 3, Chapter 50: "Vibration of Structures Induced by Seismic Waves" by George W. Housner.

Design Criteria for Abnormal Operation

The abnormal design condition assumes blowdown effects due to a reactor coolant pipe double-ended break. For this condition the criteria for acceptability are that the reactor be capable of safe shutdown and that the engineered safety features are able to operate as designed. Consequently, the limitations established on the internals for these types of loads are concerned principally with the maximum allowable deflections. The deflection criteria for critical maxima under abnormal operation are presented in Table 5A-2.

TABLE 5A-2

INTERNALS DEFLECTIONS UNDER ABNORMAL OPERATION

(Inches)

	Calculated Deflection (Preliminary)	Allowable Limit	No Loss-of-Function Limit
Upper Barrel, expansion/compression (to assure sufficient inlet flow area/and to prevent the barrel from touching any guide tube to avoid disturbing the RCC guide structure).	0.072	3	6
Upper Package, axial deflection (to maintain the control rod guide structure geometry).	0.005	1	2
RCC Guide Tube, cross section distortion (to avoid interference between the RCC elements and the guides.	0	0.0035	0.072
RCC Guide Tube, deflection as a beam (to be consistent with conditions under which ability to trip has been tested).	0.2	1.0	1.5
Fuel Assembly Thimbles, cross section distortion (to avoid interference between the control rods and the guides)	0	0.035	0.072

c) Wind and Earthquake Loads for Class I Structures, Systems and Equipment

1) Wind Force (W)

The wind loads are determined from the fastest mile of wind for a 100-year occurrence as shown in Figure 1(b) of Reference 4. This is 122 mph at the Turkey Point site. The Class I structures are designed, however, to withstand a

wind velocity at 145 mph.

In addition, Class I structures are designed to resist the effects of a tornado.

Loadings due to a tornado to be used in the design of tornado-resistant structures are as follows, the loads to be applied simultaneously:

a. Differential pressure between inside and outside of enclosed areas - 1.5 psi (bursting).

b. External forces resulting from a tornado wind velocity of 225 mph.

c. Missiles as defined in Appendix 5E.

The forces due to the wind are calculated in accordance with methods described in ASCE Paper No. 3269 entitled, "Wind Forces on Structures". Applicable pressure and shape coefficients are used. There is no variation with height or gust factor.

The forces resulting from a tornado are combined with dead loads only. Dead loads include piping and all other permanently attached or located items. There will be sufficient time after sighting a tornado to remove significant live loads such as loads on cranes. 20

Allowable stresses are limited to yield strength for structural steel and reinforced concrete. Local crushing of concrete is permitted at the missile impact zone. In all

cases, structures are reviewed to assure no loss of function for a tornado wind of 337 MPH combined with a pressure differential of 2.25 psi.

2). Earthquake Forces (E and E')

AEC Publication TID 7024, "Nuclear Reactors and Earthquakes", as amplified in this Appendix is used as the basic design guide for earthquake analysis.

Earthquake loads on structures, systems and equipment are determined by realistic evaluation of dynamic properties and the accelerations from the attached acceleration spectrum curves. These spectrum curves are corrected for the design ground accelerations. Damping factors are listed in the table below.

Earthquake forces are applied simultaneously in the vertical and any horizontal direction. The vertical component of acceleration at any level is taken as two-thirds of the horizontal ground acceleration.

DAMPING FACTORS FOR VARIOUS TYPES OF CONSTRUCTION

	% Critical Damping	
	Design Earthquake (E) (0.05g Ground Surface Acceleration)	Maximum Earthquake (E') (0.15g Ground Surface Acceleration)
Welded Steel Plate Assemblies	1	1
Welded Steel Framed Structures	2	2
Bolted Steel Framed Structures	2	2
Concrete Equipment Supports on Another Structure	2	2
Prestressed Concrete Containment Structure	2	5
Soil	5	10
Prestressed Containment Including Interior Concrete and Soil Composite	3.5	7.5
Reinforced Concrete Frames and Buildings	3	5
Composite with Soil	5	7.5
Steel Piping	0.5	0.5

d) Class III Systems and Equipment Design

Class III systems and equipment including pipe are not designed to withstand any earthquake loads. The wind loads are as per South Florida Building Code which has a basic design pressure of 37 psf. Shape Factors are applied in accordance with the Reference 4. No tornado loads are considered.

e) Miscellaneous Loads:

The units are designed for a temperature range of +30F to +95F.

No ice or snow loads are considered in the design of the various structures and equipment.

The unit is designed for a hurricane tide to an elevation of +20', with wave run up to an elevation of +22.5' on the east side of the unit.

The protection is afforded by a continuous barrier consisting of building walls, flood walls, a flood embankment as shown in Fig. 1.2-3. Door openings are protected by stop logs. The intake cooling water pumps located at the Intake Structure are protected by their elevation. Flooding from rain water is prevented by an elaborate system of storm drains, catch basins, and sump pumps. All outdoor equipment is designed for such service.

III. Method of Seismic Analysis

The method of seismic analysis for the containment structure is described in section 5.1.3.2(b). Response spectrum curves are also generated for the control building. Response curves for floors at grade and for basement are as shown in Figures 5A-1 and 5A-2.

For class I piping, floor response spectra for the connecting points are developed by the technique described in section 5.1.3.2. The pipe loop itself is also idealized as a mathematical model consisting of lumped masses connected by elastic members, and the frequencies and mode shapes for all significant modes of vibration are determined. The distance from the pipe axis to the center of gravity of the valve and operator is considered, with the mass of the valve and operator, for all motor, air, or gear operated valves. When necessary for the integrity of the piping, valve, or operation, the valve structure is externally supported. The flexibility matrix for the pipe is developed to include the effects of torsional, bending, shear and axial deformations as well as change in flexibility due to curved members and internal pressures. Flexibility factors are calculated in accordance with USAS B31.1. The spectral acceleration is determined from the response spectra.

The following equations are successively used to determine the response for each mode, maximum displacement for each mode, and the total displacement for each mass point:

(1) $$Y_n \max = \frac{R_n \, Sa_n \, D}{M_n \, w_n^2}$$

in which:

Y_n max = response of the n^{th} mode

R_n = participation factor for the n^{th} mode = $\Sigma M_i \; \phi_{in}$

Sa_n = spectral acceleration for the n^{th} mode

D = earthquake direction matrix

M_n = generalized mass matrix for the n^{th} mode = $\Sigma M_i \; \phi^2{}_{in}$ 2

(2) $$V_{in} = \phi_{in} \, Y_n \max$$

in which:

V_{in} = maximum displacement of mass i for mode n

(3) $$V_i = \sqrt{\Sigma V^2{}_{in}}$$

in which:

V_i = maximum displacement of mass i due to all modes calculated

The inertial forces for each direction of earthquake for each mode are then determined from:

$$Q_n = KV$$

in which:

Q_n = inertia force matrix for mode n

V = displacement matrix corresponding to Q_n

Each mode's contribution to the total displacements, internal forces, moments and reactions in the pipe can be determined from standard structural analysis methods using the inertia forces for each mode as an external loading condition. The total combined results are obtained by taking the square root of the sum of the squares of each parameter under consideration, in a manner similar to that done for displacements.

2 A representative number of critical piping runs have been analyzed by this method. Balance of the pipe runs have been evaluated by

(i) closeness of similarity to the runs fully analyzed,

(ii) simplicity of layout lending to a visual examination for location of seismic restraints to remove the fundamental frequency away from the resonance range, and

(iii) Static analysis based on a uniform static load equal to the peak of the pertinent response spectrum curve.

Electrical cable trays and D-C battery racks are being checked for 'g' loading obtained from the spectrum curves of the supporting floors. Motor Control Centers and Load Centers have been shaker-table tested to demonstrate no-loss-of-function capability under the maximum hypothetical earthquake. For additional information on instrumentation, see page B-37 in response to Request No. 7.3.

Mechanical and electrical equipment has been purchased under specifications that include a description of the seismic design criteria for the plant. 2

Hydrodynamic analysis of the Refueling Water Storage Tank has been performed using the methods of chapter 6 of the U.S. Atomic Energy Commission - TID 7024.

9

3. Seismic Loads

The reactor coolant loop (RCL) which consists of the reactor vessel (RV), steam generator (SG), reactor coolant pump (RCP), the pipe connecting these components, and the large component supports has been analyzed for seismic loads. The components and piping are modeled as a system of lumped masses connected by springs whose values are computed from elastic properties that are input. A simplified support model was arrived at by representing the structural support system as equivalent springs rather than as member beams and columns.

The analysis was performed by using a proprietary computer code called WESTDYN. The code uses as input, system geometry, inertia values, member sectional properties, elastic characteristics, support and restraint characteristics, and the appropriate seismic floor response spectrum for 0.5% critical damping. The floor response spectrum curves were generated at the appropriate support locations of the equipment by a time history technique described in Section 5.1.3(b). Both horizontal and vertical components of the seismic response spectrum are applied simultaneously. Two directions, namely X and Z axes, were chosen for application of the horizontal component of the seismic response spectrum. The results of the two cases were combined to determine the most severe loading condition.

With this input data, the overall stiffness matrix [K] of the three dimensional piping system is generated (including translational and rotational stiffnesses). Zero rows and columns representing restraints are deleted, and the stiffness matrix is inverted to give the flexibility matrix [F] of the system.

$$[F] = [K]^{-1}$$

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A product matrix is formed by the multiplication of the flexibility and mass matrices. This product matrix forms the dynamic matrix, [D] , from which the modal matrix is computed.

$$[D] = [F]\ [M]$$

The eigenvalues and eigenvectors representing the frequency and associated mode shape for each mode are generated using a modified Jacobi method.

$$(\omega^2 [M] - [K])\{X\} = 0$$

From this information, the modal participation factor is combined with the mode shapes and the appropriate seismic response spectrum values to give the structural response for each mode. Then the forces, moments, deflections, rotations, constraint reactions, and stresses are calculated for each significant mode. The maximum response of the system is obtained by combining the modal contributions using the root mean square method. 8

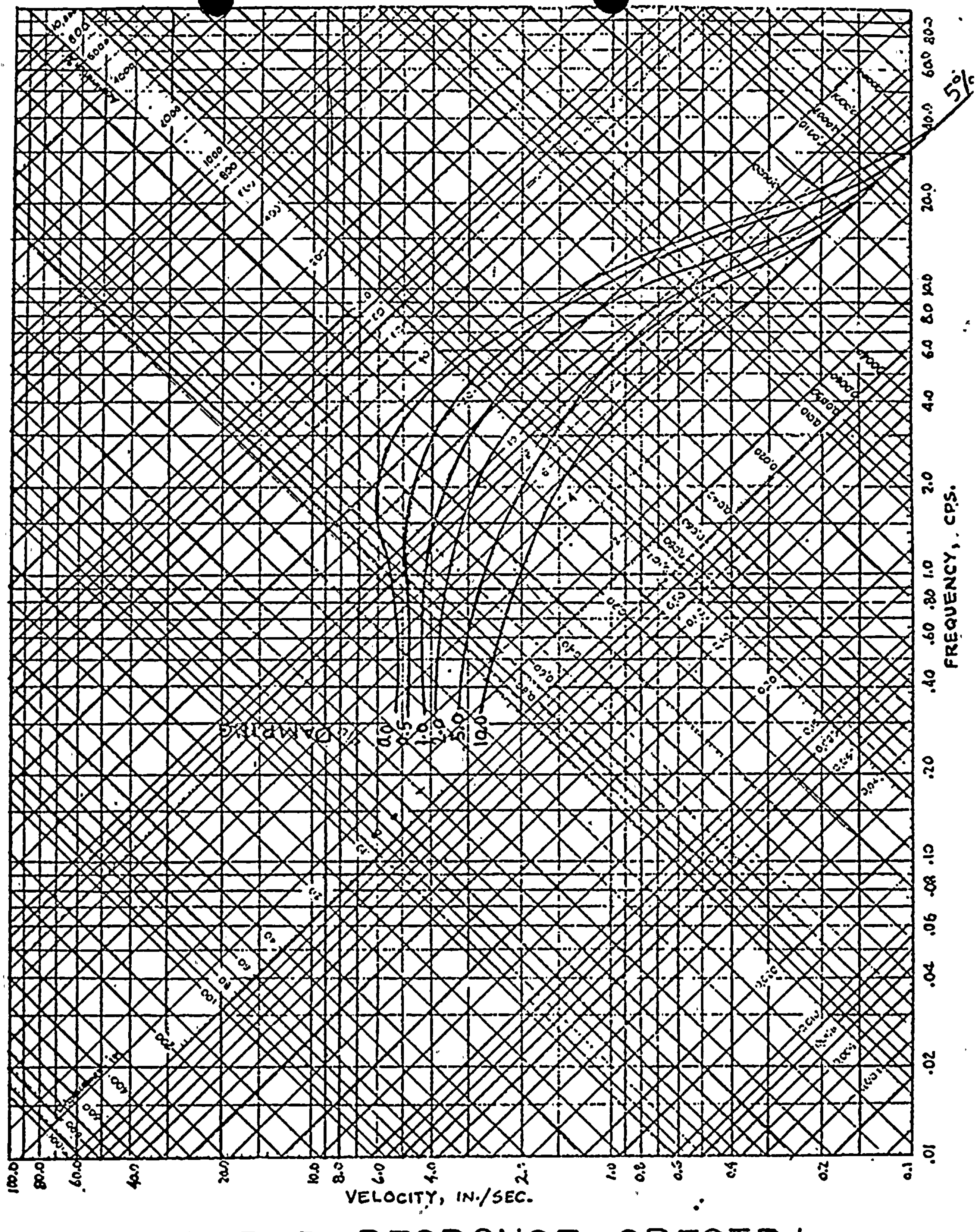
The restraints, supports, and other constraints assumed for input into the seismic computer model are given below (see Figure 5A-4 for axes orientation.). 9

Reactor Vessel	The RV is rigid.
Steam Generator	The SG at the upper support point is permitted to translate along and rotate about the X, Y, and Z axes, but translations along X and Z are resisted by the springs representing the upper support.

The SG at the lower support point is permitted to translate along and rotate about the X, Y, and Z axes, but all movements are resisted by springs representing the lower supports stiffness.

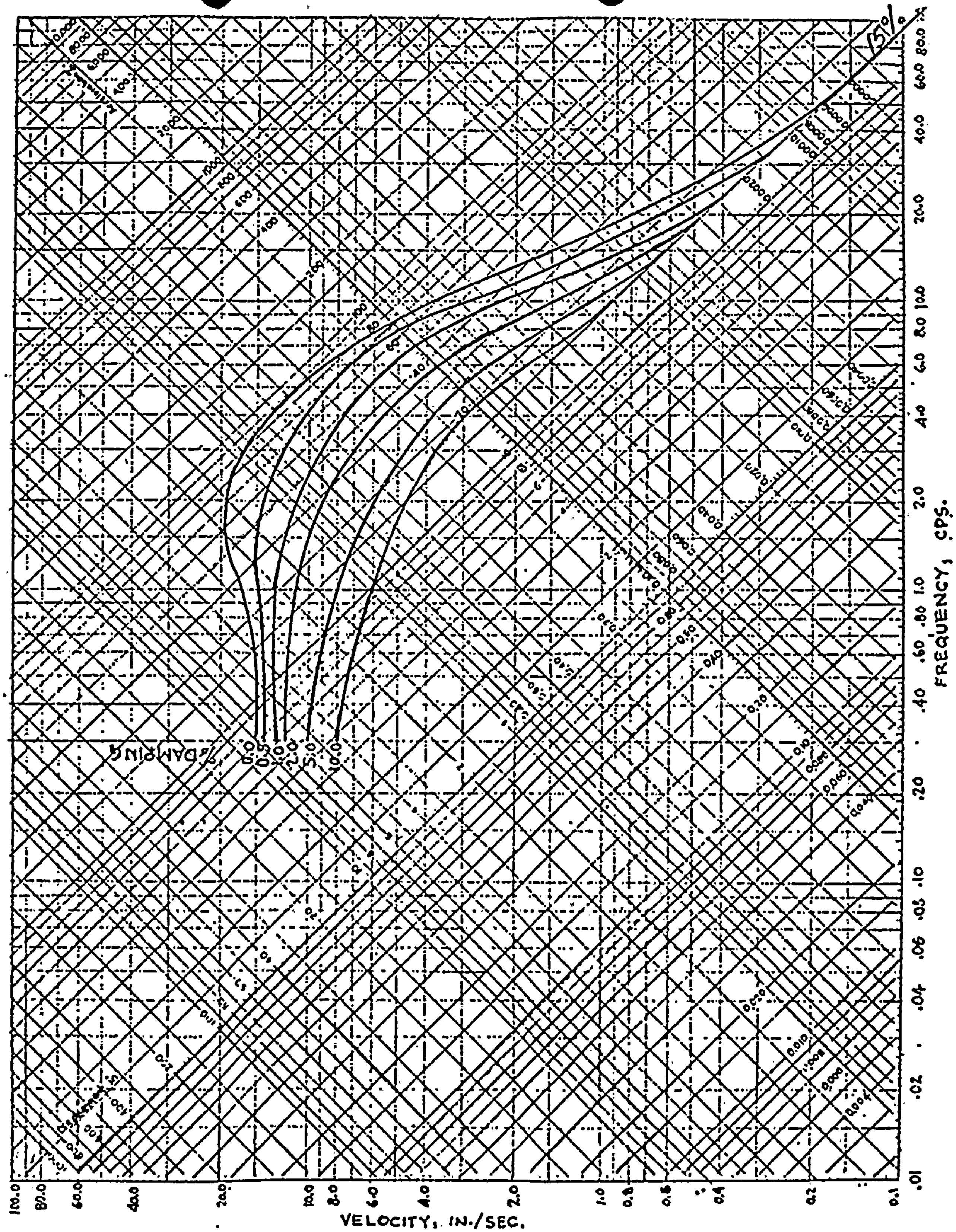
8 Reactor Coolant Pump — The RCP is permitted to translate along and rotate about the X, Y and Z axes, but all movements are resisted by springs representing the supports stiffness.

9 A summary of maximum pipe stresses is given in Table 5A-3.



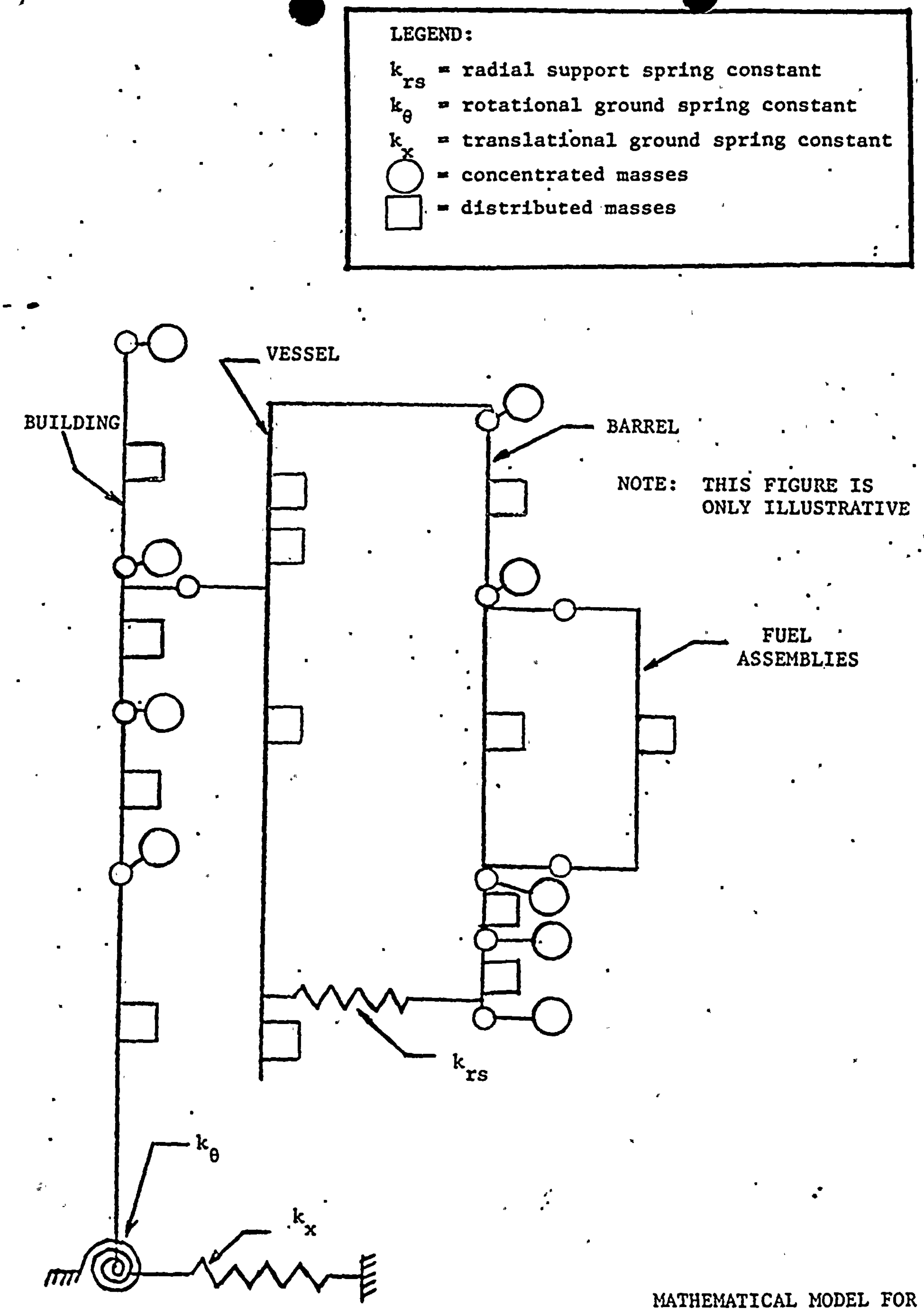
DAMPED RESPONSE SPECTRA
5% ACCELERATION

Figure 5A-1



DAMPED RESPONSE SPECTRA
15% ACCELERATION

Figure 5A-2



MATHEMATICAL MODEL FOR REACTOR VESSEL INTERNAL ANALYSIS (HORIZONTAL EXCITATION)

FIG. 5A-3

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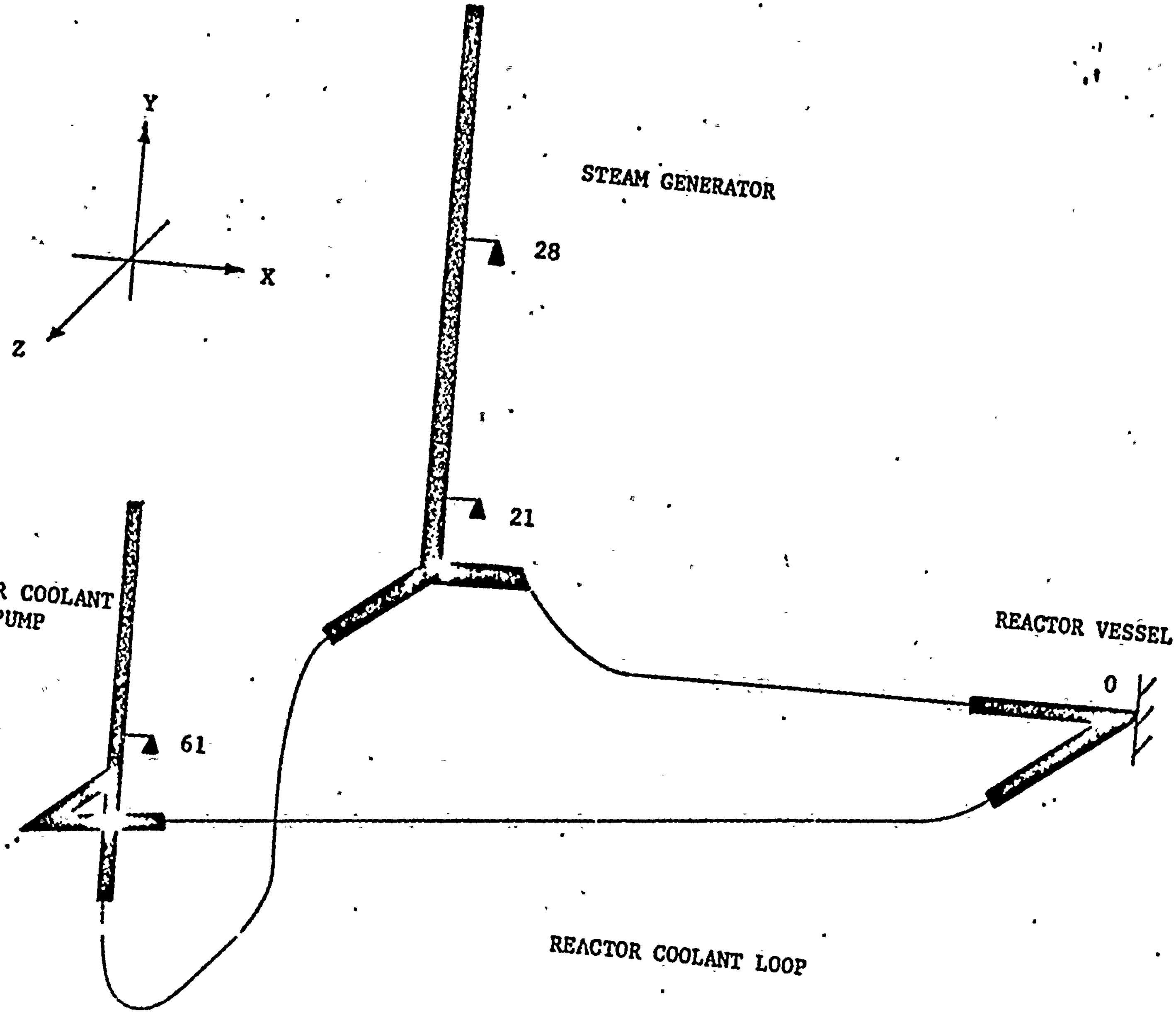

MODEL FOR SEISMIC ANALYSIS OF REACTOR COOLANT LOOP

FIG. 5A-4

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TABLE 5A-3

MAXIMUM STRESSES EXPECTED IN REACTOR COOLANT SYSTEM PIPING DUE TO THE OPERATING (.05g) EARTHQUAKE

Location	Maximum Stress (psi)
Reactor Coolant Pump Inlet	4085
Reactor Coolant Pump Outlet	3616
10 Inch Accumulator Line	3201
Steam Generator Outlet	2274
Reactor Vessel Inlet	1289
Reactor Vessel Outlet	182
Pressurizer Surge Line Connection	78
Steam Generator Inlet	71

9

Maximum Allowable Seismic Stress = 13,125 psi (This value is the result, after deadweight and pressure stresses have been subtracted from 1.2 times the material allowable stress.)

5.0 Structures

REQUEST:

5.1 In reference to Table 5A-1, which contains loading combinations and stress limits for Class I systems and equipment design, provide the following information not presently included therein:

a. Stress limits for normal reactor operating conditions plus faulted reactor operating conditions, i.e., normal + pipe rupture + design basis earthquake loads.

b. Quantitative stress limits under all loading combinations for equipment supports. Relate these limits to the requirements of paragraph N-473 of ASME Code Section III.

c. Identify the source of and the bases for the piping stress limits:

$$P_m \leq 1.2S,\ P_L + P_B \leq 1.2S, \text{ and } P_L + P_B \leq 1.2\ (1.5S)$$

RESPONSE:

a. Refer to FSAR page 5A-7 and Table 5A-1 (Rev.).

b. There is no relationship

c. Refer to ASA B31.2 1955.

REQUEST:

5.2 Identify the issue(s) of the ASME Boiler and Pressure Vessel Code Section III and addenda thereto specified for the design and fabrication of all applicable Class I components for the Turkey Point Units. Indicate any differences which exist between Units 3 and 4.

RESPONSE:

1964 and addenda thereto through October 1965. 2

REQUEST:

5.3 Failure of the bearings on a primary pump shaft could conceivably lead to the generation of missiles due to flywheel breakup. Provide the results of an analysis of the effects of applicable load combinations, including seismic loads on the bearings, and indicate the margins against failure.

RESPONSE:

Refer to answer 4.2.3 - Con Ed Docket 50-247.

REQUEST:

5.8 Appendix 5A indicates that various structures, systems and equipment, because of their special importance to public safety, are designed and built to more exacting standards than would otherwise be necessary for reliable plant operation alone. Please describe, in detail, the management reviews and approvals required in determining which portions of the plant must be of this higher classification.

RESPONSE:

The design standards and criteria selected for safety related portions of the plant were subjected to the same management review chain as shown in Figure 1.9-1 for Quality Assurance. Also, consultants (refer to FSAR page 1.7-1) were engaged. Safety Analyses Reports of other projects were used during the design as references for those evaluating the Turkey Point design.

REQUEST:

5.9 During the construction of the facility corrective concrete work was undertaken around the tendon anchorages. Provide a summary description of the outcome of this work including a discussion of any post-repair tests that were performed. What special attention will be given to monitoring for distress or failure in tendon bearing plate areas during tendon tensioning and containment structural proof testing?

RESPONSE:

Refer to reports to DRL of August 14, 1968, November 1, 1968, and February 9, 1970.

REQUEST:

5.10 Provide information on the as-built foundations regarding any unexpected foundation conditions encountered and any changes brought about by these conditions, such as changes in elevation, types of foundations, and grouting.

RESPONSE:

None.

REQUEST:

7.3 What are your seismic design bases for the reactor protection system, the emergency electric power system, and the instrumentation and controls for both the engineered safety features and the decay heat removal system? Will the systems be designed to be capable of actuating reactor trip or engineered safety feature action during the maximum peak acceleration? If a seismic disturbance occurred after a major accident, would emergency core cooling be interrupted? What tests and analyses will be performed to assure that the seismic design bases are met? What seismic specifications are employed in the instrumentation and control purchase order(s)?

RESPONSE:

The Westinghouse design bases for the protection grade equipment with respect to earthquakes is that, for design basis earthquake (DBE) or operational basis earthquake (OBE), the equipment will be designed to ensure that such equipment will not lose its capability to perform its design objective; namely, shut the unit down and/or maintain the unit in a safe shutdown condition. It is conceivable that protection grade equipment may have permanent deformation due to stresses from the maximum potential earthquake; as such, the deformation will not impede its design objective.

If a seismic disturbance occurs subsequent to an accident, the instrumentation and electrical equipment associated with emergency core cooling will not be interrupted during this disturbance.

The manufacturer states that the 4160V. switchgear, including breaker contacts, instruments and relays will withstand, and still remain operable, a seismic acceleration force of 3G in any direction.

The manufacturer of the 480V. power supply equipment made tests as follows:

1. 480V. Motor Control Center including starters, circuit breakers and relays - successful operation at accelerations of 0.5G to 1.25G at fundamental frequencies of 4 to 10 cps with shocks applied front to back and side to side. 4

2. 480V Switchgear including circuit breakers, relays and instruments - successful operation at accelerations of 0.5G to 3.0G at fundamental frequencies of 4 to 11 cps with shocks applied front to back and side to side.

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3. Power transformers withstand impacts of 4G in the vertical direction and 6G in the horizontal direction, as evidenced by impact meter records. 4

Mathematical models are not used for seismic design evaluation of instrumentation. Evaluation of such equipment for its ability to withstand the seismic condition in accordance with the design objective is done by actual vibration type testing of typical protective grade equipment. Documentation of the test program results is available in a Westinghouse proprietary document, Supplement 2 to WCAP 7397-L, Seismic Testing of Electrical and Control Equipment (WCID Process Control Equipment), E. L. Vogeding, January 1971. 2 11

No seismic specifications are employed in the instrumentation and control purchase orders. Type testing was reported as described above (WCAP 7397-1) to provide verification of the seismic design objectives.

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9.0 Auxiliary Systems

REQUEST:

9.1 Provide a P&I diagram of the intake cooling water system.

RESPONSE:

Refer to FSAR Figure 9.6-2 (new).

REQUEST:

9.2 Describe the applicable codes and standards to which the piping and components of the intake cooling water and auxiliary feedwater systems are designed.

RESPONSE:

Refer to FSAR Tables 9.6-2 (new) and 9.11-2 (new).

REQUEST:

9.3 Describe the design features which will prevent loss of the fuel pool water as a result of tornado generated winds or missiles; main turbine missiles or a dropped fuel cask. What means are provided to maintain adequate cooling of stored fuel in the event fuel pool water should be lost?

RESPONSE:

Refer to FSAR Sections 5.2 and 9.3.