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1114 CIVIL ENGINEERING BUILDING
URBANA, ILLINOIS 61801

7 August 1968

Dr. Peter A. Morris, Director
Division of Reactor Licensing
U. S. Atomic Energy Commission
Washington, D.C. 20545

Re: Contract No. AT(49-5)-2667
Oyster Creek Nuclear Power Plant - Unit No. 1
AEC Docket No. 50-219

Dear Dr. Morris:

We have reviewed the data submitted by the applicant in Amendment No. 38 concerning "Seismic Analysis Results of Feedwater Coolant Injection System." In this review we have given consideration to the fact that certain modifications are being made to permit the high pressure feedwater system to serve as an emergency core cooling system. It is clear that the system may fail to meet the requirements of Class I as a high pressure feedwater system and still be capable of supplying water for emergency core cooling, even if all parts of it do not meet Class I requirements. However, in order to investigate this problem, we have re-examined the capabilities of the various parts of the system as tabulated by the applicant in pages 5-1 to 5-12 of Amendment No. 38, and have made some simplified analyses based on the values of frequency and damping given by the applicant for the individual items in the system.

These rough calculations, made by me and reviewed by Dr. W. J. Hall, indicate that the following items, identified by the numbers given by the applicant in Amendment No. 38, in Group A, "Components Which Meet Class I Requirements" are capable of meeting the design basis earthquake of maximum ground acceleration of 0.22g:

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1. Condensate pumps
2. Steam jet air ejector intercondenser
3. Steam jet air ejector after condenser
9. Low pressure feedwater heaters
11. Feedwater pumps.

Our analysis indicates that the following items have a capability in the range of 0.11 to 0.13g maximum ground acceleration;

4. Steam packing exhauster
5. Condensate demineralizer cation tank
10. Feedwater heater drain coolers
12. Main condenser.

Our analysis also indicates that the remaining items have a capability to resist an earthquake of maximum ground acceleration of 0.15 to 0.18g;

6. Condensate demineralizer anion tank
7. Condensate demineralizer resin storage tank
8. Dilution hot water tank.

We have also looked at the group of components under the applicant's item B, "Components Which do Not Meet Class I Requirements" and verify that these have resistances in the range of 0.03g to 0.05g in general, and appear to be capable of being strengthened, in accordance with the specifications set by the applicant, for an increased capability. The applicant states in paragraph 2 on page 5-1 that such components will have a design capability no less than 0.05g ground acceleration. This implies only a minor strengthening for these elements, which in general may not be required to have a considerably higher resistance because they are in most cases parts of redundant systems, or they are parts of the system that are not required to meet emergency core cooling capabilities.

The analyses made by the applicant and those made by us are conservative in some cases. Nevertheless, it is our conclusion that the modifications of the high pressure feedwater system described by the applicant will not produce a system fully meeting Class I requirements, since a number of parts have a capability of the order of only half the maximum earthquake or just slightly greater.

It is only fair to add, however, that in our opinion the modification of the high pressure feedwater system to serve as an emergency core cooling system is well conceived and the plans to carry out the modifications are as complete and effective as seems feasible in the present stage of construction of the plant. In view of the requirements that this system will have to perform, and in view also of the redundancies involved in this part of the system, it is our opinion that the modification is a desirable and reasonable one and has a good probability of being able to perform its emergency function, even for the maximum earthquake which is considered for this plant.

Very truly yours,

N M Newmark

N. M. Newmark

bjw
cc: W. J. Hall

APPENDIX 4

Table I-4-1, FDSAR (Oyster Creek)

Principal Design Features

4 TECHNICAL DESCRIPTION OF THE FACILITY

4.1 Summary Plant Data

A summary of plant data is shown in Table I-4-1.

The remainder of this section presents a brief technical description of the power plant, its arrangement, the systems and equipment required to produce power, the auxiliary systems, and the backup safety-related systems and equipment.

TABLE I-4-1

PRINCIPAL DESIGN FEATURES

Site	
Location	Oyster Creek, New Jersey
Size of Site	Approximately 800 acres
Plant Ownership	Jersey Central Power & Light Co.
Net Electrical Output	515 MW
Reactor BWR(MARK I)	
Thermal Output, rated	1600 MW
Reactor Pressure (core exit)	1000 psig
Total Core Flow Rate	61×10^6 lb/hr
Steam Flow Rate	5.850×10^6 lb/hr
Core	
Circumscribed Core Diameter	170.55 inches
Fuel Assembly	
Number of Fuel Assemblies	560
Fuel Rod Array	7 x 7
Cladding Material	Zircaloy-2
Fuel Material	UO_2
Active Fuel Length	144 inches
Cladding Outside Diameter	0.570 inch
Cladding Thickness	0.0355 inch
Fuel Channel Material	Zircaloy-4
Control System	
Numberof Movable Control Rods	137
Shape of Movable Control Rods	Cruciform
Pitch of Movable Control Rods	12.0 inch
Control Material in Movable Control Rods	Boron Carbide
Type of Control Drives	Bottom entry, hydraulic actuated

TABLE I-4-1 (Continued)

Control System (Continued)	
Number of Temporary Control Curtains	248
Control of Reactor Power Output	Movement of control rods and variation of coolant flow rate.
Nuclear Design Data	
Initial Average Fuel Enrichment	2.10
Water/UO_2 Volume Ratio (cold)	2.38
Excess Reactivity of Clean Core (Uncontrolled) at 68°F	0.23 Δk
Total Worth of Control	0.27 Δk
Reactivity of Core with All Control Rods in	0.96 k_{eff}
Worth of Standby Liquid Control System	0.17 Δk
Reactor Vessel	
Inside Diameter	17 ft 9 in.
Overall Length (inside)	63 ft 10 in.
Design Pressure	1250 psig
Coolant Recirculation Loops	
Location of Recirculation Loops	Inside containment drywell
Number of Recirculation Loops	5
Pipe Size	26 inch
Primary Containment	
Type	Pressure absorption
Design Pressure of Drywell Vessel	62 psig
Design Pressure of Absorption Chamber Vessel	35 psig
Leakage Rate, maximum	0.5% free volume per day at 35 psig
Secondary Containment	
Type	Reinforced concrete and steel superstructure with metal siding.
Internal Design Pressure	0.25 psig
Inleakage Rate	100% free volume per day at 0.25 in. water negative pressure
Structural Design	
Seismic Resistance	0.11 g
Sustained Wind Loading	ASA design wind loadings for Ocean County, New Jersey

TABLE I-4-1 (Continued)

Station Electrical System	
Number of Incoming Power Sources	Two 34.5 kV lines, two 230 kV lines
Separate Power Sources Provided	2 startup transformers 1 auxiliary transformer 1 diesel generator 1 station battery
Reactor Instrumentation System	
Location of Neutron Monitor Sensors	In-core
Ranges of Nuclear Instrumentation	
Startup Range	Source to 0.01% rated power
Intermediate Range	0.0003% to 10% rated power
Power Range	1% to 125% rated power
Reactor Protection System	
Number of Channels in Reactor Protection System	2
Method to Prevent Unauthorized Withdrawal of Control Rods	Automatic interlocks including rod worth minimizer

Waste Disposal Systems

Liquid and gaseous waste disposed of in accordance with the requirements of 10CFR20. Solid wastes packaged for off-site storage.

Additional Engineered Safeguards - Summary of Systems and Functions	
Control Rod Velocity Limiter	In the unlikely event of a free fall rod drop from the core to limit the free fall velocity to approximately five feet per second.
Control Rod Drive Housing Support	To prevent a control rod drive mechanism from falling away from the reactor pressure vessel in the unlikely event of a failure of a drive housing.
Standby Liquid Control System	To provide a redundant, independent backup control mechanism
Flow Restrictors	A constriction in each main steam line to reduce rate of blowdown in event of a large leak from the main steam line.
2 Core Spray Systems and Automatic Depressurization System	To maintain continuity of core cooling under assumed loss of coolant accidents.

TABLE I-4-1 (Continued)

Additional Engineered Safeguards - Summary of Systems and Functions (Continued)

2 Containment Cooling Systems	To remove energy from the containment subsequent to assumed loss of coolant accident
Control of Containment Atmosphere	Provision for maintaining an inert atmosphere in the primary containment to preclude a hydrogen-oxygen reaction subsequent to a postulated coolant loss.
Isolation Valves	To effect reactor containment automatically when required.
Isolation Condenser	To avoid overheating of the reactor fuel in the event that reactor feedwater capability is lost and other normal heat removal systems which require a-c electrical power for operation are not available.
Standby Gas Treatment System	To provide a means for removal of radioactivity from reactor building air under accident conditions prior to discharge of the filtered air through the stack. It also provides a means of maintaining the reactor building at a negative pressure so that leakage is into the reactor building and thus prevents ground level release of building air under accident conditions.

APPENDIX 5

Comparison of Regulatory Guide 1.29 with the Oyster Creek Nuclear Generating Station - Supp. 6 to AM. 68, Application for FTOL, Part Two, Answers to AEC Questions.

COMPARISON OF REGULATORY GUIDE 1.29 WITH THE

OYSTER CREEK NUCLEAR GENERATING STATION

Regulatory Position	Oyster Creek Station
The following structures, systems, and components of a nuclear power plant, including their foundations and supports, are designated as Category I and should be designed to withstand the effects of the SSE and remain functional.	Class 1 - structures system 1/ and equipment whose failure could cause significant release of radioactivity or which are vital to a proper shutdown of the plant and the removal of decay heat.
a. The reactor coolant pressure boundary	All piping connections from the Reactor Vessel up to and including the first isolation valve external to the drywell. Recirculating Piping System including valves and pumps.
b. The reactor core and reactor vessel internals	The reactor core, reactor vessel internals, and the reactor vessel supports.
c. Systems or portions of systems that are required for (1) emergency core cooling, (2) post-accident containment heat removal, (3) post-accident containment atmosphere cleanup	Core Spray System Containment Spray System Standby Gas Treatment System
d. Systems or portions of systems that are required for: (1) reactor shutdown (2) residual heat removal (3) cooling the spent fuel storage pool	That portion of the Shutdown Cooling System that is required for post-incident cooling. Isolation Condenser System Fuel Pool Cooling System
e. Those portions of the steam systems of boiling water reactors extending from the outermost containment isolation valve up to but not including the turbine stop valve, and connected piping of 2-1/2 inches or larger nominal pipe size up to and including the first valve that is either normally closed or capable of automatic closure during all modes of normal reactor operation. The turbine stop valve should be designed to withstand the SSE and maintain its integrity.	This portion of the steam system has not been seismically analyzed nor designed specifically as Seismic Class I, and is therefore Seismic Class II. However, the original design and construction is such that it is the considered opinion the system satisfies Class I requirements.
f.	Not applicable.

Regulatory Position	Oyster Creek Station
g. Cooling water, component cooling, and auxiliary feedwater systems 1/ or portions of these systems that are required for (1) emergency core cooling, (2) post-accident containment heat removal, (3) post-accident containment atmosphere cleanup, (4) residual heat removal from the reactor; and (5) cooling the spent fuel storage pool.	Cooling water for the core spray system and containment spray system is supplied from the suppression chamber. The Reactor Building Closed Cooling Water System provides cooling water for the Shutdown Cooling System and the fuel pool cooling system.
h. Cooling water and seal water systems 1/ or portions of these systems that are required for functioning of reactor coolant system components important to safety, such as reactor coolant pumps.	The Reactor Building Closed Cooling Water System provides cooling water for all safety related equipment.
i. Radioactive waste treatment, handling and disposal systems 1/ except those portions of these systems whose postulated simultaneous failure would not result in conservatively calculated potential off-site exposures comparable to the guideline exposures of 10 CFR Part 100.	The Radioactive Waste Building is classified as Seismic Class II structure and the Radwaste System is classified as Class II equipment since failure of the structure and/or equipment will not cause significant release of radioactivity. (See response to Question IV-8 in Amendment 11 to the FDSAR) [illegible] as cls 1 - cat I
Systems 1/ or portions of systems that are required to supply fuel for emergency equipment.	Fuel storage tanks and associated fuel supply piping and pumps.
k. Systems 1/ or portions of systems that are required for monitoring and actuation of systems important to safety.	Reactor Pressure and Level Instrumentation Manual Reactor Control System Control Rod Position Indicating System Neutron Monitor System In-Core Neutron Monitors Area Monitors Standby Liquid Control System Instrumentation
l. The protection system	Reactor Protection System
m. The spent fuel storage pool structure, including the fuel racks.	Fuel Storage Facilities to include spent fuel and new fuel storage equipment.
n. The reactivity control systems; e.g., control rods, control rod drives, and boron injection system.	Control Rods and Drive System including equipment necessary to scram operation, Control Rod Drive Thimble Supports. Liquid Poison Systems.

Regulatory Position	Oyster Creek Station
o. The control room, including its associated vital equipment and life support systems, and any structures or equipment inside or outside of the control room whose failure could result in incapacitating injury to the operators.	Control Room (and supporting part of Turbine building)
p. Primary and secondary reactor containment.	Drywell, Vents, Torus and Penetrations, Reactor Building
q. Portions of the onsite electrical power system, including the onsite electrical power sources, that provide the emergency electrical power needed for functioning of plant features included in items 1.a. through 1.p. above.	Standby Electrical Power Systems: Station Batteries, Diesel Generators, Emergency Busses and other electrical gear and power to critical equipment including the starting transformer
r. Structures, systems, or components whose failure could reduce the functioning of any plant feature included in items 1.a. through 1.q. above to an unacceptable safety level.	Ventilation Stack, Service Water System, Isolation Valves Intake Structure
s. Category I seismic design requirements should extend to the first seismic restraint beyond the defined boundaries. Structures, systems, or components which form interfaces between Category I and non-Category I features should be designed to Category I requirements.	All other Piping and Equipment not listed under Class I has been designed as Seismic Class II.

1/ A system boundary includes those portions of the system required to accomplish the specified safety function and connected piping up to and including the first valve (including a safety or relief valve) that is either normally closed or capable of automatic closure when the safety function is required.

APPENEIX 6

Compliance with Regulatory Guide 1.48, Supp. 6 to AM. 68, Application to FTOL, Part Two, Answer to AEC Questions.

Regulatory Guide 1.48

Design Limits and Loading Combinations for Seismic Category I Fluid Systems Components

This guide establishes guidelines for design and construction codes. Oyster Creek does not strictly comply since the plant was built before the guidelines were written, but Oyster Creek was built in conformance to the best existing standards at the time of construction. The standards that were used for the design and construction codes are:

1. ASME Section I
2. ASME Section III
3. ASME Section VIII
4. Nuclear Code Cases
5. ANSI B 31.1
6. TEMA Standards

See the attachment for the design codes used for specific components.

These standards constituted the closest equivalent to the recommendations of Regulatory Guide 1.48 existing at that time and established very similar criteria.

DESIGN AND CONSTRUCTION CODES

FOR

JERSEY CENTRAL - OYSTER CREEK PROJECT

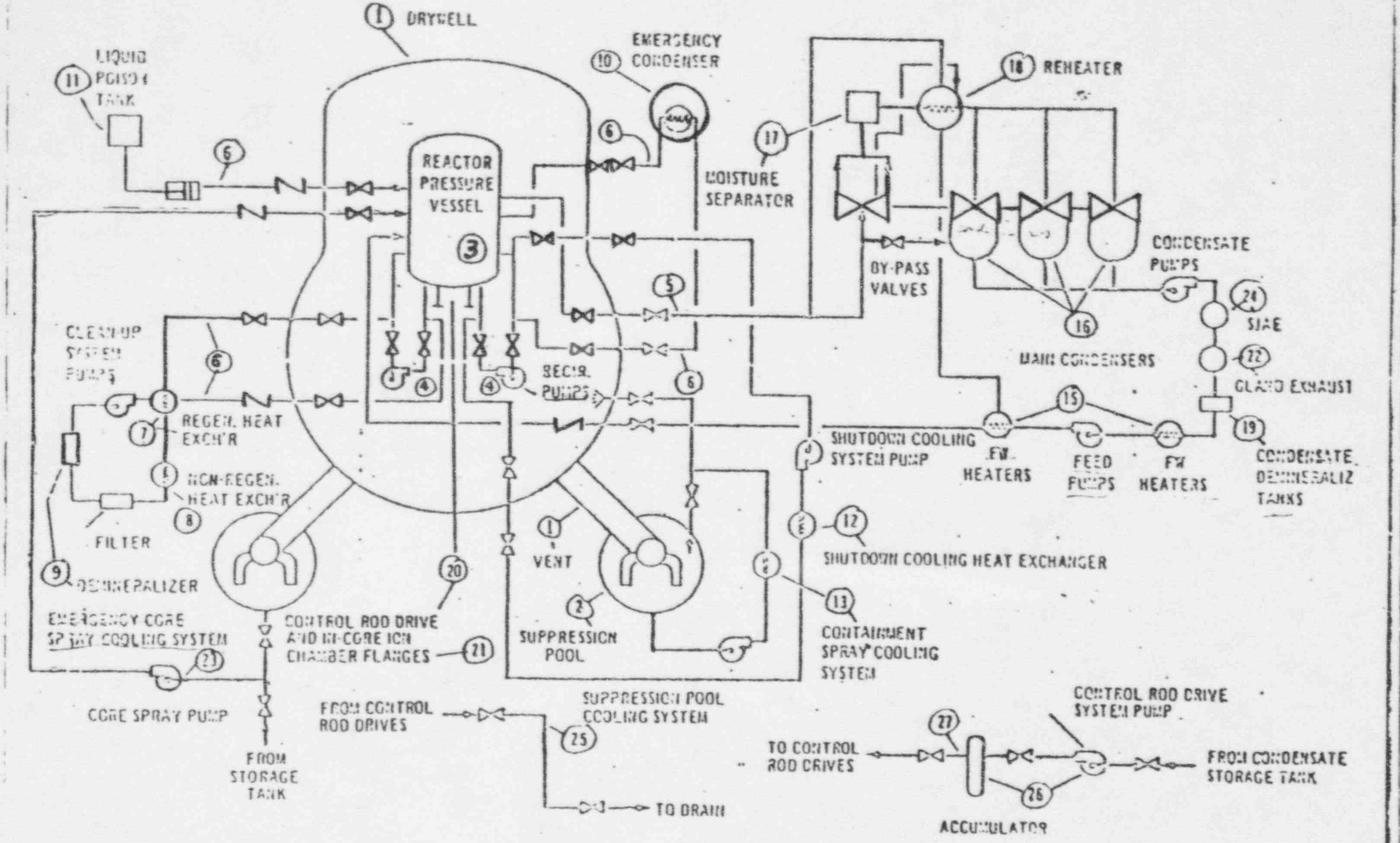
Diagram Item No.	Component	Design Code
1 & 2	Drywell & Vents & Suppression Pool See Note No. 1	ASME Section VIII Code Case 1272N-5 See Code Case 1276N-1 For ExPan. Joint
3	Reactor Pressure Vessel See Note No. 1	ASME Section I, plus Nuclear Code Case 1270
4	Recirculation Loop, Piping, Recirc Loop, Valves - See Note #2	ASME Section I, ASME Section I & Section VIII, plus G. E. Specification
	Recirc. Pump Cases - See Note #2	ASME Section VIII & Code Case 1274
5	Primary Steam Piping	ASME Section I, through the first valve outside the reactor vessel. Balance: ASA B31.1
	Primary Steam Isolation Valves	ASA B 31.1, plus G. E. Specification
	Primary Steam Safety Valves See Note #2	ASME Section I & Code Case 1271 N
6	Nuclear Steam Supply Aux. Sys. Piping & Valves - See Note #2	ASME Section I, through the first valve outside the reactor vessel Balance ASA B 31.1
7	Regenerative Hx	ASME Section III, Class C TEMA Standard Class R
8	Non-Regenerative Hx	
	Primary Side	ASME Section III, Class C
	Cooling Water Side	ASME Section VIII TEMA Standards Class R

Diagram Item No.	Component	Design Code
9	Cleanup System Veseels & Demin.	ASME Section III, Class C
10	Isolation Condenser	
	Primary Side	ASME Section III, Class A
	Cooling Water Side	ASME Section VIII
11	Liquid Poison Tank	API Standards
	Liquid Poison Pump - See Note #2	ASME Section III, Class C
12	Shutdown Heat Exchanger	
	Primary Side (Tube)	ASME Section III, Class C
	Cooling Water Side (Shell)	ASME Section VIII
	Shutdown Pump - See Note #2	ASME Section III, Class C
13	Containment Spray Cooling Sys. Equip. (See Note #2)	ASME Section VIII
14	Filters (Except those in the cleanup system)	ASME Section VIII
15	Feedwater Heaters (Including Drain Coolers)	ASME Section VIII, Plus TEMA Standards
16	Main Condenser	Heat Exchanger Institute
17	Turbine Moisture Seperator	ASME Section VIII
18	Turbine Steam Reheaters	ASME Section VIII
19	Condensate Demineralizers	ASME Section VIII
20	Control Rod Drive	
	Pressure Parts	ASME Section VIII with deviations for weld joints design covered in Code-Case 1361 (Sect. III)
	Control Rod Drive Housings	ASME Section I
21	Incore Ion Chamber Pressure Parts	ASME Section III, Class A

Diagram Item No.	Component	Design Code
22	Gland Seal Exhauster Condenser	Heat Exchanger Institute
23	Emergency Core Cooling System	
	Piping & Valves - See Note #2	ASME Section I, through the first valve outside the reactor vessel. Balance ASA B 31.1
24	Steam Jet Air Ejector & Inter & After Condensers	Heat Exchanger Institute
25	Scram Dump Piping & Valves See Note #2	ASA B 31.1, plus APED Specifications through the first valve outside the reactor vessel. Balance: ASA B 31.1
26 & 27	Control Rod Drive System	
	Pump Casing & Accumulators See Note #2	ASME Section VIII
	Piping & Valves - See Note #2	ASME Section I from control rod driv to first valve. Balance ASA B 31.1

Note #1: These pressure vessels of the Nuclear Power System were ordered prior to January 1, 1965, and therefore are designed to Codes applicable at that time.

Note #2: Pumps casings and valve bodies will be designed to Code Standards, but will not be stamped because, as machine parts, they are outside the scope of the Codes.



1 DRYWELL
EMERGENCY CONDENSER
10
REHEATER
18
LIQUID POISON TANK
11
6
REACTOR PRESSURE VESSEL
3
17
MOISTURE SEPARATOR
CONDENSATE PUMPS
BY-PASS VALVES
5
24
SJAE
16
MAIN CONDENSERS
22
GLAND EXHAUST
CLEAN-UP SYSTEM PUMPS
4
RECIR. PUMPS
15
REGEN. HEAT EXCH'R
7
NON-REGEN. HEAT EXCH'R
8
SHUTDOWN COOLING SYSTEM PUMP
FW HEATERS
FEED PUMPS
19
CONDENSATE DEMINERALIZ TANKS
FILTER
VENT
12
SHUTDOWN COOLING HEAT EXCHANGER
9
DEMINERALIZER
20
2
13
EMERGENCY CORE SPRAY COOLING SYSTEM
CONTROL ROD DRIVE AND IN-CORE ION CHAMBER FLANGES
21
SUPPRESSION POOL
CONTAINMENT SPRAY COOLING SYSTEM
23
CORE SPRAY PUMP
FROM CONTROL ROD DRIVES
SUPPRESSION POOL COOLING SYSTEM
25
CONTROL ROD DRIVE SYSTEM PUMP
27
TO CONTROL ROD DRIVES
FROM CONDENSATE STORAGE TANK
FROM STORAGE TANK
TO DRAIN
26
ACCUMULATOR

APPENDIX 7

Acceptance Criteria and Load Combination Method for Radwaste Building, Section 3.8.4, Ref. 3.

3.8.4 Other Category I Structures

3.8.4.1 Description of the Structures - Figure 3.8.1 shows the structural configuration of the Radwaste Building with the seismic Category I elements identified.

The Radwaste Building houses the facilities for solid and liquid radwaste processing. The basic functions of the building are to provide radiation protection during operating conditions and to ensure no leakage of radioactive materials to the surroundings during extreme environmental conditions.

Following is a physical description of the building:

The building is rectangular in plan. It has three main floors: grade, intermediate and operating. A large door opening for truck access is provided at grade level in the east wall. The door is designed to provide for conventional weather protection. A concrete curb is provided to retain any spillage inside the building. Reinforced concrete walls are provided to the operating level and above this level where liquid retention is required. The remaining wall area is insulated metal siding or of solid concrete block construction. The roof area is covered by insulated metal deck and roofing and by concrete slabs where radiation shielding is required. Interior shield walls of concrete block are provided for protection of operating and maintenance personnel.

3.8.4.2 Applicable Codes, Standards and Specifications - The design and construction of the Radwaste Building is based on the requirements of the following codes:

a. Uniform Building Code of International Conference of Building Officials, 1973 Edition.
b. ASCE Task Committee Report "Wind Forces on Structures", Paper No. 3269.
c. Building Code Requirements for Reinforced Concrete, ACI 318-71.

d. Specifications for Structural Concrete for Buildings, ACI 301-72.

e. AISC "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings".

f. AWS Code for Welding in Building Construction, D1.1.

g. Specifications for Structural Joints using ASTM A325 or A490 bolts.

h. The BOCA Basic Building Code.

3.8.4.3 Loads and Loading Combinations

3.8.4.3.1 Loads, Definitions, and Nomenclature - The following loads are considered in the design of this structure.

a. Normal loads - Those loads which are encountered in normal plant operation and shutdown.

D - Dead loads of the structure and all other permanent loads including bouyant pressure from design flood where applicable.

L - Live loads on floors and roof including moveable equipment loads, piping, cable trays and any other loads which vary in intensity and occurence.

T - Thermal effects and loads during normal operating or shutdown conditions.

R - Pipe reactions during normal operating or shutdown conditions based on the most critical transient or steady-state condition.

H - Lateral earth pressure and surcharge.

F - Lateral pressure from liquids including design flood.

b. Severe environmental loads -

E - Loads generated by the operating basis earthquake.

W - Loads generated by the design wind. 1

c. Extreme environmental loads -

E^1 - Loads generated by the safe shutdown earthquake.

W^1 - Loads generated by the design tornado. Tornado loads include loads due to the tornado wind pressure and tornado created differential pressure.

3.8.4.3.2 Loading Combinations - The stress resultants (axial loads, moments, and shears) obtained from each of the loads considered in design are combined to simulate the worst credible combinations of loadings. These loading combinations, including associated load factors, are as follows:

a. Concrete Structures.

Service Loads.

1. 1.4D + 1.4F + 1.7L + 1.7H
2. 1.4D + 1.4F + 1.7L + 1.7H + 1.9E
3. 1.4D + 1.4F + 1.7L + 1.7H + 1.7W
4. 0.75 (1.4D + 1.4F + 1.7L + 1.7H + 1.7T + 1.7R)
5. 0.75 (1.4D + 1.4F + 1.7L + 1.7H + 1.9E + 1.7T + 1.7R) 1
6. 0.75 (1.4D + 1.4F + 1.7L + 1.7H + 1.7W + 1.7T + 1.7R)
7. 1.2D + 1.9E
8. 1.2D + 1.7W
9. 0.9D + 1.4F
10. 0.9D + 1.7H

Factored Loads.

11. D + L + F + H + T + R + E^1
12. D + L + F + H + T + R + W^1
13. 9D + E^1
14. 9D + W^1

Both cases of "L" having its full value or being completely absent are considered.

b. Steel Structures.

Steel structures are not considered as being Seismic Category I.

3.8.4.4 Design and Analysis Procedures - The building is constructed on a foundation mat at grade resting on compacted backfill. Steel framing and metal decking are provided for support of the reinforced concrete floor slabs. Only those floor slabs within the cubicle housing the Concentrated Liquid Waste Tanks are considered to be seismic Category I. All other framing and floor slabs are conventionally designed. Exterior walls and other interior walls required for retention of spilled liquids are constructed of reinforced concrete and are treated as seismic Category I elements.

Since the building is composed of a combination of seismic Category I and non-seismic elements, both the failure and non-failure of non-seismic elements has been considered in design to determine the controlling case. Lateral loads due to wind, tornado, and earthquake are transferred to the foundation mat through the stiff reinforced concrete walls. Distribution of these lateral loads takes into account the flexural and torsional rigidities of the walls. Although not designed for these lateral loads, the floor slabs are considered adequate to effect the load transfer. The foundation mat is designed as a seismic Category I structure and its analysis takes into account the relative flexibility of the mat and the supporting soil. 1

Design of structural elements, except as modified herein, is based on the requirements of References (c) and (e) listed in 3.8.4.2.

3.8.4.5 Structural Acceptance Criteria - Referring to the loading combinations listed in 3.8.4.3.2 the following defines the allowable limits which constitutes the structural acceptance criteria.

Loading Combination	Limit
a. Concrete Structures	
All combinations	U
b. Steel Structures	
Not applicable	

Where: U - The section strength required to resist design loads based on the ultimate strength design methods described in ACI-318-71

3.8.4.6 Materials, Quality Control and Special Construction Techniques

The materials used for construction of the Radwaste Building, together with the quality control standards and inspection requirements during construction are described in 3.8.4.6.1 through 3.8.4.6.3. A summary of basis of construction is given in 3.8.4.6.4. 1

3.8.4.6.1 Concrete - All structural concrete for the Radwaste Building has a minimum compressive strength of 5000 psi at 28 days. The concrete materials and specifications confirming their suitability are discussed in the following paragraphs:

If these additional tests, show non conformance with ASTM A615 or a deviation from the mill tests by more than 10 percent, the entire lot of bars of that size produced from the heat tested are rejected.

3.8.4.6.3 Structural Steel- Since none of the structural steel is Seismic Category I this section is not applicable. 1

3.8.4.6.4 Construction- The following codes are used to establish the specifications and procedures governing the construction of the Radwaste Building:

ACI 301 - Specifications for Structural Concrete for Buildings
ACI 306 - Recommended Practice for Cold Weather Concreting
ACI 311 - Manual of Concrete Inspection 1
ACI 315 - Manual of Standard Practice for Detailing Reinforced Concrete Structures 1
ACI 318 - Building Code Requirements for Reinforced Concrete
ACI 347 - Recommended Practice for Concrete Formwork
ACI 605 - Recommended Practice for Hot Weather Concreting
ACI 614 - Recommended Practice for Measuring, Mixing and Placing Concrete 1
ACI 211 - Recommended Practice for Selecting Proportions for Concrete
ACI 214 - Recommended Practice for Evaluation of Compression Test Results of Field Concrete 1
ASME Boiler and Pressure Vessel Code, Section VIII, Pressure Vessels, Division 1
AISC - Code of Standard Practice
AWS D1.1 Structural Welding Code

3.8.4.7 Testing and Inservice Surveillance Requirements 1
Eight permanent reference bench marks are embedded in accessible locations at the top of the exterior face of the foundation mat. These bench marks will be utilized to monitor the building settlement during the life of the plant. Elevations of these bench marks are established immediately after the placement of the foundation mat.

Inside the building, movements of the mat will be measured at seven locations using the engineers' level.

Measurements will be made on a monthly to bi-monthly basis during the first year after all the dead loads have been applied. Thereafter, an appropriate interval will be selected based on the magnitude of settlement occurring.

<u>3.8.4.8 Criteria for Establishing Category I Structural Configuration</u> - The seismic Category I structural configuration of the Radwaste Building, shown on Figure 3.8.1, was established to prevent liquid wastes from being released to the environment in the event of a SSE. The walls shown are intended to retain the entire liquid inventory of the Radwaste Building with consideration given to the existence of tanks, piping and equipment located on Elevation 23'-6" and taking into account the effects of non-seismic elements of the building collapsing and displacing some of this liquid. It has been determined that a wall height of five feet is the minimum needed to meet this criteria. Wall height exceeds five feet in places either for reasons of continuity or to support the concentrated liquid waste tank cubicles (at elevation 48') which are seismic Category I. Floor drains from these cubicles are embedded in the seismic Category I floor at elevation 48' and routed until the piping is no longer over the truck bay in the north east corner of the building. Possibility of spillage into the truck bay is thus eliminated. 1

APPENDIX 8

Absorptoin Chamber (Torus) Seismic Analysis Calculations by John Blume & Associates - From Appendix III-2.4, Item 3, Amendment 15, FDSAR.

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J. P. NICOLETTI
R. J. SEXTON
R. L. SHARPE
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April 15, 1965

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Atomic Power Equipment Division
175 Curtner Street
San Jose, California

SUBJECT: Jersey Central Suppression Chamber

Dear Mr. Gile:

Transmitted herewith is one copy of the computations for the subject's response to the Jersey Central spectra curves established by Dr. Housner.

The steel torus is supported by forty columns and cross bracing between the outer columns only. Two loading conditions were considered, the most severe being that in which the chamber is filled with water. The resulting acceleration for this case was about 0.22g.

Very truly yours,

JOHN A. BLUME & ASSOCIATES, ENGINEERS

R. J. Sexton

RJS/ce
Enclosures

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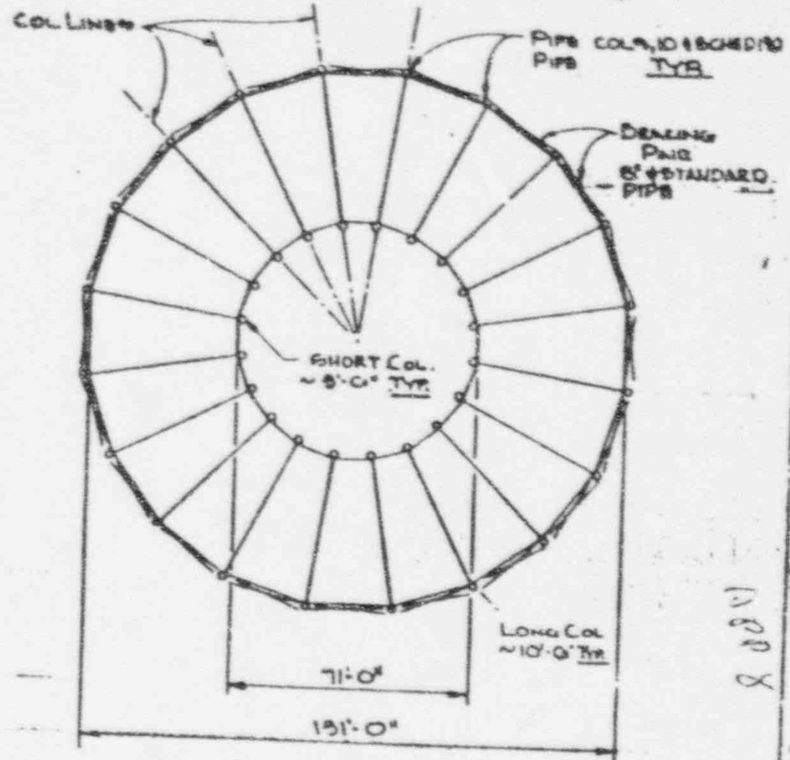
JOB NO. 54430I FOR G.E. JERSEY CENTRAL BY E.G. DATE 3-1-65
SUBJECT SUPPRESSION CHAMBER CHECKED R.K. DATE 3-1-65



PLAN OF TORUS
NO SCALE

APP 8

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JOB: JERSEY CENTRAL SUPPRESSION CHAMBER BY ML DATE 3-1-65
SUBJECT: SEISMIC ANALYSIS CHK'D EK DATE 3-8-65

Stiffness Due to Short Col's

COLUMNS ARE DESIGNED TO RESIST ONLY IN THE TANGENTIAL DIRECTION.

LET ϕ = ANGLE BETWEEN THE TANGENT LINE OF THE TORUS AT THE LOCATION OF COLUMN & THE DIRECTION OF MOTION.

ϕ	$\cos^2\phi$	
9°	0.988^2 =	.976
27°	0.891^2 =	0.794
45°	0.707^2 =	0.500
63°	0.454^2 =	0.206
81°	0.156^2 =	0.024 (+
		2.5
Four Quarters	×	4
		10.0 Equivalent Col's

SHEET NO. 2

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JOB: JERSEY CENTRAL SUPPRESSION CHAMBER BY ML DATE 3-1-65
SUBJECT: SEISMIC ANALYSIS CHK'D EK DATE 3-8-65

Stiffness due to Short Col's. Cont'd.

10" SCH 120 PIPE. 3'-6" LONG

$$k_1 = 10 \cdot \frac{12EI}{L^3}$$

$$= 10 \cdot \frac{12 \times 4.32 \times 10^6 \times \frac{344.3}{12^4}}{3.5^3}$$

$$= \underline{189 \times 10^3} \text{ kip/ft.}$$

SHEET NO. 3

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Jersey Central / Suppression Chamber BY M L DATE 3-1-65
SUBJECT Seismic Analysis CHECKED DATE 3-5-65

STIFFNESS DUE TO LONG COLUMNS

10" SCH 120, 10'-6" Long

$k_c = 10.0 \cdot \frac{12EI}{L^3}$

$= 10.0 \cdot \frac{12 \cdot 4.32 \cdot 10^6 \cdot \frac{7.2 \times}{12^4}}{10.5^3}$

$= \underline{7 \times 10^3 \text{ K}/\text{ft}}$

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Jersey Central Suppression Chamber BY M L DATE 3-1-65
SUBJECT Seismic Analysis CHECKED DATE 3-5-65

Stiffness due to Bracings

THERE ARE ONE PAIRS OF X BRACINGS IN EACH BAY. ASSUME THE BRACING MEMBERS TAKE BOTH TENSION & COMPRESSION

STIFFNESS OF THE BRACINGS IN EVERY BAY PARALLEL TO THE DIRECTION OF MOTION IS. $k = \frac{2AE}{L}\cos^2\theta$.

Number of Equivalent Bays

ϕ	$\cos^2\phi$
18°	$0.951^2 = 0.904$
36°	$0.809^2 = 0.602$
54°	$0.588^2 = 0.346$
72°	$0.309^2 = 0.095$
	1.987

4 QUARTERS × 4 = 7.95

2 BAY ∥ TO direction of motion 2 (+)

9.95 Equivalent Bays

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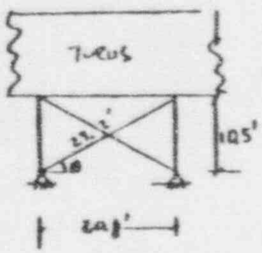
JOB: JERSEY CENTRAL SUPPRESSION CHAMBER BY: L DATE 4-14-65
SUBJECT: SEISMIC ANALYSIS CHECKED EK

Stiffness due to Bracings

8"ϕ sch 40



$A = \frac{8.399}{144} = 0.0583\ ft^2$

$\cos\theta = 0.893$

$\cos^2\theta = 0.797$

$L = \sqrt{10.5^2 + 20.8^2} = 23.3'$

$k_3 = 9.95 \times \frac{2AE}{L} \cos^2\theta$

$= 9.95 \times \frac{2 \times 0.0583 \times 4.32 \times 10^6}{23.3} \times 0.797$

$= \underline{215 \times 10^3}$ kip/ft.

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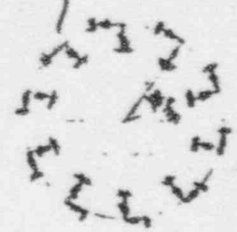
JOB: JERSEY CENTRAL SUPPRESSION CHAMBER BY: L DATE 3-1-65
SUBJECT: SEISMIC ANALYSIS CHECKED EK DATE 3-5-65

Stiffness due to Bellows

THIS ARRANGEMENT & THE SPRING CONSTANTS WERE GIVEN BY G.E.



Radial Spring const. k_r

Tangential Spring const k_t

$k_r = 8.2^k/in = \underline{98.4^k/'}$

$k_t = 41^k/in = \underline{787^k/'}$

ϕ	$\cos^2\phi$	
18°	0.951^2	= 0.905
54°	0.588^2	= 0.346
		1.251
4 Quarters		× 4
		5.00

$\rightarrow k_{R4} = 5 \times k_r = 5 \times 98.4 = \underline{\underline{492}}\ ^k/'$

Stiffness due to Bellows. Cont'd

ϕ	$\cos^2\phi$
72°	$0.305^2 = 0.095$
36°	$0.807^2 = 0.642$
	0.737

4 Quarters = 4 → 2.948

2 in the direction of motion: 2 → 4.95

$k_{rv} = 4.95\, k_r = 4.95 \times 787 = 3{,}900$ k/in

∴ Total Stiffness due to Bellows k_4 is:

$k_4 = k_{rh} + k_{rv} = 492 + 3{,}900 = \underline{4{,}390}$ k/in

Total Stiffness

$k = k_1 + k_2 + k_3 + k_4$

$= 189{,}000 + 7{,}000 + 215{,}000 + 4{,}390$

$= \underline{415{,}050}$ k/in

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JOB: Jersey Central Suppression Chamber BY: MZ DATE: 4-14-68

SUBJECT: Seismic Analysis

Seismic Response

When the chamber is full of water

$W = 15{,}100^{k}$ Given by G.E.

$\omega^2 = \frac{kg}{W} = \frac{415000 \times 32.2}{15{,}100} = 885$

$\omega = 29.8$ rad/sec

$T = \frac{2\pi}{29.8} = 0.21$ sec/cycle

From Housner's chart

$S_a = 0.22g$ for 3% damping

Max. Displacement $= \frac{S_a}{\omega^2} = \frac{0.22 \times 32.2}{885} = \underline{0.008'}$

Max. Acceleration $= S_a = \underline{0.22\ g}$

SHEET NO. 10

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JOB: Jersey Central Suppression Chamber BY: MZ DATE: 4-14-68

SUBJECT: Seismic Analysis

Seismic Response

Chamber partially filled with H_2O

$W = 1090^{k} + \underset{\text{water}}{5680^{k}} = 6{,}770^{k}$

$\omega^2 = \frac{kg}{W} = \frac{415000 \times 32.2}{6{,}770} = 1{,}970$

$\omega = 44.4$

$T = \frac{2\pi}{44.4} = 0.142$ sec.

From Housner's chart

$S_a = 0.175\ g$ for 3% damping

Max. Displacement $= \frac{S_a}{\omega^2} = \frac{0.175 \times 32.2}{1{,}970} = \underline{0.00286'}$

Max. Acceleration $= S_a = \underline{0.175\ g}$

SHEET NO. 11

APPENDIX 9

Exhibit E to AM. 16 (Reactor Pressure
Vessel Design Report)

Part I Seismic Analysis of Reactor Pressure Vessel

Part II Jet Loads Analysis (omitted)

Part III Shield Support

Part IV References

E-7

GENERAL ELECTRIC COMPANY
JERSEY CENTRAL REACTOR PRESSURE VESSEL

TABLE OF CONTENTS

F-19

E-8

E-20

-30 D-21 D-3

-8

E-9

PART V: SEISMIC ANALYSIS

E-10

RESULTS OF CALCULATIONS

D-23

E-11

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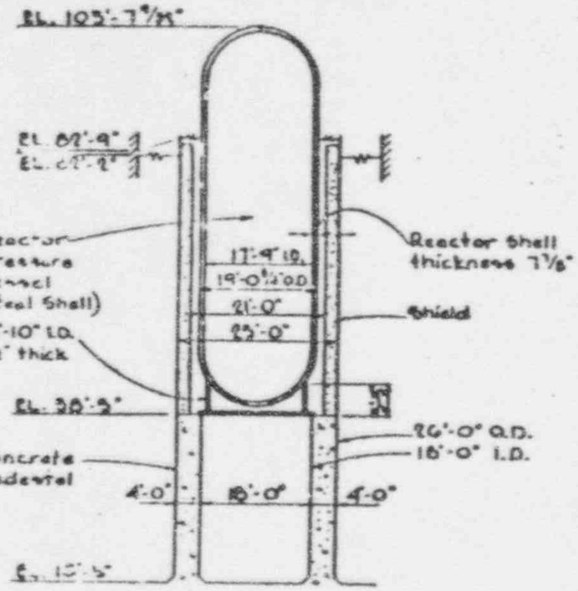
JOB NO. 9445.01 FOR Jersey Central BY S.C.L. DATE 1-8-66
SHEET G.E. SUBJECT Reactor Pressure Vessel CHKD E.K. DATE 1-8-66

GEOMETRIC PROPERTIES



GEOMETRIC FIGURE

Reactor Spring Constant = 43,900 k/ft
Shield Wall Spring Constant = 310,000 k/ft

1

D-24

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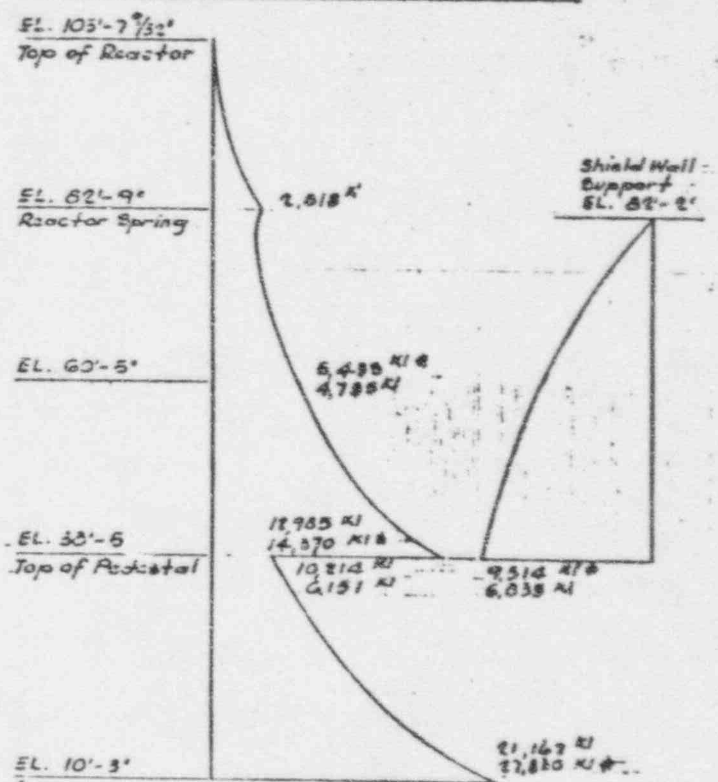
JOB NO. ... FOR JERSEY CENTRAL REACTOR PRESSURE BY S.C.L. DATE 1-8-66
SHEET S.3. SUBJECT VESSEL, SEISMIC ANALYSIS CHKD E.K. DATE 1-8-66

DESIGN MOMENT DIAGRAM



* Indicates max. design moments when modified for building movement. These values do not necessarily occur at the same time, therefore statics will not check.

2

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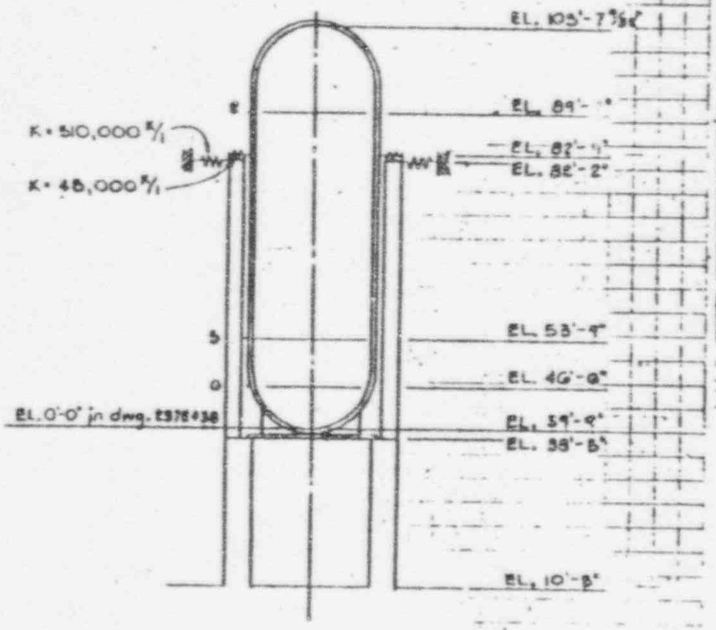
E-36

JOB NO 244501 JOB JERSEY CENTRAL BY S.C.L. DATE 9-8-66

CLIENT G.E. SUBJECT REACTOR VESSEL UNDER JET LOADING CHKD E.K. DATE 9-8-66

REACTOR VESSEL UNDER HORIZONTAL JET REACTION LOADS

CASE 1 Loading of 477 kips apply at 2
CASE 2 Loading of 580 kips apply at 5
CASE 3 Loading of 570 kips apply at 6



GEOMETRIC FIGURE

1

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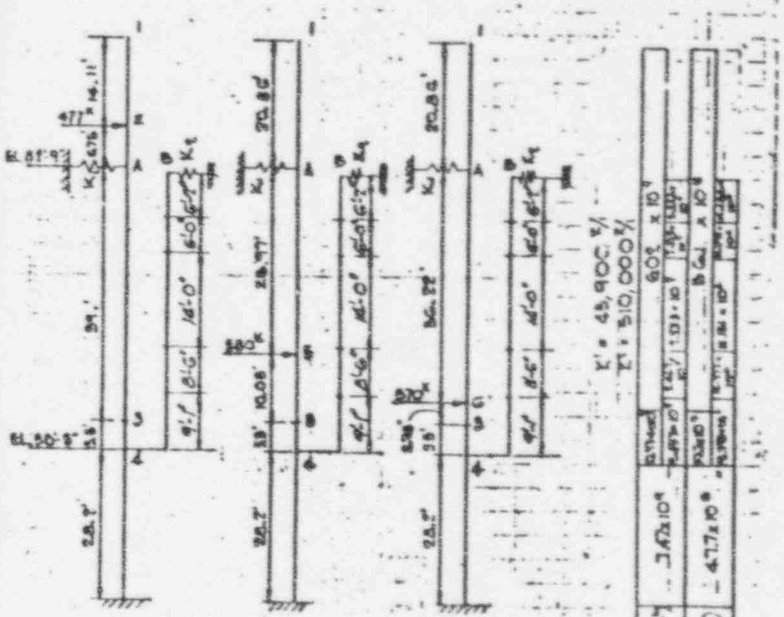
E-37

JOB NO 244051 JOB JERSEY CENTRAL BY S.C.L. DATE 9-8-66

CLIENT G.E. SUBJECT REACTOR VESSEL UNDER JET LOADING CHKD E.K. DATE 9-8-66

LOADING CONDITIONS AND SECTION PROPERTIES

Section Properties



CASE 1 CASE 2 CASE 3

2

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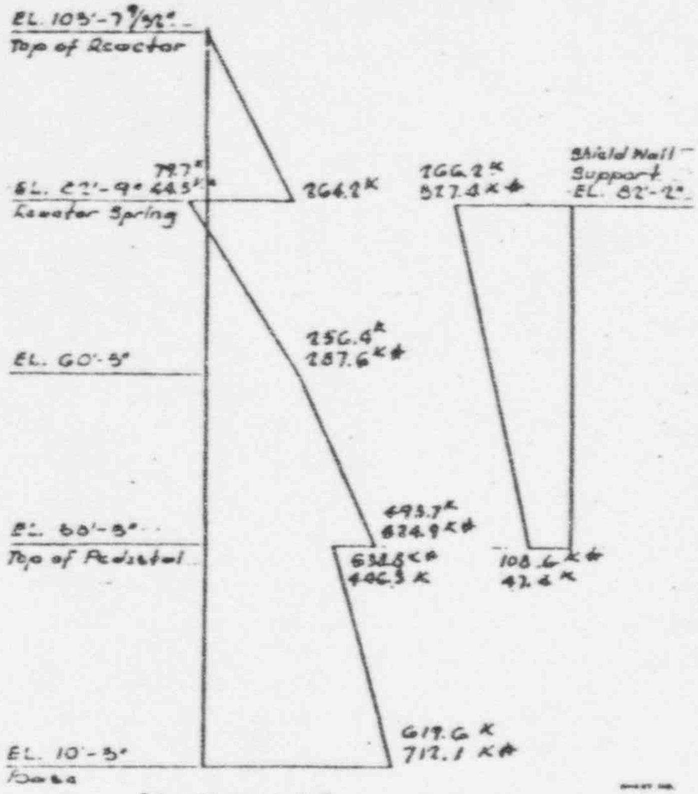
E-13

JOB NO. 244501 JOB JERSEY CENTRAL REACTOR PRESSURE BY SCL DATE 2-8-66

CLIENT G.E. SUBJECT VESSEL, SEISMIC ANALYSIS CHKD BY EK DATE 2-8-66

DESIGN SHEAR DIAGRAM



* See Note, Sht. 2

3

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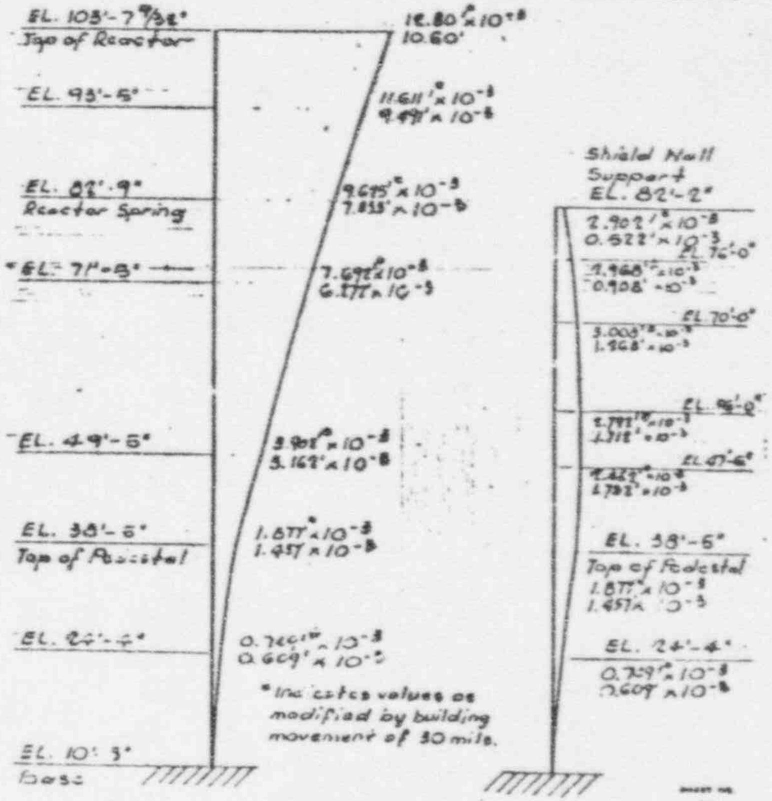
E-14

JOB NO. 244501 JOB JERSEY CENTRAL REACTOR PRESSURE BY SCL DATE 2-8-66

CLIENT G.E. SUBJECT VESSEL, SEISMIC ANALYSIS CHKD BY EK DATE 2-8-66

MAXIMUN DISPLACEMENT RESPONSE CURVE



4

E-15

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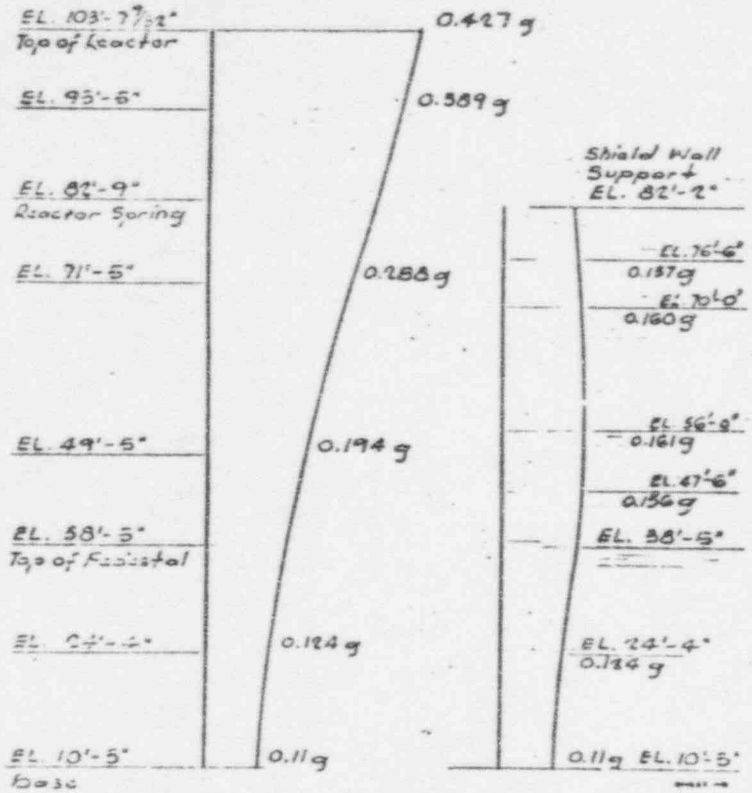
JOB NO. 24151 — JERSEY CENTRAL REACTOR PRESSURE — BY SCL — DATE 2-8-66
CLIENT G.E. — SUBJECT VESSEL, SEISMIC ANALYSIS — CHKD EK — DATE 2-8-66

MAXIMUM ACCELERATION CURVE



5

E-16

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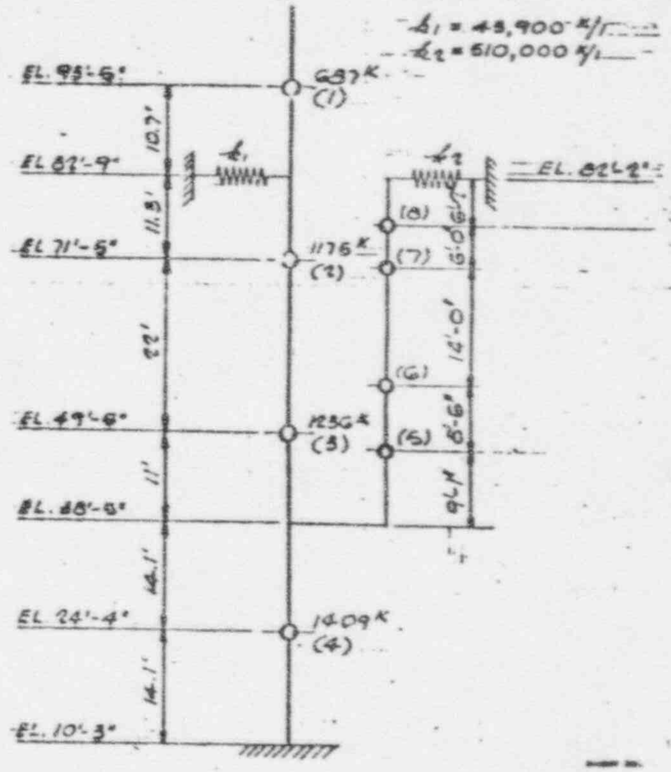
JOB NO. 24151 — JERSEY CENTRAL REACTOR PRESSURE — BY SCL — DATE 2-8-66
CLIENT G.E. — SUBJECT VESSEL, SEISMIC ANALYSIS — CHKD EK — DATE 2-8-66

MATHEMATICAL - MODEL



6

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E-17

JOB Jersey Central R.P.V. — BY ML DATE 1-29-65
CLIENT G.E. SUBJECT Seismic Analysis — CHKD BY PS DATE 2-18-65

WEIGHT OF LUMPED MASSES

STATION 1

HEAD & FLANGE		264^k
CYLIND. SHELL & NOZZLES	$902 \times \frac{12.3}{46.2}$	$= 240^k$
WATER	$780 \times \frac{13.5}{57.5}$	$= 183^k$
		687^k

STATION 2

CYLIND. SHELL & NOZZLES	$902 \times \frac{22}{46.2}$	$= 430^k$
WATER	$780 \times \frac{22}{57.5}$	$= 298^k$
INTERNAL STRUCTURE & FUEL	$375 \times \frac{2}{3}$	$= 250^k$
CONTROL ROD (EXTENDED)	$(0.25 \times 129) \times \frac{1}{2}$	$= 16^k$
FUEL	$(2.8 \times 129) \times \frac{1}{2}$	$= 181^k$
		1175^k

7

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E-18

JOB Jersey Central R.P.V. — BY ML DATE 1-29-65
CLIENT G.E. SUBJECT Seismic Analysis — CHKD BY PS DATE 2-18-65

Wt OF LUMPED MASSES (CONTINUED)

STATION 3

CYLIND. SHELL & NOZZLES	$902 \times \frac{11.9}{46.2}$	$= 232^k$
WATER	$780 \times \frac{22}{57.5}$	$= 298^k$
INTERNAL STRUCTURE & FUEL	$375 \times \frac{1}{3}$	$= 125^k$
CONTROL ROD (EXTENDED)	$(0.25 \times 129) \times \frac{1}{2}$	$= 16^k$
FUEL	$(2.8 \times 129) \times \frac{1}{2}$	$= 184^k$
GUIDE TUBES	0.385×129	$= 50^k$
DRIVE (EXTENDED)	$(0.60 \times 129) \times \frac{16.3}{25.1}$	$= 50^k$
DRIVE HOUSING	$(0.58 \times 129) \times \frac{11.6}{25.1}$	$= 35^k$
HEAD		$= 223^k$
SKIRT & FLANGE		$= 23^k$
		1236^k

8

E-19

W^t OF LUMPED MASSES (CONT'D)

STATION 4

DRIVE (EXTENDED) $(0.60 \times 129) \times \frac{88}{251} = 27^k$

DRIVE HOUSING $(0.58 \times 129) \times \frac{13.5}{251} = 40^k$

PIPING, WATER, MISCELLANEOUS $= 175^k$

PEDESTAL CONCRETE $0.15 \times 276 \times 28.2 = 1167^k$ (+

1409^k

STATION 5

CONCRETE & STEEL 430^k

STATION 6

CONCRETE & STEEL 340^k
1½" PL ½ × 60 30^k
370^k

STATION 7

CONCRETE & STEEL 270^k
1½" PL ½ × 60 30^k
300^k

STATION 8

CONCRETE & STEEL 261^k

E-20

1-3 CROSS-SECTIONAL AREAS & MOMENTS OF INERTIA

PEDESTAL SECTION

$A = \frac{\pi}{4}(26^2 - 18^2) = 276 \ ft^2$

$I = \frac{A}{16}(26^2 + 18^2) = 17300. \ ft^4$

REACTOR STEEL SECTIONS

$A = \frac{\pi}{4}(19.06^2 - 17.75^2) = \frac{\pi}{4} \times 36.81 \times 1.31 = 37.9 \ ft^2$

$I = \frac{A}{16}(19.06^2 + 17.75^2) = \frac{37.9}{16}(363.3 + 315.1) = 1610 \ ft^4$

REACTOR SKIRT STEEL SECTIONS

$A = \frac{\pi}{4}(17.08^2 - 16.83^2) = 6.66 \ ft^2$

$I = \frac{A}{16}(17.08^2 + 16.83^2) = 240 \ ft^4$

JOB NO. 245
CLIENT G.
SHIE
DETE
I_4
I_3
I_2
I_1
EL. 57'-0¼"
5
$A_3 = 16$
$A_2 = 8$
$A_1 = 8$
$A_4 = 8$

General Solution – Statics

The system is solved by strain energy methods with "1g" loading, ie, with the loadings as follows:

P_1	P_2	P_3	P_4	P_5	P_6	P_7	P_8
687ᴷ	1175ᴷ	1236ᴷ	1049ᴷ	430ᴷ	370ᴷ	370ᴷ	261ᴷ

The reactions are found to be:

$R_1 = 1159.62^K$

$R_2 = 1738.94^K$

The corresponding deflections are:

Sta.	1	2	3	4	5	6	7	8	9	13	14 & 15
δy (ft)	0.0313	0.0218	0.0178	0.0028	0.0077	0.0036	0.0059	0.0043	0.0016	0.0764	0.1063
ϕ	1	0.6968	0.3896	0.0895	[illegible]	0.2492	0.1883	0.1376	0.0767	0.8435	0.7019
ϕ^2	1	0.4851	0.1519	0.0080	0.0608	0.0621	0.0355	0.0189	0.0059	0.7116	0.0625
m	13.3155	32.69	38.383	31.5776	13.3660	11.4903	9.9378	8.1056	0	0	0
$m\phi$	13.3454	22.2193	14.9673	2.937	2.2251	2.8635	1.8755	1.1187	0	0	0
$m\phi^2$	13.3454	15.7013	6.5507	0.260	0.8079	0.7136	0.3528	0.1542	0	0	0

$\Sigma m_i \phi_i = 73.6376$, $M^* = \Sigma m_i \phi_i^2 = 47.1688 \left(\frac{K\text{-}sec^2}{FT}\right)$
(Generalized mass)

Earthquake Response (Generalized Coordinates)

Generalized Mass $M^* = 47.1655 \ \left(\frac{K\text{-}SEC^2}{FT}\right)$

Generalized Stiffness $K^* = \frac{\Sigma P_i \phi_i}{\delta_{max}} + \Sigma k_i \phi_i^2$

$K^* = \frac{1}{0.0309}$ [687 × 1 + 1175 × 0.6965 + 1236 × 0.3898 + 1049 × 0.0895 + 430 × 0.2460 + 370 × 0.2492 + 370 × 0.1886 + 261 × 0.1376] + 43900 × 0.7119 + 526000 × 0.0059 = 111,476.17

$\omega = \sqrt{\frac{K^*}{M^*}} = \sqrt{\frac{111{,}476.17}{47.1655}} = \sqrt{2363.51} = 48.62 \ \frac{rad}{sec}$

$T = \frac{2\pi}{\omega} = 0.129$ SEC.

use $\lambda = 0.02$

$S_a = 0.226g$ (from J.C.E.Q. Acce. Response Spectrum)

$\ddot{Y}_{max.} = \frac{\Sigma m_i \phi_i}{M^*} \ S_a = \frac{73.6376}{47.1655} \times 0.226g = 0.351g$

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E-25

JOB NO. 243301 JOB JERSEY CENTRAL R.P.V. BY SCL DATE 7-8-66
CLIENT G.E. SUBJECT SEISMIC ANALYSIS CHKD BY EK DATE 7-8-66

General Solution – Statics

The system is solved by strain energy methods with "1g" loading, ie, with the loadings as follows:

P_1	P_2	P_3	P_4	P_5	P_6	P_7	P_8
687^k	1178^k	1236^k	1049^k	430^k	370^k	370^k	761^k

The reactions are found to be:

$R_1 = 1139.67^k$
$R_2 = 1738.96^k$

The corresponding deflections are:

Sta.	1	2	3	4	5	6	7	8	9	13	14&15
δ_y(ft)	[illegible]	0.0718	0.0771	0.0028	0.0077	0.0036	0.0059	0.0083	0.0016	0.0164	0.0063
ϕ	1	0.6968	0.3898	0.0899	[illegible]	0.4991	0.1885	0.1376	0.0767	0.2695	0.7018
ϕ^2	1	0.4851	0.1519	0.0080	0.0605	0.0681	0.0355	0.0189	0.0059	0.7115	0.0605
m	13.3856	36.49	38.385	37.5776	13.3560	11.4907	2.9979	8.1056	0	0	0
$m\phi$	13.3464	25.7657	14.9570	2.957	3.2851	2.8695	1.8753	1.1137	0	0	0
$m\phi^2$	13.3854	17.7013	6.3807	0.7606	0.8077	0.7136	0.3378	0.1582	0	0	0

$\Sigma m_i \phi_i = 73.6376$, $M^* = \Sigma m_i \phi_i^2 = 47.1658 \left(\frac{k\text{-sec}^2}{ft}\right)$
(Generalized mass)

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E-26

JOB NO. 244301 JOB JERSEY CENTRAL R.P.V. BY SCL DATE 7-8-66
CLIENT G.E. SUBJECT SEISMIC ANALYSIS CHKD BY EK DATE 7-8-66

Earthquake Response (Generalized Coordinates)

Generalized Mass $M^* = 47.1658 \quad \left(\frac{k\text{-sec}^2}{ft}\right)$

Generalized Stiffness $K^* = \frac{\Sigma P_i \phi_i}{\delta_{max}} + \Sigma k_i \phi_i^2$

$K^* = \frac{1}{0.0309}[687 \times 1 + 1175 \times 0.6965 + 1236 \times 0.3895 + 1049 \times$

$0.0895 + 430 \times 0.0460 + 370 \times 0.2492 + 370 \times 0.1886 + 761 \times 0.1078] + [illegible] \times 0.7119 + 510000 \times$

$0.0059 = 111{,}476.17$

$\omega = \sqrt{\frac{K^*}{M^*}} = \sqrt{\frac{111{,}476.17}{47.1658}} = \sqrt{2363.51} = 48.62 \ \frac{rad}{sec}$

$T = \frac{2\pi}{\omega} = 0.129$ SEC.

use $\lambda = 0.02$

$S_a = 0.226g$ (From J.C.E.Q. Acce. Response Spectrum)

$\ddot{Y}_{max.} = \frac{\Sigma m_i \phi_i}{M^*} S_a = \frac{73.6376}{47.1658} \times 0.226g = 0.351g$

Earthquake Response

Relative Acceleration $\ddot{V}_{i,max} = \ddot{V}_{0,max} \cdot \phi_{ni}$

Building Acceleration $\ddot{U}_{bi,max}$
(From J. C. Building Seismic Report, June 18 '65, By J. A. Blume & Associates)

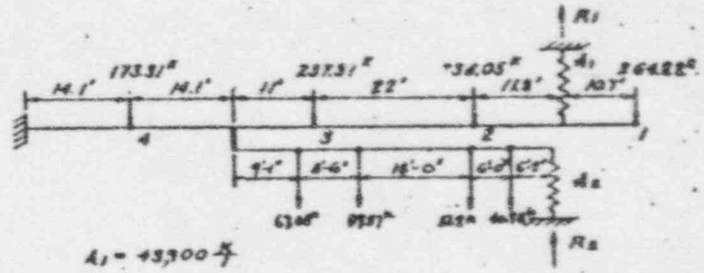
Station	1	2	3	4	5	6	7	8
$\ddot{V}_{i,max}$	0.351g	0.2463g	0.1368g	0.0316g	0.0862g	0.0875g	0.0662g	0.0682g
$\ddot{U}_{bi,max}$	0.168g	0.153g	0.137g	0.120g	0.130g	0.136g	0.145g	0.149g
* $a_{i,max}$	0.309g	0.283g	0.193g	0.124g	0.156g	0.161g	0.160g	0.157g
W_i(K)	687	1,175	1236	1,409	430	570	320	261
** P_{si}(K)	264.22	336.05	237.31	173.31	67.08	89.57	51.2	40.98

* Maximum Absolute Acceleration $= a_{max} = \sqrt{\ddot{V}_{i,max}^2 + \ddot{U}_{bi,max}^2}$

** Seismic Forces $= P_{si} = a_{i,max} \cdot W_i$

4

Seismic Response



$k_1 = 43,300$ K/"

$k_2 = 510,000$ K/"

The structure is solved by strain energy methods and the spring reactions are found to be :

$R_1 = 345.91^K$

$R_2 = 266.13^K$

E-29

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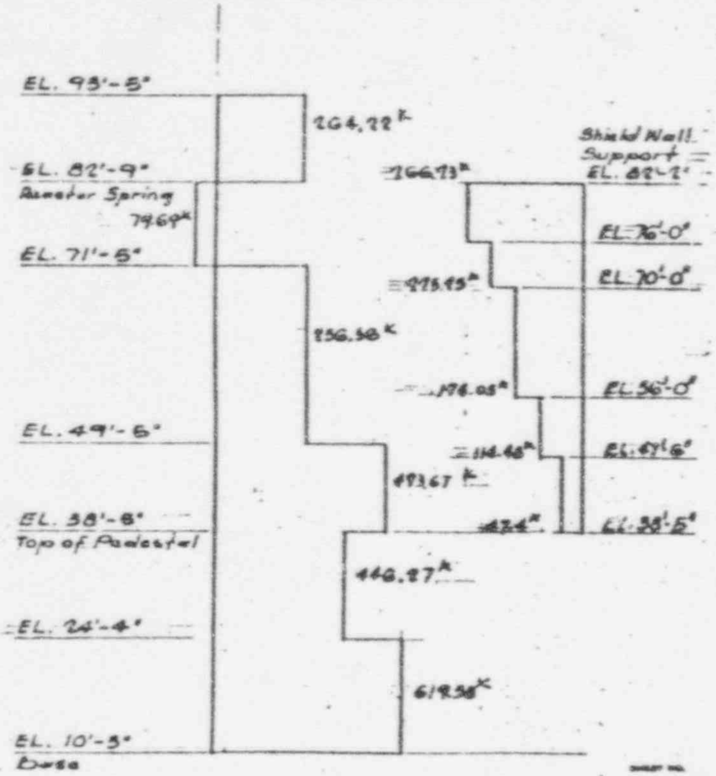
244301 JERSEY CENTRAL REACTOR PRESSURE SCL 2-8-66
G.E. VESSEL, SEISMIC ANALYSIS EK 2-8-66

DESIGN SHEAR DIAGRAM
FOR LUMPED SYSTEM



19

E-30

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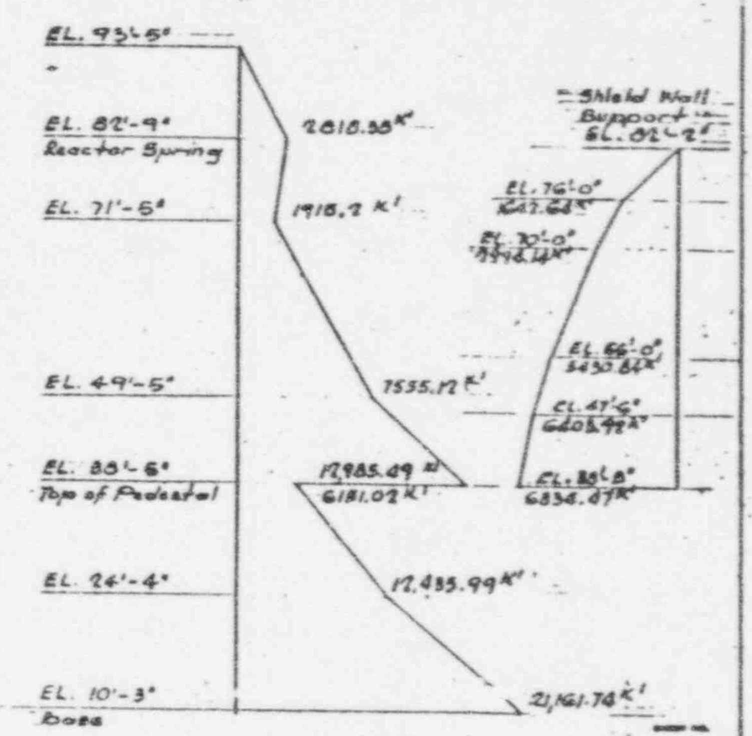
244301 JERSEY CENTRAL REACTOR PRESSURE SCL 2-8-66
G.E. VESSEL, SEISMIC ANALYSIS EK 2-8-66

DESIGN MOMENT DIAGRAM
FOR LUMPED SYSTEM



20

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E-31

244301 JERSEY CENTRAL R.R.K SCL 8-8-66

GE SEISMIC ANALYSIS EK 8-8-66

Responses Due To Building Movement

Spring Reactions Due To Building Displacement Of 0.03" @ The Elevations Of The Two Springs Can Be Obtained By Strain Energy Method; From which the Two Spring Reactions are found to be:

$R_1' = 31.238^K$

$R_2' = 61.221^K$

The moment, shear & displacement diagrams shown in the following sheets should be combined with those under seismic loads for purpose of final design.

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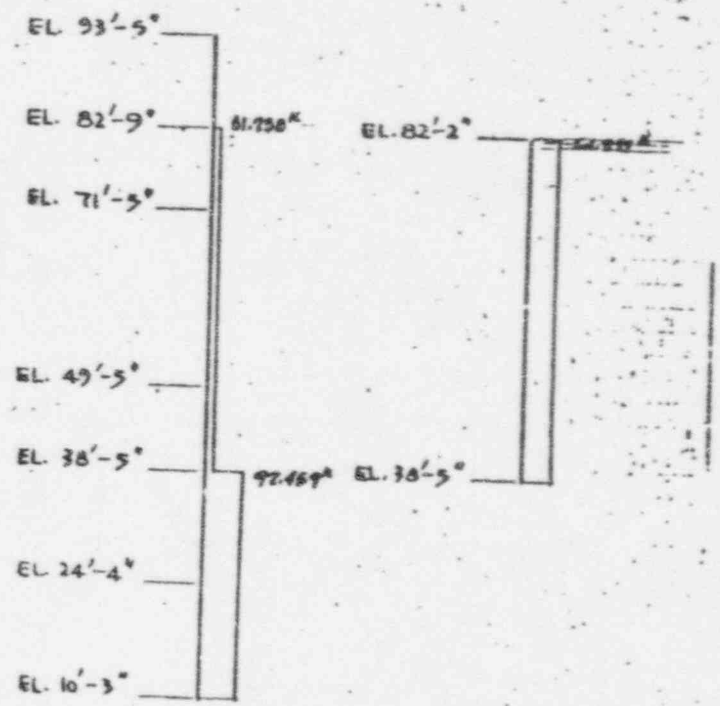
E-32

244301 Jersey Central – R.P.V SCL 8-8-66

G.E. Interactions Due to Building Movement EK 8-8-66

Shear Diagram Due to Building Movement



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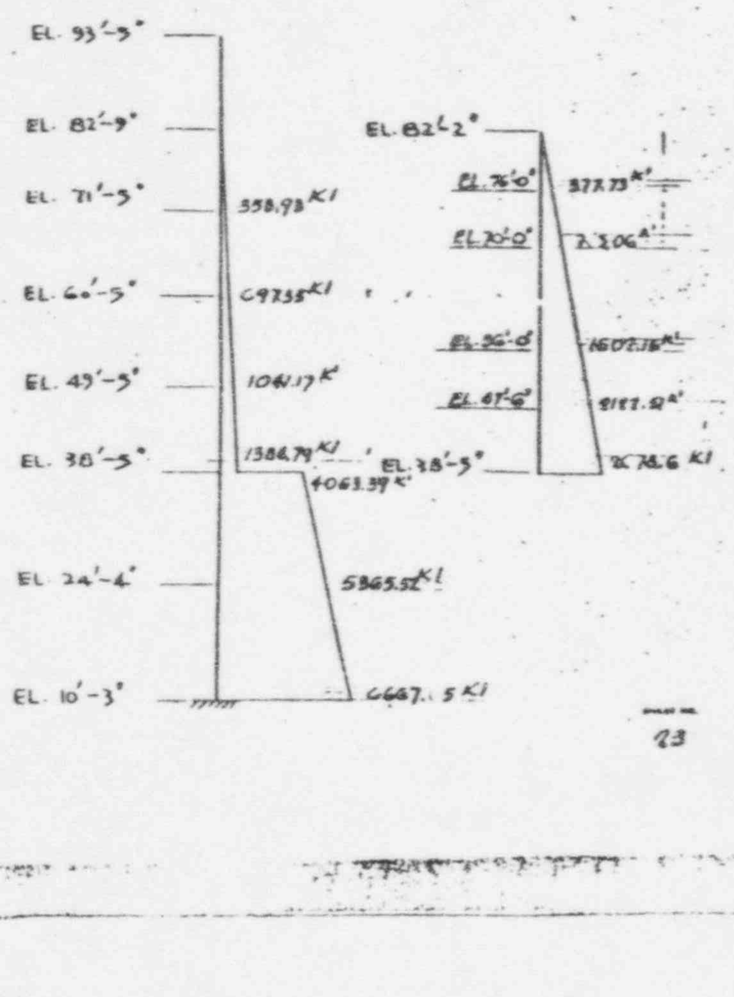
E-33

JOB NO 244301 JOB Jersey Central - R.P.V. BY SCL DATE 2-8-66

CLIENT G.E. SUBJECT Interactions Due to Building Movement DATE 2-8-66

Moment Diagram Due to Building Movement



23

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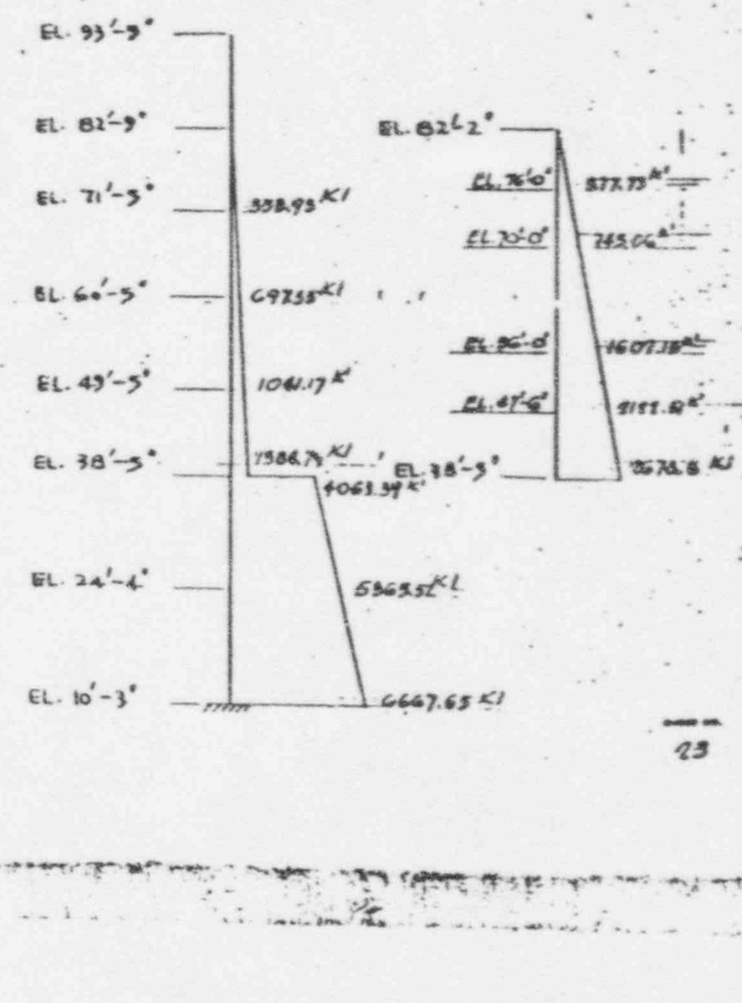
E-33

JOB NO 244301 JOB Jersey Central - R.P.V. BY SCL DATE 2-8-66

CLIENT G.E. SUBJECT Interactions Due to Building Movement DATE 2-8-66

Moment Diagram Due to Building Movement



23

E-34

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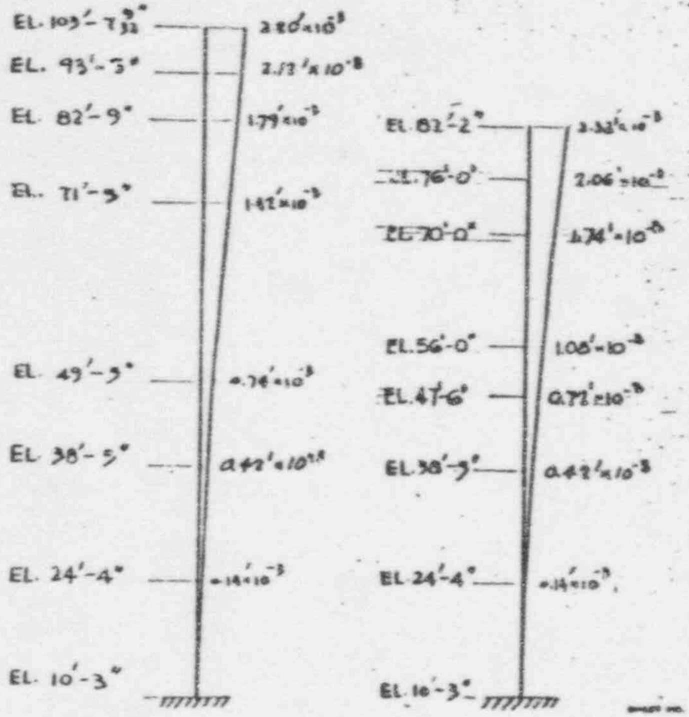
JOB 24431 — JERSEY CENTRAL — R.P.V. — BY ECL — DATE 2-8-66

G.G. — INTERACTION DUE TO BUILDING MOVEMENT — CHKD EK — DATE 2-8-66

DISPLACEMENT CURVE DUE TO BUILDING MOVEMENT



24

E-35

PART II: JET LOAD ANALYSIS

(OMITTED)

E-51

PART III: SHIELD SUPPORT

1-52

LOADS IN TRUSS MEMBERS

CAS

SUMM

EQU.

CNO

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/

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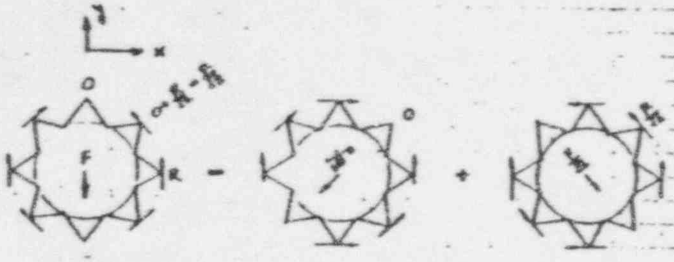
JERSEY CENTRAL, SHIELD WALL SUPPT — M L — 1-15-65
REACTIONS ON RING — PS — 2-10-65

TANGENTIAL REACTIONS

CASE 1 (ARBITRARY FORCE F)



SUMMATION OF FORCES IN "Y" DIRECTION EQUALS ZERO

EQUILIB. CONDITION IN VERTICAL DIRECTION
(NOTE THE SYMMETRICAL & ANTI-SYMMETRICAL NATURE. THIS REDUCES THE PROB TO STATICALLY DETERMINATE)

$$F = 2\left(R + 2\cdot\frac{R}{\sqrt{2}}\cos\frac{\pi}{4}\right)$$

$$F = 4R$$

$$R = 0.250F$$

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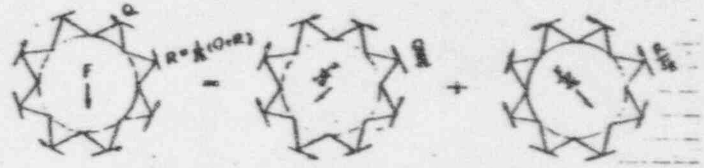
JERSEY CENTRAL, SHIELD WALL SUPPT — M L — 1-15-65
REACTIONS ON RING — PS — 2-10-65

TANGENTIAL REACTIONS

CASE 2
(ARBITRARY FORCE F)



THUS WE HAVE.

$R = \frac{1}{\sqrt{2}}(Q+R)$ — ① [SUMMATION HORIZ FORCES = 0]
$\Sigma V = 0$ GIVES.
$F = 4\left(R\cos\frac{\pi}{8} + Q\sin\frac{\pi}{8}\right)$ — ② [SUMMATION VERT FORCES = 0]

THE SOLUTION IS,
$Q = 0.0955F$
$R = 0.231F$

LOADING DIAGRAM　　MOMENT DIAGRAM　　8

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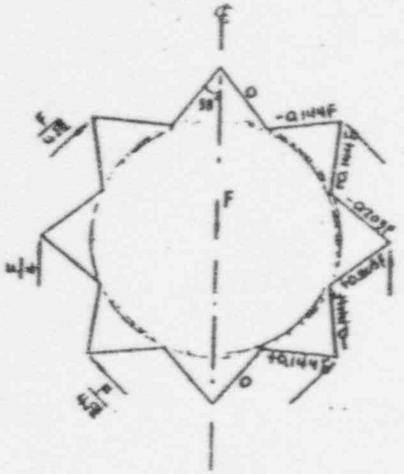
244301 Jersey Central RPV Shield Wall Lateral Support　ML　1-13-65
G.E.　Loads in Truss Members　PS　2-10-65

TRUSS LOADS

CASE 1

(Arbitrary Force F)



+ indicates tension
− indicates compression

3

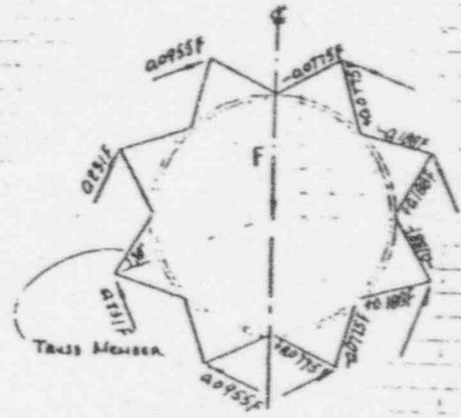
LOADING DIAGRAM　　SHEAR DIAGRAM　　9

JOHN A. BLUME & ASSOCIATES, ENGINEERS　　E-56
612 HOWARD STREET • SAN FRANCISCO 5, CALIFORNIA
244301 Jersey Central RPV Support　ML　1-13-65
G.E.　Loads in Truss Members　PS　2-10-65

TRUSS LOADS

CASE 2

(Arbitrary Force F)



+ indicates tension
− indicates compression

4

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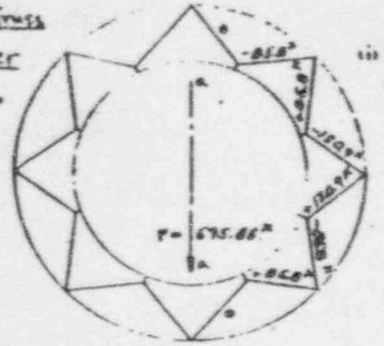
E-57

JOB NO. 244301 JOB: Jersey Central: R.P.V. Support BY: SCL DATE: 2-8-66
CLIENT: G.E. SUBJECT: Loads in Truss Members CHKD: EK DATE: 2-8-66

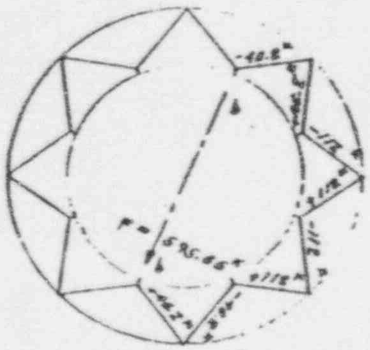
Forces in th Truss Members Under Seismic Load,



CASE 1

(i) Load in a-a Direction,

Total Load = 596.66^K



CASE 2

(ii) Load in b-b Direction,

Total Load = 596.66^K

5

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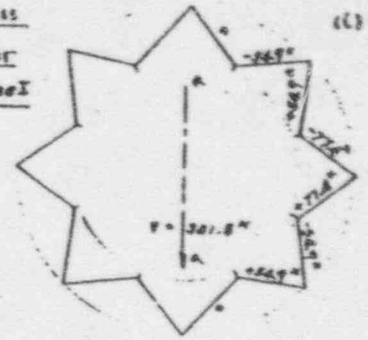
E-56

JOB NO. 244301 JOB: Jersey Central: R.P.V. Support BY: SCL DATE: 2-8-66
CLIENT: GE SUBJECT: Loads in Truss Members CHKD: EK DATE: 2-8-66

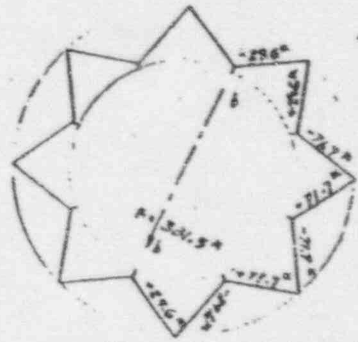
Forces in Truss Member Under Jet Load, Case I



CASE 1

(i) Load in a-a Direction,

Total Load = 381.5^K



CASE 2

(ii) Load in b-b Direction,

Total Load = 381.5^K

6

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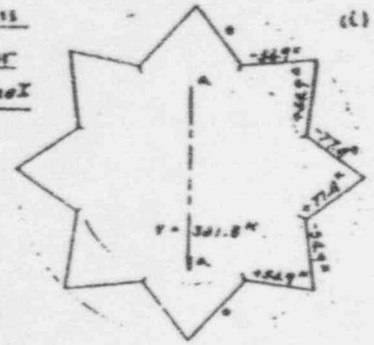
E-56

244301 Jersey Central, R.P.V. Support SCL 2-8-66
G.E. LOADS IN TRUSS MEMBERS EZ 2-8-66

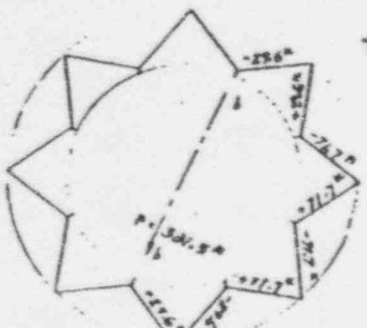
Forces in Truss
Member Under
Jet load, Case I



CASE 1

(i) Load in a-a
Direction,

Total load = 381.5[K]



CASE 2

(ii) Load in b-b
Direction,

Total load = 381.5[K]

6

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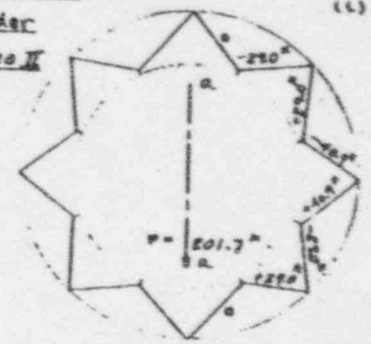
E-59

244301 Jersey Central, R.P.V. Support SCL 2-8-66
G.E. LOADS IN TRUSS MEMBERS EZ 2-8-66

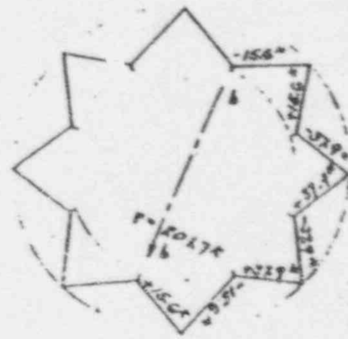
Forces in the Truss
Member Under
Jet load, Case II



CASE 1

(i) Load in a-a
Direction,

Total load = 201.7[K]



CASE 2

(ii) Load in b-b
Direction,

Total load = 201.7[K]

7

1-60

PART IV: REFERENCES

1-61

REFERENCES

1. Seismic Analysis of Reactor Pressure Vessel for the Jersey Central Reactor Project, by John A. Blume & Associates, Engineers, August 7, 1964.

2. Report on Earthquake Design Criteria for the Jersey Central Nuclear Power Plant, by G. W. Housner, March 1964, and Revisions dated May 14, 1964.

3. Seismic Analysis of Reactor Building for the Central Reactor Project, by John A. Blume & Associates, Engineers, January 26, 1965.

4. Seismic Analysis of Reactor Pressure Vessel for the Jersey Central Reactor Project, by John A. Blume & Associates, Engineers, October 13, 1964.

5. Seismic Analysis of Reactor Pressure Vessel for the Jersey Central Reactor Project, by John A. Blume & Associates, Engineers, February 18, 1965.

6. General Electric Design Drawings

237E976	December 24, 1964	(Ring Support)
237E714	October 19, 1964	(Ring Support)
237E716	December 24, 1964	(Ring Support)
237E547	May 7, 1964	Sheets 1 thru 4 (Bldg)
237E547	August 13, 1964	Sheets 3 and 4 (Bldg)
237E437	(Pressure Vessel)	
237E438	(Pressure Vessel)	

7. Burns and Roe Drawings

   SK-1418

APPENDIX 10

Exhibit A to AM. 16 (Reactor Pressure Vessel Design Report)

B.1 Code Calculation Summary

B.2 Steady State - (100% Full Power Normal Operation) Pertinent Stresses or Stress Intensities

B.3 & 5 Transient Analysis Results with respect to Material Fatigue

B.4 Pipe Reaction Stresses in RPV Nozzle Wall Junctions

B.6 Stress Results for

a. Vessel Support Skirt

b. Vessel Stabilizer Brackets

There were no deviations to the formal codes throughout the design, fabrication, inspection, and testing of the reactor vessel.

B.1 Code Calculation Summary

	Part	Section I Paragraph	Required Thickness or Reinforcing Area	Actual Thickness or Reinforcing Area
a.	Bottom Head w/Control Rod Drive Penetration	P 180	6.750 in. 23.8 in.^{2}	6.750 in. min. 41.9 in.^{2}
b.	Bottom Head	P 180	3.435 in.	6.750 in. min.
c.	Vessel Shell	P 180	7.082 in.	7.127 in. min.
d.	Vessel Flange	Section VIII	16.13 in.	16.28 in.
e.	Closure Flange	Section VIII	28.26 in.	28.80 in.
f.	Top Head	P 180	3.4975 in.	4.215 in.
g.	Nozzles	P 268		
	Recirculation			
	Inlet		85.815 in.^{2}	132.50 in.^{2}
	Outlet		223.62 in.^{2}	237.32 in.^{2}
	Steam Outlet		186.81 in.^{2}	720.64 in.^{2}
	Feedwater		75.463 in.^{2}	78.474 in.^{2}
	Emergency Cond.		70.166 in.^{2}	70.327 in.^{2}
	Core Spray		44.437 in.^{2}	48.165 in.^{2}
	CRD Hyd. Ret.		20.25 in.^{2}	20.867 in.^{2}
	Vent TH		15.24 in.^{2}	15.89 in.^{2}
	6 in. Inst. TH		24.07 in.^{2}	24.73 in.^{2}
	2 in. Inst.		21.20 in.^{2}	21.85 in.^{2}
	1 in. Inst.		14.80 in.^{2}	14.781 in.^{2}
	Core ΔP		15.02 in.^{2}	81.20 in.^{2}
	Drain		8.97 in.^{2}	17.30 in.^{2}

B.2 Steady State - (100% Full Power Normal Operation) Pertinent Stresses or Stress Intensities

Part	Material	Allowable Stress or Stress Intensity (psi)	Governing Stress or Stress Intensity (psi)
Vessel Support	SA302 Mod. Code Case 1236	20,000 60,000 (1)	-7,600 20,200 -35,410 (2)(3)
Bottom Head	SA 302B	20,000 60,000	7,800 8,900
Cyl. & PB Junction	SA302B	20,000 60,000	8,200 8,800
Shroud Support Cone	SB168	20,000 60,000	-3,300 10,400
Closure Flange To Shell	SA336 Mod. Code Case 1236	20,000 60,000	18,000 36,000
Vessel Flange To Shell	SA336 Mod. Code Case 1236	20,000 60,000	15,000 28,000
Control Rod Drive Stub Tube	SA312 TP 304	18,033 45,100	44,100 *
Feedwater Nozzle	SA336 Mod. Code Case 1236	20,000 60,000	67,120 **
Core Spray Nozzle	SA336 Mod. Code Case 1236	20,000 60,000	48,840 **
Recirculation Inlet Nozzle	SA336 Mod. Code Case 1236	20,000 60,000	21,100 ***
Recirculation Outlet Nozzle	SA336 Mod. CC 1236	20,000 60,000	30,300 **
Recirculation Nozzles SE to CS Junction	SA336F8 TP304	15,000 45,000	16,000 **
Studs	SA193 B-16 (4340)	30,000 81,000	46,000 ****

(1) First Listed Allowable Stress is the Code Allowable Stress S_m. The second listed stress is $3 S_m$ the G. E. Co., vessel specification primary plus secondary stress intensity limit.

(2) First Stress in this table is the average membrane stress through the section due to pressure and mechanical loads. The second listed stress is the calculated primary plus secondary stress intensity for the 100% full power normal operation unless otherwise stated.

(3) Third number is the primary plus secondary stress intensity that results during the automatic blowdown. (Value reported is for a six-valve-blowdown case.)

* Calculational methods yield only primary plus secondary stresses. This case includes earthquake plus a stuck control rod.

** Calculational methods yield only primary plus secondary stresses.

*** Result obtained from a photoelastic analysis and yielded only primary plus secondary stresses.

**** Allowable stress based on 1/3 of the minimum yield point at temperature for the particular stud material. The second listed allowable is only $3 S_m$ or 2/3 the minimum yield point at temperature as the primary plus secondary stress limit for studs.

APP 10

B.3 & 5 Transient Analysis Results with Respect to Material Fatigue (See G. E. Co. Drawing 137E430, Sheet 2 attached)

Transient Cycle		No. of Design Cycles (Ni)
a.	Startup - Shutdown	120
b.	Daily 3_% reduction in power	10,000
c.	Weekly 50% reduction in power	2,000
d.	Rod Worth Tests	400
e.	Loss of Feedwater Heating	80
f.	Loss of Feedwater Flow	80
g.	Scram	200
h.	Turbine Trip	40
i.	Overpressure 1250 psig	1
j.	Overpressure 1375 psig	1
k.	Blowdown (110°F in 10 min.)	1
l.	300°F/hr Emergency Cooldown	10
m.	Bolt Unbolt	80
n.	Core Spray Actuation	10
o.	CRD Penetration	
	a. + Normal + Drive Cooling Water Isolation f., g	432
p.	Automatic Blowdown (5-valves)	Same as k.

	Region of Vessel	Significant Transient Cycles	Summation Usage Factor Based on $\sum \frac{ni}{Ni} \leq 0.8$
(1)	Vessel Support	a., f., g., h., i., & p.	0.0858[1]
(2)	Shroud Support	a., f., g., h., i., & p.	0.0005
(3)	Closure Flanges	a., f., g., h., i., m., & p.	0.0230
(4)	Closure Studs	a., f., g., h., i., m., & p.	0.7934
(5)	Nozzles		
	a. Feedwater Nozzle	a., b., c., d., e., f., g., h., k., l., & p.	0.1000
	b. Steam Outlet Nozzle	a., h., l., & p.	0.0076
	c. Emergency Cond. Nozzle	a., a., l., & p.	0.0046
	d. Recirculation Outlet	a., a., l., & p.	0.008
	e. Recirculation Inlet	a., a., l., & p.	Compared to Recirc. Outlet Nozzle 0.008
	f. Core Spray Nozzle	a., h., n., l., & p.	0.0028
	g. CRD Penetration	a., h., o., l., & p.	0.3780

(1) Value includes the effect of a six-valve blowdown.

B.4 Pipe reactions stresses in the reactor pressure vessel nozzle-wall junction were compared to those stresses that would be allowed by Code. (See Exhibit D). The results are shown in the following table:

Nozzle	Maximum Stress Due to Pipe Reaction (psi)	Allowable Stress Limit (psi)
Recirculation Outlet	15,500	22,000
Recirculation Inlet	8,000	22,000
Steam Outlet	18,500	22,000
Feedwater	18,400	22,000
Emergency Condenser	18,300	22,000
Core Spray	18,300	22,000

B.6 The reactor pressure vessel earthquake analysis (Exhibit E) maximum loads are shown in the following tables:

a. Vessel Support Skirt

Loading Conditions	Base Shear (kips)	Base Moment (kip-inches)
Design Earthquake	1,100	211,024
Design Earthquake plus Jet Reaction from Rupture Pipe	1,680	353,248
Twice Design Earthquake	2,200	422,048

The maximum stress intensities in the reactor pressure vessel support skirt from all primary stresses are tabulated below.

Loading Conditions	Maximum Primary Stress Intensity (psi)
Design Earthquake	17,220
Design Earthquake plus Jet Reaction from Ruptured Pipe	23,750
Twice Design Earthquake	30,530

The limiting stress intensity for all conditions of earthquake loading is such that no gross deformations or displacements result.

b. **Vessel Stabilizer Brackets**

Loading Conditions	Total Shear (kips)
Design Earthquake	352
Design Earthquake plus Jet Reaction from Ruptured Pipe	839
Twice Design Earthquake	704

The maximum primary bending stress intensity in the stabilizer bracket and in the vessel shell are tabulated below:

Loading Conditions	Maximum Primary Bending Stress Intensity in Bracket	Maximum Primary Bending Stress Intensity in Vessel Shell
Design Earthquake	18,800 psi	17,500 psi
Design Earthquake plus Jet Reaction from Ruptured Pipe	41,400 psi	41,900 psi
Twice Design Earthquake	36,000 psi	35,000 psi

These values are to be compared to the yield strength of the bracket and shell material at temperature which is 42,860 psi at 375°F.

APPENDIX 11

Sect. V, Seismic Analysis Results
of Feedwater Coolant Injection System
Amendment 38

V. Seismic Analysis Results of Feedwater Coolant Injection System

Analyses have been performed on the components of the FWCI system to determine the capability of this system to meet the Class I requirements as defined in Section V-3 of the Facility Description and Safety Analysis Report.

The results of these analysis indicate that all but five categories of equipment meet the requirements for Class I systems. These are: (1) high pressure feedwater heaters, (2) intermediate feedwater heaters, (3) condensate mixed bed demineralizers, (4) condensate storage tank, and (5) piping. Where practical, items of equipment in these five categories will be supported or anchored such that they will meet Class I. Otherwise, such items will be evaluated as to the magnitude of ground motion that they are capable of withstanding. In no case will components be permitted to have a design capability for less than 0.5g ground acceleration.

The 0.05g ground motion is based on the USGS report for the Oyster Creek site which indicates that a range of 0.03 to 0.05g to be the highest intensity earthquake which is predicted to occur in this area. The value (0.11g) given in Section V-3 of the FDSAR represents a design basis value having additional conservatism.

John A. Blume and Associates performed evaluations on the equipment of the feedwater system using the following procedures:

1. Use the peak values of Housner's Earthquake Acceleration Response Spectrum given in his report for Jersey Central date March 6, 1964, normalized to 0.1 g ground acceleration. (See Figure V-3-1, Vol II, Facility Description and Safety Analysis Report.)

2. Using this acceleration, check the adequacy of the anchor bolts and support feet or legs if this means of support is used. If stresses are within allowable, no further analysis is required. For anchor bolts embedded in concrete, the embedment lengths were to be those required in the Uniform Building Code, 1967 Edition, for the allowable forces used in these analyses. Actual embedment lengths are all greater than those assumed.

3. If stresses exceed allowable, as the result of the analysis performed in '2' above, make a conservative estimate of the period of the equipment. Using this period and the referenced acceleration response spectrum, recheck the anchor bolts and support feet. If stresses are within allowable, no further analysis will be required.

4. If stresses exceed allowable as the result of the analysis performed in '3' above, make recommendations as to the modifications that will be required so that the equipment will meet the requirements of Class I.

It is believed that this procedure is adequate since the most of the equipment is located on the basement floor of the turbine building, and Housner's Ground Response Spectra can be used for this floor as a Floor Response Spectra.

A. Components Which Meet Class I Requirements

The results of this evaluation show that the following items meet Class I requirements:

1. Condensate Pumps

The total weight of the subject piece of equipment is 28,500 pounds. The weight of pump is assumed to include the weight of water. The anchorage arrangement consists of four 1-1/2-inch-diameter anchor bolts embedded in concrete foundation. Applying a static coefficient of 0.33g in the horizontal direction, the maximum seismically induced lateral force is calculated as 8,600 pounds. The coefficient of 0.33g represents the peak spectral response for 1.0% damping. This lateral force produces the following stresses in the anchor bolts.

(i) Shear force in one anchor bolt = 2,170 lbs.

(ii) Tensile stress in the anchor bolts = 4,300 psi.

The allowable shear force for one 1-1/2-inch-diameter concrete embedded anchor bolt is 6,500 pounds. Assuming the yield strength of steel is 40,000 psi, the allowable stress in tension is 24,000 (0.6 fy) pounds per square inch. The induced seismic forces and stresses are, therefore, well within the permissible values.

The embedment of the anchor bolts are assessed as sufficient to resist both the tensile as well as shear forces.

Based on our analysis and the above discussion, it is concluded that the anchor bolts are adequate to withstand the seismic forces.

2. Steam Jet Air Ejector Inter Condenser, and

3. Steam Jet Air Ejector After Condenser

The flooded weight of both the condensers is 30,000 pounds. This weight has been distributed for each condenser in the ratio of the cross-sectional areas of their shells. The support and anchorage arrangements consist of two 3/4-inch-thick plate-saddles for each condenser with two 7/8-inch anchor bolts in each saddle. The condensers rest on concrete foundations. Applying a static coefficient of 0.33g in the horizontal direction, the maximum seismically induced lateral force is calculated as 4330 pounds for the inter condenser. The coefficient of 0.33g represents the peak spectral response for 1.0% damping. The shear force for each bolt, tensile stress in the bolts and bending stress in the saddle plates resulting from the application of this force are noted below. These stresses represent the maximum values encountered for the lateral force acting parallel or orthogonal to the longitudinal axis of the condenser.

(i) Shear force for each bolt = 1,080 pounds.

(ii) Tensile stress in the bolts = 3,100 psi.

(iii) Bending stress in the saddle plate = 1,500 psi.

to be insulated except for thermal redistribution among the rods. Steam cooling is

The allowable shear force for one 7/8-inch-diameter concrete-embedded anchor bolt is 2,000 pounds. Using a yield strength for steel of 40,000 psi, the allowable stresses in tension and bending are calculated to be 24,000 (0.6 fy) and 26,400 (0.66 fy) psi, respectively. The induced seismic forces and stresses are, therefore, well within permissible values.

The weight of the after condenser is less than the weight of the inter condenser. Since both condensers have similar anchorage arrangements, the seismic stresses in the anchorage arrangements of the after condenser will be smaller than those in the inter condenser and, therefore, well within the permissible values.

Based on our analysis and the above discussion it is concluded that the anchor bolts and the saddle plates are adequate to withstand the seismic forces.

4. Steam Packing Exhauster

The total weight of the exhauster is 12,370 pounds. This includes the weight of the tube bundle. The supporting and anchorage arrangements consist of two 1/2-inch-thick plate saddles with two 7/8-inch-diameter anchor bolts in each saddle. The bolt holes in one saddle support are slotted in the longitudinal direction. The lateral force in the longitudinal direction is, therefore, resisted by anchor bolts in one saddle support only. The exhauster rests on a concrete foundation. Applying a static coefficient of 0.33g in the horizontal direction, the maximum seismically induced lateral force is calculated as 4,080 pounds. The coefficient of 0.33g represents the peak spectral response for 1.0% damping. This lateral force produces the following stresses in the supporting saddles and anchor bolts. These stresses represent the maximum values encountered for the lateral force acting parallel or orthogonal to the longitudinal axis of the exhauster:

(i) Shear force in one anchor bolt = 2,040 lbs.

(ii) Tensile stress in the anchor bolts = 4,890 psi.

(iii) Bending stress in saddle plate = 21,400 psi.

The allowable shear force for one 7/8-inch-diameter concrete-embedded anchor bolt is 3,000 pounds. Using a yield strength for steel of 40,000 pounds per square inch, the allowable stresses in tension and bending are calculated to be 24,000 (0.6 fy) and 26,400 (0.66 fy) psi, respectively. The induced seismic forces and stresses are, therefore, quite close to permissible values and are acceptable.

The embedment of the anchor bolts are assessed as sufficient to resist both the tensile as well as shear forces.

Based on our analysis and the above discussion, it is concluded that the anchor bolts and the saddle plates are adequate to withstand the seismic forces.

assumption that the cladding achieved by the released steam flow is no greater than that

5. Condensate Demineralizer Cation Tank

The condensate demineralizer cation tank is supported on four 4 × 4 × 3/8-inch single-angle legs welded to 6 × 6 × 3/4-inch base plates. Each base plate in turn is secured to the concrete foundation with one 7/8-inch anchor bolt. The flooded weight of the tank is 23,500 pounds and its fundamental period is calculated as 0.08 second. Applying a static coefficient of 0.20g in the horizontal direction, the maximum seismically induced lateral force is calculated as 4,700 pounds. The coefficient of 0.20g represents the spectral response corresponding to a period of 0.08 second with 1.0 percent damping. If the lateral force is applied parallel to the line joining the two diagonally opposite legs, these legs will share lesser shearing forces as compared to the other legs due to their lesser relative rigidity in this direction. The following stresses are obtained as a result of application of the seismically induced force at the mid-height of the tank shell. These stresses represent the maximum values encountered for the lateral force acting parallel or orthogonal to the direction discussed above.

(i) Shear force in one anchor bolt = 1,730 lbs.

(ii) Tensile stress in the bolts = 1,800 psi.

(iii) For each leg $\frac{fa}{Fa} + \frac{fb}{Fb} < 1$

The allowable shear force for one 7/8-inch-diameter concrete-embedded anchor bolt is 3,000 pounds. The seismically induced shear force in the anchor bolts, resultant compressive stresses in the legs and tensile stresses in the bolts are, therefore, within the permissible limits.

Based on our analysis and the above discussion, it is concluded that the legs and anchor bolts are adequate to withstand the seismic forces.

6. Condensate Demineralizer Anion Tank

The condensate demineralizer anion tank is supported on four 3 × 3 × 3/8 inch single-angle legs welded to 4-1/2 × 4-1/2 × 3/8 inch base plates. Each base plate in turn is secured to the concrete foundation with one 7/8-inch anchor bolt. The flooded weight of the tank is 10,000 pounds and its fundamental period is calculated as 0.12 second. Applying a static coefficient of 0.215g in the horizontal direction, the maximum seismically induced lateral force is calculated as 2,150 pounds. The coefficient of 0.215g represents the spectral response corresponding to a period of 0.12 second with 1.0 percent damping. If the lateral force is applied parallel to the line joining the two diagonally opposite legs, these legs will share lesser relative rigidity in this direction. The following stresses are obtained as a result of application of the seismically induced lateral force at the mid-height of the tank shell. These stresses represent the maximum values encountered for the lateral force acting parallel or orthogonal to the direction discussed above.

(i) Shear force in one anchor bolt = 900 lbs.

(ii) Tensile stress in the bolts = 3,500 psi

(iii) For each leg $\frac{fa}{Fa} + \frac{fb}{Fb} < 1$

1/4 mile, would be 1.5 × 10^{-4} rem to the thyroid gland. This exposure is based upon the assumption that the elevation achieved by the released steam cloud is no greater than that

5. Condensate Demineralizer Cation Tank

The condensate demineralizer cation tank is supported on four 4 × 4 × 3/8-inch single-angle legs welded to 6 × 6 × 3/4-inch base plates. Each base plate in turn is secured to the concrete foundation with one 7/8-inch anchor bolt. The flooded weight of the tank is 23,500 pounds and its fundamental period is calculated as 0.08 second. Applying a static coefficient of 0.20g in the horizontal direction, the maximum seismically induced lateral force is calculated as 4,700 pounds. The coefficient of 0.20g represents the spectral response corresponding to a period of 0.08 second with 1.0 percent damping. If the lateral force is applied parallel to the line joining the two diagonally opposite legs, these legs will share lesser shearing forces as compared to the other legs due to their lesser relative rigidity in this direction. The following stresses are obtained as a result of application of the seismically induced force at the mid-height of the tank shell. These stresses represent the maximum values encountered for the lateral force acting parallel or orthogonal to the direction discussed above.

(i) Shear force in one anchor bolt = 1,720 lbs.

(ii) Tensile stress in the bolts = 1,900 psi

(iii) For each leg $\frac{fa}{Fa} + \frac{fb}{Fb} < 1$

The allowable shear force for one 7/8-inch-diameter concrete-embedded anchor bolt is 3,000 pounds. The seismically induced shear force in the anchor bolts, resultant compressive stresses in the legs and tensile stresses in the bolts are, therefore, within the permissible limits.

Based on our analysis and the above discussion, it is concluded that the legs and anchor bolts are adequate to withstand the seismic forces.

6. Condensate Demineralizer Aging Tank

The condensate demineralizer anion tank is supported on four 3 × 3 × 3/8 inch single-angle legs welded to 4-1/2 × 4-1/2 × 5/8 inch base plates. Each base plate in turn is secured to the concrete foundation with one 7/8-inch anchor bolt. The flooded weight of the tank is 10,000 pounds and its fundamental period is calculated as 0.12 second. Applying a static coefficient of 0.215g in the horizontal direction, the maximum seismically induced lateral force is calculated as 2,150 pounds. The coefficient of 0.215g represents the spectral response corresponding to a period of 0.12 second with 1.0 percent damping. If the lateral force is applied parallel to the line joining the two diagonally opposite legs, these legs will share lesser relative rigidity in this direction. The following stresses are obtained as a result of application of the seismically induced lateral force at the mid-height of the tank shell. These stresses represent the maximum values encountered for the lateral force acting parallel or othogonal to the direction discussed above.

(i) Shear force in one anchor bolt = 900 lbs.

(ii) Tensile stress in the bolts = 2,500 psi

(iii) For each leg $\frac{fa}{Fa} + \frac{fb}{Fb} < 1$

The allowable shear force for one 7/8-inch-diameter concrete embedded anchor bolt is 3,000 pounds. The seismically induced shear force in the anchor bolts, resultant compressive stresses in the legs and tensile stresses in the bolts are, therefore, within the permissible limits.

Based on our analysis and the above discussion, it is concluded that the legs and the anchor bolts are adequate to withstand the seismic forces.

7. Condensate Demineralizer Resin Storage Tank

The condensate demineralizer resin storage tank is supported on four 4 × 4 × 3/8 inch single-angle legs resting on 8 × 8 × 3/4 inch base plates. Each base plate in turn is secured to the concrete foundation with one 7/8-inch anchor bolt. The flooded weight of the tank is 18,500 pounds and its fundamental period is calculated as 0.11 second. Applying a static coefficient of 0.225g in the horizontal direction, the maximum seismically induced lateral force is calculated as 4,160 pounds. The coefficient of 0.225g represents the spectral response corresponding to a period of 0.11 second with 1.0 percent damping. If the lateral force is applied parallel to the line joining the two diagonally opposite legs, these legs will share lesser shearing forces as compared to the other legs due to their lesser relative rigidity in this direction. The following stresses are obtained as a result of application of the seismically induced lateral force at the mid-height of the tank shell. These stresses represent the maximum values encountered for the lateral force acting parallel or orthogonal to the direction discussed above.

(i) Shear force in one anchor bolt = 1,230 lbs.

(ii) For each leg $\frac{fa}{Fa} + \frac{fb}{Fb} < 1$

The allowable shear force for one 7/8-inch-diameter concrete-embedded anchor bolt is 3,000 pounds. The seismically induced shear force in the anchor bolts and the resultant compressive stresses in the legs are within the permissible limits. The anchor bolts are not stressed in tension.

Based on our analysis and the above discussion, it is concluded that the legs and the anchor bolts are adequate to withstand the seismic forces.

8. Dilution Hot Water Tank

The dilution hot water tank is supported on four 4 × 4 × 3/8 inch single-angle legs welded to 8 × 8 × 3/4 inch base plates. Each base plate in turn is secured to the concrete foundation with one 7/8-inch anchor bolt. The flooded weight of the tank is 17,200 pounds and its fundamental period is calculated as 0.08 second. Applying a static coefficient of 0.20g in the horizontal direction, the maximum seismically induced lateral force is calculated as 3,440 pounds. The coefficient of 0.20g represents the spectral response corresponding to a period of 0.08 second with 1.0 percent damping. If the lateral force is applied parallel to the line joining the two diagonally opposite legs, these legs will share lesser shearing forces as compared to the other legs due to their lesser relative rigidity in this direction. The following stresses are obtained as a result of application of the seismically induced lateral force at the mid-height of the tank

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stresses represent the maximum values encountered for the lateral force acting parallel or orthogonal to the longitudinal axis of the Cooler.

(i) Shear force in one anchor bolt = 3,800 lbs.

(ii) Tensile stress in the anchor bolts = 8,100 psi

(iii) Bending stress in saddle plate = 20,800 psi

The allowable shear force for one 1-1/4 inch diameter concrete embedded anchor bolt is 3,500 pounds. Using a yield strength for steel of 40,000 psi, the allowable stresses in tension and bending are calculated to be 24,000 (0.6 fy) and 26,400 (0.66 fy) psi respectively. The induced seismic forces and stresses are, therefore, quite close to the permissible values and are acceptable.

Based on our analysis and the above discussion, it is concluded that the anchor bolts and the saddle plates are adequate to withstand the seismic forces.

11. <u>Feedwater Pumps</u>

The total weight of each feedwater pump is 35,050 pounds. The weight of the pump is assumed to include its weight of water. The anchorage arrangement consists of eight 1-1/4-inch anchor bolts. Applying a static coefficient of 0.33g in the horizontal direction, the maximum seismically induced lateral force is calculated as 11,700 pounds. The coefficient of 0.33g represents the peak spectral response in the anchor bolts. These stresses represent the maximum values encountered for the lateral force acting parallel or orthogonal to the longitudinal axis of the pump.

(i) Shear force in one anchor bolt = 1,800 lbs.

(ii) Tensile stress in the bolts = 400 psi

The allowable shear force for one 1-1/4-inch-diameter concrete-embedded anchor bolt is 3,500 pounds. Using a yield strength for steel of 40,000 psi, the allowable stress in tension is calculated to be 24,000 (0.6 fy) psi. The induced seismic forces and stresses are, therefore, well within permissible limits.

The embedment of the anchor bolts are sufficient to resist both the tensile as well as shear force.

Based on our analysis and the above discussion, it is concluded that the anchor bolts are adequate to withstand the seismic forces.

12. <u>Main Condenser</u>

The operating weight of the main condenser is 1485 kips. This value includes the estimated weights of 35 kips and 85 kips for the drain cooler and the low pressure heater, respectively.

The main condenser is welded to four 3-1/2-inch-thick steel base plates, each anchored to the foundation by six 1-1/2-inch and six 1-1/4-inch-diameter bolts. At the surface of the foundation, the seismically induced lateral force is, therefore, resisted by twenty-four 1-1/2-inch and twenty-four 1-1/4-inch-diameter bolts. The condenser is held down to each base plate by six 1-1/2-inch-diameter bolts passing through the holes slotted in both longitudinal and transverse directions in the hold-down brackets which, in turn are welded to the condenser. The condenser foot pads (2-1/4-inch-thick plates) butt against triangular centering lugs welded to the steel base plates. The positioning of these lugs is such as to make them act as seismic restraints resisting the lateral force at the surface of the steel base plates coated with lubricant and also permit free thermal expansion of the condenser in both longitudinal and transverse directions.

The main condenser is a very rigid piece of equipment resting directly on the foundation is rigidly coupled to it, and would, therefore be subjected to a maximum horizontal acceleration equal to that of the ground. Applying a static coefficient of 0.11g to the horizontal direction, the maximum seismically induced lateral force is calculated as 164 kips. The coefficient of 0.11g represents the maximum ground acceleration at the site.

The seismic lateral force is resisted by welds connecting the centering lugs to the steel base plates at the surface of the base plates, and by twenty-four 1-1/2-inch-diameter and twenty-four 1-1/4-inch-diameter bolts at the surface of the foundation.

Assuming that the allowable stress in the welds is 13.6 kips per square inch, the centering lugs welds can withstand a lateral force equal to 28x kips. Further, using an allowable shear force of 3.5 kips for one 1-1/4-inch-diameter and 4.6 kips for one 1-1/2-inch-diameter concrete-embedded bolt, the anchor bolts have the capability to withstand a lateral force equal to 180 kips at the surface of the foundation. These values are greater than the induced lateral force of 164 kips. The anchor bolts are not stressed in tension.

Based on our analysis and the above discussion, it is concluded that the anchor bolts and the centering lugs welds are adequate to withstand the required seismic forces.

B. <u>Components which Do Not Meet Class I Requirements</u>

The following components have been analyzed and do not meet Class I requirements:

1. <u>Condensate Demineralizer Mixed Bed Tank (3 Tanks)</u>

Each condensate demineralizer mixed bed tank is supported on four 6 × 6 × 3/4 inch single-angle legs resting on 8-1/2 × 8-1/2 × 3/4 inch base plates. Each base plate in turn is secured to the concrete foundation with one 7/8 inch anchor bolt. The flooded weight of the tank is 38,000 pounds and its fundamental period is calculated as 0.20 second. Applying a static coefficient of 0.30g in the horizontal direction, the maximum seismically induced lateral force is calculated as 12,150 pounds. The coefficient of 0.30g represents the spectral response corresponding to a period of 0.20 second with 1.0 percent damping. If the lateral force is applied parallel to the line joining the two diagonally opposite legs, those legs will share lesser shearing forces as compared to the other legs due to their lesser relative rigidity in this direction. The following stresses are obtained as a result of application of the seismically induced lateral force at the

...id height of the tank shell. These stresses represent the maximum values encountered for the lateral force acting parallel or orthogonal to the direction discussed above.

(i) Shear force in one anchor bolt = 4,000 lbs.

(ii) Tensile stress in the bolts = 3,370 psi

(iii) For each leg $\frac{fa}{Fa} + \frac{fb}{Fb} < 1$

The allowable shear force for one 7/8 inch-diameter concrete-embedded anchor bolt is 3,000 pounds. The seismically induced shear force in the anchor bolts is, therefore, more than the permissible limits. The resultant compressive stresses in the legs and tensile stresses in the bolts are within the permissible limits.

Based on our analysis and the above discussion, the anchorage arrangement will be redesigned and made strong enough to withstand the seismic forces.

2. Intermediate Pressure Feedwater Heaters (3 heaters)

The total flooded weight of the intermediate pressure feedwater heaters is 114,000 pounds. Each heater rests on four supports. Three of these supports have rollers attached to their bases and the fourth is secured to the foundation with four 1-3/8 inch diameter anchor bolts. In the absence of any guides for the rollers in the longitudinal direction, it is assumed that the seismically induced lateral forces and moments both in the longitudinal and transverse directions will be resisted by the anchored support only. Applying a static coefficient of 0.33g in the horizontal direction, the maximum seismically induced lateral force is calculated as 37,620 pounds. The coefficient of 0.33g represents the peak spectral response for 1.0 percent damping.

The shear force in each anchor bolt resulting from the lateral force acting parallel to the longitudinal axis of the heater is 9,400 pounds which is very high as compared to the permissible value of 4,000 pounds for one 1-1/8 inch-diameter concrete-embedded anchor bolt. The shear force in each anchor bolt is increased to 73,900 pounds as a result of direct shear and reactions developed at the bolts due to moment produced by the seismically induced lateral force acting orthogonal to the longitudinal axis at the mid-length of the heater.

Based on our analysis and the above discussion, the supporting and anchorage arrangements will be redesigned and made strong enough to resist the seismically induced forces and moments.

3. High Pressure Feedwater Heaters (3 Heaters)

The total flooded weight of each high pressure feedwater heater is 155,000 pounds. The heater is resting on four supports. Three of these supports have rollers attached to their bases and the fourth is secured to the foundation with four 1-3/8 inch diameter anchor bolts. In the absence of any guides for the rollers in the longitudinal direction, it is assumed that the seismically induced lateral forces and moments both in the longitudinal and transverse directions will be resisted by the anchored support only. Applying a static coefficient of 0.33g in the

horizontal direction, the maximum seismically induced lateral force is calculated as 51,150 pounds. The coefficient of 0.33g represents the peak spectral response for 1.0 percent damping.

The shear force in each anchor bolt resulting from the lateral force acting parallel to the longitudinal axis of the heater is 12,800 pounds which is very high as compared to the permissible value of 4,000 pounds for one 1-3/8-inch-diameter concrete-embedded anchor bolt.

The shear force in each anchor bolt is increased to 81,000 pounds as a result of direct shear and reactions developed at the bolts due to moment produced by the seismically induced lateral force acting orthogonal to the longitudinal axis at the mid-length of the heater.

Based on our analysis and the above discussion, the supporting and anchorage arrangements will be redesigned and made strong enough to resist the seismically induced forces and moments.

4. Condensate Storage Tank

The condensate storage tank is a circular tank having an outside diameter of 45.0 feet. The tank is made of aluminum plates varying in thickness from 0.081 inch at the base to 0.250 inch at the top. The height of the tank is 45.0 feet. The height of the liquid in the tank has been assumed to be 43.5 feet which corresponds with the invert of the 12-inch overflow shell nozzle. The tank rests on a concrete foundation placed directly on the ground.

For the purpose of this analysis the tank was idealized as a two mass system to include the hydrodynamic effects. The fundamental period for the oscillating water in the tank was found to be 3.88 seconds, and the tank with the constrained liquid was assumed to be rigidly coupled to the ground through the tank walls, undergoing a maximum horizontal acceleration equal to that of the ground. The tank was subjected to the spectral accelerations corresponding to 0.11g ground acceleration.

The induced forces due to the design earthquake, including the hydrodynamic effects of the liquid in the tank are:

(i) Maximum Overturning Moment at the base = 12,000 k-ft.

(ii) Maximum Bending Moment at the base = 9,400 k-ft.

(iii) Maximum Shear Force at the base = 484 kips

The anchorage of the tank to the foundation will be designed to resist the overturning moment and shear force noted above. The friction between the tank and the foundation was not relied upon to resist the seismically induced forces. The tank is secured to the concrete foundation by twelve 1-1/4 inch-diameter anchor bolts. The seismically induced shear force for each bolt is calculated to be 40.3 kips. The maximum recommended allowable shear force for one 1-1/4-inch diameter concrete-embedded anchor bolt is 3.5 kips. In addition, these anchor bolts will be subjected to bending moments caused by shearing forces acting at approximately 8 inches above the surface of the concrete foundation.

Some alternative anchorage arrangements such as a restraining skirt will be installed to resist the seismic lateral force. The bottom plate will remain unstiffened but an anchorage arrangement will provide positive resistance against overturning.

The maximum bending stress in the shell at the base when subjected to the above bending moment is 0.835 kips per square inch. Assuming yield strength of the aluminum as 34 kips per square inch and its modulus of elasticity as 10.3×10^3 kips per square inch, the buckling stress is calculated as 5,890 kips per square inch. The shell is, therefore, safe against buckling.

Sloshing of the fluid inside the tank was found to be 0.78 foot for the 0.11g ground acceleration and 1.88 feet for the 0.22g ground acceleration. Considering 1.50 feet of freeboard, the roof will be subjected to an upward pressure corresponding to 1.88 - 1.50 = 0.38 foot of water near the tank wall for the shutdown earthquake. For the design earthquake the freeboard being more than the sloshing of the liquid, the roof will not be subjected to an upward pressure. The intensity of pressure corresponding to 0.18 foot of water is 10 lb/sq ft. This intensity being low and considering the fact that the tank is provided with a 12-inch diameter overflow nozzle, the upward forces because of sloshing would be negligible.

Rigid connection of the 12 inch drain pipe at the penetrations has been avoided.

Based on our analysis and above discussions, the following is concluded:

(i) The anchorage arrangement to secure the tank to the concrete foundation will be designed to withstand the seismic forces.

(ii) The compressive stresses in the tank shell are such that buckling will not be induced.

(iii) The upward forces on the roof because of the sloshing liquid will be negligible and can be ignored.

5. Piping

The FWCI piping in the turbine building has been evaluated for meeting Class I standards by using a procedure described below. The response acceleration used for this evaluation are those presented in Section II-5 and V-3 of the Facility Description and Safety Analysis Report.

The Procedure was as follows:

a. Knowing the period of the supporting building or structure, establish the period of the piping under conditions wherein the piping is rigid, flexible or resonant.

b. Establish the maximum span for piping of various diameters to sustain a loading of 0. g and not be stressed more than 1500 psi. (See Table 121.4 in Power Piping USAS B 31.10, 1967.)

c. Establish the resonant limits for piping of various diameters by using the curves of John A. Blume and Associates (Plate V-1), which gives natural periods as a function of pipe size and span.

d. After selection of the length of pipe span, Plates V-2 and V-3 are used to determine the deflection and reaction at the supports.

The displacement and support reactions are increased, where required, by a factor of three due to magnification of response of the equipment over ground acceleration.

Span lengths are reduced to account for valves or branch lines. For 90-degree bends, either leg shall be not more than L/2 where L is the sum of the span length for both legs.

Supports are located in a manner so as to avoid the resonant range. Pipe spans that fall in the rigid range as indicated by Plate V-1 will have acceptable deflections, reactions and stresses and will require no further analysis. Pipe spans that fall in the flexible range will be analyzed for seismic loads based on Plate V-1 and Housner's Acceleration Response Spectrum. Stresses induced from these loads will be held to acceptable values.

The supports and snubbers for piping in this category will be designed to meet the above criteria.

6. Others

The ventilation system for the feedwater pump room is being evaluated with respect to Class I requirements.

The turbine building, radwaste building and pipe tunnels are being evaluated on the basis that failure of these structures would not prevent proper functioning of the FWCI system.

TEMPERATURE

DN-50-363

Jersey Central

TELEPHONE REPORT

DATE May 14, 1979 TIME 11:10 Am

FACILITY Forked River DOCKET NO. 50-363

LICENSEE'S OCCURRENCE IDENTIFICATION NO. (IF ANY) NONCONFORMANCE REPORT #0856

BRIEF SUBJECT: Verbal Report / Concrete Record Deficiency

DESCRIPTION OF OCCURENCE, DEFICIENCY OR INCIDENT:

Two concrete columns in Auxiliary Building Engineering Safety Features Pit, ~~were~~ placed in August 1978, were very recently found to lack documentation of the required cure period. GPU's Mr. Raychech reported this by telephone and inquired if it required a written report under 10 CFR 50.55e. I advised him it appears not to be a significant Deficiency.

NOTIFICATION RECEIVED BY A. C. Varela

RO:I Form 50
June 74

Zerox copies: SDE, RWMcG, LN

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(79-63)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

June 11, 1979

Docket No. 50-219

Mr. I. R. Finfrock, Jr.
Vice President - Generation
Jersey Central Power & Light Company
Madison Avenue at Punch Bowl Road
Morristown, New Jersey 07960

Dear Mr. Finfrock:

RE: IMPLEMENTATION OF CATEGORY 2 AND 3 REGULATORY GUIDES IN THE SYSTEMATIC EVALUATION PROGRAM - OYSTER CREEK NUCLEAR GENERATING STATION

In the Systematic Evaluation Program for your facility we plan to address Regulatory Guides and other staff positions that have been classified as Category 2 or 3 for implementation by our Regulatory Requirements Review Committee.

Category 2 and Category 3 are defined as follows:

Category 2: Further staff consideration of the need for backfitting appears to be required for certain identified items of the regulatory position - these individual issues are such that existing plants need to be evaluated to determine their status with regard to these safety issues to determine the need for backfitting.

Category 3: Clearly backfit. Existing plants should be evaluated to determine whether identified items of the regulatory position are resolved in accordance with the guide or by some equivalent alternative.

For your information, a list of the Category 2 and 3 positions that we currently plan to address is provided in Enclosure 1. The SEP topics under which these issues will be considered is also shown.

A copy of one Category 3 position, Regulatory Guide 1.114, Revision 1, "Guidance on Being Operator at the Controls of a Nuclear Power Plant", is provided in Enclosure 2. To complete our evaluation of this specific guide you are requested to provide a commitment to meet the recommendations of the guide.

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Please contact the assigned Project Manager for your facility if you have any questions or comments about these matters.

Sincerely,

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. List of Category 2 and 3 Positions
2. Regulatory Guide 1.114

cc w/enclosures:
See next page

cc w:enclosures:
G. F. Trowbridge, Esquire
Shaw, Pittman, Potts and Trowbridge
1800 M Street, N. W.
Washington, D. C. 20036

GPU Service Corporation
ATTN: Mr. E. G. Wallace
Licensing Manager
260 Cherry Hill Road
Parsippany, New Jersey 07054

Anthony Z. Roisman
Natural Resources Defense Council
917 15th Street, N. W.
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Toms River, New Jersey 08753

Joseph W. Ferraro, Jr., Esquire
Deputy Attorney General
State of New Jersey
Department of Law and Public Safety
1100 Raymond Boulevard
Newark, New Jersey 07012

Ocean County Library
Brick Township Branch
401 Chambers Bridge Road
Brick Town, New Jersey 08723

K M C, Inc.
ATTN: Richard E. Schaffstall
1747 Pennsylvania Avenue, N. W.
Suite 1050
Washington, D. C. 20006

ENCLOSURE 1

CATEGORY 2 AND 3 MATTERS TO BE ADDRESSED IN THE SEP

CATEGORY 2 MATTERS

Document Number	Revision	Date	Title	Topic
RG 1.27	2	1/76	Ultimate Heat Sink for Nuclear Power Plants	II-3.C
RG 1.52	1	7/76	Design, Testing, and Maintenance Criteria for Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants (Revision 2 has been published but the changes from Revision 1 to Revision 2 may, but need not, be considered.	VI-8
RG 1.59	2	8/77	Design Basis Floods for Nuclear Power Plants	II-3.B
RG 1.63	2	7/78	Electric Penetration Assemblies in Containment Structures for Light Water Cooled Nuclear Power Plants	III-12
RG 1.91	1	2/78	Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plant Sites	II-1.C
RG 1.102	1	9/76	Flood Protection for Nuclear Power Plants	II-3.B
RG 1.108	1	8/77	Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants	VIII-2
RG 1.115	1	7/77	Protection Against Low-Trajectory Turbine Missiles	III-4.B
RG 1.117	1	4/78	Tornado Design Classification	III-4.A
RG 1.124	1	1/78	Service Limits and Loading Combinations for Class 1 Linear Type Component Supports	III-9 III-11
RG 1.130	0	7/77	Design Limits and Loading Combinations for Class 1 Plate- and Shell-Type Component Supports	III-9 III-11

(Continued)

CATEGORY 2 MATTERS (CONT'D)

Continued

Document Number	Revision	Date	Title	Topic
RG 1.137	0	1/78	Fuel Oil Systems for Standby Diesel Generators (Paragraph C.2)	VIII-2
BTP ASB 9.5-1	1		Guidelines for Fire Protection for Nuclear Power Plants (See Implementation Section, Section D)	IX-6
BTP MTEB 5-7		4/77	Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping	V-4
RG 1.141	0	4/78	Containment Isolation Provisions for Fluid Systems	VI-4

CATEGORY 3 MATTERS

Document Number	Revision	Date	Title	Topic
RG 1.99	1	4/77	Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials (Paragraphs C.1 and C.2.	V-6
RG 1.101	1	3/77	Emergency Planning for Nuclear Power Plants	XIII-1
RG 1.114	1	11/76	Guidance on Being Operator at the Controls of a Nuclear Power Plant	ORB #2
RG 1.121	0	8/76	Bases for Plugging Degraded PWR Steam Generator Tubes	V-8
RG 1.127	1	3/78	Inspection of Water-Control Structures Associated with Nuclear Power Plants	III-3.C
RSB 5-1	1	1/78	Branch Technical Position: Design Requirements of the Residual Heat Removal System	VII-3 V-II.B
RSB 5-2	0	3/78	Branch Technical Position: Reactor Coolant System Overpressurization Protection	V-6
RG 1.97	1	8/77	Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident (Paragraph C.3 - with additional guidance on paragraph C.3.d to be provided later)	VII-5
RG 1.56	1	7/78	Maintenance of Water Purity in Boiling Water Reactors	V-12.A

MATTERS THAT ARE NOT TO BE ADDRESSED

1. Regulatory Guide 1.105 "Instrument Setpoints" - This matter was included in the SEP Topic VII-1.B and was resolved on the basis of the NUREG-0138 discussion of the topic. However, DOR generic action on specific instrumentation such as LPRM drift in BWRs is continuing (such action was noted in the NUREG) and these generic reviews may be expanded if the need to do so is identified.

2. Regulatory Guide 8.8 "Information Relevant to Ensuring That Occupational Radiation Exposure at Nuclear Power Stations Will Be As Low As is Reasonably Achievable (Nuclear Power Reactors)" - This matter is presently being proposed for resolution by DOR/EEB on a generic basis. Since ALARA issues were not included in the initial SEP scope it would be inappropriate to include this one unique issue now.

3. Regulatory Guide 1.68.2 "Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants" -- This matter was determined, by Regulatory Requirements Review Committee, to be applicable to plants in the OL stage of licensing.

ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION

Revision 1
November 1976

REGULATORY GUIDE

OFFICE OF STANDARDS DEVELOPMENT

REGULATORY GUIDE 1.114

GUIDANCE ON BEING OPERATOR AT THE CONTROLS OF A NUCLEAR POWER PLANT

A. INTRODUCTION

Paragraph (k) of §50.54, "Conditions of Licenses," of 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires that an operator or senior operator licensed pursuant to 10 CFR Part 55, "Operators' Licenses," be present at the controls at all times during the operation of a facility. General Design Criterion 19, "Control Room," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 requires, in part, that a control room be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain the nuclear power plant in a safe condition under accident conditions. As defined in 10 CFR §50.2(t), the term "controls," when used with respect to nuclear reactors, means apparatus and mechanisms, the manipulation of which directly affects the reactivity or power level of the reactor. This guide describes a method acceptable to the NRC staff for complying with the Commission's regulations that require an operator to be present at the controls of a nuclear power plant.* The Advisory Committee on Reactor Safeguards has been consulted concerning this guide and has concurred in the regulatory position.

B. DISCUSSION

Operating experience has shown that there is a need for guidance with regard to acceptable methods of complying with the Commission's requirement for the presence of an operator at the controls of a facility. The operator at the controls of a nuclear power plant has many responsibilities, which include but are not limited to (1) adhering to the plant's technical specifications, plant operating procedures, and NRC regulations; (2) reviewing operating data, including data logging and review, in order to ensure safe operation of the plant; and (3) being able to manually initiate engineered safety features during various transient and accident conditions.

In order for the operator at the controls of a nuclear power plant to be able to carry out these and other responsibilities in a timely fashion, he must give his attention to the condition of the plant at all times. He must be alert in order to ensure that the plant is operating safely and must be capable of taking action to prevent any progress toward a condition that might be unsafe. This is facilitated by control room design and layout in which all controls, instrumentation displays, and alarms required for the safe operation, shutdown, and cooldown of the unit are readily available to the operator in the control room.

C. REGULATORY POSITION

1. The operator at the controls of a nuclear power plant should have an unobstructed view of and access to the operational control panels,[2] including instrumentation displays and alarms, in order to be able to initiate prompt corrective action, when necessary, on receipt of any indication (instrument movement or alarm) of a changing condition.

2. The operator at the controls should not normally leave the area where continuous attention (including visual surveillance of safety-related annunciators and instrumentation) can be given to reactor operating conditions and where he has access to the reactor controls. For example, the operator should not routinely enter areas behind control panels where

* Line indicates substantive change from previous issue.

Current Standard Technical Specifications require a licensed operator to be present in the control room at all times while fuel is in the reactor.

Operational control panels are control panels that enable the operator at the controls to perform required manual safety functions and equipment surveillance and to monitor plant conditions under normal and accident conditions. Operational control panels include instrumentation for the reactor, reactor coolant system, containment, and safety-related process systems.

USNRC REGULATORY GUIDES

Regulatory Guides are issued to describe and make available to the public methods acceptable to the NRC staff of implementing specific parts of the Commission's regulations, to delineate techniques used by the staff in evaluating specific problems or postulated accidents, or to provide guidance to applicants. Regulatory Guides are not substitutes for regulations, and compliance with them is not required. Methods and solutions different from those set out in the guides will be acceptable if they provide a basis for the findings requisite to the issuance or continuance of a permit or license by the Commission.

Comments and suggestions for improvements in these guides are encouraged at all times, and guides will be revised, as appropriate, to accommodate comments and to reflect new information or experience. This guide was revised as a result of substantive comments received from the public and additional staff review.

Comments should be sent to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Section.

The guides are issued in the following ten broad divisions:

1. Power Reactors
2. Research and Test Reactors
3. Fuels and Materials Facilities
4. Environmental and Siting
5. Materials and Plant Protection
6. Products
7. Transportation
8. Occupational Health
9. Antitrust Review
10. General

Copies of published guides may be obtained by written request indicating the divisions desired to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Office of Standards Development.

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plant performance cannot be monitored. The operator at the controls should not under any circumstances leave the surveillance area defined by regulatory position 3 for any nonemergency reason (e.g., to confer with others or for personal reasons) without obtaining a qualified relief operator at the controls. In the event of an emergency affecting the safety of operations, the operator at the controls may momentarily be absent from the defined surveillance area in order to verify the receipt of an annunciator alarm or initiate corrective action, provided he remains within the confines of the control room.

3. Administrative procedures should be established to define and outline (preferably with sketches) specific areas within the control room where the operator at the controls should remain. The procedures should define the surveillance area and the areas that may be entered, in the event of an emergency affecting the safety of operations, by the operator at the controls to verify receipt of an annunciator alarm or initiate corrective action.

4. Prior to assuming responsibility for being operator at the controls, the relief operator should be properly briefed on the plant status. In order to ensure that proper relief occurs, administrative procedures should be written to describe what is required. The administrative procedure should include, as a minimum, a definition of proper relief (e.g., what information is required to be passed on and acknowledged between the two operators).

5. A single operator should not assume the operator-at-the-controls responsibility for two or more nuclear power units at the same time.

D. IMPLEMENTATION

The purpose of this section is to provide information to license applicants and licensees regarding the NRC staff's plans for using this regulatory guide.

This guide reflects current NRC staff practice. Therefore, except in those cases in which the applicant or licensee proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein is being and will continue to be used in the evaluation of submittals for operating license or construction permit applications and the performance of licensees until this guide is revised as a result of suggestions from the public or additional staff review.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON D.C. 20555

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

June 6, 1979

Docket No. 50-219

Mr. I. R. Finfrock, Jr.
Vice President - Generation
Jersey Central Power & Light Company
Madison Avenue at Punch Bowl Road
Morristown, New Jersey 07960

Dear Mr. Finfrock:

RE: REQUEST FOR ADDITIONAL INFORMATION
TOPIC VI-1
OYSTER CREEK NUCLEAR GENERATING STATION

To continue our review of Systematic Evaluation Program Topic VI-1, "Organic Materials and Post Accident Chemistry", we request that you submit the information described in the enclosure.

Your response is requested within 30 days of receipt of this letter so that we can maintain our review schedule.

Sincerely,

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosure:
Request for Additional Information

cc w/enclosure:
See next page

8/26

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79-99

cc w/enclosure:
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ATTN: Mr. E. G. Wallace
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Parsippany, New Jersey 07054

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Brick Township Branch
401 Chambers Bridge Road
Brick Town, New Jersey 08723

K M C, Inc.
ATTN: Richard E. Schaffstall
1747 Pennsylvania Avenue, N. W.
Suite 1050
Washington, D. C. 20006

REQUEST FOR ADDITIONAL INFORMATION

INFORMATION NEEDED TO EVALUATE TOPIC VI-1,
"ORGANIC MATERIALS AND POST ACCIDENT CHEMISTRY"

The following information is necessary to complete our evaluation of this topic:

1. Estimate the areas and thicknesses of the major protective coating systems inside containment, including aluminum and zinc base paints, epoxy paints, acrylic lacquer, etc.

2. Indicate whether these coating systems and their methods of application were qualified according to the recommendations of Regulatory Guide 1.54.

3. If not, describe the QA provisions which were used to assure proper application.

4. For coatings not qualified with Regulatory Guide 1.54, describe the present condition of the coatings, including estimates of the amounts of flaking, peeling, cracking, bubbles, etc.

5. Estimate the quantity of other miscellaneous coating materials (such as on snubbers) not qualified according to Regulatory Guide 1.54.

6. Estimate the types and amounts of other organic materials such as electrical insulation inside containment.

7. In addition to the information described above for organic coating systems, we will need the following information with respect to the related topic VI-5, "Combustible Gas Control": Estimate the surface area and thickness of aluminum, zinc and galvanized steel inside containment.

We have reviewed the information available in the docket files for some of the SEP plants and found that the information described above was not available. Since this topic is a fairly recent addition to the review areas for CP's and OL's, we do not expect to find the information in any of the SEP plant docket material.

Accordingly, you are requested to provide the information described above.

June 6, 1979

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUN 5 1979

Docket No. 50-363

MEMORANDUM FOR: Robert L. Baer, Chief, Light Water Reactors Branch No. 2, DPM

FROM: R. A. Benedict, Project Manager, Light Water Reactors Branch No. 2, DPM

SUBJECT: MEETING WITH JERSEY CENTRAL POWER AND LIGHT COMPANY - FORKED RIVER NUCLEAR STATION

DATE & TIME: Wednesday, June 20, 1979 - 10:00 a.m.

LOCATION: Room P-422, Phillips Building
Bethesda, Maryland

PURPOSE: To discuss with the applicant the financial, environmental and radiological safety considerations concerning the JCP&L application of August 31, 1978 for extension of the construction permit (CPPR-96) latest completion date.

PARTICIPANTS:

APPLICANT
R. C. Arnold, Vice President Generation, GPU
J. Graham, Treasurer, GPU
R. W. Heward, Manager, Nuclear Projects, GPU
J. R. Thorpe, Manager, Environmental Affairs, GPU
E. G. Wallace, Manager, Licensing, GPU
G. F. Trowbridge, Attorney, Shaw, Pittman, Potts and Trowbridge

NRC - STAFF
R. S. Boyd, Director, Division of Project Management
D. B. Vassallo, Assistant Director, Light Water Reactors, Division of Project Management
R. L. Baer, Chief, Light Water Reactors Branch No. 2, Division of Project Management
R. A. Benedict, Licensing Project Manager
R. Gilbert, Environmental Project Manager
J. Petersen, Senior Financial Analyst
J. Cutchin, Attorney, OELD

R A Benedict

B/27

R. A. Benedict
Light Water Reactors Branch No. 2
Division of Project Management

ccs: See next page

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ccs:
M. Kenneth Pastor, Project Manager
GPU Service Corporation
260 Cherry Hill Road
Parsippany, New Jersey 07054

Mr. E. G. Wallace
Licensing Manager
GPU Service Corporation
260 Cherry Hill Road
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Steven P. Russo
248 Washington Street
P. O. Box 1060
Toms River, New Jersey 08753

Mr. Ivan R. Finfrock, Jr.
Vice President
Jersey Central Power and Light Company
Madison Avenue at Punch Bowl Road
Morristown, New Jersey 07960

Meeting Notices

Docket File
NRC PDR
Local PDR
TIC
LWR #2 File
NRR Reading
H. Denton
E. Case
D. Crutchfield
D. Bunch
R. Boyd
D. Ross
R. Mattson
R. DeYoung
D. Muller
D. Vassallo
D. Skovholt
W. Gammill
F. Williams
J. Stolz
O. Parr
S. Varga
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C. Heltemes
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Project Manager - R. A. Benedict
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SD(7)
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Receptionist - Phillips Building
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V. Moore
W. Kreger
M. Ernst
R. Denise
R. Ballard
B. Youngblood
W. Regan
G. Chipman
R. Houston
J. Collins
G. Lear
M. Spangler
V. Benaroya
R. Jackson
L. Hulman
H. Ornstein
J. LeDoux, IE
Principal Staff Participants:

R. Gilbert, J. Petersen,
J. Cutchin

OPA

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUN 5 1979

Docket No. 50-363

MEMORANDUM FOR: Robert L. Baer, Chief, Light Water Reactors Branch No. 2, DPM

FROM: R. A. Benedict, Project Manager, Light Water Reactors Branch No. 2, DPM

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R. L. Baer, Chief, Light Water Reactors Branch No. 2, Division of Project Management
R. A. Benedict, Licensing Project Manager
R. Gilbert, Environmental Project Manager
J. Petersen, Senior Financial Analyst
J. Cutchin, Attorney, OELD

R A Benedict
R. A. Benedict
Light Water Reactors Branch No. 2
Division of Project Management

ccs: See next page

~~7907240424~~ 2pp.

Jersey Central Power & Light Company - -

ccs:
M. Kenneth Pastor, Project Manager
GPU Service Corporation
260 Cherry Hill Road
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Mr. E. G. Wallace
Licensing Manager
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Mr. Ivan R. Finfrock, Jr.
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JCP&L / GPU

Jersey Central Power & Light Company
Madison Avenue at Punch Bowl Road
Morristown, New Jersey 07960
(201) 455-8200

June 1, 1979

Director
Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Sir:

Subject: Oyster Creek Nuclear Generating Station
Docket No. 50-219
License Change Request No. 71

By letter dated March 14, 1979, we requested a schedule change for certain fire protection modifications. In subsequent conversations with your staff, it was requested that we submit a License Change Request in order to allow for appropriate review and approval.

Enclosed is the requested License Change. Attachment 1 presents a requested change to Table 3.1 of the NRC Fire Protection Safety Evaluation Report. Attachment 2 presents a discussion of currently planned changes to our program and of the safety significance of the requested change in schedule.

This License Change Request has been reviewed and approved by the Station Superintendent, the Plant Operations Review Committee, and an Independent Safety Review Group in accordance with Section 6 of the Oyster Creek Technical Specifications.

The enclosed submittal has been evaluated and classified in accordance with 10CFR170.22. The change to paragraph 3.E of the Operating License is deemed Pro Forma in nature and therefore is a Class II Amendment. As per 10CFR170.22 enclosed is a check for $1,200.00.

Very truly yours,

Ivan R. Finfrock Jr

Ivan R. Finfrock, Jr.
Vice President

la

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79-101

JERSEY CENTRAL POWER & LIGHT COMPANY
OYSTER CREEK NUCLEAR GENERATING STATION

Provisional Operating
License No. DPR-16

Operating License
Change Request No. 71
Docket No. 50-219

Applicant submits, by this Operating License Change Request No. 71 to the Oyster Creek Nuclear Generating Station Operating License, changes to incorporate supplemental information to the NRC Safety Evaluation Report on the Oyster Creek Nuclear Generating Station's Fire Protection Program.

JERSEY CENTRAL POWER & LIGHT COMPANY

BY Ivan R. Finfrock
VICE PRESIDENT

STATE OF NEW JERSEY)
)
COUNTY OF MORRIS)

Sworn and subscribed to before me this day of , 1979.

Notary Public

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF)
) DOCKET NO. 50-219
JERSEY CENTRAL POWER & LIGHT COMPANY)

CERTIFICATE OF SERVICE

This is to certify that a copy of Operating License Change Request No. 71 for the Oyster Creek Nuclear Generating Station Operating License, filed with the U. S. Nuclear Regulatory Commission on June 1, 1979 has this 1st day of June, 1979, been served on the Mayor of Lacey Township, Ocean County, New Jersey by deposit in the United States Mail, addressed as follows:

The Honorable Mary Lou Smith
Mayor of Lacey Township
P. O. Box 475
Forked River, New Jersey 08731

JERSEY CENTRAL POWER & LIGHT COMPANY

BY Ivan R. Finfrock Jr.
Vice President

DATED: June 1, 1979

Jersey Central Power & Light Company
Madison Avenue at Punch Bowl Road
Morristown, New Jersey 07960
(201) 455-8200

June 1, 1979

The Honorable Mary Lou Smith
Mayor of Lacey Township
P. O. Box 475
Forked River, New Jersey 08731

Dear Mayor Smith:

Enclosed herewith is one copy of License Change Request No. 71 for the Oyster Creek Nuclear Generating Station Operating License.

This document was filed with the U. S. Nuclear Regulatory Commission on June 1, 1979.

Very truly yours,

Ivan R. Finfrock, Jr.

Ivan R. Finfrock, Jr.
Vice President

la

Enclosure

Oyster Creek Nuclear Generating Station
Provisional Operating License DPR-16
(Docket No. 50-219)

Applicant hereby requests the Commission to change Provisional Operating Licensing DPR-16 as follows:

1. Sections to be changed:
   Paragraph 3.E of the License.

2. Extent of changes:
   Addition of SER supplements.

3. Changes Requested
   Change paragraph 3.E of the license to read as follows: (Changes Underlined)

E. The licensee may proceed with and is required to complete the modifications identified in paragraphs 3.1.1 through 3.1.23 of the NRC's Fire Protection Safety Evaluation (SE), and supplements thereto, on the facility dated March 3, 1978. These modifications shall be completed as specified in Table 3.1 of the SE, and supplements thereto. In addition, the licensee shall submit the additional information identified in Table 3.2 of the SE in accordance with the schedule contained therein. In the event these dates cannot be met, the licensee shall submit a report, explaining the circumstances, together with a revised schedule.

4. Discussion:

By letter dated March 14, 1979, JCP&L requested a schedule change for completion of the proposed sprinkler systems and hose station installations required by Table 3.1 of the NRC Safety Evaluation. In the same letter JCP&L also requested that the requirements for the installation of thermally actuated, self closing valves in the diesel generator fuel oil lines be changed to the installation of a second fuel oil line to diesel generator no. 2. In subsequent discussions with the NRC staff, JCP&L was informed that a change to the Operating License would be necessary in order to allow proper consideration of the requested changes. The license change requested herein will allow proper consideration by the NRC staff and provides a mechanism for granting necessary changes provided that there is no undue risk to the health and safety of the public.

Attachment 1

Requested Change to Tabl· 3.1 of the NRC Safety Evaluation

TABLE 3.1

IMPLEMENTATION DATES FOR LICENSE
PROPOSED MODIFICATIONS

Item		Date
3.1.1	Fire Barriers	December 1979
3.1.2	Fire Barrier Penetrations	December 1979
3.1.3	Dampers	December 1979
3.1.4	Fire Detectors	December 1979
3.1.5	Halon Suppression Systems	December 1979
3.1.6	Water Spray Systems	December 1979
3.1.7	Sprinkler Systems	September 15,1979***
3.1.8	Carbon Dioxide Suppression System	December 1979
3.1.9	Hose Stations	September 15, 1979
3.1.10	Aqueous Film Forming Foam	Completed
3.1.11	Portable Extinguishers	Completed
3.1.12	Emergency Breathing Apparatus	Completed
3.1.13	Removal of Combustible Material	Completed
3.1.14	Transformer Dike	December 1979
3.1.15	Diesel Generator Fuel Oil Line	December 1979
3.1.16	Ventilation System Changes	December 1979
3.1.17	Loss of Ventilation Alarm-Battery Room	Completed
3.1.18	Suppression System Valve Control	Completed
3.1.19	Portable Smoke Removal Equipment	Completed
3.1.20	Alternate Water Supply to the Yard Loop	July 1980
3.1.21	Protection From Water Damage	December 1979
3.1.22	New Battery Room and Rerouting Battery Cables	Completed
3.1.23	Remote Shutdown Station	**

**Schedule dependent on equipment availability (not to exceed end of 1980 refueling outage)

***Except for the system extensions in the turbine building condenser bay which shall be completed during the next refueling outage.

Attachment 2

Discussion of Requested Changes

Introduction

In the course of implementing the Oyster Creek Fire Protecton Program, new information and further evaluation have led to some changes in order to better the fire protection facilities. Also, some changes were necessary in order not to adversely affect existing safety systems. These changes include a second fuel oil line to diesel generator no. 2 in lieu of the proposed thermally operated, self closing valves; installation of an automatic water spray system for the cable spread room in lieu of the proposed halon system; installation of an automatic sprinkler system in the south end of the turbine building basement in lieu of the indicated (NRC SE Item 3.1.6) water spray system; and installation of a modified sprinkler system in the fire water pump house.

In addition to the changes indicated above, it is also necessary to request a change in schedule for the completion of the proposed sprinkler systems described in Section 3.1.7 of the NRC Safety Evaluation. This change is necessitated by the procurement situation described in the letter of March 14, 1979, from Mr. I. R. Finfrock, Jr. to the Director, Nuclear Reactor Regulation.

The following presents a discussion of the proposed changes to the Oyster Creek Fire Protection Program and the justification for each change. Also presented is the basis for the requested delay in schedule and the safety significance of such a delay.

DIESEL GENERATOR FUEL OIL LINES

As part of the Oyster Creek Fire Protection Program a commitment was made to install thermally operated, self closing valves in the fuel oil supply lines to the Emergency Diesel Generators. At that time it was believed that the supply lines were in a parallel configuration; therefore, the installation of the self closing valves would allow operation of the redundant diesel in the event of a fire. Further investigation has shown that the fuel oil supply line is common to both diesels. The installation of a thermally operated, self-closing valve to this line would render both diesel genertors inoperable upon closure; therefore, diesel generator reliability would be compromised to an unacceptable level in that onsite electric power must have sufficient independence, redundancy, and testibility to perform their safety function assuming a single active failure. Installation of a self closing valve in the common fuel oil line would introduce a single active failure event that would incapacitate both onsite power sources.

As an alternative, it was decided to install a second fuel oil line to feed diesel generator No. 2 which will be isolable and independent of the fuel oil line to diesel generator No. 1. Also, it was decided to install manual isolation capability rather than thermally acutuated, self closing valves since the installation of this type of valve may result in a decrease in diesel generator reliability. Installation of the second fuel oil line will allow continued operation of one diesel generator should the other become involved in a fire. The new line will provide a means of isolating the main fuel oil storage tank from a fire in either diesel generator while still supplying fuel to the operable diesel generator. This will not only assure diesel generator reliability but also fulfills the intent of the thermally actuated, self closing valve commitment.

CABLE SPREADING ROOM FIRE PROTECTION

Further study of the Fire Hazards Analysis for the Oyster Creek cable spreading room has resulted in a change in approach to fire protection in this area. Originally, it was proposed to install an automatic total flooding halon system in order to provide rapid extinguishment of fires without presenting a hazard to personnel either in the area or those responding to a fire emergency.

During the subsequent design stage of this system, it was found that in order to provide protection in the event of deep-seated fires within the cable trays a higher concentration of halon would be required than was previously planned. Consequently, the principal advantage of the halon system, personnel safety, would be negated. Also it was found that total flooding halon is not recommended for deep-seated fires by NFPA 12-A.

In light of the above and after consultation with our fire protection consultant it was decided to install an automatic, zoned water spray system tracing the existing cable trays in lieu of the halon system. This system will be more effective than the halon and will maintain the personnel safety objective. The effects of water runoff due to actuation of this system either in response to a fire or by inadvertant operation will be addressed in the design phase.

This change will increase the level of protection in this area and further assure the overall safety of the plant. As per the commitment to the NRC staff, design particulars (i.e. water spray density, number and spacing of nozzles, etc.) will be transmitted as this information is developed.

TURBINE BUILDING BASEMENT SPRINKLER SYSTEM

The Oyster Creek Fire Protection Plan indicates that the south end of the turbine building basement will be protected by an automatic sprinkler system in order to protect cables in this area from the effects of an exposure fire. The NRC safety evaluation lists this area under proposed water spray systems; however, it was never intended that this system would be water spray.

Since safety related power cables traversing this area are routed in conduit, the major hazard is exposure to an area fire; therefore, a water sprinkler system, being more effective on this type hazard, was proposed.

By utilizing directional heads at specific junctions of cable trays, this system will also provide the protection committed to for the New Battery Room Installation. Since the main hazard in this area is exposure, the proposed sprinkler system will provide the required protection for cabling in this area.

FIRE PUMP HOUSE FIRE PROTECTION

Originally, it was proposed to install a sprinkler system in the fire water pump house in order to protect the diesel driven fire pumps and their associated fuel oil tanks (located outside the building) from the effects of fire. During the subsequent evaluation of this approach in the design stage, it was decided to install a pre-action sprinkler system inside the building for added reliability and a dry pipe deluge system over the fuel oil tanks in order to eliminate the possibility of freezing in cold weather.

These systems will be installed by the June, 1979 date; however, since the planned detection system will not be installed at this time, the pre-action and deluge valves will be installed in the open position. The system will then be used as a wet pipe sprinkler system (closed fusible heads will be installed over the fuel oil tanks) until the proposed detection system is installed. At this time the pre-action and deluge valves will be connected to the detection system and the closed fusible heads over the fuel oil tanks will be replaced with open heads. The detection system will then actuate the pre-action and deluge valves upon sensing a fire condition thereby providing water to the sprinkler and deluge system piping for fire suppression.

This modified approach maintains the level of protection committed to in the Fire Protection Program. Since measures are being taken to provide the required protection to this area by the June, 1979 date, no delay in schedule is involved and the health and safety of the public is not affected.

SCHEDULE CHANGE

The implementation of the Oyster Creek Fire Protection Program has resulted in major changes in plant procedures and administrative controls which have not only significantly increased control of combustibles and ignition sources but also increased personal awareness of fire hazards and the measures taken to reduce such hazards. New and modified procedures require a permit for welding and cutting operations, fire watches under hazard conditions, restriction of smoking to designated areas, non-combustion generated smoke for leak testing, and the use of flame retardant materials where possible. These measures have been effective in reducing fire hazards throughout the plant.

Even though the probability of a fire occurring has been reduced, other measures have been taken to mitigate the consequences should a fire occur. The station fire brigade has been increased from 3 to 5 members thus providing sufficient personnel to rapidly extinguish the fire, replace any member that is incapacitated, and to supply vital support functions (i.e., breathing air, communications, smoke removal, etc.). The fire brigade members undergo a program of instruction designed to impart the necessary expertise to effectively extinguish fires in the most efficient manner possible. Should a fire be detected by station personnel, the fire brigade can be dispatched immediately which provides assurance that fire damage will be limited to the immediate area.

The measures taken above serve to reduce the probability of fire occurrance by reducing the accumulation of combustibles and controlling ignition sources throughout the plant as well as providing competant and rapid response should a fire occur. Although these and other administative controls (i.e. control of fire doors, staff augmentation, assignment of organizational responsibilities, etc.) in conjunction with passive fire protecton features have provided a significant level of protection, it is recognized that additional means of detection/suppression are required. JCP&L has committed to major installations which will provide this additional protection. Consequently, it was agreed that JCP&L will accomplish these modifications in accordance with the schedule contained in the NRC Safety Evaluation of the Oyster Creek Fire Protection Program and designated Table 3.1.

The schedule contained in Table 3.1 of the NRC safety evaluation indicates that the proposed sprinkler system and hose station installations will be completed by June, 1979; however, due to unforeseen circumstances these systems will not be complete by the indicated date. Being a Public Utility, JCP&L obtains services from vendors through a competitive bidding process in order to assure that services are obtained as economically as possible in the interest of our consumers. In obtaining a vendor for the proposed sprinkler an hose station installations, it became necessary to ask the interested suppliers to submit additional information in order to permit a responsible evaluation of their bids. At the present time a vendor has been selected and is proceeding with the modifications; however, due to the time lost in the bidding process, the completion of these modifications will require more time than allowed by the June, 1979 date.

Sprinkler systems (and other water systems) have been proposed for the (1) Fire Water Pump House (pre-action and deluge), (2) outside of the west wall of the turbine building (deluge), (3) turbine building basement, (4) upper trays in the condenser bay, (5) Monitor & change area, (6) Reactor building elev. 119', and (7) Reactor building elev. 75' in the area of the spent fuel cooling pumps. It is expected that items 1,2, 3, and 5 will be complete by the June, 1979 date; however, the completion of items 6 & 7 as well as the hose station installations are dependent on the installation of a fire water header in the reactor building. Item 4 is an extension to an existing system and is scheduled for completion after the above modifications have been installed. Those systems that may not be completed by the June, 1979 date are the condenser bay extension to upper trays, the reactor building 119' & 75' elev., and several hose station installations.

The planned sprinkler systems for the 119' and 75' elevations of the reactor building will provide overall area protection for the 119' level and protection of the spent fuel pool cooling pumps on the 75' level. There is no safety related equipment on the 119' elevation of the reactor building; therefore, a fire in this area will not affect the safe shutdown of the reactor. Ignition sources in this area are minimal during periods of normal plant operation since there are no process lines through this area, there is no operating machinery, and little if any maintenance activity. This area poses the greatest hazard during periods of plant shutdown with attendent refueling activities. The sprinkler system planned for the 75' elevation provides protection against the loss of both fuel pool cooling pumps. No other safety related equipment is involved so that a fire in this area will not affect the safe shutdown of the reactor. Should a fire occur which would incapacitate the pumps, sufficient time would be available to initiate cooling via the redundant, higher capacity system installed for the core off-loading during the 1977 outage period. A delay in the installation of these sprinkler systems will not present an undue risk to the health and safety of the public since the system for the 119' elevation will be complete prior to refueling operations and adequate means are available to assure cooling of spent fuel in the event of a fire.

In conjunction with the proposed sprinkler systems, hose stations are scheduled to be installed on each level of the reactor building, outside the cable spread room, and outside the control room by the June, 1979 date. All of these hose stations will be installed by this date with the possible exception of those on the 75', 95', and 119' elevations of the reactor building. The probability of fires in these areas has been significantly lessened by the aforementioned administrative controls. Even if such a fire were to occur the continuity of combustibles, location of safety related equipment, and actions required to mitigate the event are such that safe shutdown of the reactor would not be impeded. As mentioned above there is no safety related equipment on the 119' level of the reactor building. Credible fires would be of a class A type and sufficient portable extinguishers are available in this area to provide for extinguishment. The only safety related equipment on the 95' elevation are the liquid poison system and the isolation condensers. Due to the lack of continuity of combustibles, the maximum credible fire would not involve both of these systems, and furthermore, neither of these systems is required for safe shutdown.* Safety related equipment on the 75' elevation consists of the fuel pool cooling system (discussed above), reactor protection instrument racks, and isolation condenser valves. The only combustible material in the area of the spent fuel pool cooling system is the cable insulation above the pumps. As discussed above a redundant system exists approximately 20 feet away. A fire engulfing the cables above the pumps would not affect the redundant system since the cables for this system are run in rigid conduit by a different route; therefore, fuel pool cooling is assured. Although a fire could incapacitate the isolation condenser system, this system is not required for safe shutdown.* The reactor protection instrument racks are separated by distance and lack of continuity of combustibles such that a fire would not incapacitate both racks. The loss of either rack would not prevent safe shutdown. It is therefore concluded that a delay in the installation of the aforementioned hose stations will not adversely affect the health and safety of the public.

The remaining system that may not be installed by the June, 1979 date is the additional sprinkler piping for the upper cable trays along the west wall of the condenser bay. This area is protected by an existing sprinkler system at present and the fire loading in this area is low; therefore, the major hazard to these cables is a fire originating within the tray itself. A fire in these cables may affect the availability of emergency power; however, the normal shutdown systems for the reactor would not be affected. The loss of the cables in this area would not affect rod insertion and various means of heat removal would be available (i.e. bypass to main condenser, shutdown cooling, emergency condensers).

*In relation to a fire emergency

Another problem associated with the installation of additional sprinkler piping in this area is radiation exposure to personnel. An analysis based on radiation surveys have shown that exposures could run as high as 160 man-rem although it is more probable that they would be in the area of 50-75 man-rem. If this work were to be accomplished during a shutdown period radiation exposure could be reduced to very low levels. In light of the ALARA objectives and since a fire in these cables would not affect the safe shutdown of the plant,* assuring the health and safety of the public, it is proposed to install this system during the next refueling outage.

In summary, the requested schedule changes will not adversely affect the health and safety of the public. Administrative controls and an augmented fire brigade assure that the occurrance of a fire is a low probability event and that should a fire occur adequate personnel are available to mitigate the consequences. Finally, even though a fire might develop in these areas where system installation may be delayed, the ability to safely shutdown the plant will not be jeopardized and therefore the health and safety of the public is protected.

*In relation to a fire emergency

50-363

March 20, 1979

MEMORANDUM FOR: Chase R. Stephens, SECY

FROM: Karl Abraham, PAO RI

SUBJECT: LETTER DATED MARCH 7, 1979 FROM SEA ALLIANCE OF OCEAN COUNTY

Please place the enclosed letter in the docket file for the Forked River site.

Karl Abraham
Public Affairs Officer

Enclosure: Ltr dated 3/7/79 from Sally Rush, SEA Alliance of Ocean County

PAO
Abraham/dih
3/20/79

B/30

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79-04

500 Sloop Court [illegible] 1978.
Toms River, N.J. 08753

KA - [illegible]
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Nuclear Regulatory Commission
Office of Public Affairs
Washington DC 20555

Dear Sirs:

Jersey Central Power & Light plan to build a salt water cooling tower at Forked River in New Jersey. This will be the tallest structure of New Jersey and will be on very flat land in the middle of a resort area. This cooling system, as proposed, will have very disasterous affects on the natural beauty of this area. The people of Ocean County wish to protect the beauty of their coast line and feel this tower would definitely destroy the vista of this area. Please respond to this need to preserve the beauty of the environment here as you did in New York. Thank you for your consideration.

Sincerely, Sally Rush
for SEA Alliance of Ocean County

put in FORKED RIVER DOCKET FILE

~~7905150293~~ 2p.