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

September 4, 1979

Docket No. 50-219

LICENSEE: Jersey Central Power & Light Company (JCP&L)

FACILITY: Oyster Creek Nuclear Generating Station

SUBJECT: SITE VISIT OF SEISMIC REVIEW TEAM TO OYSTER CREEK NUCLEAR GENERATING STATION

Members of the Nuclear Regulatory Commission (NRC) staff and their consultants visited the Oyster Creek Nuclear Generating Station on July 17 and 18, 1979. The attendees are listed in Attachment 1. The agenda and outline of the meetings and discussions are provided as Attachment 2.

On the first morning, JCP&L personnel and their consultants described the plant arrangement and showed the locations of the components, systems, and structures identified in the NRC letter to JCP&L dated June 13, 1979. In addition, they described the seismic design of the facility and presented the scope of the seismic review for Oyster Creek, this is included as Attachment 3. They also presented the remaining response to our June 13, 1979 letter which is included as Attachment 4.

In the afternoon a plant tour was conducted so the NRC personnel and their consultants could observe the structures and the supports for piping, mechanical and electrical components associated with those systems identified in the NRC letter as being required for safe shutdown in the event of a postulated seismic event.

On the second day, the NRC staff discussed the results of the previous day's presentation and plant tour. The items identified by JCP&L for additional evaluation and the temporary supports already in place were acknowledged and those items are included in Attachment 5. The NRC staff and their consultants then indicated what additional information is needed to document the seismic resistance capability of the Oyster Creek Nuclear Generating Station. These are listed in Attachment 6. The review team also indicated that supplemental information may be necessary relative to mechanical details of the main steam and feedwater piping. This information will be requested at a later time when our needs are fully assessed.

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MEETING SUMMARY DISTRIBUTION:

Docket 50-219
NRC PDR
Local PDR
ORB#2 Reading
NRR Reading
H. R. Denton
E. G. Case
D. Eisenhut
R. Vollmer
B. Grimes
W. Gammill
J. Miller
L. Shao
T. Carter
D. Crutchfield
D. Ziemann
V. Noonan
Seismic Review Group
A. Schwencer
T. Ippolito
R. Reid
G. Lainas
P. Check
R. Clark
F. Pagano
G. Knighton
T. Wambach
H. Smith
OELD
OI&E(3)
R. Fraley, ACRS(16)
T. M. Cheng
H. W. Lee
H. Levin
K. N. Jabbour
W. J. Hall (Consultant)
J. D. Stevenson (Consultant)
J. R. Buchanan
TERA

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JCP&L promised to review these items and provide us with a schedule for submitting the information. They stated that the schedule would depend upon the availability of the information.

Thomas V. Wambach

Thomas V. Wambach, Systematic
Evaluation Program Manager
Division of Operating Reactors

Attachments:
As stated

cc w/attachments:
See next page

cc
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Ocean County Library
Brick Township Branch
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Brick Town, New Jersey 08723

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

Attachment 1

LIST OF ATTENDEES

JULY 17 AND 18, 1979

NRC

T. V. Wambach
T. M. Cheng
H. W. Lee
H. Levin
K. N. Jabbour
W. J. Hall, Consultant
J. D. Stevenson, Consultant

K M C, Inc.

R. E. Schaffstall

JCP&L

J. Knubel
E. Riggle
T. E. Tipton
I. R. Finfrock*
D. N. Grace*
A. F. Hertz
P. F. Wells*
G. W. Busch*
M. W. Laggart
E. G. Roome
Y. Nagai

URS/Blume

L. E. Malik
R. P. Gallagher

EG&G

S. M. Ma

LLL

R. C. Murray

MPR Assoc.

W. R. Schmidt
J. E. Nestell

GE

A. B. Fife
A. L. Armitage
R. L. Boone
G. C. Nelson*

Burns & Roe

J. C. Archer
S. Chou
J. R. Clapp
J. Saranga
A. S. Dam
R. S. Gagliardo

*Attended July 17, 1979 meeting only

July 16, 1979

AGENDA AND OUTLINE OF PRESENTATION FOR SSRT REVIEW OF OYSTER CREEK

July 17, 1979

8:30 a.m. I. Introduction at Energy Spectrum - JCP&L

8:45 a.m. II. Plant Orientation & Description - JCP&L

9:15 a.m. III. Scope of SSRT Review - JCP&L

A. List of systems required for

(1) RC system integrity and
(2) Safe shutdown

B. Class 1 structures

9:45 a.m. IV. Health Physics Orientation - JCP&L

10:15 a.m. V. Summary of Seismic Design Basis - MPR and Criteria

A. Design Earthquakes (OBE, SSE) and Bases

B. Seismic Design Criteria

(1) Analysis methods
(2) Damping
(3) Allowables

10:45 a.m. VI. Summary of Seismic Design Analysis Methods Used

A. Class 1 Structures

(1) Dynamic analyses - J. A. Blume (10-15 minutes)
(2) Structural design - B&R (10-15 minutes)

B. Piping Systems - MPR (10-15 minutes)

C. Drywell & Torus - MPR (5-10 minutes)

D. Components & Equipment - MPR/GE/B&R (10-15 minutes)

Attachment 2

July 16, 1979

SCOPE OF SEISMIC REVIEW
OYSTER CREEK NUCLEAR GENERATING STATION

The following systems, components and structures are considered to fall within the scope of the seismic re-evaluation of Oyster Creek:

A. Reactor Coolant Pressure Boundary

1. Reactor vessel and support
2. Reactor internals, including fuel
3. Recirculation system, including bypass lines
4. CRD housings
5. CRD piping
6. Connected piping inside drywell, including non-isolable portions of:
   - Main steam
   - Feedwater
   - Emergency condenser
   - Core spray
   - Reactor cleanup
   - Shutdown cooling
   - Liquid poison
   - RV head cooling
   - CRD hydraulic return

B. Safe Shutdown Systems

1. Emergency condenser system, including isolation condenser, piping and valves.

2. Core spray system, including piping, pumps and valves to torus.

3. Auto depressurization system.

4. Condensate transfer system (portion that supplies condensate to isolation condensers) including piping, pumps, valves and condensate storage tank.

5. Emergency service water system, including pumps, piping, valves and heat exchangers.

6. Containment spray system, including pumps, piping and valves.

7. Emergency power system, including:

    - Diesel generators
    - Station batteries
    - Distribution lines
    - Switchgear
    - Control boards
    - Motor control centers
    - Diesel auxiliaries (fuel, starting batteries)

8. Reactor control and protection system.

9. Safe shutdown system instrumentation and control.

C. <u>Class I Structures</u>

1. Reactor building
2. Drywell, torus and vents and directly connected piping (core spray suction header, vacuum breaker piping, etc.)
3. Turbine building (control room & portions supporting Class I equipment).
4. Spent fuel storage facilities
5. Ventilation stack
6. Intake structure (Designed as Class II)
7. Diesel generator building

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III. RESPONSE TO NRC QUESTIONS

12. NRC Information Request

L. Provide a summary of changes to the facility as described in the FSAR which affect seismic category I structures, systems, and components.

Response

Listed below are the changes to the facility in chronological order which affect seismic category I structures, systems, and components.

1969 Modifications

(1) Diesel Generator Engine Replacement

During a core spray surveillance test, No. 1 diesel generator started as required but tripped off. Immediate visual examination disclosed that the engine could not be repaired at the site. Arrangements were made to replace this engine with a new engine. The new engine was installed, tested, and returned to service. The cause of the failure was a cracked cylinder head due to insufficient water flow through it.

(2) Reactor Recirculation Pump Repair

During plant heat-up "D" recirculation pump tripped and failed to restart. After an inspection of the pump seal and cooling water impeller, it was decided to remove the entire pump seal from its casing. Further examination revealed failure of the shaft axial bearing and severe galling in the area of the auxiliary impeller and the thermal barrier. The pump was returned to the manufacturer for detailed examination and repair. Foreign material was found in the pump seal area in the form of chips containing a high percentage of Inconel 600. As the pump design did not call for any of this material, the source of the material was unknown. The repaired pump was returned to the site, reinstalled and was ready for service.

On numerous occasions it was impossible to restart the recirculation pumps while the reactor was at full pressure. In order to restart the pumps, it was necessary to reduce reactor pressure. To correct this situation the generator field forcing voltage was increased in an attempt to overcome the starting torque of the idle pump. In addition, the fluid coupling oil temperatures were increased with some degree of success. The final correction was the addition of a second field forcing transformer in parallel combined with increased fluid coupling oil temperature.

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1970 Modifications

(1) Indexing Mechanism Modification

Modifications were made to the indexing tube mechanism as a result of receiving a field and deposition instruction directive from the General Electric Company. This modification involved the use of new gasket material and contact cemend in conjunction with a new check valve and relief valve for the indexing mechanism to insure its leak tightness.

(2) Travelling Incore Probe Shielding Room

Additional shielding material consisting of lead and concrete was added to the TIP Room for the purpose of minimizing variations in background radiation levels in the counting room. The safety evaluation associated with this change required that the shielding structure be designed to accommodate Class I seismic standards. No other safety significance was associated with this change.

1971 Modifications

(1) Core Spray System Fill Pump

A fill pump and associated valves and piping were installed on core spray system II The purpose of this modification is to keep the discharge piping of "B" and "D" core spray and core spray booster pumps filled with water so as to eliminate a water hammer condition which is set up when the water in that piping drains back to the suppression chamber through the check valves.

1972 Modifications

(1) Fuel Oil Transfer Line

In order to overcome operational difficulties associated with the biweekly surveillance of the emergency diesel generators, a fuel oil transfer line was installed between the package boiler oil tanks and the diesel generator tank.

The system is composed of a 25 gpm gear type oil pump which takes a suction on the 75,000 gallon package boiler oil tank and pumps fuel oil via an underground pipe line to the normal fill line of the diesel generator fuel oil tank. The fuel oil transfer line is connected to the fill line through a flexible connection.

(2) Torus Baffle Removal

In response to reports of damage to torus components incurred at Northern States Power Company's Monticello Nuclear Power Station and Niagra Mohawk's Nine Mile Point - Unit I, a plan was formulated for a detailed inspection of torus componenets at Oyster Creek Unit No. 1. This inspection revealed that five (5) baffles in the vicinity of the discharges of the auto-relief valves were displaced and had fallen to the bottom of the torus. Upon further investigation, it was discovered that numerous baffle support bolts were either broken or missing at scattered locations throughout the torus.

To preclude the change of further baffle displacement due to auto-relief actuation, the three (3) sets of baffles in line with each auto-relief discharge nozzle were removed. These include the sets at torus azimuths 108°, 122°, 130°, 230°, and 252°.

(3) Electromatic Relief Valve Discharge Piping

During the 1972 Spring Refueling Outage, a system of snubbers and springs was installed on the electromatic relief valve discharge piping. The snubber system was installed after a stress analysis of the discharge piping revealed the existence of unacceptable reaction loads during the valve opening transient. The snubbers were installed to restrain the reaction forces while allowing unrestricted thermal expansion and contraction of the plant.

(4) Electromatic Relief Valve Discharge Pipe Canal Fitting Support Modification

As delineated in Mr. I. R. Finfrock's August 22, 1972 letter to Mr. D. Skovholt on the Oyster Creek Station Safety and Relief Valve Piping Design Report (Docket No. 50-219), the electromatic relief valve discharge pipe canal fitting restraints were temporarily modified to help restrain the reaction and steady state forces experienced at the canal fitting during electromatic relief valve blowdown.

1973 Modifications

(1) Control Rod Drive Scram Air Supply System

A modification was made to the CRD scram air supply system in order to provide redundancy, increased reliability, and improved maintainability. The air supply system provides for filtering and reducing in pressure plant instrument air which is used to maintain the scram valves in the closed position during normal operation. The components were installed such that the redundant components could be cross-connected to the primary components, thereby allowing selected components to be taken out of service for maintenance.

(2) Baffle Removal

During the routine torus baffle inspection, the middle and lower baffles at Azimuth 324° were found dislodged from their outside support brackets and lying on the torus bottom.

General Electric conducted a study which concluded that there are no adverse consequences in removing any or all the baffles in the torus. In addition to this, Jersey Central Power & Light Company contacted General Electric specifically for the baffles at 324° and they reaffirmed their report of May 5, 1972. The middle baffle at Azimuth 324° was removed.

(3) Electromatic Relief Valve Discharge Piping Restraints

Jersey Central Power & Light Company conducted a design review of the electromatic relief valve inlet and discharge piping in both the drywell and torus. Based on this review, there have been three successive modifications. At this time, the transient and steady state stress conditions have been analyzed (and reduced where necessary by means of modification) at each point in the electromatic relief valve system. The result of designed modifications has been to reduce the stress at all points, under all conditions, to levels within the allowable stresses of USAS B31.1 (the original code used for the electromatic relief valve system design).

The major modification accomplished during the Spring 1973 shutdown was to remove the temporary cables on the torus canal fittings and install permanent restraints to reduce both vertical and horizontal stress to within code limits. The temporary cable installation was completed during August 1972. In addition to the torus installation, an additional restraint was installed on the south header at the 14-inch to 12-inch reduced to help reduce the stress at the 8-inch to 14-inch lateral. Finally, a spring hanger on the 8-inch line of the south header was replaced with a stronger one to reduce stress on the same lateral mentioned above. The remaining hangers and restraints were installed during May 1972.

(4) Refueling Bridge Modification

In order to improve the reliability of the refueling platform, two modifications were implemented. The first modification provides for relocating the fuel grapple load cell from the cable to a modified sheave support structure. The modification provides for measuring the load on the sheave rather than the load on the cable. This has eliminated the reverse bending of the cable due to passing through the load cell roller which has been identified as a cause of cable fraying.

The modification consisted of rebuilding the sheave support structure such that it is pivoted on one end by two pillow blocks and supported on the other end by the load cell. The second modification provided for modifying the refueling platform to accommodate a newly designed fuel grapple. The changes required provided for moving the hole, through which the grapple cable passes, forward three inches and moving the sheave support structure forward by the same amount.

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1974 Modifications

(1) Main Steam Isolation Valve (MSIV) Control System

In order to improve the reliability of the Main Steam Isolation Valve Control System, the Numatics Company spool valves were replaced with Automatic Valve Company manifold assembled poppet-valves. The spool valves consisted of a selectively fit (about 0.0005" clearance) noncontacting metal which was susceptible to binding due to particles carried by the pressurizing media. The Control Schematic is the same as the original design with one exception. A redundant exhaust port for fast closure was added to allow for fast closure of the MSIV's when required during slow closure exercising of the MSIV's.

(2) Baffle Removal

During an inspection of the torus performed during the April refueling outage, it was observed that the lower baffle at azimuth 014 was bent slightly and that its holdown bolts were missing. The bolt holes could not be properly aligned due to the slight bend whereupon it was decided to remove the baffle after an unsuccessful straightening attempt. As reported in a letter by Mr. I. R. Finfrock, Jr., to Mr. A. Giambusso dated June 2, 1972, General Electric performed an analysis which concluded that any or all of the baffles could be removed with no adverse safety consequences.

(3) Torus Vent Valve Modification

As a result of Local Leak rate testing during the Spring 1974 refueling outage, the 12" Torus Vent Valves, V-28-17 and 18, were found to leak at a rate of approximately 400 scfh at the 35 psig test pressure. This leakage rate was well in excess of the Technical Specification limit.

As a result, replacement valves of different manufacture were purchased, qualified and installed. Leak rate testing after installation revealed zero (0) leakage.

(4) Vacuum Breaker Modifications

A modification was made to the torus to drywell vacuum breaker check valve lever arms to increase the valve closing moment. The modification consisted of rotating the lever arms 31.5° (6° beyond the pivot point) and adding a 7.1 lb. weight on each of the two valve lever arms at a moment arm of 15.75 inches. The modification adds a high degree of assurance that the valves will be fully closed thus minmizing the possibility for excessive leakage from the drywell to torus under design basis loss of coolant accident conditions. In addition, five of the valves were modified by replacing the aluminum disc pins with ones fabricated from stainless steel. The change was made necessary by the discovery of two broken disc pins. A subsequent dye penetrant examination revealed three other defective disc pins. The three affected valve discs were bored and the replacement stainless steel pins inserted. This acted to increase the weight of each disc and pin assembly by. approximately 12.5 lbs. which consequently increased the valve closing moment. The valve 0.5 psi differential pressure opening requirement is met for these valves even with the lever arm modification.

(5) Torus to Drywell Vacuum Breakers Alarm System

An alarm system was installed on each of the fourteen (14) torus to drywell vacuum breakers to alert the control room operator if any vacuum breaker is opened more than a specified allowable limit.

(6) Modification to Steam Flow and Steam Pressure Transmitter Sensing Lines

This modification was incorporated into the steam flow and steam pressure sensing lines to assist performance of preventive and corrective maintenance.

Tee's were installed between the instrument block valves and the steam flow and steam pressure transmitters to allow instrument line snubber replacement without disturbing the transmitter or the block valves. In addition, one end of the tee is capped to provide a means of flushing sensing lines during maintenance and also to provide a limited reservoir for any undesirable deposits which may clog the instrument snubbers.

Additional valves were installed in the steam flow transmitter instrument lines upstream of the existing block valves. These valves provide finer control during valving operations, therefore minimizing the change of upsetting the steam break sensors.

(7) Reactor Vessel Instrument Tube Anti-Ejection Device

A device was manufactured and installed which will prevent ejection of instrument tube 28-05 should the tube-to-vessel weld fail and the tube become "free" of the vessel. The device is installed in such a manner that it is supported cross ways on the upper edges of two adjoining members of the CRD supports. When in place, the top surface of the impact plate will be approximately one (1) inch below the lower surface of the instrument flange with the instrument extension within the device. Clearance holes are provided for this accommodation. The position of the impact plate will easily accommodate any thermal growth of the instrument tube.

In the event of tube ejection, the device will restrict the movement of the instrument tube flange to approximately 1/2 inch and thus maintain the tube within the confines of the reactor vessel penetration annulus. All forces will be transmitted in a downward direction and absorbed by the CRD restricting mechanism. It is assumed that this mechanism is capable of sustaining this force since it is designed to withstand the impact of an ejected CRD estimated to be approximately nine (9) times as great as that of the instrument tube ejection forces.

1975 Modifications

(1) Fuel Channel Modification

During fuel channeling operations prior to the Spring 1975 Refueling Outage,it was determined that the fuel channel design was incompatible with the new Exxon Nuclear 8 x 8 Fuel Design in that the rivet head of the channel spacer pads interfered with two(2) of the fuel assembly tie rod adjusting nuts. To correct the interference problem and to provide adequate spacing between the components, modification to the fuel channel rivet head was proposed. The modification consisted of machining off the top of the rivet head so that it protruded from the fuel channel only 0.155± .010 inches. All future channel purchase specifications will include this dimension.

(2) Reactor Drain Line Modification

The purpose of this modification was to relocate the reactor bottom drain line (ND-1) closer to the reactor pedestal wall allowing for the installation of a new control rod drive handling system. As originally installed, the drain line fell inside the circular beam and track which supports the platform assembly and therefore would interfere with the rotation of the unit. Also removed, as part of this modification, were four(4) valves which branched off the drain line. These valves were no longer in use and had become crud traps and consequently, sources of high radiation.

(3) Inner Filter Conversion

A modification was made to all CRD's as they are rebuilt during refueling outage. During the past refueling outage, thirty-three (33) CRD's were modified. The modification consists of moving the inner filter from the top of the index tube to the stop piston in order to eliminate a potential slowing of the rod insertion time due to plugging of the inner filter.

(4) Spent Fuel Cask Drop Protection System

An analysis of the postulated dropping of a spent fuel shipping cask indicated that the resulting damage to the spent fuel pool could threaten the capability of the water make-up systems to keep any spent fuel in the pool submerged.

As a result of investigation into the alternative solutions to the problem, a Cask Drop Protection System was developed and installed. The CDPS consists of the following.

1. A guide structure which properly guides and restrains the falling cask in the event it is dropped into the spent fuel pool.
2. A hydraulic dashpot in the lower section of the guide structure which retards the falling cask such that impact loads are kept

well within acceptable values.

(5) In-Core Instrument Guide Tube Plugs

The objective of this modification is to reduce personnel exposure under the reactor vessel by decreasing or eliminating the exposure contribution from non-instrumented guide tubes. Following decontamination of the spare guide tube, an in-core plug is installed in each of these guide tubes. During the 1975 refueling outage these plugs were installed in four non-instrumented guide tubes.

1977 Modifications

(1) Torus to Drywell Differential Pressure Instrumentation

The purpose of the modification is to satisfy an NRC staff technical position which requires two independent systems with a specified accuracy for monitoring drywell to torus differential pressure.

The portion of the modificaton consists of the installation of valving and taps to existing drywell and torus pressure sensing lines. A pressure gage was installed at the drywell pressure tap and another at the torus pressure tap. By using these new gages and the existing drywell and torus pressure recorders in the Control Room, two independent systems that meet the NRC staff technical position are available.

(2) Feedwater Nozzle Cladding Removal and Sparger Replacement

The modification was a result of cracking which was found in the feedwater nozzles of several operating BWR's. Initiation of the cracks has been attributed to high frequency thermal cycling in the annulus between the feedwater sparger thermal sleeve and the nozzle. It is thought that this thermal cycling is caused by bypass flow of cold feedwater into the nozzle/thermal sleeve annulus mixing with hot reactor water and also hot reactor water sweeping into the relatively cold annulus area. In order to prevent this mixing phenomena from occurring, a sparger of a new design was developed and installed.

The replacement feedwater sparger assembly consists of a carbon steel 5" x 12" rectangular cross-section sparger and a nominally 10" diameter Inconel thermal sleeve. The thermal sleeve directs feedwater from the reactor vessel feedwater nozzle to the arms of the sparger where it is distributed to the reactor vessel by 52 uniformly spaced 1" diameter holes in the sparger.

The sparger design also includes a flow baffle arrangement consisting of two Inconel X-750 flow baffles which cover the annular opening between the sparger thermal sleeve and the reactor vessel feedwater nozzles and a piston ring seal assembly on the thermal sleeve.

The purpose of the flow baffles is to prevent mixing of hot recirculation water with the significantly cooler feedwater which leaks past the thermal sleeve into the annular space between the sleeve and the feedwater nozzle. The purpose of the piston ring seal assembly is to minimize bypass leakage of feedwater into the annular space between the sleeve and the nozzle.

Complementing the new sparger design was the removal of stainless steel cladding from the inside diameter and blend radius regions of the feedwater nozzles. This removal was considered an integral part of an overall solution to the feedwater nozzle cracking problem. Removal of the cladding, among other things, reduce thermal stresses in the nozzle and eliminate the possibility of cladding cracks propagating in the base metal. The cladding was removed utilizing a special, single point boring machine which was developed for this purpose.

(3) Backup Scram Solenoid Valve Air Piping Modification

This modification was required to eliminate a pressure decrease in the scram valve pilot air header during periods of a half scram on R_x protection system #2. The decrease in pressure was caused by the combination of excessive leakage of fittings and scram solenoid valve seals along with the fact that at high demand rates, the NC16B valve restricts flow.

The piping modification installed a section of piping parallel to solenoid valve NC16B. The new section of piping contains two gate valves and two check valves, all in series. The gate valves isolate the new section when it is not needed. The check valves allow increased air flow to the scram valve pilot air header from NC16A when NC16B is de-energized but prevent venting the header through the new section when NC16A is de-energized. Two check valves were installed for redundancy.

(4) Spent Fuel Pool Cooling System Augmentation

A modification consisting of one (1) heat exchanger and two(2) parallel pumps was added to the existing SFPCS to increase its cooling capacity to handle the heat generated by a full core unload through the entire expected operating range of cooling water temperatures.

(5) Torus Level Instrumentation

This modification was the result of an NRC staff technical position which required a redundant narrow range torus water level instrument channel. The staff technical position is based upon operational experience which indicates that pressure transducers tend to drift and water level indicators tend to stick.

The modification consisted of installing an additional differential pressure transmitter, a level indicator and the associated piping and electrical wiring to provide a second means of monitoring the torus water level.

(6) Torus Water Storage Tank and Torus Water Removal Piping

Modifications were necessary to facilitate the removal of chromated water from the torus to allow completion of required internal inspections and modifications. The tank, with a storage capacity of 750,000 gallons, is large enough to store the entire volume of torus water (613,000 gallons, minimum) plus water used for flushing purposes.

1978 Modifications

(1) Containment Spray Loop Seal

While replacing the containment spray heat exchangers during the 1978 refueling outage, the emergency service water discharge piping on System 1 was rerouted to produce a loop seal ,thereby preventing the heat exchangers from draining while the system is in standby. The change was made to reduce water hammer previously experienced when starting the system for test.

In addition, 150 psig and 250 psig relief valves were installed on the containment spray and emergency service water sides of the new heat exchanger respectively, replacing the lower rated valves installed to protect the degraded heat exchangers.

(2) 125-V DC Separation

In order to provide increased physical separation of 125-V DC power supplies, the 125-V DC system was extensively modified during the 1978 refueling outage. The modification consists of an additional 1200 amp-hour 125-V battery, two (2) chargers, a new distribution center (DC-C), a new distribution panel (DC-F) and a new motor control center (DC-2). A majority of the equipment was installed in the 4160-V switchgear room, a considerable distance from existing A/B batteries. Increased separation was ensured by running the "C" battery loads in new conduit.

(3) A/B Battery Room Fire Protection

An automatic fire suppression system was installed to protect the "A" and "B" 125-V DC station batteries, battery room switchgear and M-G sets, the electric tray room and the tunnel connecting the battery room with the electric tray room. The system uses Halon 1301 total flooding and is actuated by products of combustion sensors. Automatic damper isolation was included. The ventilation system to the battery room was modified to detect loss of air flow.

(4) Torus to Drywell Differential Pressure Instrumentation

Two (2) digital (LED) differential pressure instruments were installed on RK03 in the reactor building to indicate torus to drywell differential pressure. The instruments share common pressure taps from both the drywell and torus but have indpendent power supplies fed by redundant emergency buses. These instruments are used to satisfy Technical Specification requirements. Previously, presssure indication on the drywell and on the torus were used to determine the pressure differential.

ITEMS IDENTIFIED BY JCP&L FOR FURTHER EVALUATION

1. Control room panels - evaluation of bracing and anchorage
2. 4160 switch gear panels - evaluation of anchorage
3. RK04 instrument panel supported on grating - evaluation of adequacy of support
4. Motor control centers - evaluation of anchorage
5. Block wall in cable spreading room - evaluation of structural integrity (equipment is mounted on wall)
6. Distribution panel in battery room - evaluation of anchorage
7. 4160 isolation phase ductwork - evaluation to verify that ducts will not fall on vital equipment
8. Office building - evaluation of portion that is founded on reactor building (vital cables pass through this building)
9. Intake structure - evaluation to document that it can meet seismic Category I requirements
10. "B" battery rack temporary modification (complete)

ADDITIONAL INFORMATION REQUIRED TO DOCUMENT THE SEISMIC RESISTANCE CAPABILITY OF THE OYSTER CREEK FACILITY

The NRC staff and its consultants have completed a review of the Oyster Creek docket for information pertinent to the seismic design bases of the facility and its capability to withstand the effect of potential earthquakes. The information currently on the docket is not sufficiently complete to adequately quantify such capability. It is anticipated that pertinent information is available from other sources such as the original NSSS, A-E, or plant files.

The following information is necessary to proceed with the seismic review; however, at this time it is not required that new information be generated to satisfy this request.

A. Seismic Analysis and Design Criteria for Structures and Input to Equipment

1. Shear areas of original reactor building and turbine building models

2. Structural drawings (including foundation details) in sufficient detail to permit verification analyses of the following buildings:

   - reactor building
   - turbine building
   - office building (portion founded on reactor building)
   - ventilation stack

3. Reinforcement details of block wall in containment spray corner room (near stairway)

4. Confirmation of soils design information; shear wave velocity or shear moduli, poissons ratio, bearing capacity

B. Seismic Qualification of Mechanical and Electrical Equipments and Fluid and Electrical Distribution Systems

Seismic qualification documentation based upon analysis or testing of actual or similar equipment to that installed at the Oyster Creek facility is required.

MECHANICAL COMPONENTS

1. Emergency condenser - original Foster Wheeler computations including arrangement drawings showing support and anchorage details

a. Isolation condenser piping - from containment penetration to the condenser

b. 2" control rod drive return piping (50')

ELECTRICAL COMPONENTS

13. RK01 - 5 instrument racks - Blume analysis

14. 4160-480 transformer - arrangement drawings including weights, center of gravity and tie down details or qualification test data

15. Instrumentation test reports - GE

16. Battery racks - computations supporting evaluation of new racks and qualification testing data for jars when available

ELECTRICAL DISTRIBUTION SYSTEM

17. Cable trays including lateral supports - original supporting criteria and/or later qualification testing data

SPECIAL COMPONENTS

18. Crane over service water pumps - provisions for assuring that crane will not overturn and damage the emergency service water pumps and/or supporting evaluations

UNDERGROUND PIPING

19. Emergency Service Water - Any additional existing information that can be provided to address the state of stress at entry points into the buildings or other discontinuity points.

2. Reactor pressure vessel stress report - Combustion Engineering

3. Reactor vessel supports - Burns and Roe computations (1965)

4. Stabilizer - analytical techniques used to evaluate the stabilizer and overall RPV support system

5. Heat exchangers - containment spray and shutdown (arrangement drawings including support and anchorage details or qualification data)

6. Pumps - emergency service water and recirculation (arrangement drawings including support and anchorage details and/or qualification test data)

7. Tanks - condensate storage and 7 day fuel oil (arrangement drawings including support and anchorage details and/or original seismic design computations)

8. Diesel generator support - any original computations that evaluate the sliding stability of the diesel generator unit and the effect of relative displacement of attached lines or information that would permit such an evaluation

9. Motor operated valves - effect of eccentric mass on 1" to 4" diameter lines

10. Control rod drive hydraulic control units

    a. computations addressing the lateral support of tubing

    b. anchorage criteria for the CRD modules including supporting computations and field verification that the modules are supported in accordance with the design criteria

11. M-G set in cable spreading room - arrangement drawings including support and anchorage details, weights and/or computations, qualification data

FLUID DISTRIBUTION SYSTEMS

12. Piping isometric drawings including support locations, support stiffnesses (or necessary drawings of supports to enable a determination of support stiffnesses), valve locations, valve masses and center of gravity and other pertinent details needed to independently verify and audit the seismic capability of the following piping systems