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December 13, 2006

Attached is a collection of documents for the ACRS License Renewal Subcommittee meeting on January 18, 2007. The documents are associated with the Oyster Creek drywell.

OPTIONAL FORM NO. 10
MAY 1962 EDITION
GSA FPMR (41 CFR) 101-11.6

UNITED STATES GOVERNMENT

Memorandum

O'Reilly
Mosley
Carlson
Sears + Nolan

TO : R. S. Boyd, Chief, Research & Power Reactor Safety Branch, Division of Reactor Licensing, HQ

DATE: April 15, 1966

FROM : J. P. O'Reilly, Senior Reactor Inspector
Region I, Division of Compliance

J. P. O'Reilly

SUBJECT: JERSEY CENTRAL POWER & LIGHT COMPANY
DOCKET NO. 50-219

The attached report by our field inspector of a visit to the subject facility on March 22 and 23, 1966, is forwarded for information.

The construction activities at the site are estimated to be 38% complete, based on money expended. The present status, according to GE personnel, indicates that they are 2 to 3 months behind schedule. The contributing causes, jurisdictional labor disputes, a steel shortage and an alignment problem with a vent header, are outlined in the attached report.

Attachment:
CO Rpt. No. 219/66-2
by R. T. Carlson
dtd 4/15/66

cc: L. Kornblith, Jr., CO:HQ
E. G. Case, DRL
R. G. Page, SLR
CO:HQ File
M. L. Ernst, CO:II
H. D. Thornburg, CO:III
J. W. Flora, CO:IV
R. G. Engleken, CO:V

A/S

U. S. ATOMIC ENERGY COMMISSION
REGION I
DIVISION OF COMPLIANCE

April 15, 1966

CO REPORT NO. 219/66-1

Title: JERSEY CENTRAL POWER & LIGHT COMPANY
LICENSE NO. CPPR-15
Dates of Visit: March 22 and 23, 1966

By : R. T. Carlson, Reactor Inspector
J. P. O'Reilly

SUMMARY

The status of construction activities is discussed in the report. Overall construction is estimated to be 38% complete, based on money expended.

The installation, overload and initial leak rate tests of the dry well and torroidal chamber were completed satisfactorily.

A problem with an expansion joint located in one of the vent headers that joins the dry well and torroidal chamber, that resulted in both the replacement of the joint and a repetition of the overload test on the dry well, is discussed in the report.

Adequate quality control measures appear to be in effect for reinforced concrete.

A 400' meteorological tower has been installed and data are being accumulated.

A fatality, the first at this site, resulted from injuries received by a construction worker in a fall.

(continued)

DETAILS

I. Scope of Visit

Mr. R. T. Carlson, Reactor Inspector, Region I, Division of Compliance, visited the construction site of the Jersey Central Power & Light Company's reactor facility at Oyster Creek, New Jersey, on March 22 and 23, 1966. The visit included the following:

A. A review of the construction organization.

B. A review of the status of the containment system.

C. A review of the quality control measures in effect for reinforced concrete.

D. A review of the status of construction and the timetable of significant events.

E. A tour of the construction site.

The principal persons contacted were as follows:

Jersey Central Power & Light Company (Jersey Central)

Mr. Ivan Finfrock, Nuclear Project Engineer
Mr. Norman M. Nelson, Plant Maintenance Supervisor, Designee

General Electric Company (GE)

Mr. Willard C. Royce, Resident Manager
Mr. Abel B. Dunning, Construction Engineer, Mechanical
Mr. Glen C. Brockmeir, Construction Engineer, Civil

(continued)

II. Results of Visit

A. Organization

1. Jersey Central

Jersey Central currently has two people at the site on a full-time basis - Mr. Nelson, the designated Plant Maintenance Supervisor, and Mr. Fred Kossatz, the designated Plant Mechanical Maintenance Foreman under Mr. Nelson. Both are present for on-the-job training relating to plant construction and operation.

Mr. Finfrock, the Nuclear Project Engineer, operates out of the Company Office in Morristown, New Jersey, and spends much of his time at the site, 3 to 4 days per week. His principal concern at this time relates to site meteorology.

Both Messrs. Nelson and Finfrock report to Mr. Donald Rees, the Project Engineer, who is located in the Company Office in Morristown.

2. General Electric

GE, the prime contractor for the Oyster Creek Project, currently has six people at the site. These personnel are: Mr. Royce; Messrs. Dunning and Brockmeir - the men most actively engaged in following day-to-day construction; Mr. Stibers, Office Engineer; Mr. Ryan, Site Auditor; and a clerical worker. According to Mr. Royce, the staff will be increased to eight in the near future.

Mr. Royce reports to Mr. R. A. Huggins, Project Engineer, Atomic Power Equipment Department (APED), San Jose, California.

(continued)

Results of Visit (continued)

3. Burns and Roe, Inc. (B&R)

B&R is the Architect-Engineer and the direct Supervisor of Construction for this project. The senior site representative for B&R is Mr. Giles Willis, who reports to Mr. David Kregg, the Project Manager. The principal channel of communication between GE and B&R is through Messrs. Huggins and Kregg.

4. Other Principal Contractors

Other principal contractors associated with this project, and their responsibilities, are listed below:

Contractor	Responsibility
American Bridge	Structural steel on Turbine Building, and on bridge crane
American Dewatering Corp.	Site dewatering
Chicago Bridge & Iron Co.	Containment system
Eastern Transit Mix Co.	Concrete
Hatzel & Buehler, Inc.	Miscellaneous electrical work
McBride Plumbing Co.	Miscellaneous piping
Poirier & McLane Corp.	Superstructure

(continued)

Results of Visit (continued)

Contractor	Responsibility
United Roofing & Waterproofing	Concrete waterproofing
U. S. Testing Laboratory	Construction related testing
White Construction Co.	Reactor Building
Worthington Corp.	Turbine condensers

B. Construction Status

Overall construction was estimated by Mr. Dunning to be 38% complete, based on expenditures, as of March 1, 1966. A picture reflecting the construction status as of early February is shown in Figure 1 of this report. The reported status of the major subdivisions of the facility, as of March 1, 1966, is provided below:

Subdivision	Percent Complete
Containment system	100%
Reactor Building, structural portion	35%
Turbine Building, structural portion	80%
Intake and discharge structures, structural portions	98%
Intake and discharge canals, excavation	5%
Waste Disposal Building, excavation	90%

(continued)

Results of Visit (continued)

Construction activities at the site are estimated by GE to be 2 to 3 months behind schedule. The principal delay being the result of labor jurisdictional disputes. Mr. Royce told the inspector that this was not a current cause for delay; however, it was still a sensitive subject area and could result in further delays in the future.

C. Containment System

The installation, overload and initial leak rate tests of the containment system, the dry well and torroidal pressure suppression chamber, by CB&I have been completed. Significant aspects of these operations were reviewed by the inspector and are discussed in the following paragraphs:

1. General

The installation and testing of the system was completed several months behind schedule. Mr. Dunning told the inspector that a major contributing factor, in addition to the problem of labor jurisdictional disputes, was the upset in material delivery schedules caused by the then impending strike in the steel industry. Late deliveries of large quantities of material necessitated the hiring of additional welders, a shortage of which resulted in the acceptance of some welders that would not have been hired otherwise. As a result, the percentage of welds requiring repair increased from 0.5% to 50 - 75%. When asked by the inspector what assurance he had that all faulty welds were repaired, Mr. Dunning stated that this assurance was provided by the fact that all welds on the containment system were 100% X-rayed, and that the results were reviewed by qualified representatives of the following organizations: CB&I, B&R, The Hartford Steel Boiler Inspection and Insurance Company, and GE.

(continued)

Results of Visit (continued)

2. Expansion Joint Problem

The expansion joint in one of the ten vent lines that join the dry well to the torroidal chamber, the fourth going clockwise from the personnel airlock, was found to be distorted when a temporary protective cover was removed from the joint during the initial phase of post-installation testing*, i.e., a low pressure soap bubble test immediately preceding the pneumatic overload test on the dry well. The faulty joint was subsequently replaced.

According to Mr. Dunning, the distortion in the joint, the last to be installed, was the result of torsional and radial stresses imposed during installation when compensating for misalignment between the vent line and the torroidal chamber. He said that the distortion was inadvertently overlooked by construction supervision at the time of installation and that its discovery was delayed because of the presence of the protective cover. Mr. Dunning told the inspector that the original misalignment problem was corrected by proper mitering during replacement of the joint. He said that the remaining joints were subsequently inspected and found to be satisfactory.

The decision to replace the joint was made subsequent to the completion of the pneumatic overload and leak rate tests on both the dry well and the torroidal chamber. Post-replacement pressure testing included a repeat of the pneumatic overload test on the dry well, and the performance of hydropneumatic overload and leak rate tests on the torroidal chamber as originally planned.

(continued)

*Containment testing, including results, discussed further in paragraph II.C.3.

Results of Visit (continued)

Mr. Dunning told the inspector that a report of the expansion joint problem was being prepared by him and would be submitted to Jersey Central.

The inspector's review of the expansion joint problem indicated that the corrective measures taken were adequate and in accordance with good engineering practice.

3. Overload and Leak Rate Test Program

The inspector discussed with Mr. Dunning the scope and results of the overload and leak rate test programs. The sequence of significant tests conducted, as told to the inspector, was as follows:

a. Pneumatic overload test of dry well and vent system at 71.3 psig, 1.15 times the design pressure of 62 psig*.

b. Pneumatic leak rate test of dry well and vent system at design pressure.

c. Pneumatic overload test of torroidal chamber at 40.25 psig, 1.15 times the design pressure of 35 psig.

d. Pneumatic leak rate test of torroidal chamber at design pressure.

e. Repeat of the test described in paragraph 3.a. because of the replacement of the faulty expansion joint.

(continued)

*Witnessed performance and results discussed in CO REPORT NO. 219/65-3, paragraph II.A.

<u>Results of Visit</u> (continued)

f. Hydro-pneumatic overload test of torroidal chamber at 40.25 psig. The chamber contained 91,000 cubic feet of water to simulate operating conditions.

g. Hydro-pneumatic leak rate test of torroidal chamber at design pressure, with the same water present as described in paragraph 3.f.

The preliminary results of the leak rate tests were stated by Mr. Dunning to be as follows:

Test	Leak Rate, % Per Day
Dry well and vent system at 62 psig	0.064
Torroidal chamber at 35 psig, dry	0.078
Torroidal chamber at 35 psig, wet	~0.1 (computations incomplete)

According to Mr. Dunning, Jersey Central representatives were present throughout the significant phases of containment testing and will be provided with a report of the test results from CB&I, the group responsible for the performance of the tests, through GE.

D. <u>Reinforced Concrete - Quality Control Program</u>

The inspector reviewed the quality control program for reinforced concrete. Included in the review were the following: An examination, on a selective basis, of pertinent

(continued)

Results of Visit (continued)

records including contracts and specifications, testing programs and results; a visual examination of construction field activities; and discussions with cognizant site personnel. It appears to the inspector, as a result of the review, that adequate measures are in effect to assure that the reinforced concrete will meet the minimum requirements of applicable American Society for Testing and Materials (ASTM) and American Concrete Institute (ACI) codes.

E. Site Meteorology

A 400' meteorological tower has been erected about 1500' southwest of the facility stack. Mr. Finfrock is overseeing this aspect of the Oyster Creek Project. According to Mr. Finfrock, the accumulation of data was started on February 14, 1966, and includes the following:

1. Wind velocity and direction at 75' and 400'.

2. Ambient temperature at 10'.

3. Thermal stability data as reflected by the differences between the temperature at 10' and at 75', 200' and 400'.

4. Rainfall.

Mr. Finfrock said that the tower installation was completed ten months behind schedule because of delays encountered in his dealings with State officals, FAA officials, and the contractor. He said that as a result, the submission to DRL of the desired one year's accumulation of data from the site will be made subsequent to the submission of the Final Safety Analysis Report (FSAR), tentatively scheduled for July 1966.

(continued)

Results of Visit (continued)

F. Miscellaneous

1. Expansion Gap, Dry Well - Biological Shield

The inspector reviewed a letter from Mr. Kregg to Mr. Huggins, dated October 26, 1965, in which a method of attaining the desired expansion gap between the dry well and its surrounding biological shield was discussed. The method discussed proposed the application to the exterior of the dry well, prior to the pouring of the biological shield, of a layer of an inelastic, compressible, asbestos-magnesite cement product. A layer of polyethylene sheeting would then be installed as a bond breaker at the concrete interface, and the concrete pours made. The letter stated that the material would compress about 0.150" during the pouring and curing of the concrete. Subsequently, the dry well would be filled with steam and heated to 280°F. The resultant pressures from the expansion of the dry well would be sufficient to compress the heated cement product an additional amount sufficient enough to attain the desired gap, 3/8".

This subject area will be reviewed further during future inspection visits.

2. Progress Reports

The inspector reviewed monthly progress reports from GE to Jersey Central for the period since September 1965. One item of interest noted, as extracted from the report for January 1966, is as follows:

(continued)

Results of Visit (continued)

"An informal meeting was held with AEC Licensing and Regulatory Staff representatives in Washington to update that group on the details of the metal-water reaction design basis and cooling system design approach to be utilized in the Oyster Creek Station. The information presented was well received by the Staff, and as a result of these discussions, the cooling system design was firmed on the basis of providing 4-loop cooling. Space allocation previously held for the 5th and 6th loops was released for other system requirements."

3. Timetable for Significant Events

The latest timetable for significant events as obtained from site personnel and supplemented by information obtained at an information meeting* between DRL and Jersey Central, and attended by the inspector, is as follows:

Event	Date
Initiation of erection of turbine-generator	4/66
Submission of FSAR	7/66
Submission of technical specifications	9/66
Receipt of reactor pressure vessel at site	9/66

(continued)

*Meeting held at Headquarters on March 24, 1966.

Results of Visit (continued)

Event	Date
Completion of installation of reactor pressure vessel within dry well	11/66
Issuance of Notice of Proposed Issuance of Operating Permit	3/67
Initiation of significant preoperational tests	3-4/67
Initiation of loading	7/67
Attainment of full power and plant turnover	12/67

G. Exit Interview

A formal exit interview was not held because of the nature of the visit. Significant comments by those interviewed during the course of the visit are contained within the body of the report.

Attachment:
Figure 1

JERSEY CENTRAL POWER & LIGHT COMPANY
(CO REPORT NO. 219/66-1)

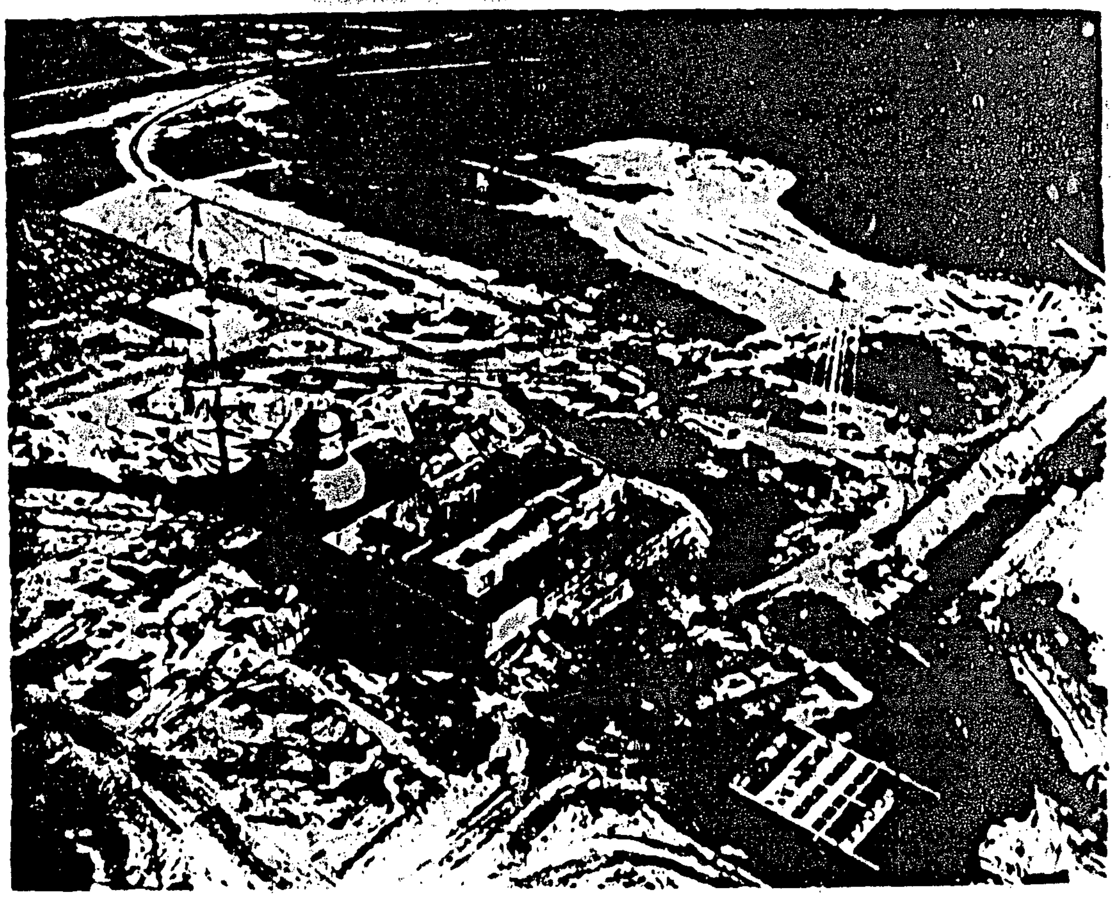

Figure 1

Picture Showing Construction Status
as of February 1966

OPTIONAL FORM NO. 10
MAY 1962 EDITION
GSA FPMR (41 CFR) 101-11.6

UNITED STATES GOVERNMENT

Memorandum

H. O'Reilly
cm Moseley
for Carlson
cy Sears
Nolan

TO : B. H. Grier, Senior Reactor Inspector
Division of Compliance, HQ

DATE: December 6, 1966

J. P. O'Reilly

FROM : J. P. O'Reilly, Senior Reactor Inspector
Region I, Division of Compliance

SUBJECT: JERSEY CENTRAL POWER & LIGHT COMPANY
DOCKET NO. 50-219

The attached report by our field inspector of a visit to the subject facility on November 15, 1966, is forwarded for information.

Attachment:
CO Rpt. No. 219/66-5
by J. R. Sears
dtd 12/6/66

cc: L. Kornblith, Jr., CO:HQ
E. G. Case, DRL (2)
R. S. Boyd, DRL (2)
R. G. Page, SLR
CO:HQ File

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U. S. ATOMIC ENERGY COMMISSION
REGION I
DIVISION OF COMPLIANCE

December 6, 1966

CO REPORT NO. 219/66-5

Title: JERSEY CENTRAL POWER & LIGHT COMPANY
LICENSE NO. CPPR-15
Date of Visit: November 15, 1966

By. : J. P. O Reilly for
J. R. Sears, Reactor Inspector

SUMMARY

The pouring of concrete in the reactor building around the dry well has progressed to the next-to-the-top floor level. The compressible material between the dry well and the concrete shield was observed.

Major mechanical equipment in the turbine building is in place.

The operating staff is now on-site.

DETAILS

I. Scope of Visit

A visit was made to the Jersey Central Power & Light Company reactor, under construction at Oyster Creek, New Jersey, by Mr. John R. Sears, Reactor Inspector, Region I, Division of Compliance, on November 15, 1966. The visit included a tour of the construction site and discussions with the following:

(continued)

Scope of Visit (continued)

Mr. Abe Dunning, Site Representative, General Electric (GE)
Mr. Tom McCluskey, Plant Superintendent, Jersey Central Power & Light Company (Jersey Central)
Mr. Ivan Finfrock, Project Engineer, Jersey Central
Mr. Donald Hettrick, Project Engineer, Jersey Central

II. Results of Visit

A. Tour

The inspector toured the construction site in company with GE and Jersey Central representatives. It was observed that major pieces of equipment had been installed in the turbine building, e.g., the turbine shell, the condenser, some tanks. The installation of some larger sized piping is in progress.

During the tour, a concrete floor slab was being poured for the next-to-the-top floor of the reactor building. The concrete biological shielding around the dry well had been poured to this level. The inspector observed that compressible material, which appeared to be similar to mineral asbestos insulation, had been applied to the sides of the dry well. This was covered by thin polyethylene sheets. Mr. Dunning stated that after all the concrete is placed around this material and has set, the atmosphere in the dry well will be raised to 280°F and 20 psig in order to compress the compressible covering. He stated that GE engineers have calculated that when the dry well atmosphere then returns to ambient conditions, the shrinkage should leave a one half inch gap between the dry well and the concrete. Mr. Dunning described the alternate methods being used at Niagara Mohawk and at Tarapur to allow for dry well expansion at MCA conditions, and said that simple economics of installation costs will determine which method will be used for future facilities.

(continued)

Results of Visit (continued)

B. Interview

Mr. Tom McCluskey, Plant Superintendent, stated that the following Jersey Central people are now resident at the site: Mr. Tom McCluskey, Plant Superintendent, Mr. Richard Doyle, Chemistry Supervisor; Mr. Donald Kaulback, Radiation Protection Engineer; Mr. Norman Nelson, Maintenance Engineer; Mr. Woody Riggle, Electrical Foreman; Mr. Fred Cassady, Mechanical Foreman; and also two chemical technicians, two radiation protection technicians, ten operators and four shift foreman.

He stated that the Technical Engineer and two assistants are at GE, San Jose, California, for training.

Mr. McCluskey said that the operators had been chosen from conventional plant operators who had bid for the job. Successful candidates were selected on the basis of a series of screening tests. They were given a six week course in reactor engineering at Morristown, followed by ten months of practical on-the-job training at Saxton. Each operator on-site has been assigned a system of the plant and is presently reading manufacturer's literature on system components toward the goal of writing operating procedures and cautions. Mr. McCluskey stated that it is not standard practice in Jersey Central's conventional plants to operate via written procedure, but he affirmed that the Oyster Creek plant will be operated via written procedure because of its newness and the operator's lack of familiarity with such a facility.

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

April 28, 1989

Docket No. 50-219

Mr. E. E. Fitzpatrick
Vice President and Director
Oyster Creek Nuclear Generating Station
Post Office Box 388
Forked River, New Jersey 08731

Dear Mr. Fitzpatrick:

SUBJECT: DRYWELL CONTAINMENT - OYSTER CREEK NUCLEAR GENERATING STATION (TAC NO. 72029)

In a letter dated September 12, 1988, GPU Nuclear Corporation (GPUN/the licensee) committed to provide an assessment of the drywell corrosion to date (12 Refueling outage) and the projected corrosion rate for the following operating cycle. In a letter dated February 9, 1989, GPUN provided the staff with this information. The pertinent information as given by GPUN is summarized in Table 1 (enclosed).

On the basis of the corrosion rate listed in Table 1, the licensee concluded that the most limiting condition is in the sand bed region of the drywell shell and the drywell shell thickness is projected to be acceptable until June 1992. In an attempt to reduce the corrosion rate, the licensee has (1) installed cathodic protection in selected sand bed locations, (2) taken steps to eliminate water leakage from reactor building equipment and refueling cavity, and (3) drained water from sand bed region. In order to assure the structural integrity of the drywell, the licensee has committed periodic UT thickness measurements of the drywell shell at all outages of opportunity. The licensee emphasized that the projection to June 1992 was based on conservative approaches.

Based on our review of the information provided by GPUN, we concur with the licensee that with the actions taken and to be taken by the licensee to ensure drywell integrity, and that plant operation can continue to the 13R refueling outage. In the event that efforts to arrest corrosion are not successful the licensee has argued that existing conservatism would still allow operation. However, the staff has reservations due to the fact that such conservatisms are not easily quantifiable and are required in assuring drywell adequacy for the protection of public health and safety. The licensee is required to perform

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thickness measurements and reconfirm the adequacy of the containment integrity at future outages of opportunity, including forced outages requiring drywell entry during the next cycle, but no later than prior to the resumption of power operation following the 13R refueling outage.

Sincerely,

/s/

Alexander W. Dromerick, Project Manager
Project Directorate I-4
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosure:
Table 1

cc w/enclosure:
See next page

DISTRIBUTION
Docket File
NRC & Local PDRs
PDI-4 Reading
SVarga, 14/E/4
BBoger, 14/A/2
SNorris
ADromerick
OGC (for info. only)
E. Jordan, 3302 MNBB
B. Grimes, 9/A/2
ACRS (10)

[OYST CRK DRYWELL CONT 72029]

LA:PDI-4	PM:PDI-4	D:PDI-4
SNorris	ADromerick:bd	JStolz
04/27/89	04/28/89	04/28/89

ENCLOSURE

TABLE 1

Location (Elevation)	Thickness (Inch) Nominal	Code Req'd	UT Measured	Corrosion Rate (MPY)
8'-11 3/4" to 12'-3" (sand bed region)	1.15	0.700	0.838	-27.6 ± 6.1
50' - 2"	0.77	.725	0.750	-4.3 ± .03
87' - 5"	0.64	.639	0.620*	0

*Accepted on the basis of data from certified material test reports (CMTRs) and no corrosion after plant operation (corrosion occurred during erection).

U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-219/90-03

Docket No. 50-219

License No. DPR-16

Licensee: GPU Nuclear Corporation

1 Upper Pond Road

Parsippany, New Jersey 07054

Facility Name: Oyster Creek Nuclear Generating Station

Inspection Conducted: January 7, 1990, - February 17, 1990

Participating Inspectors: M. Banerjee, Resident Inspector
E. Collins, Senior Resident Inspector
D. Lew, Resident Inspector

Approved By: Ronald W. Hernan 3/14/90
R. Hernan, Acting Section Chief, Date
Reactor Projects Section 4B

Inspection Summary:
Inspection Report No. 50-219/90-03 for January 7, 1990 - February 17, 1990

Areas Inspected: The inspection consisted of 240 hours of direct inspection hours by resident inspectors. The areas inspected included observation and review of plant operational events (paragraph 1.0), the fire protection deluge system (paragraph 2.0), main steam isolation valve leak repair (paragraph 3.0), drywell wall thinning measurements (paragraph 4.0), recirculation pump discharge valve failure (paragraph 5.0), recirculation pump "A" seal failure (paragraph 6.0), isolation condenser steam leak (paragraph 7.0), core spray keep fill pumps (paragraph 8.0), engineered safeguard feature system walkdown (paragraph 9.0), monthly maintenance observation (paragraph 10.0), monthly surveillance observation (paragraph 11.0), review of the Fitness For Duty Initial Training Program (paragraph 12.0), and onsite review of Licensee Event Reports (paragraph 15.0).

Results: The plant was operated in a safe manner during this inspection period. Licensee discovery of an inoperable deluge system 16 days after a trouble alarm is an unresolved item. The absence of documentation of material used in a valve repair is an unresolved item. Licensee evaluation of recirculation pump seal problems was thorough, and the subsequent removal of the pump from service was well planned and executed. Recirculation pump discharge valve problems may have contributed to seal failure. The Standby Gas Treatment System (SGTS) was evaluated as able to perform its intended safety function. Initial training sessions for the Fitness For Duty Program were well presented.

The licensee changed the date for their estimate to reach minimum code wall thickness in the drywell from June 1992 to June 1991.

The steam leak was also repaired during the 12U-8 unplanned outage by Leak Repair Company. The inspector reviewed the work package. A clamp was installed around the bonnet flange, the inside of which was injected with Fermanite 2X material to seal the body to bonnet area leak. The installation of the clamp was not considered a temporary variation as the installation of the clamp did not affect system function or operation. The licensee determined that because of the weight of the clamp, the additional loading was acceptable and seismic qualification was not affected. A final injection of Fermanite was made during startup at 1000 psig reactor pressure. The leak was minimized to a very small value. The licensee evaluated it as acceptable. A permanent repair is scheduled to be made during 13R outage.

The work package did not include any QA paperwork documenting the acceptability of the vendor supplied Fermanite material. The licensee later identified that a QA receipt inspection was not performed before the Fermanite material was accepted for installation. This item is unresolved. (UNR 50-219/90-03-02).

4.0 Drywell Wall Thinning

During outage 12U-8, the licensee performed ultrasonic measurements of the drywell wall thickness. The results showed that the most limiting portion of the drywell had shifted from the sand bay area to the 51-foot elevation and the most conservative estimate of the time when minimum code wall thickness would be reached had changed from June 1992 To June 1991.

A telephone conference was initiated by the licensee to inform the NRC about their preliminary findings. During the conference, the licensee stated that a copy of the revised safety evaluation will be provided to the NRC Project Manager and the resident inspectors.

5.0 "A" Recirculation Pump Discharge Valve

On 1/10/89, a plant shutdown was commenced when the "A" recirculation discharge valve failed to close and the recirculation loop was placed in an isolated condition. When the licensee was able to place the loop in an idle configuration, the plant shutdown was secured and the plant returned to full power.

Technical specifications allow continued plant operation with one loop in an idle configuration. In an idle loop configuration, the recirculation pump is stopped with the discharge valve shut and the discharge bypass valve and the suction valve open. If the suction valve is shut, the recirculation loop is considered isolated and the plant must be in cold shutdown within 24 hours.

The "A" recirculation loop was isolated during an evolution to remove the "A" recirculation pump motor generator from service for maintenance. The sequence to remove the motor generator from service required shutting the

Nuclear

GPU Nuclear Corporation
One Upper Pond Road
Parsippany, New Jersey 07054
201-316-7000
TELEX 136-482
Writer's Direct Dial Number:
(201) 316-7246

April 11, 1990
5000-90-1910

U. S. Nuclear Regulatory Commission
Mail Station P1-137
Washington, DC 20555

Attention: Document Control Desk

Dear Sir:

SUBJECT: Oyster Creek Nuclear Generating Station
Docket No. 50219, Licensing No. DPR-16
Oyster Creek Drywell Containment

References: GPUN Letters 5000-88-1633 dated September 12, 1988, 5000-89-1717 dated February 9, 1989, NRC Letter dated April 28, 1989, and GPUN Letter 5000-89-1820, dated September 29, 1989

In our conference call on March 8, GPUN committed to provide the staff with information on our recent findings regarding the Oyster Creek drywell wall thickness and our plans to assess and maintain the structural integrity of that vessel. This letter satisfies that commitment.

It has been our practice for several years to monitor the drywell thickness at selected representative locations by periodic ultrasonic (UT) inspection during plant outages where a drywell entry is made. One such UT inspection was made during the brief Oyster Creek outage in February of this year. The wall thickness data obtained during that inspection suggested that corrosion rates in some locations were higher than previously projected. As a result of those findings, we prepared an update (Revision 4) to our safety evaluation, concluding that, based on present analyses and observed corrosion rates, the drywell's service life can be conservatively confirmed to extend beyond our current operation cycle, that is, mid-1991.

Because the February database was somewhat limited, we decided also to conduct a more extensive examination at the next opportunity. That examination was conducted during the 12UJ outage (March 26 through April 3, 1990), and the results are reported below. We chose also to expand significantly our other ongoing activities to abate the drywell corrosion, to analyze the drywell, and to develop methods for any needed drywell repair. Our plans in these areas are also discussed in this letter.

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12UJ INSPECTION PROGRAM

For the 12UJ outage, GPUN conducted the following inspections:

1. We re-inspected areas in the sand bed and the sphere where the February 1990 data indicated apparent changes in corrosion rate. These additional confirmatory data were obtained from Bay 5 in the Sphere (El. 50'-2") and from Bay 13A in the sand bed region, the same locations where the February 1990 data was taken.

2. We inspected additional areas in the sand bed region and El. 50'-2" which previously had exhibited lower corrosion rates. This included eight locations in the sand bed region which had not been inspected since October 1988 and performance of an A-scan of the accessible portions of the drywell circumference at El. 50'-2". We also took grids of 7 x 7 UT readings at the three thinnest points found by A-scan of the 50'-2" circumference.

3. We re-inspected the three regions in the upper cylinder (El. 87'-5") where we had previously not observed ongoing corrosion. (Had this examination showed ongoing corrosion, the plan called for an A-scan of accessible segments of the drywell circumference at El. 87'-5" followed by taking grids of 7 x 7 UT readings at the three thinnest locations found by A-scan. This expanded examination at 87'-5" proved unnecessary.)

An evaluation ot the data taken in 12UJ as described above indicates that the conclusions of the safety evaluation (SE 000243-002, R4) are unchanged. This is based upon the determination that corrosion rates in the sand bed and sphere are about the same as those rates calculated from the February 1990 data. The areas inspected in the sand bed which had shown low corrosion when last examined in October 1988 have not changed with the exception of one location in Bay 13D. We plan to redesignate this location as a Priority 1 location for frequent monitoring pending completion of our evaluation of the 12UJ data. Our evaluation of the three thinnest locations on El. 50'-2" found by the A-scan shows that the minimum thickness around the circumference is consistent with that at the Priority 1 location currently being monitored. Finally, our evaluation of the three regions in the upper cylinder showed no ongoing corrosion.

In addition to the UT inspections during 12UJ, we also extracted a 2" diameter sample (core plug) from drywell Bay 13A in the sand bed. Bay 13A is an area of apparent significant corrosion (based on February 1990 data) which is not cathodically protected. This core plug was removed and will be chemically and metallurgically examined to determine if significant corrosion is occurring and to identify the corrosion mechanism. Removal of the plug also permitted removal of a sample of sand for chemical analysis to assess the condition of the sand bed.

While the lab results of the core plug and surrounding sand are not yet in hand, by visual inspection the core plug looked similar to those removed in 1986, and the surrounding sand appeared relatively dry.

ONGOING WORK

Based on our conclusions from the drywell inspection activities in February and March of this year, we are proceeding with several parallel work paths on a very high priority. Our ultimate objective is to ensure that the drywell is, and remains, structurally adequate to meet its intended safety function. Our workplan includes several main elements, as follows:

- Augmented data acquisition
- Corrosion mitigation tasks
- Structural analysis
- Drywell modification/repair.

Our plan of attack in each of these areas is outlined in the following sections.

Augmented Data Acquisition

Our approach here is to build on the existing database of UT wall thickness measurements and other examinations already conducted, and to continue on an aggressive data acquisition program. We are considering an augmented effort to include measurements at locations not yet interrogated in order to provide high statistical confidence that our program does in fact characterize the entire drywell vessel. Our feasibility study of the expanded plan will take into account both accessibility and radiation exposure implications. Our target is to complete our evaluations by September 1990. Until implementation of any augmented program, we will continue the current program.

Corrosion Mitigation

This involves several activities. The primary one is to evaluate the effectiveness of the existing cathodic protection (CP) system and to consider design and/or operational changes to enhance its performance. The system installed at Oyster Creek is quite extensive and was the result of a major engineering effort. We have been monitoring the effectiveness of this system since placed in operation in 1988. So far it appears to be less effective than we had hoped. A system performance test has recently been concluded, and the results are currently being evaluated by GPUN and Corrosion Services Co., Ltd. (the consultant who designed the system). The results of this evaluation should indicate the level of protection being afforded the drywell and potential enhancements to the operating system.

Also, actions will continue to prevent or retard intrusion of water into the gap and the sand bed. During the 12R outage, a strippable coating was applied to the refueling cavity prior to refueling in order to eliminate this source of water into the sand bed. For the 13R outage, the application of this type of coating will be expanded to include both the refueling cavity and the equipment storage pool, which is another presumed source of water.

In parallel with the above evaluation of CP and because of the uncertainties in its effectiveness, we are reconsidering other mitigation methods we previously evaluated, including the use of drying systems, addition of chemical corrosion inhibitors, and chemical inhibitors in combination with CP.

Our target in corrosion mitigation is to develop a course of action by October 1990 with implementation as soon as possible thereafter. Over the long term, the effectiveness of the installed CP system or any other selected methods will be monitored by ongoing UT measurements.

Structural Analysis

Our objective is to develop a more comprehensive understanding of the dynamic structural performance of the drywell vessel under varying conditions in order to ensure that the drywell is structurally adequate for continued use. This will include application of state-of-the-art techniques for modelling and analyzing the vessel, review of the design basis loading conditions, and consideration of the actual material properties of the Oyster Creek vessel. Our target in this activity is to conclude our structural analysis work by September 1990.

Drywell Modification/Repair

Our approach here is to build on previous evaluations of potential structural repair of corrosion damaged areas of the drywell. This will include review of the previous study performed by CB&I Services to define conceptually various options for structural repair in the sand bed region. This study evaluated selected plate replacement, doubler plates, weld overlay, and stiffener structures as potential repair methods. This study will be expanded to consider Oyster Creek plant-specific constructability requirements, radiation dose estimates, decontamination and contamination control requirements, radwaste disposal requirements, schedule development, cost estimates, and locations in the drywell most likely to require repair. Options for repair of elevations above the sand bed will also be evaluated.

Our target is to select a preferred repair option before the 13R outage, and then take steps to be ready to implement that option if and when it is required. During the 13R outage, drywell walkdowns will be performed to assess physical aspects of the job and to compile the information required to complete selection of and planning for a repair option.

In summary, GPUN's Safety Evaluation 000243-002 (Rev. 4) has conservatively confirmed safe operation of Oyster Creek through the 13R outage until August 1991. Our current actions, including continued inspections, structural analysis, and corrosion mitigation will establish the basis for continued operation until the 14R outage. Over the longer term, the repair contingency plan will be developed to the extent that it is available to support a timely decision by GPUN regarding steps necessary to ensure drywell serviceability.

We will continue to keep you informed of our progress in this area. If you have any questions or you wish to schedule a meeting for further discussion, please contact M. W. Laggart, Manager, BWR Licensing at (201) 316-7968.

Sincerely,

J. C. DeVine, Jr.
Vice President, Technical Functions

JCD:mes

cc: Administrator
Region I
U. S. Nuclear regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

NRC Resident Inspector
Oyster Creek Nuclear Generating Station
Forked River, NJ 08731

Mr. Alex Dromerick, Jr.
U. S. Nuclear Regulatory Commission
Mail Station P1-137
Washington, DC 20555

MISC/17

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

July 10, 1990

Docket No. 50-219

Mr. E. E. Fitzpatrick
Vice President and Director
Oyster Creek Nuclear Generating Station
P.O. Box 388
Forked River, New Jersey 08731

Dear Mr. Fitzpatrick:

SUBJECT: REVIEW OF OYSTER CREEK DRYWELL CONTAINMENT STRUCTURAL INTEGRITY (TAC No. 75064)

References:
1. GPU Nuclear "Safety Evaluation on Steel Shell Plate Thickness Reduction" SE No. 00243-002, Rev. 4, Dated February 16, 1990
2. Letter to NRC from J. C. Devine of GPU Nuclear, Dated April 11, 1990

In February 1990 GPU Nuclear Corporation (GPUN), the licensee of the Oyster Creek Nuclear Generating Station informed NRC staff that based on UT measurements made in February there is evidence of possible ongoing corrosion at elevation 50' - 2", which is above the sand bed region, at a rate greater than previously estimated. As a result GPUN projected that the minimum thickness of the drywell shell will be reached in August of 1991 instead of in June 1992 as previously projected. In addition the licensee has found the cathodic protection system (CPS) is less effective than expected. In view of these findings the staff was concerned with the continuous deterioration of the drywell shell and its effect on the structural integrity of the drywell. On March 8, a conference call between GPUN representatives and NRC staff was held. The staff requested GPUN to submit a plan for short and long term actions to address the degraded condition of the drywell. In response to the staff's request GPUN submitted Reference 2 which is summarized as follows.

(A) During the 12 UJ outage (March 26 through April 3, 1990), GPUN conducted a more extensive examination than than performed in February 1990. It consisted of inspecting areas in the sand bed and at elevation 50' - 2" and additional areas in these same regions previously found to exhibit lower corrosion rates, and areas in the upper cyclinder at elevation 8 7' - 5". In addition to UT measurements, a 2" diameter core plug was taken from drywell bay 13A in the sand bed together with the removal of a sand sample. From the evaluation of

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the UT measurements taken in the UJ outage, GPUN found that the safe operation of Oyster Creek through August 1991 can be assured as indicated in Reference 1 is still valid.

(B) GPUN has formulated a work plan which consists of the following elements:

(1) Augmented Data Acquisition - Measurements will be made at locations not yet inspected in order to augment the data acquired as a measure to provide high statistical confidence that the inspection program instituted does in fact characterize the entire drywell.

(2) Corrosion Mitigation - The existing cathodic protection system is being evaluated for its effectiveness and for possible enhancement. Other methods of mitigation are under consideration. Measures to prevent or retard intrusion of water into the gap and sand bed are being taken.

(3) Structural Analysis - Use of state-of-the-art techniques for modelling and analyzing the vessel is being considered in conjunction with the use of the actual material properties.

(4) Drywell Modification/Repair - A study has been made for various options such as selected plate replacement, doubled plates, weld overlay or stiffeners for structural repair in the sand bed region and other areas. Factors such as constructability and radiation exposure are to be taken into consideration. A preferred repair option will be selected before the 13R outage, and steps will be taken to be ready for implementing that option if and when it is required.

In accordance with GPUN, the effort outlined above under (A) is to confirm safe operation of Oyster Creek through the 13R outage until August 1991 as indicated in Reference 1, and the effort under (B) (1), (2) and (3) is to establish the basis for continued operation until the 14R outage. The effort in (B)(4) above for longer term is formulated as a repair contingency plan.

From the information provided by GPUN under (A) above it can be stated with reasonable confidence that the Oyster Creek drywell minimum thickness will not be violated until at least August 1991 as indicated in Reference 1. However the staff has some reservations on GPUN's use of the effort in (B) (1), (2) and (3) as a basis for continued operation until the 14R outage, especially GPUN's intention to use strength values of the drywell steel in the certified material test reports (CMTRS). GPUN's rationale for such an approach is that in the evaluation of the cylinder portion (EL 8 7' - 5") GPUN used the allowable stress derived from CMTRs with the approval of NRC. However, from an AISI survey of test results for thousands of individual product samples, it has been found that strength levels vary as much as 20% from the CMTR test values. Therefore it is the staff's position that minimum specified strength values (e.g, ASME Code minimum strength values) should be used as the basis for allowable stresses in the stress re-evaluation of degraded components. Consequently GPUN cannot predicate drywell integrity on CMTR values.

We believe that plans should be made for the implementation of the drywell repair by August 1991 not relying on favorable results of the effort in (B)(1), (2) and (3) to justify continued operation until the 14R outage. If the reanalysis effort includes considering changes to the design basis of the plant, a license amendment will be required. At the same time GPUN should continue the inspection as presently instituted.

We will arrange a meeting with your staff during late August or early September at our Rockville office to hear the status of your work plan and discuss the NRC staff's concerns noted in this letter.

Sincerely,

Original signed by

Alexander W. Dromerick, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

cc: See next page

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

October 3, 1990

Docket No. 50-219

LICENSEE GPU NUCLEAR CORPORATION
JERSEY CENTRAL POWER & LIGHT COMPANY

FACILITY: OYSTER CREEK NUCLEAR GENERATING STATION

SUBJECT: Summary of September 19, 1990 Meeting With GPU Nuclear Corporation (GPUN) to Discuss Matters Related to Oyster Creek Drywell Corrosion.

On Wednesday, September 19, 1990, a meeting was held at the NRC, One White Flint North, Rockville, Maryland with GPUN, the licensee, to discuss the drywell corrosion problem at the Oyster Creek Nuclear Generating Station. Enclosure 1 is the list of participants that attended the meeting.

Enclosure 2 is the licensee's morning session agenda. Enclosure 3 is the licensee's afternoon session agenda. The following is a summary of the significant items discussed.

The Licensee indicated that the Oyster Creek Drywell (1) has been examined thoroughly: its present condition and the ongoing corrosion problem are well understood, (2) is a rugged, conservatively designed pressure vessel; it has ample margin to permit continued safe plant operation for several years while corrective action is being taken, and (3) program is a very high priority, resource intensive, and multifaceted one and that GPUN intends to arrest the drywell corrosion by positive means and ensure containment integrity for the full licensed life of the plant. During the discussion the licensee described a three phase program to address the drywell corrosion problem.

The licensee stated that based on analysis performed during the first phase of the program, GPUN concluded that:

1) current best estimates of corrosion rates at the worst areas of the drywell sphere indicate Code allowable stresses will not be exceeded for at least three years, even if corrosion extended over its entire surface.

2) Taking into account actual conditions, the Oyster Creek Drywell will be in full compliance with the ASME code for at least three years even at very conservatively projected (95% confidence level) corrosion rates.

3) The Oyster Creek design basis pressure (62 psig) is conservative by a significant margin.

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The licensee stated that he will submit the details of his program including the structural analysis by December 1990. The staff advised the licensee that GPUN should expedite the submittal including plans to arrest corrosion.

Alexander W Dromerick

Alexander W. Dromerick, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc w/enclosures:
See next page

ENCLOSURE 2

OYSTER CREEK DRYWELL CORROSION

GPU Nuclear / NRC Meeting

September 19, 1990

MEETING OBJECTIVES

- To communicate the scope, depth, objectives, and expectations of GPUN's Oyster Creek Drywell Program.
- To permit technical exchange among GPUN and NRC technical staff members on engineering and analysis issues.
- To obtain feedback from NRC on GPUN's course of action
- To agree on subsequent steps and their timing.

THREE KEY POINTS

1. The Oyster Creek Drywell has been examined thoroughly; its present condition and the ongoing corrosion problem are well understood.

2. The Oyster Creek Drywell is a rugged, conservatively designed pressure vessel; it has ample margin to permit continued safe plant operation for several years while corrective action is being taken.

3. GPUN's drywell program is a very high priority, resource intensive, and multifaceted one; we intend to arrest the drywell corrosion by positive means and ensure containment integrity for the full licensed life of the plant.

OYSTER CREEK DRYWELL PROGRAM

Phase:	Phase 1	Phase II	Phase III
Objective:	Develop Success Path	Solve the Problem	Keep It Solved
Timing:	Through 1990	Through 1992	Long Term
Focus:	• Examine all information in-hand. • Confirm shell integrity through Phase II. • Develop detailed plan and engineering for full solution. • Continue corrosion prevention activities.	• Implement plans/engineering developed in Phase I to: — Fully characterize shell. — Complete analysis of shell strength and margin. — Arrest corrosion.	• Implement life-of-plant monitoring program. • Other work as needed.

BASES FOR OYSTER CREEK DRYWELL SAFETY DETERMINATION DURING PHASE II

1. Based on current best estimates of corrosion rates at the worst areas of the drywell sphere, Code allowable stresses will not be exceeded for at least three years, even if that corrosion extended over its entire surface.

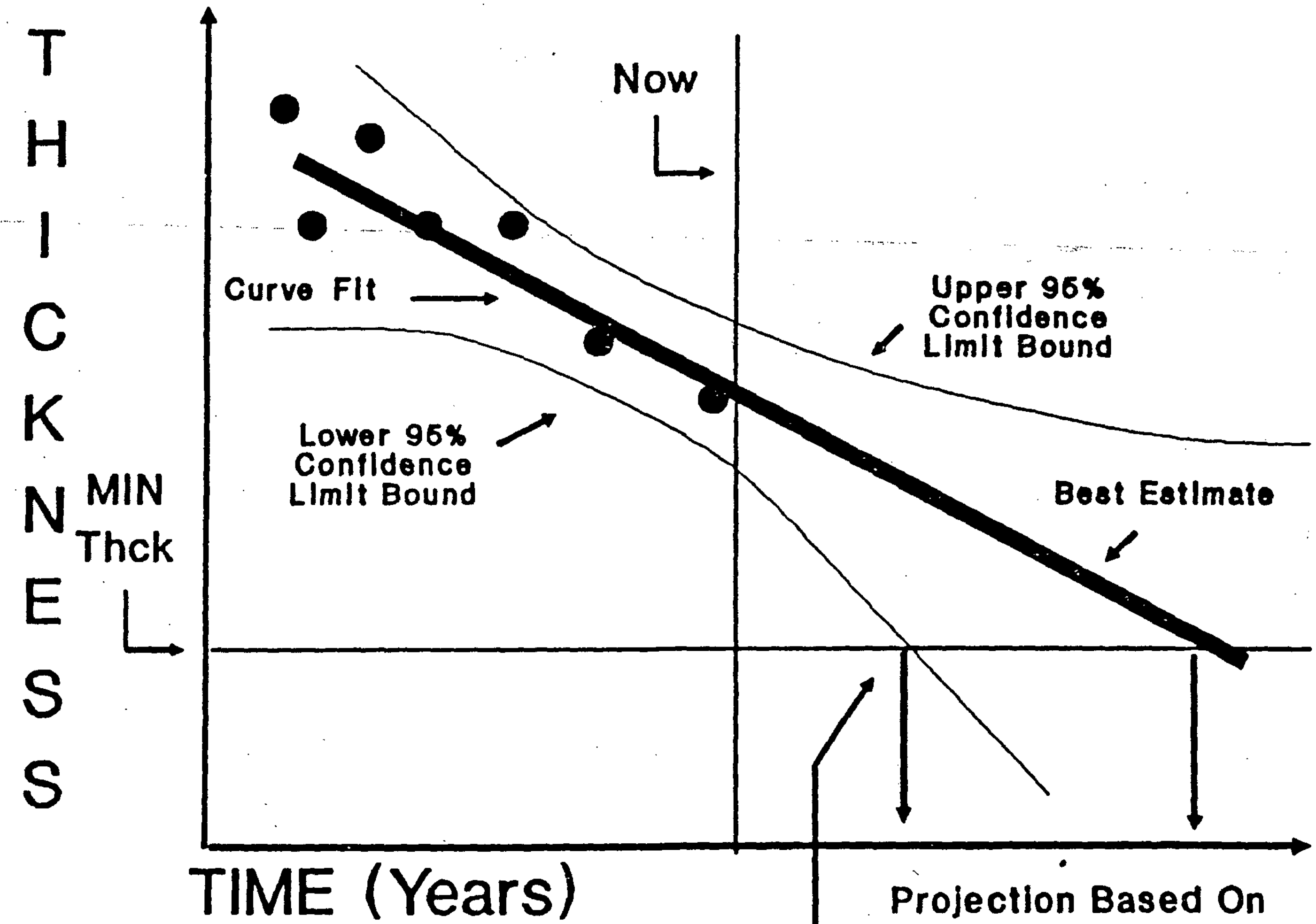
2. Taking into account actual conditions, the Oyster Creek Drywell will be in full compliance with the ASME Code from at least three years, even at very conservatively projected (95 percent confidence level) corrosion rates.

3. The Oyster Creek design basis pressure (62 psig) is known to be conservative by a significant margin.

SUMMARY OF DRYWELL ACTIVITIES

	DEC '86	OCT/ NOV '87	JUL/ OCT '88	NOV '88 FEB '89 (12R)	CYCLE 12	'90
SANDBED REGION	– UT Readings – Core Samples		– UT Readings	– Installed cathodic protection (CP) – Removed water – UT Readings	– Energized CP system 3/89 – UT Readings 6/89 – UT Readings 9/89	– UT Readings (Feb, March & April) – Core Samples
SPHERICAL REGION (50' - 2")		– Core Samples – UT Readings	– UT Readings	– UT Readings – Steps taken to reduce water sources	– UT Readings 6/89 – UT Readings 9/89	– UT Readings (Feb, March & April) – Expanded to elevation 51' - 10" (April)
CYLINDRICAL REGION (87' - 5")		– UT Readings		– UT Readings – Steps taken to reduce water sources	– UT Readings 6/89	– UT Readings (March)

Projections Based On Inverse Regression
(SCHEMATIC)



SHELL EXAMINATION SUMMARY

Sand Bed Region

- Extensive measurement data in-hand.
- Corrosion rate is highest of 3 regions ($\approx$39 mils/yr.)
- This is the thickest part of the shell (initially 1.154").
- Several years margin remain based on:
  - — Best estimate corrosion rate in worst area.
  - — Original design basis.
  - — Code requirements.

SHELL EXAMINATION SUMMARY (Cont'd)

Spherical Region

- 2.5 years data in-hand (although less extensive than sandbed).
- Observed corrosion rate is low (≈4.6 mils/yr.)
- Initial shell thickness is .722" and .770".
- Several years margin remain based on:
  - — Best estimate corrosion rate in worst area.
  - — Original design basis.
  - — Code requirements.

SHELL EXAMINATION SUMMARY
(Cont'd)

Cylinder Region

- 2.5 years data in-hand (although less extensive than sandbed).
- No ongoing corrosion observed.
- Environmental conditions make region less prone to corrosion.
- Area of least margin.

ASME III - SUBSECTION NE EVALUATION

- Code defines "local primary membrane stress intensity" to be greater than 1.1 S_{mc} and less than 1.5 S_{mc}.

- This 10% variation in allowable stress was provided because of the "beam on elastic foundation; effects, i.e., stress decays but remains greater than zero for significant distances.

- Clearly not intended to design for 1.1 S_{mc}, however, given a design that satisfies the code intent, it is not a violation of the code for the membrane stress to be between 1.0 S_{mc} and 1.1 S_{mc} for significant distance.

- Largest exceedance of S_{mc} is 3%. Therefore, drywell currently complies with the code.

DETAILED INSPECTIONS

- Sandbed Region, EL. 11' - 3" (1986)
- Cylindrical Region, El. 87' - 5" (1987)
- Spherical Region, El. 50' - 2" (1987 & 1990)
- Spherical Region, El. 51' - 10" (1990)

ONGOING INSPECTION PROGRAM

- Outage of opportunity

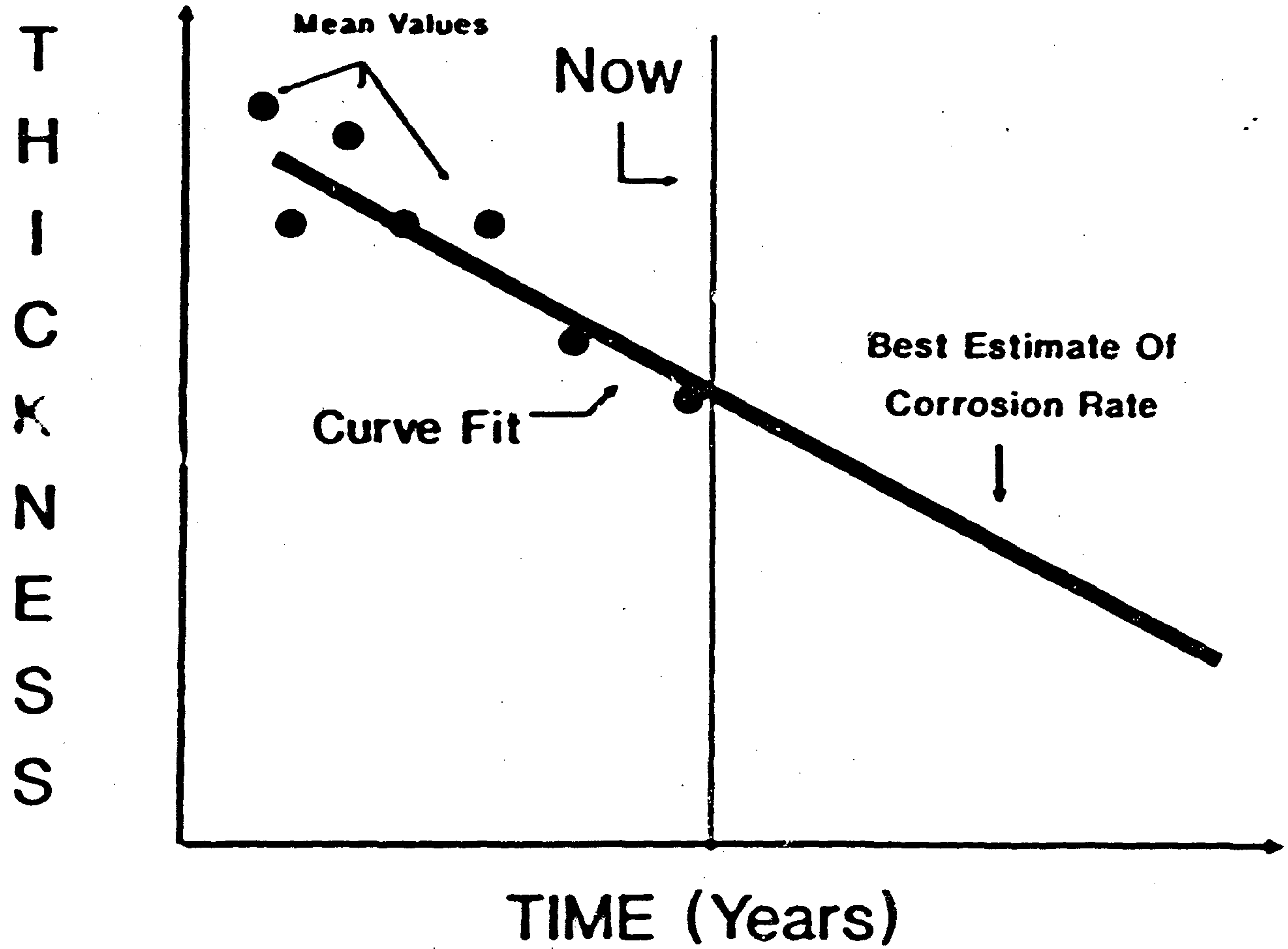
(and)

- Drywell entry for reasons other than program inspection
- Priority #1 Locations - $\geq$ 3 month frequency
- Priority #2 Locations - $\geq$18 month frequency

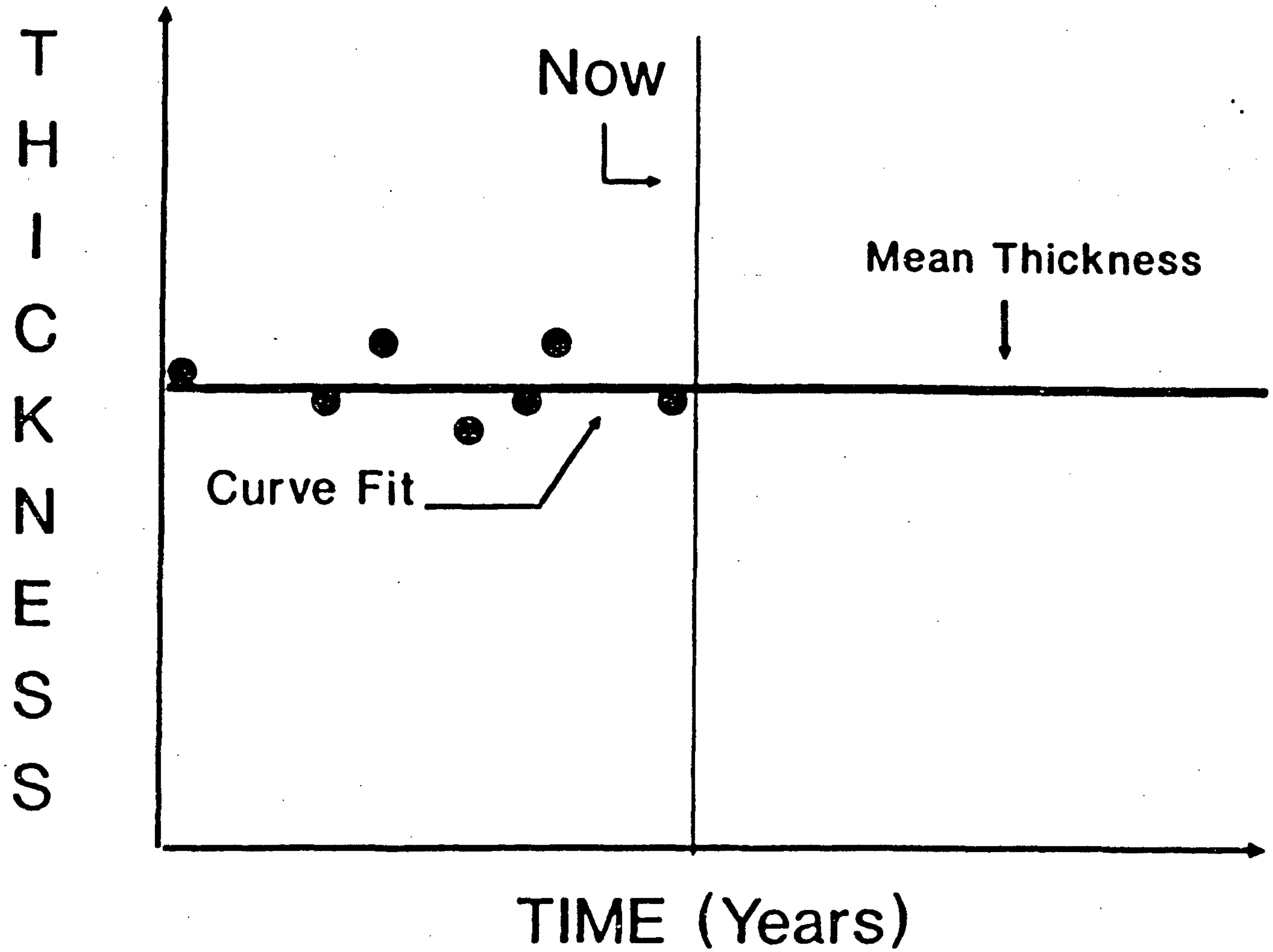
CORROSION RATE CALCULATION

- Mean of 49 points
- Mean is plotted over time
- Linear regression model/curve fit?
  - — Slope of curve - calculated corrosion rate
- Mean model/curve fit?
  - — No slope - No corrosion rate

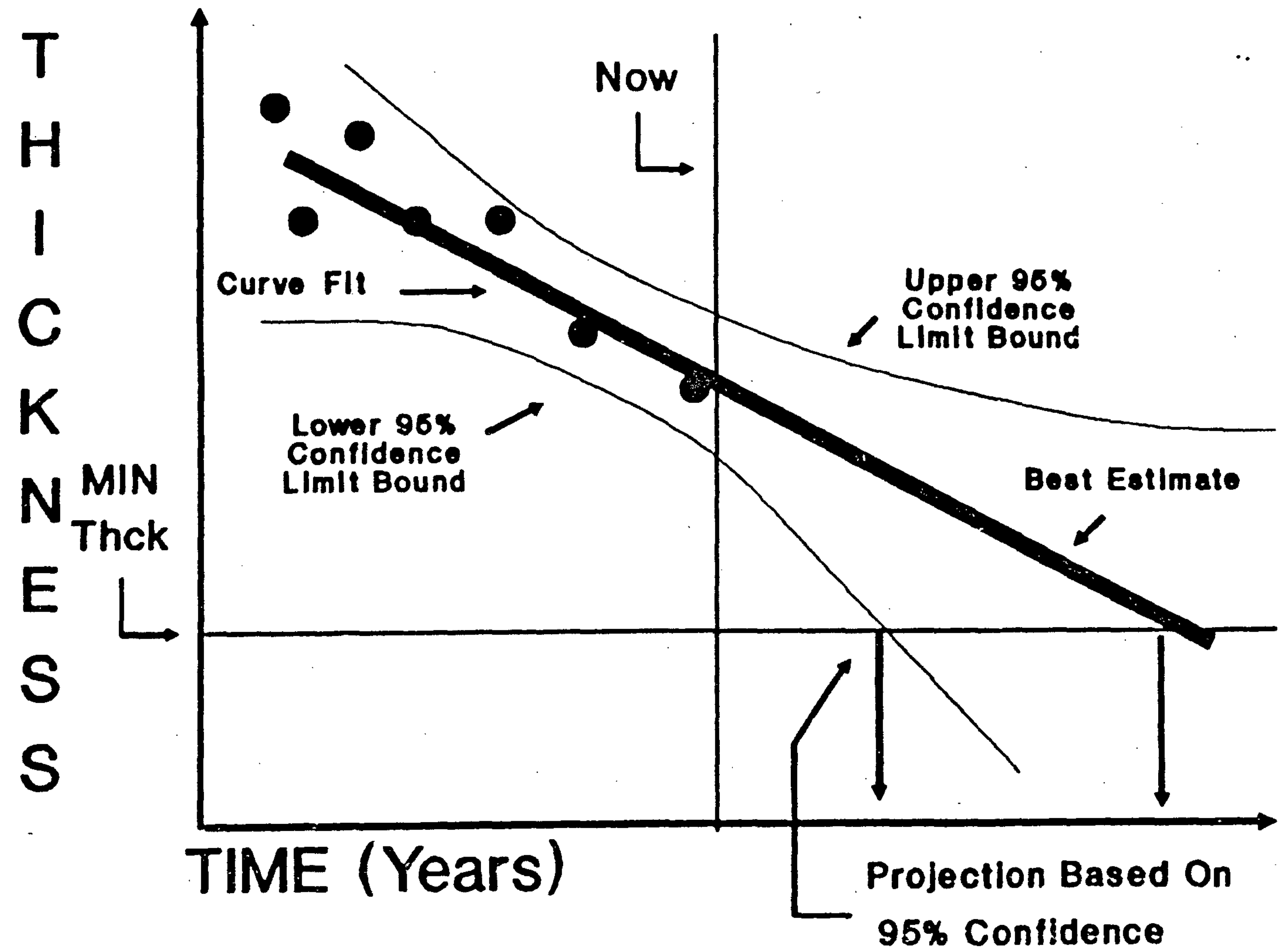
Curve Fit Based On Linear Regression



Curve Fit Based On Mean Model



Projections Based On Inverse Regression
(SCHEMATIC)

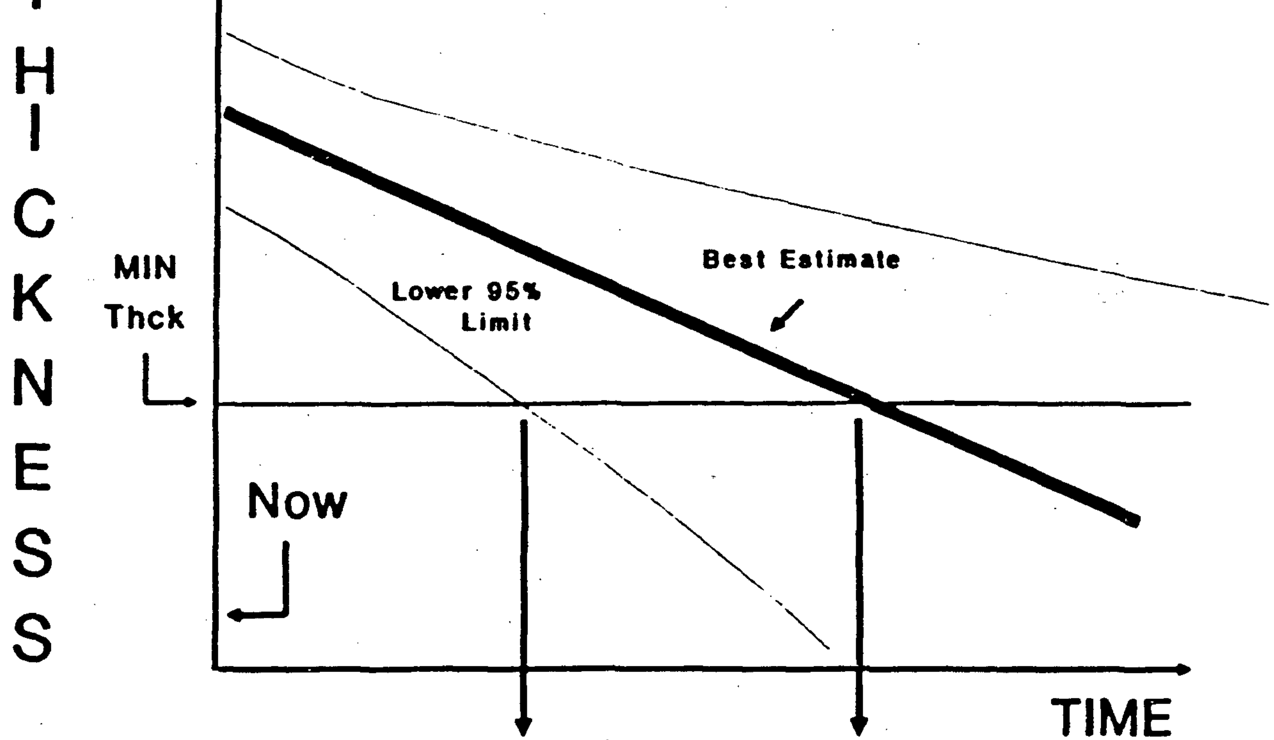


SIGNIFICANT CORROSION RATE CONCLUSION (AS OF APRIL, 1990)

- Spherical region, elevation 50' - 2"
  - — Bounding calculated corrosion rate = 4.6 ±1.6 MPY
- Sandbed region, elevation 11' - 3"
  - — Bounding calculated corrosion rate = 39.1 ±3.4 MPY
- Sandbed, cathodically protected regions
  - — No significant corrosion rate reduction (15 to 25 MPY)
- Cylindrical region elevation 87' - 5"
  - — No observed ongoing corrosion
- Spherical region elevation 51' - 10"
  - — Calculated corrosion rate not available

Current Projections

(Based on Data Up To April 1990)



Sandbed	Jul 1993	Mar 1994
Elev. 50'-2"	Jun 1992	Sep 1994
Elev. 51'-10"	Oct 1991 •	Jul 1993

•NOTE: Projection Based on Elev. 50'-2" Corrosion Rate

SUMMARY:

This augmented inspection plan, using 60 locations selected at random, provides a statistically based characterization of the drywell. The inspection plan provides a sensitive test for unacceptable observations. Measurements of the region adjacent to a low area, should one be found, will be made in order to show that the condition of the plate is, in general, much better.

CONTROLLING LOAD CASES FOR VARIOUS LOCATIONS

(Reported from analyses completed from 1986 to 1988)

- CYLINDRICAL REGION – (Design t=0.640", min. as found t=0.619") Accident Condition – Primary membrane stress caused by design pressure dominates
- SPHERICAL REGION (Design t=0.722") – Accident Condition – Primary membrane stress caused by design pressure dominates
- SPHERICAL REGION (Design t=0.770") – Accident Condition – Primary membrane stress caused by design pressure dominates
- SPHERICAL REGION SANDBED (Design t=1.154", min. as found t=0.808", assumed t=0.700") – Refueling Condition – Buckling due to compressive stresses caused by deadweight and water in refueling cavity + 2 psi external pressure dominate

REVIEW OF RESTART EVALUATIONS (1986/1987)

- CYLINDRICAL REGION:

  Established minimum as found thickness of 0.619" accepted using CMTR data and the fact that there is no ongoing corrosion.

- SPHERICAL REGION SANDBED:

  The stress analysis was performed to ensure structural integrity for the shell assumed to be 0.700" thick. This configuration subjected to the combined load cases yielded the following conclusions:

  - The tensile stresses were less than the specified allowable stress from the 1962 issue of the ASME Code, Section VIII, including the Summer 1964 Addendum plus Code Cases 1270N-5 and 1272N-5 (1.1 Sm= 19,250 psi).

  - The compressive stresses were less than the specified allowable stress computed according to rules of Code Case N-284.

ADDITIONAL STRUCTURAL ANALYSIS

- COMPARISON WORK FOR BUCKLING EVALUATION:

  Capacity margin (buckling) in the sandbed is improved by including the details of vent pipe and its reinforcing plates.

  Stability analysis comparing 3-D FEM methods and BOSOR techniques (shell of revolution) using a similar Mark I drywell has been performed using the same percentage reduction in wall thicknesses as observed at Oyster Creek in the sandbed region.

  Loads were adjusted to produce a stress state at the midpoint of the sandbed equal to that computed for the Oyster Creek stability analysis.

  The ratio of the FEM results divided by the BOSOR results was computed and is equal to 2.1. Hence, the previously computed capacity margin of 1.00 is very conservative.

CORROSION ASSESSMENT CONCLUSIONS

- Different local environments most likely exist within the drywell annular space which would explain various corrosion rates observed.

- Aqueous corrosion is primarily responsible for the metal loss. Galvanic action, oxygen, pH and temperature are most likely influencing the rate.

- Corrosion mitigators must be aimed at changing local environments as well as global environments, ie. we must utilize a mitigative scheme which deals with the bulk environment in the sandbed or insulation material and with the environment in the oxide crust.

- Corrosion rates are within the bounds discussed in the literature for aqueous corrosion. Therefore, we do not expect to find regions of the drywell with more extensive metal loss than that already observed.

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

October 16, 1990

Docket No. 50-219

Mr. J. D. DeVine, Jr.
Vice President and Director
Technical Functions
GPU Nuclear Corporation
One Upper Pond Road
Parsippany, New Jersey 07054

Dear Mr. DeVine:

SUBJECT: DRYWELL CORROSION PROGRAM - OYSTER CREEK NUCLEAR GENERATING STATION

On September 29, 1990, GPU Nuclear Corporation (GPUN) met with the NRC staff to discuss the Oyster Creek Nuclear Generating Station's Drywell Corrosion Program. During the meeting, GPUN requested that the staff provide feedback regarding the Drywell Corrosion Program. As a result of the discussions held during the meeting the staff so far has identified the following aspects of GPUN's presentation that call for staff feedback. These are: 1) sampling of shell surfaces for UT measurements, 2) appropriateness of the use of ASME Section III Subsection NC, and 3) the need for detailed review of preliminary results of the stress analysis presented by GPUN. The Enclosure provides details of the required clarification.

If during our ongoing review of your program additional items requiring further clarification are identified we will notify you.

If you have any questions regarding the above, please contact me.

Sincerely,

Alexander W Dromerick

Alexander W. Dromerick, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc w/enclosure:
See next page

ENCLOSURE

REQUESTED CLARIFICATION REGARDING

OYSTER CREEK CORROSION OF DRYWELL SHELL

DOCKET NO. 50-219

There are several aspects of the licensee presentation that call for staff feed back, these are: i) sampling of shell surfaces for UT measurements, ii) appropriateness of the use of ASME Section III Subsection NC, and iii) need for detailed review of preliminary results of the stress analysis presented by the licensee.

i) Sampling plan for monitoring drywell corrosion: The licensee presented a statistically based inspection program of the entire shell surface not embedded in concrete. However, based on the results of observation so far, the licensee presented a correlation between corrosion and presence of moisture for example, in the sand region the plug samples 15A and 11A-H were dry and had corrosion rates equal to zero. It is not clear to the staff how the licensee plans to locate sensors for on-line monitoring of drywell corrosion rate at those places where the presence of moisture is likely. The staff needs to review the statistically based sampling plan.

ii) The original design code for the Oyster Creek shell is ASME Section VIII. Should the licensee choose to use a more recent code, there will be a burden on the licensee to clearly establish that the material selection, design, fabrication, inspection and surveillance in service are all in accordance with the requirements of the current code which should be the ASME Section III, Subsection NE, and Section XI.

iii) It is clear that through the corrosion process, the margin for over pressure capacity of the containment has been reduced (see GDC#50 and 51). Therefore, the staff judgment as to the adequacy of the drywell shell margin must be based on a detailed review of the stress calculations and the stress allowables.

iv) In your presentation you indicated that there has been leakage from refueling cavity liner, equipment pool and spent fuel pool. Describe the actions you will take to prevent leakage from these structures into the drywell gap and the effect of the leakage on other structures or equipment.

U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-219/90-21

Docket No. 50-219

License No. DPR-16

Licensee: GPU Nuclear Corporation
P.O. Box 388
Forked River, New Jersey 08731

Facility Name: Oyster Creek Nuclear Generating Station

Inspection At: Forked River, New Jersey

Inspection Conducted: October 29-31, 1990

Inspector: H. Kaplan 12-4-90
H. Kaplan, Sr. Reactor Engineer, Materials and Processes Section, EB, DRS — date

Approved by: E. H. Gray 12/11/90
E. H. Gray, Chief, Materials and Processes Section, EB, DRS — date

Inspection Summary: Inspection on October 29-31, 1990 (Report No. 50-219/90-21)

Areas Inspected: An announced inspection of the licensee's activities involving the drywell corrosion problem activities. The scope of this inspection included review of ultrasonic thickness procedures and records, inspection and repairs of suspected sources of leakage, review of metallurgical reports and a facility tour.

Results: On the basis of this inspection, it was concluded that the licensee's program for monitoring, repairing and evaluating the corrosion problem was comprehensive and was being conducted in a systematic manner in accordance with prescribed procedures. Of the area inspected, no violations were identified. The licensee has presented substantial evidence that the plant can be operated safely until the 14R refuel outage provided that thickness measurements are taken in the prescribed intervals, and show no significant loss in wall thickness.

DETAILS

1.0 Persons Contacted

1.1 GPU Nuclear Corporation

*E. E. Fitzpatrick, Vice President and Director
*J. A. Martin, Mechanical Engineer
*J. D. Amramovici, Manager, Pressure Vessels
*R. Zak, Licensing Engineer
*S. Gicobbi, Manager, Materials Engineering

1.2 U.S. Nuclear Regulatory Commission (NRC)

*G. Bagchi, Office of Nuclear Reactor Regulation (NRR), ESGB
*E. Collins, Sr. Resident Inspector

*Denotes attendance at exit meeting on October 30, 1990.

2.0 Scope

The objective of this inspection was to review the licensee's continuous on site activities regarding the drywell corrosion problem. The results of a plant walkdown of accessible areas and an evaluation of the licensee's analytical methodology by NRR will be reported separately by Mr. Goutam Bagchi. The overall strategy to monitor and control drywell corrosion had been presented by the licensee in a meeting held in Headquarters on September 19, 1990.

3.0 History

Corrosion was initially discovered by the licensee on the outside surface of the drywell in the sand cushion region of the drywell in late 1986. Since then, the licensee has carried out an extensive program to ensure the short and long term integrity of the drywell. The program includes continuous monitoring of the corrosion as reflected by frequent thickness measurements, inspection and repair of suspected sources of leakage which are believed to be responsible for the leaks, reanalysis of the drywell stresses, and a study of feasible corrective actions.

The corrosion apparently was caused by moisture trapped inside the thermal insulation surrounding the drywell and in the sand cushion around its base. The highest corrosion rate has occurred in the sand bed area (39 mils/year) followed by the spherical region (4.6 mils/year). No recent corrosion has been observed in the upper cylinder region. Although the calculated stresses based on thickness measurements and corrosion rates indicate a marginal condition from the standpoint of code allowable stresses, the licensee has concluded that the drywell will still be in compliance with the code at refuel outage 14R on the basis of assuming that the major source of leakage has been eliminated.

4.0 Findings

4.1 Ultrasonic Thickness Measurements

The inspector reviewed the methods and appropriate records associated with ultrasonic thickness determinations. The measurements are obtained from the inside of the drywell using a calibrated ultrasonic instrument (D METER) in accordance with GPUN Procedures 6150-QAP-7209.07 Rev. 0 and IS-328227-004 Rev. 2. Forty-nine (49) individual readings are taken in 11 discrete areas using a 6 inch x 8 inch grid template. The 11 areas covered 7 areas in the sand bed area, 3 in the cylinder region (87' level) and 1 in the spherical (51') level. To assure validity of the data, the instrument is calibrated before each set of data is taken. In the presence of the inspector, the licensee demonstrated the accuracy of the instrument using the specified stepped calibration standard. The inspector reviewed 2 recent data sheets 87-026-135 and 87-026-143 representing Bay No. 19 Area C (sand bed) and Bay No. 13 Area 6 (52'). Except for three anomalous points in 87-026-135, the inspector found no discrepancies. The three points were subsequently attributed to a welded plug in an area in which a core bar had been previously removed. The data is subsequently sent to GPU Engineering in Parsippany, New Jersey for analysis. Basically, the data points for each sector are averaged, statistically analyzed and compared with previous data to calculate conservative stress values as determined by corrosion rates and wall thickness measurements.

In addition to performing wall thickness measurements during the last outage (12R), the licensee removed a core sample from the sand bed Area 13A as part of his continuous effort to monitor the drywell corrosion. The inspector reviewed the GE metallurgical report covering evaluation of core bar 13A. The report concluded that the findings were similar to those generated in previous core bar evaluations and that no basic changes occurred in the conditions driving the corrosion of the drywell.

4.2 Repair Activities

The inspector reviewed certain aspects of the licensee's activities regarding the inspection and/or repair of the suspected sources of leakage. The major source of leakage which appears to be responsible for the corrosion of the drywell shell is the reactor cavity liner. The cavity is filled with demineralized water during refueling and thus provides a direct leak path to the outside surface of the drywell if there were defects in the liner. The inspector reviewed comprehensive visual and liquid penetrant inspection reports as documented in Material Nonconformance Report 87-240 which showed that the .109" thick type 304 stainless steel liner exhibited numerous cracks on its I.D. surface in addition to 2 severely damaged areas which were reported have been caused by movement of equipment used in refueling. The cracks showed no preferred orientation or preferred location with regard to base metal or welds. The inspector reviewed a metallurgical report (General Electric 88-178-006) which covered an evaluation of two through-wall samples which were removed from the cavity liner to include the cracks. The investigation did not disclose any material deficiencies or anomalies associated with the failure. Although the cracks were found to be transgranular, no detrimental anions such as Cl or F which are known to cause transgranular stress corrosion cracking were found to be associated with the cracking.

The report concluded that because of the wetted surface and thermal fluctuations, the most likely cause of failure was corrosion fatigue. The source of stress was believed to have occurred during initial welding and the restraint caused by welding to backing strips embedded in the concrete. The fluctuations may have been higher than anticipated because the liner was found to be .109" instead of the specified .250". The conclusions in the subject report appear to be valid.

Because of the excessive number of defects found in the cavity liner, the licensee opted to employ a unique, temporary system that covered 100% of the I.D. surface. The system consisted of a combination of stainless steel adhesive tape covered by two coats of a Latex barrier (ISOLOCK 300). The licensee provided the inspector a report (TDR-938) which showed that the tape-coating had been qualified for 125° F-10 week immersion service using both adhesion, pressure and leachate testing. The system is designed to be removed after refueling and is applied with the reactor head in place.

The inspector reviewed other documents pertaining to the inspection and repair of the suspected sources of leakage. These are listed below:

IS-328 257-001 - Repair of Reactor Cavity Concrete Trough

Material Nonconformance Report 85-034 Weld Repair and Inspection of Weld Defects in Equipment Storage Pool

Technical Specification - SP-1302-22-006 of Reactor Cavity - Repair of Reactor Cavity and Storage Pool Lining

Material Nonconformance Report 87-240

Installation Specification for Replacement of Drywell Vessel Core Sample Plugs

The inspector's review of these documents indicated that the prescribed activities were performed in accordance with appropriate procedures: Repair welds were inspected using various NDE procedures (magnetic particle, liquid penetrant and vacuum box). Documents included Quality Assurance requirements including inspection points and records. A sampling of welding activities indicated the use of appropriate ASME Section IX qualified procedures.

The licensee is currently exploring methods for removing the wet sand and possible repairs to reinforce the drywell if required. The cathodic protection system which has been in operation for several years has not been effective apparently because the major source of leakage has been eliminated.

5.0 Conclusions

On the basis of the above findings, the inspector concluded that the licensee's program for monitoring, repairing and evaluating the corrosion problem was being conducted in a systematic manner in accordance with prescribed procedures. Since the major sources of leakage has been found and corrected, no significant leakage has been observed as indicated by frequent inspections of five sand bed drains.

6.0 Management Meetings

Management was informed of the scope and purpose of the inspection at the entrance meeting at the start of the inspection. The findings of the inspection were discussed with licensee representatives during the course of the inspection and presented to licensee management at the October 30, 1990 exit interview (see Paragraph 1 for attendees).

At no time during the inspection, was written material provided to the licensee by the inspector. The licensee did not indicate that proprietary information was involved within the scope of this inspection.

February 14, 1991

Docket No. 50-219

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Mr. John J. Barton, Director
Oyster Creek Nuclear Generating Station
P. O. Box 388
Forked River, New Jersey 08731

Dear Mr. Barton:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION ON OYSTER CREEK DRYWELL STRESS AND STABILITY ANALYSIS (TAC NO. 79166)

The staff has reviewed the GE reports Index No. 9-1 and 9-2, "An ASME Section VII Evaluation of the Oyster Creek Drywell Stress and Stability Analysis" and our comments and request for additional information are contained in the enclosure.

We request that the information be provided within 30 days of receipt of this letter. If you have any questions regarding this request, please contact me.

The requirements of this letter affect fewer than 10 respondents and therefore, are not subject to Office of Management and Budget review under P.L. 97-511.

Sincerely,

/ /

Alexander W. Dromerick, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc w/enclosure:
See next page

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NAME	:SNorris	:ADromerick:cn	:JFStolz	:	:
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Document Name: TAC NO. 79166

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Comments on
GE Reports Index No. 9-1 and 9-2
An ASME Section VIII Evaluation of the Oyster Creek
Drywell Stress and Stability Analysis

PART I - Stress Analysis

1. Page 2-3, first paragraph

Reference is made to Table 2-1 which shows the 95 percent confidence thickness values in the locally corroded areas of the drywell. The basis and method of calculating these projected thicknesses should be explained. Furthermore, the anticipated date for reaching these projected thicknesses should be specified.

2. Page 2-5, first paragraph

The last sentence states that "given a design which satisfies the general code intent, as the Oyster Creek drywell does as originally constructed, it is not a violation of Subsection NE requirements for the membrane stress to be between $1.0S_{mc}$ and $1.1S_{mc}$ over significant distances." Further justification for the licensee's position should be provided. Under what conditions would this become a code violation? In other words, at what point does the "local" region become a "general membrane" region? Has the opinion of the Code Committee been solicited regarding this matter? If reference is to be made to Code Case N-480, the specific portions of the Code Case as it applies to the Oyster Creek drywell situation should be fully explained.

3. Page 5-2, Section 5.4

This section states that "the membrane stresses for the degraded thickness condition were obtained by scaling upwards the calculated stresses for the nominal thickness case (Table 5-2) by the thickness ratio." It should also be explained how the primary membrane plus bending stresses shown in Table 5-3 were obtained. It appears that the combined stress was scaled upwards linearly by the thickness ratio. However, the bending portion of the stress should be scaled by the square of the thickness ratio. Also, the effect of stress concentrations due to the change of thickness should be addressed.

4. Appendix A, page 21, second paragraph

The last sentence states that "impact testing would not be required by the present code rules unless the LST (lowest metal service temperature) were less than 30°F, and the Oyster

Creek drywell material would not require impact testing." Earlier in this section it is stated that an LST of 30°F was used for the Oyster Creek design basis. Is the LST for the drywell monitored by any plant operating procedures or the Technical Specifications? Have studies and plant operating history demonstrated that the drywell shell temperature is not expected to be lower than 30°F for all loading conditions?

5. Appendix F, page 1, first paragraph

What is the basis for performing the sand sensitivity study with a nominal sand stiffness of 366 psi/inch and a sand stiffness of 80 percent of the nominal value? Were studies and/or tests performed to support these assumptions? Otherwise, the sensitivity study should be conducted further with lower stiffness values. The licensee's letter of December 5, 1990 indicates that structural calculations assuming the sand removed would be completed by December 31, 1990. The results of these studies should be provided to demonstrate the sensitivity of the stresses to the assumed sand stiffness.

PART 2 - Stability Analysis

6. Page 2-3, Section 2.3

This section states that the method described in Reference 2-5 was used to quantify the effect that the orthogonal tensile stress has on reducing the effect of imperfections on the buckling strength. The sensitivity of the results should be studied by using other methods which also address this effect.

7. Page 2-4, Section 2.4

This section states that Reference 2-6 was used to calculate the plasticity reduction factor for the meridional direction elastic buckling stress. Since this approach apparently has not been incorporated into Code Case N-284, the sensitivity of the results should be studied by using other methods which address this effect.

8. Page 3-3, second paragraph

For the stability analysis the stiffness for the sandbed was assumed to be 366 psi/inch and no sensitivity studies are reported. As described in Question 5, the results of the stability analysis with the sand removed should be provided.

9. **Page 3-6, Section 3.5.3**

The first sentence states that "the 2 psi external pressure load for the refueling case is applied to the external faces of all of the drywell and vent shell elements." Unless it can be demonstrated that this pressure actually is present at all times during normal operation and refueling, the effect on the buckling analysis results of assuming no external pressure for these two load cases should be reported. Furthermore, is it possible to have an external pressure greater than 2 psi on the drywell shell? If so, an enveloping pressure case should be considered in the analysis.

PART 3 - General

10. Justification for the use of ASME Section III, Subsection NE has been provided to evaluate the Oyter Creek Steel drywell, taking into consideration DESIGN, material's, fabrication inspection and testing with exception of the comments indicated above, the justification appears to be reasonable. Since the present-day quality assurance and quality control requirements for the design and construction of nuclear power were in the formative stage at the time when the Oyster Creek Plant was designed and constructed, indicate what quality assurance and quality control programs were implemented for the Oyster Creek drywell. Indicate if documentation of the programs is available.

11. In GPU's presentation to the staff in September, 1990, it was indicated that GPU would have an on-line thickness measurement capability in the critical areas of thickness measurement. GPU has a current commitment to make UT measurements at outages of opportunity. State clearly what on-line thickness measurement program GPU will have during the fuel cycle starting in early 1991.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

February 14, 1991

NRC INFORMATION NOTICE NO. 86-99, SUPPLEMENT 1: DEGRADATION OF STEEL CONTAINMENTS

Addressees:

All holders of operating licenses or construction permits for nuclear power reactors.

Purpose:

This supplement to Information Notice (IN) 86-99 is intended to alert addressees to additional information about a potential degradation problem regarding corrosion in steel containments. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this supplement to the information notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Discussion:

IN 86-99 was issued on December 8, 1986, in response to the discovery of significant corrosion on the external surface of the carbon steel drywell in the sand bed region of the Oyster Creek plant. This supplement updates the status of Oyster Creek containment corrosion and the licensee's mitigation program.

Since drywell corrosion was detected in 1986, the licensee instituted periodic wall thickness measurements by the ultrasonic testing (UT) technique to determine corrosion rates. The most severe corrosion was found in the sand bed region at a nominal elevation of 11'-3". The highest corrosion rate determined was 35.2±6.8 mils per year. To mitigate the corrosion in the sand bed region, water was drained from the sand bed and cathodic protection (CP) was installed in the bays with the greatest wall thinning in early 1989. Subsequent UT thickness measurements in these bays indicated that CP was ineffective. The licensee's consultants indicated that it would be necessary to flood the sand bed and to install CP in all the bays to make the CP system effective. The licensee decided that large amounts of water in the sand bed would be counterproductive.

In the spherical portion of the drywell above the sand bed region, the highest corrosion rate determined was 4.6±1.6 mils per year at a nominal elevation of 51'. In the cylindrical portion of the drywell above the spherical portion, where minor corrosion was discovered and was thought to have originated mostly during construction, no significant wall thinning was detected (at a nominal elevation of 87'). However, this is the region in which the nominal thickness of the wall has the least margin, thus requiring periodic monitoring of actual thickness.

The licensee has instituted a drywell program to arrest corrosion and to ensure containment integrity for the full licensed term of the plant. The licensee has taken action to investigate, identify, and correct leak paths into the drywell gap and plans to take more action to survey leakage and prevent it. The stainless steel liners in the refueling cavity and the equipment pool developed cracks along the perimeter of the liner plates where they were welded to embedded channels. For the refueling cavity, all potential leakage pathways have been thoroughly checked and liner cracks are sealed with adhesive stainless steel tape before a strippable coating is applied. Since the refueling cavity is flooded only during refueling, no leakage concerns exist at other times. At the end of an outage, the refueling cavity is drained, and the tape and strippable coating are removed. The licensee found leaks related to the equipment pool and stopped them with liner weld repairs. The equipment pool also will be protected with a strippable coating during flooded periods of operation.

The licensee believes that a thorough program has been established for managing leakage that could affect drywell integrity due to corrosion from moisture ingress into the drywell gap. Recent surveillance of the sand bed drains indicates that the sand bed is free of water. To further mitigate drywell corrosion, the licensee is considering removing the sand, insulation, gap filler material, and corrosion film and applying a protective coating to the exterior drywell surface. The licensee is proceeding with the analysis, engineering and planning to support removing the sand from the drywell sand bed region in the near future. Removal of the insulation and gap filler material from the drywell gap is being evaluated for future consideration.

The BWR Owners Group is surveying its members to determine whether other plants are experiencing water leakage into the drywell gap and possible corrosion of the exterior surfaces in the sand bed region as well as in the spherical and cylindrical parts of the drywell.

This supplement requires no specific action or written response. If you have any questions about the information in this notice, please contact one of the technical contacts listed below or the appropriate NRR project manager.

Charles E. Rossi

Charles E. Rossi, Director
Division of Operational Events Assessment
Office of Nuclear Reactor Regulation

Technical Contacts: Frank J. Witt, NRR
(301) 492-0767

C.P. Tan, NRR
(301) 492-3315

Attachment: List of Recently Issued NRC Information Notices

MAY 23 1991

MEMORANDUM FOR: John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects I/II

FROM: Goutam Bagchi, Chief
Structural and Geosciences Branch
Division of Engineering Technology

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION - REVIEW OF OYSTER CREEK CORRODED DRYWELL ANALYSIS

Plant Name: Oyster Creek Nuclear Power Plant
Licensee: GPU Nuclear Corporation
Request Status: Request for Additional Information
Tac No.: M79166

The staff of the Structural and Geosciences Branch has reviewed the licensee's responses to the staff's previous request for information (GPU March 20, 1991 Letter) and the information provided on the drywell analysis with the sand removed (GPU March 4, 1991 Letter). In order to complete our review, we find more information is required. The required information is contained in the enclosure. The review was performed by C. P. Tan of the Geosciences Section with the assistance of consultants from Brookhaven National Laboratory.

/S/

Goutam Bagchi, Chief
Structural and Geosciences Branch
Division of Engineering Technology

Enclosure: As stated

cc: J. E. Richardson
B. D. Liaw
A. Dromerick

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ENCLOSURE

ESGB
REQUEST FOR ADDITIONAL INFORMATION
ON
OYSTER CREEK CORRODED DRYWELL ANALYSES

1. Your response to question 2 states "For a vessel that originally complied with the code, increases beyond 1.0 Smc in localized areas of undefined size are acceptable." This statement is loose and you have applied it throughout the drywell as shown in tables 5-1b, 5-2a and 5-2b of GE Report Index No. 9-3. One may conclude from what you have stated and implemented that a corroded drywell has increased its structural capability. Your interpretation of section NE 3213.10 of the ASME Code is questionable. Without corrosion, you consider the drywell when subjected to internal pressure to be under general primary membrane stress (tensile) and with corrosion you consider it to be under local primary membrane stress (tensile). NE 3213.10 considers a membrane stress to be local primary if it is produced by pressure or other mechanical loading and associated with a primary or discontinuity effect, resulting in excessive distortion in the transfer of load to other portions of the structure. NE 3213.10 specifies the region to be considered local over which the membrane stress intensity exceeds 1.1 Smc. The code gives an example of the discrete regions of local primary membrane stress. We realize that there is no code limit for the extent of the region in which the membrane stress exceeds 1.0 Smc but is less than 1.1 Smc. Logical judgement is to be exercised in the interpretation, and the basis for your judgment should be clearly defined. Even if your interpretation of NE 3213.10 for application to the corroded Oyster Creek drywell is acceptable for localized areas, it should be demonstrated that the present and projected corroded condition of the Oyster Creek drywell falls within the boundaries established in accordance with NE. 3213.10. Unless and until the staff's concerns as indicated above are satisfactorily resolved, the staff has reservations on, your use of 1.1 Smc as indicated in GE Report Index No. 9.3. This means that the allowable stresses indicated in tables 5-1b, 5-2a and 5-2b should be based on 1.0 Smc for primary membrane.

2. The response to Question 3 does not fully address the question regarding possible stress concentrations resulting from the corroded condition of the drywell. This issue should be fully discussed, noting that at corrosion locations the change in the plate thickness is not likely to be tapered as assumed in your analyses.

3. In GE Report Index No. 9-3, Section 5.2.2, comparisons of circumferential and meridional stress magnitudes with the large and small displacement options should be provided from the sandbed region up to the knuckle region of the drywell. The amount of stress reduction obtained as a

result of the large displacement method appears to be too high for the small deflection calculated; the results of these calculations should be further investigated. Also show mathematically as in the case of beams and flat plates, that consideration of large deflection decreases the stress in the drywell shell which is in membrane tension under internal pressure for regions of the shell away from the discontinuity.

4. In GE Report Index No. 9-3, Tables 3-3 and 3-4 indicate the large concentrated loads considered in the analysis; however, these loads are uniformly distributed along the circumference of the pie slice finite element model at various elevations. Since the stresses in the corroded regions of the drywell are close to the allowables, what effect would a more refined treatment of these loads have on the stress evaluation? This question should be addressed for all drywell regions (i.e., cylinder, knuckle, upper sphere, middle sphere, lower sphere, and sandbed). The response should consider stresses directly under the load (if corrosion in this area is present), as well as the effect on the stress distribution at further distances from the load.

5. In GE Report Index No. 9-3, Section 3.2.3 indicates that the seismic loads are imposed on the pie slice model by applying forces at four elevations of the model and matching stresses at selected elevations with those from the axisymmetric model. How sensitive are the calculations to the location and number of elevations chosen to match the stresses? How well do the stresses compare at other elevations in the drywell?

6. In order to examine your analysis in more detail, the staff requests that you provide the ANSYS input file for both the axisymmetric and pie slice models. This information should be provided on a high density 5 1/4 in. floppy disc for an IBM PC.

September 3, 1991

Docket No. 50-219

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Mr. John J. Barton
Vice President and Director
Oyster Creek Nuclear Generating Station
Post Office Box 388
Forked River, New Jersey 08731

Dear Mr. Barton:

SUBJECT: OYSTER CREEK NUCLEAR GENERATING STATION - STAFF POSITION ON EVALUATION OF STRUCTURAL INTEGRITY OF A DEGRADED STEEL CONTAINMENT (TAC NO. 79166)

At a meeting held on July 24, 1991, the NRC staff advised GPU Nuclear Corporation (GPUN) that they would inform GPUN on the staff's position on the application of the ASME Code in the evaluation of degraded steel containments.

Enclosed is the staff's position regarding this matter. We request that you respond within 21 days of receipt of this letter indicating your intent to comply with our position.

The requirements of this letter affect fewer than 10 respondents, and therefore, are not subject to Office of Management and Budget review under P.L. 96-511.

Sincerely,

/s/

Alexander W. Dromerick, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
Staff Position

cc w/enclosure:
See next page

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NAME	:SNorris	:ADromerick:cn	:JFStolz	:	:
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ENCLOSURE

STAFF POSITION

ON

EVALUATION OF STRUCTURAL INTEGRITY OF A

DEGRADED STEEL CONTAINMENT

OYSTER CREEK NUCLEAR GENERATING STATION

ASME Section XI Subsections IWE-3519.3 and IWE-3122.4 state that a steel containment is acceptable if the thickness of the area of degradation discovered is reduced by not more than 10%. This is acceptable only on the basis of considering the area of degradation as a form of discontinuity as stipulated in ASME Section III Division I Subsection NE-3213.10. The area of degradation, where the stress intensity exceeds 1.1 Smc, is stipulated in NE-3213.10 in terms of the square root of the product of R and t as defined therein. The code requires such a discontinuity be localized. This is due to the fact that the load on a highly stressed and localized area will be transferred to the adjacent area. If the area of degradation is localized, the effect on the overall behavior of the containment will be minimal or negligible.

The code does not specify the limit of the extent of the support region in which the stress intensity varies from 1.0 Smc to 1.1 Smc. However, the limit can be determined from the analysis for load combinations with the internal pressure as the major load. On the basis of the above observation, the staff has established the following position:

1. The corroded or degraded area with a reduction in thickness of not more than 10% should be considered in accordance with NE-3213.10 as a discontinuity with the limits of its extent as prescribed therein.

2. For a corroded containment shell where the thicknesses of the corroded areas are obtained through UT measurements, the extent of each corroded area should be determined as accurately as practical.

3. Except in the support zone of the discontinuity where the stress intensity value may vary from 1.0 Smc to 1.1 Smc, the primary membrane stress should be in accordance with the stress intensity limits as stipulated in Table NE-3221-1, Summary of Stress Intensity Limits.

Docket File

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

November 19, 1991

Docket No. 50-219

Mr. John J. Barton, Vice President
and Director
GPU Nuclear Corporation
Oyster Creek Nuclear Generating Station
Post Office Box 388
Forked River, New Jersey 08731

Dear Mr. Barton:

SUBJECT: CLARIFICATION OF STAFF POSITION ON EVALUATION OF STRUCTURAL INTEGRITY OF A DEGRADED STEEL CONTAINMENT (TAC NO M79166)

References:
1. Letter to J. J. Barton from A. W. Dromerick providing the subject staff's position dated September 3, 1991.
2. Letter to NRC from GPU Nuclear Corporation providing the response to staff's position dated October 9, 1991.

In a letter of October 9, 1991 (Reference 2), GPU Nuclear Corporation (GPUN) provided responses to the staff position on the evaluation of the structural integrity of a degraded steel containment. It appears from the responses that GPUN differs with the staff's position, specifically on the application of ASME subsection NE-3213.10. Enclosed is the staff's review of GPUN's response. It clarifies the staff's position and requires GPUN to provide additional information to aid in a final resolution of staff's concerns.

We request that the information be provided within 30 days of receipt of this letter. If you have any questions regarding this request, please contact me.

DF01
111

The requirements of this letter affect fewer than 10 respondents, and therefore, are not subject to Office of Management review under P.L. 97-511.

Sincerely,

/s/

Alexander W. Dromerick, Sr. Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc: See next page

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REVIEW OF GPUN'S RESPONSE OF OCTOBER 9, 1991
RELATED TO THE
STAFF'S POSITION ON EVALUATION OF
DEGRADED STEEL CONTAINMENT
AT OYSTER CREEK

The staff has reviewed GPU Nuclear Corporation's (GPUN) response of October 9, 1991 to the staff's position on the evaluation of the structural integrity of a degraded steel containment. It is to be noted that this staff position is to be applied generically in the evaluation of steel containments which are degraded, not specifically to the Oyster Creek steel drywell. The staff's position is based on technical criteria that conform to the spirit and intent of ASME subsection NE-3213.10. NE is the design part of the ASME code and cannot be directly applied to the situation of inservice degradation without the exercise of engineering judgment. By considering the corroded area as equivalent to a discontinuity as indicated in NE-3212.10, great caution must be exercised. It should be understood that the discontinuity as created by corrosion is not the same as the "designed" discontinuity such as a change in shell thicknesses, the presence of a bracket or a penetration as envisioned in the code. The basic characteristic of the discontinuity due to corrosion is irregularity, e.g. variation in thickness and extent of corroded areas. In view of the above observation, the NE 3312.10 stipulation cannot be applied indiscriminately to a corroded steel containment. NE-3312.10 specifies the limit of the discontinuity region in which the stresses can be greater than 1.1 Smc. The code does not specify the outside limit of the region which is contiguous to and supports the discontinuity and in which the stresses vary from 1.1 Smc to 1.0 Smc. This should be expected because this outside limit varies with the configuration of the discontinuity and the loading. Therefore, the lack of specific stipulation in the code in this respect should be understood and should not be construed to allow the stress limit of 1.1 Smc to be applied universally throughout the containment shell. The staff position is not, in any way, more restrictive than the stipulation in the ASME Code.

The staff is well aware of the extensive examinations and analysis performed on the Oyster Creek drywell as reported by GPUN. GPUN has repeatedly claimed that the Oyster Creek drywell has been examined thoroughly and the condition of the drywell is fully understood with a 95% confidence level. On the basis of this claim, the staff has requested GPUN to determine the extent of each corroded area. The staff is not requesting any additional physical examination. However, on the basis of the information available, GPUN should present in a figure the known areas of corrosion with the critical stresses (general primary membrane stress or local primary membrane stress) identified. The purpose of such an action is to determine the behavior of the drywell especially at and around the corroded areas. By comparing the calculated stresses of the drywell shell at and around corroded areas with the code allowables the staff can reasonably determine the adequacy of the licensee's proposed actions.

APR 09 1992

MEMORANDUM FOR: John F. Stolz, Director
Project Directorate 1-4
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

FROM: Goutam Bagchi, Chief
Structural and Geosciences Branch
Division of Engineering Technology
Office of Nuclear Reactor Regulation

SUBJECT: EVALUATION REPORT ON STRUCTURAL INTEGRITY OF THE OYSTER CREEK DRYWELL

Plant Name: Oyster Creek Nuclear Generating Station
Applicant: GPU Nuclear Corporation
Docket No.: 50-219
Review Status: Complete
Tac No.: M79166

The Structural and Geosciences Branch (ESGB) has completed the review and evaluation of the stress analyses and stability analyses reports of the corroded drywell with and without the sand bed. Our evaluation report together with a SALP is contained in the enclosure. The licensee used the analyses to justify the removal of the sand from the sand bed region. Even though the staff, with the assistance of consultants from Brookhaven National Laboratory (BNL), concurred with licensee's conclusion that the drywell meets the ASME Section III Subsection NE requirements, it is essential that the licensee continue UT thickness measurements at refueling outages and at outages of opportunity for the life of the plant.

The review is performed by C. P. Tan of Geosciences Section of ESGB with the assistance of BNL.

/S/
Goutam Bagchi, Chief
Structural and Geosciences Branch
Division of Engineering Technology

Enclosure:
As stated

cc: J. E. Richardson
B. D. Liaw
A. Dromerick

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SAFETY EVALUATION REPORT
OYSTER CREEK NUCLEAR GENERATING STATION
DRYWELL STRUCTURAL INTEGRITY
STRUCTURAL AND GEOSCIENCES BRANCH

I. INTRODUCTION

In 1986 the steel drywell at Oyster Creek Nuclear Generating Station (OCNGS) was found to be extensively corroded in the area of the shell which is in contact with the sand cushion around the bottom of the drywell. Since then GPU Nuclear, the Licensee of OCNGS, has instituted a program of periodic inspection of the drywell shell sand cushion area through ultrasonic testing UT thickness measurements. The inspection has been extended to other areas of the drywell and some areas above the sand cushion have been found to be corroded also. From the UT thickness measurements, one can conclude that corrosion of the drywell shell in the sand cushion area is continuing. In an attempt to eliminate corrosion or reduce the corrosion rate, the licensee tried cathodic protection and found it to be of no avail. An examination of the results of consecutive UT measurements, confirmed that the corrosion is continuing. There is concern that the structural integrity of the drywell cannot be assured. Since the root cause of the corrosion in the sand cushion area is the presence of water in the sand, the licensee has considered sand removal to be an important element in its program to eliminate the corrosion threat to the drywell integrity.

In the program, the licensee first established the analysis criteria and then performed the analyses of the drywell for its structural adequacy with and without the presence of the sand. The licensee performed stress analyses and stability analyses for both with and without the sand cases and concluded the drywell with or without the sand to be in compliance with the criteria established for the reevaluation. It is to be noted that the original purpose of the sand cushion is to provide a smooth transition of stresses from the fixed portion to the free-standing portion of the steel drywell.

II. EVALUATION

The staff with the assistance of consultants from Brookhaven National Laboratory (BNL) has reviewed and evaluated the information (Refs. 1,2,3,4,5) provided by the licensee.

1. Re-Analysis Criteria

The drywell was originally designed and constructed to the requirements of ASME Section VIII code and applicable code cases, with a contract date of July 1, 1964. The section VIII code requirements for nuclear containment vessels at that time were less detailed than at any subsequent date. The evolution of the ASME Section III code for metal containments and its relation with ASME Section VIII code were reviewed and evaluated by Teledyne Engineering Services (TES). The evaluation criteria used are based on ASME Section III Subsection NE code through the 1977 summer addenda. The reason for the use of the code of this vintage is that it was used in the Mark I containment program to evaluate the steel torus for hydrodynamic loads and that the current ASME Section III Subsection NE Code is closely related to that version. The following are TES's findings relevant to Oyster Creek application:

a) The steel material for the drywell is A-212, grade B, Firebox Quality (Section VIII), but it is redesignated as SA-516 grade in Section III.

b) The relation between the allowable stress (S) in Section VIII and the stress intensity (Smc) in Section III for metal containment is 1.1S = Smc.

c) Categorization of stresses into general primary membrane, general bending and local primary membrane stresses and membrane plus bending stresses is adopted as in Subsection NE.

d) The effect of a locally stressed region on the containment shell is considered in accordance with NE-3213.10.

In addition to ASME Section III Subsection NE Code, the licensee has also invoked ASME Section XI IWE Code to demonstrate the adequacy of the Oyster Creek drywell. IWE-3519.3 and IWE-3122.4 state that it is acceptable if either the thickness of the base metal is reduced by no more than 10% of the normal plate thickness or the reduced thickness can be shown by analysis to satisfy the requirements of the design specification.

The staff has reviewed the licensee's adoption of ASME Section III Subsection NE and Section XI Subsection IWE in its evaluation of the structural adequacy of the corroded Oyster Creek drywell, and has found it to be generally reasonable and acceptable.

By adopting the Subsection NE criteria, the licensee has treated the corroded areas as discontinuities per NE-3213.10, which was originally meant for change in thicknesses, supports, and penetrations. These discontinuities are highly localized and should be designed so that their presence will have no effect on the overall behavior of the containment shell. NE-3213.10 defines clearly the level of stress intensity and the extent of the discontinuity to be considered localized. A stress intensity limit of 1.1 Smc is specified at the boundary of the region within which the membrane stress can be higher than 1.1 Smc. The region where the stress intensity varies from 1.1 Smc to 1.0 Smc is not defined in the code because of the fact that it varies with the loading. In view of this, the licensee rationalized that the 1.1 Smc can be applied beyond the region defined by NE-3213.10 for localized discontinuity without any restriction throughout the drywell. The staff disagreed with the licensee's interpretation of the code. The staff pointed out that for Oyster Creek drywell, stresses due to internal pressure should be used as the criterion to establish such a region. The interpretation of Section XI Subsections IWE-3519.3 and IWE-3122.4 can be made only in the same context. It is staff's position that the primary membrane stress limit of 1.1 Smc not be used indiscriminately throughout the drywell.

In order to use NE-3213.10 to consider the corroded area as a localized discontinuity, the extent of the reduction in thickness due to corrosion should be reasonably known. UT thickness measurements are highly localized; however, from the numerous measurements so far made on the Oyster Creek drywell, one can have a general idea of the overall corroded condition of the drywell shell and it is possible to judiciously apply the established re-analysis criteria.

2. Re-analyses

The re-analyses were made by General Electric Company for the licensee, one reanalysis considered the sand present and the other considered the drywell without the sand. Each re-analysis comprises a stress analysis and stability analysis. Two finite element models, one axisymmetric and another a 36° pie slice model were used for the stress analysis. The ANSYS computer program was used to perform the analyses. The axisymmetric model was used to determine the stresses for the seismic and the thermal gradient loads. The pie slice model was used for dead weight and pressure loads. The pie slice model includes the vent pipe and the reinforcing ring, and was also used for buckling analysis. The same models were used for the cases with and without sand, except that in the former, the stiffness of sand in contact with the steel shell was considered. The shell thickness in the sand region was assumed to be 0.700" for the with-sand case and to be 0.736" for the without-sand case. The 0.70" was, as claimed by the licensee, used for conservatism and the 0.736" is the projected thickness at the start of fuel cycle 14R. The same thicknesses of the shell above the sand region were used for both cases. For the with-sand case, an analysis of the drywell with the original nominal wall thicknesses was made to check the shell stresses with the allowable values established for the re-analyses.

The licensee used the same load combinations as specified in Oyster Creek's final design safety analysis report (FDSAR) for the re-analyses. The licensee made a comparison of the load combinations and corresponding allowable stress limits using the SRP section 3.8.2 and concluded they are comparable.

The results of the re-analyses indicated that the governing thicknesses are in the upper sphere and the cylinder where the calculated primary membrane stresses are respectively 20,360 psi and 19,850 psi vs. the allowable stress value of 19,300 psi. There is basically no difference, in the calculated stresses at these levels, between the with and without sand cases. This should be expected, because in a steel shell structure the local effect or the edge effect is damped in a very short distance. The stresses calculated exceed the allowable by 3% to 6%, and such exceedance is actually limited to the corroded area as obtained from UT measurements. However, in order to perform the axisymmetric analysis and analysis of the pie slice model, uniform thicknesses were assumed for each section of the drywell. Therefore, the calculated over-stresses may represent only stresses at the corroded areas and the stresses for areas beyond the corroded areas are less and would most likely be within the allowable as indicated in results of the analyses for nominal thicknesses. The diagram in Ref. 6 indicated such a condition. It is to be noted that the stresses for the corroded areas were obtained by multiplying the stresses for nominal thicknesses by the ratios between the corroded and nominal thicknesses.

The buckling analyses of the drywell were performed in accordance with ASME Code Case N-284. The analyses were done on the 36° pie slice model for both with-sand and without-sand cases. Except in the sand cushion area where a shell thickness of 0.7" for the with-sand case and a shell thickness of 0.736" for the without-sand case were used, nominal shell thicknesses were considered for other sections. The load combinations which are critical to buckling were identified as those involving refueling and post accident conditions. By applying a factor of safety of 2 and 1.67 for the load combinations involving refueling and the post-accident conditions respectively, the licensee established for both cases the allowable buckling stresses which are obtained after being modified by capacity and plasticity reduction factors. It is found that the without-sand, case for the post-accident condition is most limiting in terms of buckling with a margin of 14%. The staff and its BNL consultants concur with the licensee's conclusion that the Oyster Creek drywell has adequate margin against buckling with no sand support for an assumed sandbed region shell thickness of 0.736 inch.

A copy of BNL's technical evaluation report is attached to this SER.

III. CONCLUSION

With the assistance of consultants from BNL, the staff has reviewed and evaluated the responses to the staff's concerns and the detailed re-analyses of the drywell for the with-sand and without-sand cases. The reanalyses by the licensee indicated that the corroded drywell meets the requirements for containment vessels as contained in ASME Section III Subsection NE through summer 1977 addenda. This code was adopted in the Mark I containment program. The staff agrees with the licensee's justification of using the above mentioned code requirements with one exception, the use of 1.1 Smc throughout the drywell shell in the criteria for stress analyses. It is the staff's position that the primary membrane stress limit of 1.1 Smc not be used indiscriminately throughout the drywell. The staff accepted the licensee's reanalyses on the assumption that the corroded areas are highly localized as indicated by the licensee's UT measurements. The stresses obtained for the case of reduced thickness can only be interpreted to represent those in the corroded areas and their adjacent regions of the drywell shell. In view of these observations, it is essential that the licensee perform UT thickness measurements at refueling outages and at outages of opportunity for the life of the plant. The measurements should cover not only areas previously inspected but also areas which have never been inspected so as to confirm that the thicknesses of the corroded areas are as projected and the corroded areas are localized. Both of these assumptions are the bases of the reanalyses and the staff acceptance of the reanalysis results.

References:

1. "An ASME Section VIII Evaluation of the Oyster Creek Drywell Part 1, Stress Analysis", GE Report No. 9-1 DRF #00664 November 1990, prepared for GPUN (with sand).

2. "Justification for use of Section III, Subsection NE, Guidance in Evaluating the Oyster Creek Drywell" TR-7377-1, Teledyne Engineering Services, November 1990 (Appendix A to Reference 1).

3. "An ASME Section VIII evaluation of the Oyster Creek Drywell, Part 2, Stability Analysis", GE Report No. 9-2 DRF #00664, Rev. 0, & Rev. 1. November 1990, prepared for GPUN (with sand).

4. "An ASME Section VIII Evaluation of Oyster Creek Drywell for without sand case, Part I, stress analysis," GE Report No. 9-3 DRF #00664, Rev. 0, February 1991. Prepared for GPUN.

5. "An ASME Section VIII Evaluation of Osyter Creek Drywell, for without sand case, Part 2 Stability Analysis", GE Report No. 9-4, DRF #00664 Rev. 0, Rev. 1 November 1990, prepared for GPUN.

6. Diagram attached to a letter from J. C. Devine Jr. of GPUN to NRC dated January 17, 1992 (C321-92-2020, 5000-92-2094).

SALP INPUT

FACILITY NAME: Oyster Creek Nuclear Generating Station

LICENSEE: GPU Nuclear Corporation

SUMMARY OF REVIEW

Since the discovery of corrosion in the sand cushion area of the drywell, the licensee has performed UT thickness measurements at outage of opportunity and at refueling outages from the results of the UT measurements it can be concluded that corrosion is still continuing in view of this, the licensee has considered sand removal to be an important element in its program to eliminate the corrosion threat to the drywell integrity. Since removal of the sand may affect the behavior of the drywell, the licensee had General Electric performed stress and stability analyses of the drywell for both with and without sand conditions taking into consideration the reduction in thickness in the sand cushion region. The criteria for the re-analyses are based on ASME Section VI Code Subsection NE. The use of subsection NE was examined and justified by the licensee's consultant from Teledyne Engineering
Services. The staff with the assistance of consultants from Brookhaven National Laboratory reviewed the reanalyses and the criteria used and found them to be acceptable.

NARRATIVE DISCUSSION OF LICENSEE PERFORMANCE - FUNCTIONAL AREAS ENGINEERING/TECHNICAL SUPPORT

Since the discovery of the corrosion of the drywell, the licensee has been working diligently to monitor the state of the corrosion, to stop the source of leakage and to eliminate further aggravation. Even though in the review process differing opinion and disagreement with staff's position arose, the licensee has been co-operative and forthcoming in striving to resolve staff's concerns.

TECHNICAL EVALUATION REPORT

ON

STRUCTURAL ANALYSES OF THE CORRODED OYSTER CREEK STEEL DRYWELL

1. Introduction

An inspection of the steel drywell at the Oyster Creek Nuclear Generating Station in November 1986 revealed that some degradation due to corrosion had occurred in the sandbed region of the shell. Subsequent inspections also identified thickness degradations in the upper spherical and cylindrical sections of the drywell. The licensee, GPU Nuclear Corporation, has performed structural analyses to demonstrate the integrity of the drywell for projected corroded conditions that may exist at the start of the fourteenth refueling outage (14R). This outage is expected to start in October 1992. In an attempt to arrest the corrosion, the licensee plans to remove the sand from the sandbed region. Consequently, they have submitted structural analyses of the drywell both with and without sand for drywell wall thicknesses projected to exist at the start of 14R outage.

2. Summary of Licensee's Analyses

The analyses performed by the licensee utilized the drywell wall thicknesses summarized in Table 1.

Table 1
Drywell Wall Thicknesses

Drywell Region	As-Designed Thicknesses (in.)	Projected 95% Confidence 14R Thicknesses (in.)
Cylindrical Region	0.640	0.619
Knuckle	2.5625*	2.5625*
Upper Spherical Region	0.722	0.677
Middle Spherical Region	0.770	0.723
Lower Spherical Region Except Sand Bed Area	1.154	1.154
Sand Bed Region	1.154	0.736

*NOTE: Table 2-1 of both References 1 and 3 indicates that the knuckle thickness is 2.625". This appears to be a mistake since the knuckle thickness is shown to be 2-9/16" in Figure 1-1 of the same report.

The stress analysis for the "with sand" case is described in Reference 1. For this analysis the licensee utilized the as-designed thicknesses, except for the sandbed region where a thickness of 0.70" was used. The stress results were obtained from a finite element analysis which utilized axisymmetric solid elements and the ANSYS computer program. Later, the stress results were scaled to address the local thinning in areas other than the sandbed region (the projected 95% confidence 14R thicknesses in Table 1). The loads and load combinations considered in the analysis are based on the FSAR Primary Containment Design Report and the 1964 Technical Specification for the Containment. Appendix E of Reference 1 compares the load combinations considered in the analysis with those given in Section 3.8.2 of the NRC Standard Review Plan, Rev. 1, July 1981.

The stress analysis for the "without sand" case is described in Reference 3. For this analysis the licensee also utilized the as-designed thicknesses, except for the sandbed region where a thickness of 0.736" was used. In this case, two finite element models, an axisymmetric and a 36° pie slice model, were used. The axisymmetric model is essentially the same as that used in Reference 1; however, the elements representing the sand stiffness were removed. This model was used to determine the seismic and thermal stresses. The pie slice model was used to determine the dead weight and pressure stresses, as well as the stresses for load combinations. The pie slice model included the effects of the vent pipes and the reinforcing ring in the drywell shell in the vicinity of each vent pipe. The drywell and vent shell were modeled using 3-dimensional elastic-plastic quadrilateral shell elements. At a distance of 76 inches from the drywell shell, beam elements were used to model the remainder of the ventline. The loads and load combinations are the same as those considered in Reference 1.

The code of record for the Oyster Creek drywell is the 1962 Edition of the ASME Code, Section VIII with Addenda to Winter 1963, and Code Cases 1270N-5, 1271N and 1272N-5. The licensee utilized these criteria in evaluating the stresses in the drywell, but also utilized guidance from the NRC Standard Review Plan with regard to allowable stresses for service level C and the post-accident condition. The licensee also used guidance from Subsection NE of Section III of the ASME Code in order to justify the use of a limit of $1.1S_{mc}$ in evaluating the general membrane stresses in areas of the drywell where reduced thicknesses are specified. Based on these criteria the licensee has concluded that the stresses in the drywell shell are within code allowable limits for both the "with sand" and "without sand" cases.

The licensee also performed stability analyses of the drywell for both the "with sand" case (Reference 2) and the "without sand" case (Reference 4). For the "with sand" case the licensee utilized the as-designed thicknesses shown in Table 1, except in the sandbed region where a thickness of 0.700 inch was used. For the "without sand" case the same thicknesses were used , except in the sandbed region where a thickness of 0.736 inch was used. The buckling capability of the drywell for both the "with sand" and "without sand" cases was evaluated by using the 36° pie slice finite element model discussed above. For the "with sand" case spring elements were used in the sandbed region to model the sand support. For the "without sand" case these spring elements were removed. The most limiting load combinations which result in the highest compressive stresses in the sandbed region were considered for the buckling analysis. These are the refueling condition (Dead Weight + Live Load + Refueling Water Weight + External Pressure + Seismic) and the post-accident condition (Dead Weight + Live Load + Hydrostatic Pressure for Flooded Drywell + External Pressure + Seismic).

The buckling evaluations performed by the licensee follow the methodology described in ASME Code Case N-284, "Metal Containment Shell Buckling Design Methods, Section III, Class MC", Approved August 25, 1980. The theoretical elastic buckling stress is calculated by analyzing the three dimensional finite element model discussed above. Then the theoretical buckling stress is modified by capacity and plasticity reduction factors. The allowable compressive stress is obtained by dividing the calculated buckling stress by a factor of safety. In accordance with Code Case N-284 the licensee used a factor of safety of 2.0 for the refueling condition and 1.67 for the post-accident condition. The capacity reduction factors were also modified to take into account the effects of hoop stress. Originally the licensee based the hoop stress modification on data related to the axial compressive strength of cylinders (References 2 and 4). Later the licensee revised the approach based on a review of spherical shell buckling data and recalculated the drywell buckling capacities for both the "with sand" and "without sand" cases (Reference 8). For the "with sand" case, the licensee reports a margin above the allowable compressive stress of 47% for the refueling condition and 40% for the post-accident condition. For the "without sand" case, the licensee reports margins of 24.5% for the refueling condition and 14% for the post-accident condition.

3. Evaluation of Licensee's Approach

The analyses performed by the licensee as summarized in Section 2 and discussed more fully in References 1 through 4 have been reviewed and found to provide an acceptable approach for demonstrating the structural integrity of the corroded Oyster Creek drywell. The finite element analyses performed for both the stress and stability evaluations are consistent with industry practice. Except for the use of a limit of $1.1S_{mc}$ in evaluating the general membrane stress in areas of reduced drywell thickness, the loads, load combinations and acceptance criteria used by the licensee are consistent with the guidance given in Section 3.8.2 of the NRC Standard Review Plan, Rev. 1, July 1981. To further support their position, the licensee has provided two appendices to Reference 1.

Appendix A provides a detailed justification for the use of Section III, Subsection NE as guidance in evaluating the Oyster Creek drywell. Appendix E compares the load combinations given in the Final Design Safety Analysis Report (FDSAR) with the load combinations given in SRP 3.8.2 and demonstrates that the load combinations used in the analysis envelop those given in the SRP.

In the areas of the drywell where reduced thicknesses are specified, the licensee has used a limit of $1.1S_{mc}$ to evaluate the general membrane stresses. In support of this position the licensee has cited the provisions of NE-3213.1 of the ASME Code concerning local primary membrane stresses. In effect, the licensee's criteria would treat corroded or degraded areas as discontinuities. For such considerations the code places no limit on the extent of the region in which the membrane stress exceeds $1.0S_{mc}$ but is less than $1.1S_{mc}$. In support of this position the licensee has provided the opinion of Dr. W.E. Cooper, a well known expert on the development of the ASME Code. Dr. Cooper concluded that "given a design which satisfies the general Code intent, as the Oyster Creek drywell does as originally constructed, it is not a violation of Subsection NE requirements for the membrane stress to be between $1.0S_{mc}$ and $1.1S_{mc}$ over significant distances". The licensee has also cited the provisions of IWE-3519.3 which accepts up to a 10% reduction in the thickness of the original base metal.

The licensee's position has merit, but great caution must be exercised to assure that such a position is not applied indiscriminately. In the case of the Oyster Creek drywell the licensee has concluded that "there are very few locations where the calculated stress intensities for design basis conditions, would exceed $1.0S_{mc}$, and in these cases only slightly" (Reference 7). The licensee has provided additional information in Reference 9 to support this conclusion. Based on the information provided by the licensee which demonstrates that the use of the $1.1S_{mc}$ criteria is limited to localized areas, it is concluded that the Oyster Creek drywell meets the intent of the ASME Code.

As discussed in Section 2, the capacity reduction factors used in the buckling analysis are modified to take into account the beneficial effects of tensile hoop stress. As a result of a question raised during the review regarding this matter, the licensee submitted additional information in Reference 5 to support the approach. This information included a report prepared by C.D. Miller entitled "Effects of Internal Pressure on Axial Compression Strength of Cylinders" (CBI Technical Report No. 022891, February 1991). The report presented a design equation which was the lower bound of the test data included in the report. It also demonstrated that the equation used in References 2 and 4 was conservative relative to the proposed design equation. The report presented further arguments that the rules determined for axially compressed cylinders subjected to internal pressure can be applied to spheres. Subsequently the licensee has submitted Reference 8, which indicates that the original approach was not conservative with regard to its application to spherical shapes and recommends a new equation. However, the documentation supporting the use of this equation is not included in Reference 8, but apparently is contained in a referenced report prepared by C.D. Miller entitled "Evaluation of Stability Analysis Methods Used for the Oyster Creek Drywell" (CBI Technical Report Prepared for GPU Nuclear Corporation, September 1991). This report was subsequently submitted and reviewed by the NRC staff. As discussed in Section 2, the use of the revised equation still results in calculated capacities in compliance with the ASME Code provisions; however, the margins beyond those capacities are reduced from those reported by References 2 and 4.

It is noted that the licensee may have "double-counted" the effects of hoop tension, since the theoretical elastic instability stress was calculated from the finite element model using the ANSYS Code. The elastic instability stress calculated by the ANSYS Code may have already taken into account the effects of hoop tensile stress. However, by comparing the theoretical elastic instability stress and the corresponding circumferential stress predicted by the licensee for the refueling and post-accident cases, it appears that the effect of hoop tension in the ANSYS calculations is small and there is sufficient margin in the results to compensate for the potential "double-counting". Furthermore, it is judged that there is sufficient capacity in the drywell to preclude a significant buckling failure under the postulated loading conditions since the licensee's calculations: (a) incorporate factors of safety of 1.67 to 2.0, depending upon the load condition, and (b) utilize a conservative assumption by considering the shell wall thickness to be severely reduced for the full circumference of the drywell throughout the sandbed region.

During the course of the review of the licensee's submittals, a number of other issues were raised regarding the approach. These included: (a) the basis and method of calculating the projected drywell thicknesses, (b) the scaling of the calculated stresses for the nominal thickness case by the thickness ratio, (c) the effect of stress concentrations due to the change of thickness, (d) monitoring of the drywell temperature, (e) sensitivity of stresses due to variations in the sand spring stiffness, (f) sensitivity of the plasticity reduction factor in the buckling analysis, (g) use of the 2 psi design basis external pressure in the buckling analysis, (h) effect of the large displacement method, (i) the treatment of the large concentrated loads considered in the analysis, and (j) the method of applying the seismic loads to the pie slice model. These issues were adequately addressed by the additional information provided by the licensee in References 5 and 6.

4. Conclusions

The licensee has demonstrated that the calculated stresses in the Oyster Creek drywell (both with and without the sandbed), as a result of the postulated loading conditions, meet the intent of the ASME Code for projected corroded conditions that may exist at the start of the fourteenth refueling outage. However, if the actual thickness in the sandbed region at 14R is close to the projected thickness of 0.736", there may not be adequate margin left for further corrosion through continued operation unless it is demonstrated that removal of sand will completely stop further thickness reductions. The licensee has also demonstrated that there is sufficient margin in the drywell design (both with and without the sandbed) to preclude a buckling failure under the postulated loading conditions.

It should be recognized that the conclusions reached by the licensee have been accepted for this particular application with due regard to all the assumptions made in the analysis and the available margins. The use of the $1.1S_{mc}$ criteria for evaluating general membrane stress in corroded or degraded areas should be investigated further by the NRC staff and the ASME Code Committee and appropriate bounds established before it is accepted for general use. The licensee's buckling criteria regarding the modification of capacity reduction factors for tensile hoop stress and the determination of plasticity reduction factors should also be investigated in a similar manner.

5. References

1. GE Report Index No. 9-1, "An ASME Section VIII Evaluation of the Oyster Creek Drywell - Part 1 - Stress Analysis", November 1990.

2. GE Report Index No. 9-2, "An ASME Section VIII Evaluation of the Oyster Creek Drywell - Part 2 - Stability Analysis," November 1990.

3. GE Report Index No. 9-3, "An ASME Section VIII Evaluation of the Oyster Creek Drywell for Without Sand Case - Part 1 - Stress Analysis," February 1991.

4. GE Report Index No. 9-4, "An ASME Section VIII Evaluation of the Oyster Creek Drywell for Without Sand Case - Part 2 - Stability Analysis," February 1991.

5. GPU Nuclear letter dated March 20, 1991, "Oyster Creek Drywell Containment."

6. GPU Nuclear letter dated June 20, 1991, "Oyster Creek Drywell Containment".

7. GPU Nuclear letter dated October 9, 1991, "Oyster Creek Drywell Containment"

8. GPU Nuclear letter dated January 16, 1992, "Oyster Creek Drywell Containment".

9. GPU Nuclear letter dated January 17, 1992, "Oyster Creek Drywell Containment".

April 24, 1992

Docket No. 50-219

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Mr. John J. Barton
Vice President and Director
GPU Nuclear Corporation
Oyster Creek Nuclear Generating Station
Post Office Box 388
Forked River, New Jersey 08731

Dear Mr. Barton:

SUBJECT: EVALUATION REPORT ON STRUCTURAL INTEGRITY OF THE OYSTER CREEK DRYWELL (TAC NO. M79166)

The staff has completed the review and evaluation of the stress analyses and stability analyses reports of the corroded drywell with and without the sand bed. Our evaluation report is contained in the enclosure. GPUN used the analyses to justify the removal of the sand from the sand bed region. Even though the staff, with the assistance of consultants from Brookhaven National Laboratory (BNL), concurred with GPUN's conclusion that the drywell meets the ASME Section III Subsection NE requirements, it is essential that GPUN continue UT thickness measurements at refueling outages and at outages of opportunity for the life of the plant. The measurements should cover not only areas previously inspected but also accessible areas which have never been inspected so as to confirm that the thickness of the corroded areas are as projected and the corroded areas are localized.

We request that you respond within 30 days of receipt of this letter indicating your intent to comply with the above requirements as discussed in the Safety Evaluation.

The requirements of this letter affect fewer than 10 respondents, and therefore, are not subject to Office of Management and Budget review under P.L. 96-511.

Sincerely,

/s/

Alexander W. Dromerick, Sr. Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc w/enclosure:
See next page

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Document Name: M79166

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NAME	:SNorris	:ADromerick:cn	:JStolz	:	:	:
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

DRYWELL STRUCTURAL INTEGRITY

OYSTER CREEK NUCLEAR GENERATING STATION

GPU NUCLEAR CORPORATION

DOCKET NO. 50-219

I. INTRODUCTION

In 1986 the steel drywell at Oyster Creek Nuclear Generating Station (OCNGS) was found to be extensively corroded in the area of the shell which is in contact with the sand cushion around the bottom of the drywell. Since then GPU Nuclear Corporation, (GPUN, the licensee of OCNGS), has instituted a program of periodic inspection of the drywell shell sand cushion area through ultrasonic testing (UT) thickness measurements. The inspection has been extended to other areas of the drywell and some areas above the sand cushion have been found to be corroded also. From the UT thickness measurements, one can conclude that corrosion of the drywell shell in the sand cushion area is continuing. In an attempt to eliminate corrosion or reduce the corrosion rate, the licensee tried cathodic protection and found it to be of no avail. An examination of the results of consecutive UT measurements, confirmed that the corrosion is continuing. There is concern that the structural integrity of the drywell cannot be assured. Since the root cause of the corrosion in the sand cushion area is the presence of water in the sand, the licensee has considered sand removal to be an important element in its program to eliminate the corrosion threat to the drywell integrity.

In the program, the licensee first established the analysis criteria and then performed the analyses of the drywell for its structural adequacy with and without the presence of the sand. The licensee performed stress analyses and stability analyses for both with and without the sand cases and concluded the drywell with or without the sand to be in compliance with the criteria established for the reevaluation. It is to be noted that the original purpose of the sand cushion is to provide a smooth transition of stresses from the fixed portion to the free-standing portion of the steel drywell.

II. EVALUATION

The staff with the assistance of consultants from Brookhaven National Laboratory (BNL) has reviewed and evaluated the information (Refs. 1,2,3,4,5) provided by the licensee.

1. Re-Analysis Criteria

The drywell was originally designed and constructed to the requirements of ASME Section VIII code and applicable code cases, with a contract date of July 1, 1964. The Section VIII Code requirements for nuclear containment vessels at that time were less detailed than at any subsequent date. The evolution of the ASME Section III Code for metal containments and its relation with ASME Section VIII Code were reviewed and evaluated by Teledyne Engineering Services (TES). The evaluation criteria used are based on ASME Section III Subsection NE Code through the 1977 summer addenda. The reason for the use of the Code of this vintage is that it was used in the Mark I containment program to evaluate the steel torus for hydrodynamic loads and that the current ASME Section III Subsection NE Code is closely related to that version. The following are TES's findings relevant to Oyster Creek application:

a) The steel material for the drywell is A-212, grade B, Firebox Quality (Section VIII), but it is redesignated as SA-516 grade in Section III.

b) The relation between the allowable stress (S) in Section VIII and the stress intensity (Smc) in Section III for metal containment is 1.1S = Smc.

c) Categorization of stresses into general primary membrane, general bending and local primary membrane stresses and membrane plus bending stresses is adopted as in Subsection NF.

d) The effect of a locally stressed region on the containment shell is considered in accordance with NE-3213.10.

In addition to ASME Section III Subsection NE Code, the licensee has also invoked ASME Section XI IWE Code to demonstrate the adequacy of the Oyster Creek drywell. IWE-3519.3 and IWE-3122.4 state that it is acceptable if either the thickness of the base metal is reduced by no more than 10% of the normal plate thickness or the reduced thickness can be shown by analysis to satisfy the requirements of the design specification.

The staff has reviewed the licensee's adoption of ASME Section III Subsection NE and Section XI Subsection IWE in its evaluation of the structural adequacy of the corroded Oyster Creek drywell, and has found it to be generally reasonable and acceptable.

By adopting the Subsection NE criteria, the licensee has treated the corroded areas as discontinuities per NE-3213.10, which was originally meant for change in thicknesses, supports, and penetrations. These discontinuities are highly localized and should be designed so that their presence will have no effect on the overall behavior of the containment shell. NE-3213.10 defines clearly the level of stress intensity and the extent of the discontinuity to be considered localized. A stress intensity limit of 1.1 Smc is specified at the boundary of the region within which the membrane stress can be higher than 1.1 Smc. The region where the stress intensity varies from 1.1 Smc to 1.0 Smc is not defined in the Code because of the fact that it varies with the loading. In view of this, the licensee rationalized that the 1.1 Smc can be applied beyond the region defined by NE-3213.10 for localized discontinuity without any restriction throughout the drywell. The staff disagreed with the licensee's interpretation of the Code. The staff pointed out that for Oyster Creek drywell, stresses due to internal pressure should be used as the criterion to establish such a region. The interpretation of Section XI Subsections IWE-3519.3 and IWE-3122.4 can be made only in the same context. It is staff's position that the primary membrane stress limit of 1.1 Smc not be used indiscriminately throughout the drywell.

In order to use NE-3213.10 to consider the corroded area as a localized discontinuity, the extent of the reduction in thickness due to corrosion should be reasonably known. UT thickness measurements are highly localized; however, from the numerous measurements so far made on the Oyster Creek drywell, one can have a general idea of the overall corroded condition of the drywell shell and it is possible to judiciously apply the established re-analysis criteria.

2. Re-analyses

The re-analyses were made by General Electric Company for the licensee, one reanalysis considered the sand present and the other considered the drywell without the sand. Each re-analysis comprises a stress analysis and stability analysis. Two finite element models, one axisymmetric and another a 36° pie slice model were used for the stress analysis. The ANSYS computer program was used to perform the analyses. The axisymmetric model was used to determine the stresses for the seismic and the thermal gradient loads. The pie slice model was used for dead weight and pressure loads. The pie slice model includes the vent pipe and the reinforcing ring, and was also used for buckling analysis. The same models were used for the cases with and without sand, except that in the former, the stiffness of sand in contact with the steel shell was considered. The shell thickness in the sand region was assumed to be 0.700" for the with-sand case and to be 0.736" for the without-sand case. The 0.70" was, as claimed by the licensee, used for conservatism and the 0.736" is the projected thickness at the start of fuel cycle 14R. The same thicknesses of the shell above the sand region were used for both cases. For the with-sand case, an analysis of the drywell with the original nominal wall thicknesses was made to check the shell stresses with the allowable values established for the re-analyses.

The licensee used the same load combinations as specified in Oyster Creek's final design safety analysis report (FDSAR) for the re-analyses. The licensee made a comparison of the load combinations and corresponding allowable stress limits using the Standard Review Plan (SRP) section 3.8.2 and concluded they are comparable.

The results of the re-analyses indicated that the governing thicknesses are in the upper sphere and the cylinder where the calculated primary membrane stresses are respectively 20,360 psi and 19,850 psi vs. the allowable stress value of 19,300 psi. There is basically no difference, in the calculated stresses at these levels, between the with and without sand cases. This should be expected, because in a steel shell structure the local effect or the edge effect is damped in a very short distance. The stresses calculated exceed the allowable by 3% to 6%, and such exceedance is actually limited to the corroded area as obtained from UT measurements. However, in order to perform the axisymmetric analysis and analysis of the pie slice model, uniform thicknesses were assumed for each section of the drywell. Therefore, the calculated over-stresses may represent only stresses at the corroded areas and the stresses for areas beyond the corroded areas are less and would most likely be within the allowable as indicated in results of the analyses for nominal thicknesses. The diagram in Ref. 6 indicated such a condition. It is to be noted that the stresses for the corroded areas were obtained by multiplying the stresses for nominal thicknesses by the ratios between the corroded and nominal thicknesses.

The buckling analyses of the drywell were performed in accordance with ASME Code Case N-284. The analyses were done on the 36° pie slice model for both with-sand and without-sand cases. Except in the sand cushion area where a shell thickness of 0.7" for the with-sand case and a shell thickness of 0.736" for the without-sand case were used, nominal shell thicknesses were considered for other sections. The load combinations which are critical to buckling were identified as those involving refueling and post accident conditions. By applying a factor of safety of 2 and 1.67 for the load combinations involving refueling and the post-accident conditions respectively, the licensee established for both cases the allowable buckling stresses which are obtained after being modified by capacity and plasticity reduction factors. It is found that the without-sand, case for the post-accident condition is most limiting in terms of buckling with a margin of 14%. The staff and its Brookhaven National Laboratory (BNL) consultants concur with the licensee's conclusion that the Oyster Creek drywell has adequate margin against buckling with no sand support for an assumed sandbed region shell thickness of 0.736 inch.

A copy of BNL's technical evaluation report is attached to this safety evaluation.

III. CONCLUSION

With the assistance of consultants from BNL, the staff has reviewed and evaluated the responses to the staff's concerns and the detailed re-analyses of the drywell for the with-sand and without-sand cases. The reanalyses by the licensee indicated that the corroded drywell meets the requirements for containment vessels as contained in ASME Section III Subsection NE through summer 1977 addenda. This Code was adopted in the Mark I containment program. The staff agrees with the licensee's justification of using the above mentioned Code requirements with one exception, the use of 1.1 Smc throughout the drywell shell in the criteria for stress analyses. It is the staff's position that the primary membrane stress limit of 1.1 Smc not be used indiscriminately throughout the drywell. The staff accepted the licensee's reanalyses on the assumption that the corroded areas are highly localized as indicated by the licensee's UT measurements. The stresses obtained for the case of reduced thickness can only be interpreted to represent those in the corroded areas and their adjacent regions of the drywell shell. In view of these observations, it is essential that the licensee perform UT thickness measurements at refueling outages and at outages of opportunity for the life of the plant. The measurements should cover not only areas previously inspected but also accessible areas which have never been inspected so as to confirm that the thicknesses of the corroded areas are as projected and the corroded areas are localized. Both of these assumptions are the bases of the reanalyses and the staff acceptance of the reanalysis results.

References:

1. "An ASME Section VIII Evaluation of the Oyster Creek Drywell Part I, Stress Analysis" GE Report No. 9-1 DRF #00664 November 1990, prepared for GPUN (with sand).

2. "Justification for use of Section III, Subsection NE, Guidance in Evaluating the Oyster Creek Drywell" TR-7377-1, Teledyne Engineering Services, November 1990 (Appendix A to Reference 1).

3. "An ASME Section VIII evaluation of the Oyster Creek Drywell, Part 2, Stability Analysis" GE Report No. 9-2 DRF #00664, Rev. 0, & Rev. 1. November 1990, prepared for GPUN (with sand).

4. "An ASME Section VIII Evaluation of Oyster Creek Drywell for <u>without sand case</u>, Part I, stress analysis" GE Report No. 9-3 DRF #00664, Rev. 0, February 1991. Prepared for GPUN.

5. "An ASME Section VIII Evaluation of Oyster Creek Drywell, for <u>without sand case</u>, Part 2 Stability Analysis" GE Report No. 9-4, DRF #00664 Rev. 0, Rev. 1 November 1990, prepared for GPUN.

6. Diagram attached to a letter from J. C. Devine Jr. of GPUN to NRC dated January 17, 1992 (C321-92-2020, 5000-92-2094).

Principal Contributor: C.P. Tan

Date: April 24, 1992

Attachment:
BNL Technical Evaluation
Report

GPU Nuclear

GPU Nuclear Corporation
One Upper Pond Road
Parsippany, New Jersey 07054
201-316-7000
TELEX 136-482
Writer's Direct Dial Number:

May 26, 1992
5000-92-3026
C321-92-2163

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Gentlemen:

Subject: Oyster Creek Nuclear Generating Station (OCNGS)
Docket No. 50-219
Facility Operating License No. DPR-16
Oyster Creek Drywell Containment

References: (1) NRC Letter dated April 24, 1992, "Evaluation Report on Structural Integrity of the Oyster Creek Drywell (TAC No. M79166)."

(2) GPUN Letter C320-92-264 dated November 26, 1990, "Oyster Creek Drywell Containment."

In response to the Reference 1 request, GPU Nuclear commits to continue taking UT drywell measurements at refueling outages and at other outages of opportunity. The measurements will be at areas previously inspected and also at other accessible areas not previously inspected. Drywell thickness measurements will continue for the life of the plant.

The following is our current plan for Oyster Creek drywell UT thickness measurements.

(1) During the 14R outage, GPU Nuclear will take UT thickness measurements in the drywell sandbed region, from the torus room side (outside the drywell), at shell locations not readily accessible from inside the drywell. These are areas not previously inspected. The specific locations selected for inspection will be identified once we have direct access to the sandbed region.

Assuming that these measurements confirm that we have bounded the corrosion problem with our current inspection locations, we currently do not plan to make repeat measurements at these specific locations.

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(2) Now through the 15R outage, GPU Nuclear will continue taking UT thickness measurements in accordance with the priority method described in Reference 2, Attachment I, "GPUN Specification IS-328227-004, Functional Requirements for Drywell Containment Vessel Thickness Examination".

(3) After the 15R outage, GPU Nuclear will assess the condition of the drywell by evaluating the then current UT thickness measurements and will formulate an extended inspection plan. The plan will identify measurement locations including frequency of inspection for the remaining life of the plant.

If you have any questions or comments on this submittal or the overall drywell corrosion program, please contact Mr. Michael Laggart, Manager, Corporate Nuclear Licensing at (201) 316-7968.

Very truly yours,

G. R. Bond

for J. C. DeVine, Jr.
Vice President and Director
Technical Functions

JCD/RZ/amk

cc: Administrator, Region 1
Senior Resident Inspector
Oyster Creek NRC Project Manager

JUN 0 2 1992

Docket No. 50-219

Mr. John J. Barton
Vice President and Director
GPU Nuclear Corporation
Oyster Creek Nuclear Generating Station
P.O. Box 388
Forked River, New Jersey 08731

Dear Mr. Barton:

SUBJECT: NRC INSPECTION REPORT NO. 50-219/92-08

This letter transmits the report of the resident safety inspection conducted by Mr. D. Vito for the period March 29, 1992, through May 2, 1992, at the Oyster Creek Nuclear Generating Station. The inspection consisted of document reviews, personnel interviews and observations of activities. Inspectors discussed the findings with Mr. D. Ranft, Plant Engineering Director, and members of your staff after the inspection.

Inspector observations during this report period indicate that activities conducted were safe and conservative. However, we are concerned about the inadvertent actuation of the containment spray system and the spray of the drywell with approximately 825 gallons of water. This event was caused by a licensed operator's failure to follow a containment spray system surveillance procedure and is a violation of NRC requirements as specified in the enclosed Notice of Violation. You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be placed in the NRC Public Document Room.

The responses directed by this letter and the enclosed Notice are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, Public Law No. 96.511.

IE01

We appreciate your cooperation.

Sincerely,
Original Signed By:

A. Randolph Blough, Chief
Projects Branch No. 4
Division of Reactor Projects

Enclosures:
1. Notice of Violation
2. NRC Report No. 50-219/92-08

cc w/encls:
M. Laggart, Manager, Corporate Licensing
G. Busch, Licensing Manager, Oyster Creek
K. Abraham, PAO, (2)
Public Document Room (PDR)
Local Public Document Room (LPDR)
Nuclear Safety Information Center (NSIC)
NRC Resident Inspector
State of New Jersey

ENCLOSURE 1

NOTICE OF VIOLATION

GPU Nuclear Corporation
Oyster Creek Nuclear Generating Station

Docket No. 50-219
License No. DPR-16

During an NRC inspection conducted March 29, 1992, through May 2, 1992, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2, Appendix C, (1992), the violation is listed below:

> Technical Specification 6.8.1 requires that written procedures shall be established, implemented, and maintained that meet or exceed the requirements of Regulatory Guide (Reg Guide) 1.33, revision 2, "Quality Assurance Program Requirements (Operation)." Reg Guide 1.33, Appendix A requires that procedures be written for surveillance testing of the containment spray system.
>
> Station procedure 604.4.007, revision 13, "Containment Spray and Emergency Service Water System 1 Pump Operability and Inservice Test," step 6.20, requires the containment spray and emergency service water (ESW) pumps to be secured if inservice testing (IST) is not required to be performed.
>
> Contrary to the above, on April 20, 1992, the control room operator failed to implement procedure 604.4.007 in that the containment spray and ESW pumps were not secured when performance of IST was not required before proceeding to the next step in the procedure. As a result of this action the system was aligned to spray the containment when the operator placed the system control switch in the AUTO 1 position and approximately 825 gallons of water were sprayed into the containment.

This is a Severity Level IV violation (Supplement I).

Pursuant to the provisions of 10 CFR 2.201, GPU Nuclear Corporation is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Regional Administrator, Region I, and a copy to the NRC Resident Inspector, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Where good cause is shown consideration will be given to extending the response time.

Dated at: King of Prussia, PA
this 2d day of Jun 1992

U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No.	92-08
Docket No.	50-219
License No.	DPR-16
Licensee:	GPU Nuclear Corporation 1 Upper Pond Road Parsippany, New Jersey 07054
Facility Name:	Oyster Creek Nuclear Generating Station
Inspection Period:	March 29, 1992 - May 2, 1992
Inspectors:	David Vito, Senior Resident Inspector John Nakoski, Resident Inspector

Approved By: John Rogge, Section Chief, Reactor Projects Section 4B — Date

Inspection Summary: This inspection report documents the safety inspections conducted during day shift and backshift hours of station activities including: plant operations; radiation protection; maintenance and surveillance; engineering and technical support; security; and safety assessment/quality verification.

Results: Overall, GPUN operated the facility in a safe manner. A violation was identified as the result of an operator error which caused the inadvertent spray of the drywell with approximately 825 gallons of water from containment spray system 1. This operator error was contrary to the associated containment spray system surveillance procedure.

Two starting failures on emergency diesel generator (EDG) No. 2 were determined to be the result of a broken prop spring on the EDG output breaker. The licensee's apparent lack of corrective action in response to generic correspondence related to this failure mechanism was addressed as part of a separate inspection of the preventive maintenance area (see Inspection Report 50-219/92-07).

Initially, the March 23, 1992, EDG No. 2 start failure was not considered to be reportable because it was believed that the problem was within the automatic synchronization portion of the EDG testing circuitry and did not affect the fast starting capability of the diesel. The results of the root cause assessment performed after the April 5 failure found that this was not the most probable cause. The most probable cause (the prop spring failure) would have affected diesel fast start capability.

The failure mode of the prop spring was originally identified in NRC Information Notice 90-41, "Potential Failure of General Electric Magne-Blast Circuit Breakers and AK Circuit Breakers," dated June 12, 1990. General Electric also distributed a service action letter (SAL) dated December 7, 1990, which discussed the prop spring failure mechanism and the availability of a newly designed spring with a considerably longer service life. The licensee had not taken correction action related to this generic correspondence prior to the discovery of the broken prop spring on the EDG No. 2 output breaker. The licensee's apparent lack of effective corrective action related to this generic correspondence was reviewed in detail during a separate inspection of the preventive maintenance area (see Inspection Report 50-219/92-07).

It should be noted that the EDG No. 2 start failures, caused by the broken prop spring, provide additional information related to the generic correspondence. While not specifically stated, the related Information Notice and GE SAL imply that the prop spring is necessary to ensure breaker closure. However, the results of the GE tests on the removed EDG No. 2 output breaker (breaker latched closed on 3 of 20 tries) with a broken prop spring and the March 23, 1992, and April 5, 1992, EDG No. 2 load test results (i.e., the output breaker successfully latched and remained closed on the second attempt in each case) show that it is possible for the breaker to remain closed, even with a broken prop spring. Thus, a breaker closure test, by itself, may not necessarily reveal a spring failure.

The licensee's decision to call the March 23, 1992, EDG No. 2 start failure a reportable event after completion of their root cause assessment was appropriate. However, these events could have been precluded had appropriate corrective action been taken on the related generic correspondence.

EDG No. 2 was declared operable on April 9, 1992 after successful completion of post-maintenance testing. The EDG No. 1 output breaker was replaced with a refurbished breaker on April 27, 1992 and was returned to service later the same day after successful post-maintenance testing. The EDG No. 1 output breaker had considerably fewer cycles on it (1700) than the EDG No. 2 output breaker (3000). The generic correspondence had indicated that prop spring failures were seen to occur at around 2000 cycles.

1.4 Inadvertent Spray of the Drywell

On April 20, 1992, at 12:54 p.m., approximately 825 gallons of water were sprayed into the drywell during performance of containment spray system 1 surveillance testing. A control room operator (CRO) was performing surveillance procedure 607.4.004, revision 13, "Containment Spray and Emergency Service Water System 1 Pump Operability and Inservice Test." The testing was required by technical specifications (TS) because system 2 was out of service for preventive maintenance. The plant was operating at 100% power and nitrogen makeup to the torus was in progress before the expected decrease in torus pressure during the surveillance.

In the process of completing the surveillance the CRO inadvertently repositioned the system control switch from the DYNAMIC TEST I position to the AUTO I position before securing the containment spray and emergency service water (ESW) pumps as required. When the control switch was placed in the AUTO I position, the discharge to containment spray valve (V-21-11) went open and the dynamic test flow return valve (V-21-17) went closed as designed. The CRO recognized that the pumps were still running and secured the pumps about 30 seconds after placing the system control switch in the AUTO I position.

At 1:02 p.m. the DRYWELL HI LEAK RATE alarm was received indicating that the unidentified leak rate had increased substantially. The group shift supervisor (GSS) and STA reviewed the emergency plan implementing procedures (EPIPs) to determine the need to enter an emergency condition. Based on their review and knowledge of the source of the water (the inadvertent spray of the drywell), they determined that entry into an emergency action level based on excessive unidentified leakage rate was not required. TS 3.3.D requires the licensee to reduce the leakage rate to within acceptable limits within 8 hours or place the reactor in the shutdown condition within the next 12 hours and be in cold shutdown within the following 24 hours. The leak rate returned to normal levels (about 0.9 gpm unidentified leakrate) within 40 minutes of initiation of spray.

The licensee has experienced two other occasions when the containment spray system was inadvertently used to spray the drywell. The first occurrence was in December 1982 when a CRO mistakenly started a containment spray pump aligned to the drywell and sprayed about 2000 gallons of water into the drywell (see NRC inspection report 50-219/82-29 section 7.5). A more recent occurrence on August 6, 1990, involved the leakage of 313 gallons into the drywell during an automatic actuation test. A design configuration deficiency for the position indication of valve V-21-5 resulted in the operators leaving the valve partially open even though it indicated closed (see NRC inspection report 50-219/90-12 section 1.2). The licensee had conducted thorough reviews of the effects of the 1982 and 1990 drywell spray events. Based on the testing and analysis performed in response to the 1982 and 1990 events and a review of the environmental qualification of the equipment in the drywell, the licensee determined that testing of the main steam line (MSL) safety and electro-matic relief valves (EMRV) acoustic monitors and thermocouple monitors was warranted. The remaining equipment was determined not to be adversely affected by the small amount of water introduced into the drywell.

Plant response was reviewed by a post transient review group (PTRG 92-136A) consisting of the shift technical advisor (STA) and plant operations and engineering department personnel.

Drywell pressure initially decreased by about 0.13 psig during the containment spray. Once the spray was stopped and pumps were secured, drywell pressure began to increase as the water came into contact with hot components in the drywell and flashed to steam. Drywell pressure remained below any trip setpoints or required action levels, however the DRYWELL PRESSURE HI/LO alarm was received at 12:58 p.m. with a maximum drywell pressure reading of about 1.39 psig. As a result of the containment spray, a minor power transient, lasting about 2 minutes, was experienced causing reactor power to increase from 1930 MWth to a maximum of 1938 MWth. The cause of the minor power transient was attributed to thermal effects on the sensing lines for reactor water level. When sprayed with the relatively cooler containment spray water, the differential pressure sensed by the reactor water level instrumentation increased due to the cooling of the reference legs. This resulted in an indicated reactor water level less than actual and caused a momentary increase in feedwater flow. The observed reactor power transient was the result of this feedwater flow transient. No unexpected plant response was noted during the transient based upon PTRG review of plant response data obtained from instrument traces and computer data.

The human performance issues identified during the surveillance test were the subject of an April 20, 1992, Operations Critique (number 2100-92-006). During the critique, the licensee determined that the CRO performing the surveillance did not perform the surveillance as written. Specifically, he failed to secure the pumps before placing the system control switch in the AUTO I position as required by step 6.20 of procedure 607.4.004. Contributing to the event was a weakness in the procedural instructions of step 6.20. This step required the operator to perform a specific set of actions if inservice test (IST) data was to be obtained and a different set of actions if no IST data was being taken. The intermixing of instructions enhanced the probability of the operator missing a required action before proceeding to the next step. Complicating the response to this event was that the CRO performing the surveillance did not inform the other CROs and the group shift supervisor (GSS) until about 10 minutes into the response that he had secured the pumps after placing the system control switch in the AUTO I position.

One of the short-term corrective actions to prevent this event from recurring was to issue a temporary procedure change (TPC) to both system surveillance procedures that separated the individual actions of the IST performance paragraph into discrete steps. The procedures had yet to be updated to the procedure writer's guide and were cumbersome to use. As a long-term corrective action, the licensee plans to submit the system 1 and 2 containment spray/ESW surveillance procedures for review and rewriting to meet the requirements of the procedure writer's guide.

Review of the involved CRO's response to his failure to follow the procedure and the time required for him to provide the information to the others on shift resulted in operations management removing him from licensed duties. The CRO was required to complete a requalification program before returning to licensed duties. By the end of the inspection period the CRO had not yet completed his requalification program and was not performing licensed duties. The requalification program involved retraining on self-checking; review of procedure 106 "Conduct of Operations;" review of procedure compliance standards; a system checkout on the containment spray system that included a review of the system control logic; participation in a crew teamwork and leadership session concentrating on individual and team self-checking and intra-crew communication; and interviews with Plant Operations Managers and Directors. Upon completion of the requalification program, the GSS must make a recommendation for requalification followed by an interview with the Vice President and Director, Oyster Creek who will make the determination to return the operator to licensed duties. When the involved CRO returns to licensed duties, he must interview other plant personnel affected by his actions and develop a presentation emphasizing the cost to the company due to the adverse effects on the plant and the potential for more severe adverse effects.

The inspector reviewed procedure 607.4.004; Operation Critique 2100-92-006; a draft version of PTRG report 92-136A; PTRG report for the 1990 event (PTRG 90-135A); NRC inspection reports 50-219/82-29 and 50-219/90-12; observed performance of the MSL safety/EMRV acoustic monitoring surveillance (see section 1.4 of this report); discussed the event with the involved CRO and operations supervision; and monitored plant conditions shortly after the transient had occurred. No abnormal plant response was noted. Control room response to the event was significantly hampered by the failure of the involved CRO to inform the rest of the operating crew of his actions. However, the response was appropriate by the other members of the crew based on the available information. Discussions with the involved CRO were unable to determine the reason for the 10 minute delay in providing the information on continued operation of the containment spray pump while the discharge valve was going open.

The inspector concluded that inadequate self-checking by the involved CRO, contrary to his training, resulted in the operator missing the requirement to secure the containment spray and ESW pumps. The licensee's actions to remove the operator from licensed duties and the development of a detailed individual requalification program were appropriate. The inspector was particularly concerned with the CRO's failure to inform the others on shift of the error he had made. The timely and accurate communication of information between onshift crew members is vital to ensure the safe operation of the plant. The corrective actions specified by Operation Critique 2100-92-006 were adequate to address the immediate and long term concerns identified by this event. The PTRG was thorough in reviewing the effects on equipment in the drywell from this event.

Failure to secure the containment spray and ESW pumps as required by procedure 607.4.004, step 6.20 was determined to be a violation of NRC requirements. Specifically, TS 6.8.1 requires procedures to be established, implemented, and maintained that meet the requirement of Reg Guide 1.33, revision 2, "Quality Assurance Program Requirements (Operation)." Appendix A, to Reg Guide 1.33 requires that procedures be written for surveillance testing of the containment spray system. This event was caused by the failure of the operator to adequately perform self-checking resulting in the procedural noncompliance. Previous events have been caused by similar personnel errors (specifically closure of all five recirculation loop suction valves that occurred in August of 1991). As such this violation does not meet the criteria for non-citing as described in 10 CFR Part 2, Appendix C (1992) (VIO 50-219/92-08-01).

1.5 Reactor Building Ventilation Trips

On April 17, 1992, at 2:30 p.m., a trip of the reactor building (RB) ventilation system occurred. Following the trip the licensee determined that a sticking relay (XC) associated with a high RB pressure sensor located on the 119 foot elevation resulted in the trip. With strong or gusting wind a pressure transient is sensed by this sensor which can cause a large enough spike to generate a trip condition. If the relay sticks the trip condition will not reset before the RB ventilation trips. However, prior to replacing the relay, a second RB ventilation trip occurred on April 22, 1992, at 7:10 p.m.

High RB pressures were not observed during both trips using other indications available for monitoring RB conditions. To correct the problem the licensee replaced the XC relay on April 26, 1992. Since the relay replacement no additional RB ventilation trips associated with the XC relay or strong or gusting winds have occurred.

The inspector observed the response to the RB ventilation trip that occurred on April 17. Control room operators (CROs) responded to the event by reviewing the RB pressure instrumentation in the control room, determining which of the trip relays caused the trip, and starting the standby gas treatment system (SGTS) to ensure the RB pressure remained negative. The group shift supervisor (GSS) and electrical maintenance supervisor reviewed the electrical drawings to verify the source of the signal that had caused the trip.

Overall, the inspector found the response by the CROs was good when the RB ventilation tripped on April 17, 1992. Evaluation of the trip was adequate in determining that a faulty XC relay had caused the RB ventilation to trip on both occasions. The inspector concluded that the licensee's response and corrective actions were adequate to address the tripping of the reactor building ventilation system.

1.6 Facility Tours

The inspectors observed plant activities and conducted routine plant tours to assess equipment conditions, personnel safety hazards, procedural adherence and compliance with regulatory requirements. Tours were conducted of the following areas:

- control room
- cable spreading room
- diesel generator building
- new radwaste building
- old radwaste building
- transformer yard
- intake area
- reactor building
- turbine building
- vital switchgear rooms
- access control points

Docket File

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

June 30, 1992

Docket No. 50-219

Mr. John J. Barton
Vice President and Director
GPU Nuclear Corporation
Oyster Creek Nuclear Generating Station
Post Office Box 388
Forked River, New Jersey 08731

Dear Mr. Barton:

SUBJECT: OYSTER CREEK DRYWELL CONTAINMENT (TAC NO. M79166)

In our letter of April 29, 1992, regarding Oyster Creek drywell containment, we requested that GPU Nuclear Corporation (GPUN), continue ultrasonic testing (UT) thickness measurements at refueling outages and at outages of opportunity for the life of the plant. The measurements should cover not only areas previously inspected but also accessible areas which have never been inspected so as to confirm that the thicknesses of the corroded areas are as projected and the corroded areas are localized. We also requested that you indicate your intent to comply with the above requirements as discussed in the Safety Evaluation.

In your letter of May 26, 1992, GPUN committed to continue taking UT drywell measurements at refueling outages and at other outages of opportunity. The measurements will be at areas previously inspected and also at other accessible areas not previously inspected. Drywell thickness measurements will continue for life.

You also indicated that the following is your current plan for Oyster Creek drywell UT thickness measurement.

(1) During the 14R outage, GPU Nuclear will take UT thickness measurements in the drywell sandbed region, from the torus room side (outside the drywell), at shell locations not readily accessible from inside the drywell. These are areas not previously inspected. The specific locations selected for inspection will be identified once GPU has direct access to the sandbed region.

Assuming that these measurements confirm that GPU has bounded the corrosion problem with current inspection locations, GPU does currently not plan to make repeat measurements at these specific locations.

(2) Now through the 15R outage, GPU Nuclear will continue taking UT thickness measurements in accordance with the priority method described in Reference 2, Attachment I, "GPUN Specification IS-328227-004, Functional Requirements for Drywell Containment Vessel Thickness Examination."

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(3) After the 15R outage, GPU Nuclear will assess the condition of the drywell by evaluating the then current UT thickness measurements and will formulate an extended inspection plan. The plan will identify measurement locations including frequency of inspection for the remaining life of the plant.

We have reviewed the above information and find that your program commitments regarding UT inspection of the Oyster Creek drywell containment are acceptable. This closes TAC No. M79166.

Sincerely,

/s/

Alexander W. Dromerick, Sr. Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

cc: See next page

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GPU Nuclear

GPU Nuclear Corporation
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201-316-7000
TELEX 136-482
Writer's Direct Dial Number:

April 19, 1994
C321-94-2048

U. S. Nuclear Regulatory Commission
Att: Document Control Desk
Washington, DC 20555

Gentlemen:

Subject: Oyster Creek Nuclear Generating Station (OCNGS)
Docket 50-219
SEP Topic III-7B, Drywell Shield Wall Integrity

Our letter dated November 19, 1993 transmitted ABB Impell Corporation Report No. 0037-00196-01, Rev.0 which provides calculated stresses in the concrete and reinforcing bars in the drywell shield wall above elevation 95 ft. The results indicate that stresses are well below allowables taking into consideration the existing (cracked) condition of the shield wall.

During refueling outages, the reactor cavity is flooded and the inside surface of the drywell shield wall is exposed to some water due to leakage past the steel plate covering the cavity surface. This water could enter the pre-existing cracks in the concrete wall and wet the surface of the steel reinforcing. However, during normal operation, very little moisture is present in the vicinity of the drywell shield wall due to the relatively high temperatures and the fact that the cavity is not flooded.

In our recent phone conversation concerning the subject matter, the NRC staff requested GPU Nuclear to establish a crack monitoring program for the drywell concrete shield wall to provide confirmatory information regarding shield wall conditions. During the phone conversation, GPU Nuclear informed the NRC staff that a structural systems engineer is assigned to the Oyster Creek site. The systems engineer is responsible for ensuring that the structures at Oyster Creek are monitored and evaluated. The NRC expressed a desire that a formal program to monitor cracks in the drywell shield wall be established.

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GPU Nuclear agrees to perform the periodic inspections of the drywell shield wall as requested by the NRC staff. Therefore, GPU Nuclear is developing a program to ensure monitoring of concrete conditions during each refueling outage and a formal guideline for performing the monitoring (e.g. visual inspections for crack growth and/or staining of the concrete). The program and guideline will be in place prior to refueling outage 15R.

Sincerely,

RWKeaten

R. W. Keaten
Director, Technical Function

/lt

cc: Administrator Region I
NRC Resident Inspector
Oyster Creek NRC Project Manager

GPU Nuclear

GPU Nuclear Corporation
One Upper Pond Road
Parsippany, New Jersey 07054
201-316-7000
TELEX 136-482
Writer's Direct Dial Number:

September 15, 1995
C321-95-2235
5000-95-0088

U. S. Nuclear Regulatory Commission
Att: Document Control Desk
Washington, DC 20555

Gentlemen:

Subject: Oyster Creek Nuclear Generating Station (OCNGS)
Docket No. 50-219
Facility Operating License No. DPR-16
Drywell Corrosion Monitoring Program

References:

(1) GPU Nuclear Letter C321-92-2163, "Oyster Creek Drywell Containment," May 26, 1992.

(2) NRC Letter dated June 30, 1992, "Oyster Creek Drywell Containment."

(3) GPU Nuclear Letter C321-93-2100, "Oyster Creek Drywell Inspection," March 25, 1993.

In compliance with Item (3) of References 1 and 2, and Reference 3, GPU Nuclear has (1) assessed the condition of the drywell based upon inspections performed at Oyster Creek during the 15R Outage and is (2) submitting an extended drywell inspection plan for the remaining life of the plant. GPU Nuclear remains committed, as stated in Reference 1, to continue taking drywell thickness measurements for the life of the plant.

Through the 15R Outage, GPU Nuclear's drywell containment vessel thickness monitoring program, Item (2) of References 1 and 2, consisted of ultrasonic thickness (UT) measurements taken at the sandbed region and upper elevations (cylinder, sphere) of the drywell during refueling outages and other outages of opportunity.

Assessment of the most recent UT data taken during the 15R Outage has determined that there is no evidence of ongoing corrosion in the upper elevations of the drywell and that corrosion has been arrested in the sandbed region of the drywell which was cleaned of sand and rust and coated during the 14R Outage (December 1992). The attached table summarizes the 15R Outage UT inspection results for both the sandbed region and upper surfaces indicates that, after 21 months of service, the coating is performing satisfactory with no signs of deterioration such as blisters, flakes, discoloration, etc.

GPU Nuclear's extended inspection plan for the Oyster Creek drywell containment vessel covers both the upper elevations of the drywell and the coated sandbed region.

For the upper elevations of the drywell, this program will perform UT measurements during the 16R Outage (currently scheduled to begin September, 1996) and, as a minimum, again during every other refueling outage (18R, 20R, etc.). The UT measurement locations will be the nine areas identified as most severely corroded. Assessment of the most recent UT data taken during the 15R Outage has determined (and will be reconfirmed by the 16R inspections) that there is no evidence of ongoing corrosion in the upper elevations of the drywell. After each inspection, a technical assessment of the drywell condition will be made, any appropriate corrective action will be taken, and any necessary additional inspections would be scheduled to ensure that drywell integrity is maintained for the remaining life of the plant.

For the sandbed region of the drywell, this program will perform visual inspection of the external epoxy coating during the 16R Outage and, as a minimum, again during the 18R Outage (year 2000). The epoxy coating has an estimated life of 8-10 years which makes the current projected end of life between December, 2000 and December, 2002. Coating inspection shall be by direct (physical) and/or remote methods on a sample basis. Based upon these inspections, a technical assessment of the coating condition will be made, any appropriate corrective action will be taken, and the need for additional (post 18R) inspections will be determined to ensure that drywell integrity is maintained for the remaining life of the plant. In addition, while not technically required based upon the performance of the epoxy coating, UT thickness measurements will be taken one more time in the sandbed region during the 16R Outage, to the same extent as the 15R Outage inspections.

In compliance with Reference 3, GPU Nuclear remains committed to inform the NRC prior to implementing any changes to this drywell inspection program.

Very truly yours,

G. R. Bond

for R. W. Keaten
Vice President and Director
Technical Functions

Attachment
RTZ/plp

c: Administrator, Region I
Senior Resident Inspector
Oyster Creek NRC Project Manager

TABLE 1

ACCEPTABLE MEAN DRYWELL THICKNESSES				
	15R OUTAGE INSPECTION DRYWELL THICKNESSES			
LOCATION	NOMINAL	UT MEASURED MINIMUMS (1)	CODE REQUIRED	MARGIN
Sandbed Region	1.154"	0.806"	.736" (2)	.070" (3)
Sphere (el. 50' - 2")	0.770"	0.733"	0.541"	0.192"
Sphere (el. 51' - 10")	0.722"	0.695"	0.518"	0.177"
Sphere (el. 60' - 11")	0.722"	0.709"	0.518"	0.191"
Cylinder (el. 87' - 5")	0.640"	0.613"	0.452"	0.161"

(1) Thinnest Location as measured during the 15R outage, September, 1994.
(2) Controlled by buckling.
(3) Corrosion arrested (sandbed region coated in 14R outage).

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 1, 1995

Mr. John J. Barton
Vice President and Director
GPU Nuclear Corporation
Oyster Creek Nuclear Generating Station
P.O. Box 388
Forked River, NJ 08731

SUBJECT: CHANGES IN THE OYSTER CREEK DRYWELL MONITORING PROGRAM (TAC NO. M93658)

Dear Mr. Barton:

In a letter dated September 15, 1995, GPU Nuclear Corporation (GPUN) stated that they assessed the condition of the drywell based upon inspections performed at Oyster Creek during the 15R refueling outage (15R) and submitted an extended drywell inspection plan for the remaining life of the plant. GPUN also stated that they remain committed, as stated in their letter of May 26, 1992, to continue taking drywell thickness measurements for the life of the plant.

The staff has reviewed the information provided by GPUN and concludes that changes in the drywell corrosion monitoring program as planned by GPUN is acceptable if GPUN commits to additional inspection within approximately 3 months after discovery of water leakage from the pools above the reactor cavity. Our safety evaluation is enclosed.

Within 30 days of the date of this letter, we request that you provide your intent to perform additional inspection within approximately 3 months after discovery of water leakage.

This requirement affects nine or fewer respondents and, therefore, is not subject to the Office of Management and Budget review under P.L. No. 96-511.

Sincerely,

Alexander W. Dromerick

Alexander W. Dromerick, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure: Safety Evaluation

cc w/encl: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

DRYWELL MONITORING PROGRAM

GPU NUCLEAR CORPORATION

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

GPU Nuclear Corporation (GPUN), the Oyster Creek Nuclear Generating Station licensee, previously, in a letter dated May 26, 1992, committed to conduct ultrasonic thickness (UT) measurements of the drywell at refueling outages (RO) and at other outages of opportunity. The areas to be monitored are the upper elevations and the sandbed regions of the drywell where corrosion had been detected. During the 14th RO (December 1992) the sandbed region of the drywell was cleaned of sand and rust, and coated. During the 15th RO the licensee made UT measurements at the sandbed region and at the upper elevations (cylinder and sphere) of the drywell. In a letter dated September 15, 1995, GPUN stated that they assessed the results of the inspection and determined: (1) there is no evidence of ongoing corrosion in the upper elevations and (2) the corrosion of the sandbed region has been arrested. On the basis of this finding the licensee has proposed to reduce their inspection program as follows:

1. For the upper elevations, UT measurements will be made during the 16th RO (September 1996) and during every second RO, thereafter. After each inspection, a determination will be made if additional inspection is to be performed.

2. For the sandbed region visual inspection of the coating as well as UT measurement of the shell will be made during the 16th RO. The coating will be inspected again during the 18th RO (year 2000). Based on the results of inspection of the coating, determinations will be made for additional inspections.

The licensee has provided a table of UT measurement results from the 15th RO inspection. This table shows the locations of the measurements, the nominal as-constructed thickness, the minimum as measured thickness, the ASME Code required thickness and the corrosion margin available.

On the basis of the information provided, the staff finds the proposed change to the licensee's previous inspection commitment to be reasonable and acceptable. However, since water leaking from the pools above the reactor cavity has been the source of corrosion, the licensee should make a commitment to the effect that an additional inspection of the drywell will be performed about 3 months after the discovery of any water leakage.

Principal Contributor: C. P. Tan

Date: November 1, 1995

ATTACHMENT

EVALUATION REPORT
Oyster Creek Nuclear Generating Station
Docket No. 50-219
Civil Engineering and Geosciences Branch

GPU Nuclear, the Oyster Creek Nuclear Generating Station licensee, previously (May 1992) committed to conduct ultrasonic thickness (UT) measurements of the drywell at refueling outages (RO) and at other outages of opportunity. The areas to be monitored are the upper elevations and the sandbed regions of the drywell where corrosion had been detected. During the 14th RO (December 1992) the sandbed region of the drywell was cleaned of sand and rust, and coated. During the 15th RO the licensee made UT measurements at the sandbed region and at the upper elevations (cylinder and sphere) of the drywell. The licensee assessed the results of the inspection and determined: (1) there is no evidence of ongoing corrosion in the upper elevations and (2) the corrosion of the sandbed region has been arrested. On the basis of this finding the licensee has proposed to reduce their inspection program as follows:

1. For the upper elevations, UT measurements will be made during the 16th RO (September, 1996) and during every second RO, there after. After each inspection, a determination will be made if additional inspection is to be performed.

2. For the sandbed region visual inspection of the coating as well as UT measurement of the shell will be made during the 16th RO. The coating will be inspected again during the 18th RO (year 2000). Based on the results of inspection of the coating, determinations will be made for additional inspections.

The licensee has provided a table of UT measurement results from the 15th RO inspection. This table shows the locations of the measurements, the nominal as-constructed thickness, the minimum as measured thickness, the ASME Code required thickness and the corrosion margin available.

On the basis of the information provided, the staff finds the proposed change to the licensee's previous inspection commitment to be reasonable and acceptable. However, since water leaking from the pools above the reactor cavity has been the source of corrosion, the licensee should make a commitment to the effect that an additional inspection of the drywell will be performed about three months after the discovery of any water leakage.

GPU Nuclear

GPU Nuclear Corporation
One Upper Pond Road
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TELEX 136-482
Writer's Direct Dial Number:

December 15, 1995
5000-95-098
C321-95-2360

U. S. Nuclear Regulatory Commission
Att: Document Control Desk
Washington, DC 20555

Gentlemen:

Subject: Oyster Creek Nuclear Generating Station (OCNGS)
Docket No. 50-219
Facility Operating License No. DPR-16
Drywell Corrosion Monitoring Program

References: (1) NRC Letter dated November 1, 1995, "Changes in the Oyster Creek Drywell Monitoring Program."

(2) GPU Nuclear Letter C321-95-2235, "Drywell Corrosion Monitoring Program," September 15, 1995.

Reference 1 requested GPU Nuclear to make a commitment, as part of the proposed extended Oyster Creek Drywell Monitoring Program (Reference 2), to perform "...additional inspection within approximately 3 months after discovery of water leakage from pools above the reactor cavity." Subsequent discussion with the NRC Staff provided clarification that this request was made to address contingency actions should water leakage be discovered during power operation between scheduled drywell inspections. The requirement was not meant to apply to minor leakage associated with normal refueling activities.

Accordingly, GPU Nuclear proposes to commit to take the following actions should water leakage not associated with normal refueling outage activities be discovered during power operation.

(1) The Oyster Creek NRC Resident Inspector will be notified of the discovery of leakage.

(2) The source of leakage will be investigated and appropriate corrective actions taken.

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(3) An evaluation of the impact of the leakage on drywell structural integrity will be performed to ensure sufficient structural margin is maintained for operation to the next scheduled drywell inspection.

(4) In the unexpected event that the evaluation of the impact of the leakage on drywell structural integrity does not ensure sufficient structural margin will be maintained for operation to the next scheduled outage, an additional drywell inspection will be performed within approximately 3 months after discovery of water leakage.

If you have any questions or comments on this submittal, please contact Mr. Ron Zak, Corporate Regulatory Affairs at (201) 316-7035.

Very truly yours,

R W Keaten

R. W. Keaten
Vice President and Director
Technical Functions

c: Administrator, Region 1
Senior Resident Inspector
Oyster Creek NRC Project Manager

February 15, 1996

Mr. Michael B. Roche
Vice President and Director
GPU Nuclear Corporation
Oyster Creek Nuclear Generating Station
P.O. Box 388
Forked River, NJ 08731

SUBJECT: CHANGES IN THE DRYWELL CORROSION MONITORING PROGRAM (TAC NO. M92688)

Dear Mr. Roche:

In a letter dated November 1, 1995, NRC informed GPU Nuclear Corporation (GPUN) that the changes to the previously committed Drywell Corrosion Monitoring Program as delineated in GPUN's letter dated September 15, 1995, are acceptable. However, GPUN is required to make a commitment to perform additional inspections of the drywell 3 months after the discovery of any water leakage. GPUN felt such a requirement is too broad to be cost effective. In a letter dated December 15, 1995, GPUN clarified its commitment and an understanding between the NRC staff and GPUN has been reached. The requirement is to address water leakage discovered during power operation between scheduled drywell inspections. The requirement was not meant to apply to minor leakage associated with normal refueling activities where minor leakage is defined as less than 12 GPM (gallons per minute). GPUN indicated that prior to each refueling outage, a refueling cavity and equipment pool inspection and leak assessment plan is put in place and the plan has been found to be successful in prior outages. For leakages not associated with refueling activities, GPUN will investigate the source of leakage, take corrective actions, evaluate the impact of the leakage and, if necessary, perform an additional drywell inspection about 3 months after the discovery of the water leakage.

Based on the additional information provided by GPUN, the staff finds GPUN's commitment to perform the inspections acceptable.

Sincerely,

Original signed by:

Alexander W. Dromerick, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-219
cc: See next page

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