50-219

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

March 6, 1995

The Honorable Bill Bradley
United States Senate
Washington, DC 20510

Dear Senator Bradley:

I am replying to your letter of January 25, 1995, which requested that the NRC respond to a letter you received from the Nuclear Information and Resource Service (NIRS) dated January 6, 1995. This letter also addresses the identical issues raised by NIRS in its January 10, 1995, press release.

On December 16, 1994, I sent you three separate, but similar letters giving information on inspections of reactor vessel internal components conducted at the Oyster Creek Nuclear Generating Station. These letters responded to three independent requests from you, each of which included a letter from one of your constituents regarding Oyster Creek. Subsequently, in a press release dated January 10, 1995, NIRS stated that the Nuclear Regulatory Commission (NRC) had misled you and Senator Lautenberg (who had received a copy of a December 15, 1994, letter addressed to Chairman Helen Richmond of the Berkeley Township Environmental Commission) about cracking of safety-related components at Oyster Creek. The major issue raised by NIRS was that even though the NRC had become aware of the cracking of the top guide, a safety-related reactor internal component, in August 1991, I had failed to report this fact to you or Senator Lautenberg in any of the above-mentioned correspondence. NIRS also asserted that I failed to report the presence of additional cracking that had been recently detected in Oyster Creek's top guide, that the NRC failed to admit that Oyster Creek did not inspect their core plate, and that the NRC has not required an analysis of the synergistic effects of cracking in multiple reactor internal components.

In my letters, I did not attempt to identify all the components in which cracking has been identified during inspections of the reactor vessel internal components at Oyster Creek. I did identify the range of safety-related reactor pressure vessel (RPV) components inspected, including the top guide assembly and core support plate holddown bolts, and provided information on cracking of the core shroud, which was the most significant finding of the inspections performed by GPU Nuclear Corporation (the licensee) during its most recent outage. Indeed, cracking of other RPV internal components had been found during previous inspections, as well as during the most recent inspection at Oyster Creek. Though NIRS expressed concerns about cracking of the top guide, this cracking, as I discuss later in this letter, is relatively insignificant. Of more interest is the cracking that was detected in prior inspections in the core spray piping

The core spray pipe cracking was initially identified in 1978 and the extent of this cracking has been evaluated by the NRC and the licensee during each subsequent outage. The core spray piping was repaired in 1978 and 1980. Since that time, results of additional visual inspections have not identified any significant degradation of the piping or the repairs made to the piping.

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Regarding the cracking of the top guide, the licensee first detected cracking of this reactor internal component in 1991 and has closely monitored it in successive outages. NIRS indicates that NRC regional and national officials were only recently made aware of this problem. However, the regional manager to whom NIRS representatives spoke had not recently reviewed the previous inspection reports and therefore was not familiar with the specific inspection results at the time the conversation took place. The record shows that this is not reflective of the NRC's involvement in this issue. The NRC staff conducted an inspection in June 1991, and the results were reported in Inspection Report 50-219/91-21 issued on August 9, 1991. In this inspection report, the staff concluded that the licensee's disposition of the top guide crack as "acceptable as is" was adequate. In an NRC inspection conducted between December 1992 and January 1993, the staff evaluated the results of a remote visual inspection of the top guide conducted by General Electric Corporation for GPU Nuclear Corporation. The results of this evaluation were given in NRC Inspection Report 50-219/92-22. In this inspection report, the staff reported on the quality of the licensee's visual evaluation of the top guide and noted the licensee's determination that it was acceptable to "use as is." The reports referenced above are enclosed for your information and have been available to the public at the Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753 since they were issued.

With respect to the new top guide cracking found in the most recent outage, it was during a conference call with the NRC staff on October 11, 1994, that the licensee stated additional cracking in the top guide had been found. The licensee also reported that cracks found in earlier inspections did not show any measurable growth. In addition, the licensee has assessed all the cracks that have been identified to ensure they do not jeopardize the structural integrity or function of the top guide. It should be noted that the location of the cracks detected in the Oyster Creek top guide is different from that in the foreign reactor cited in the NIRS press release and that both the top guide and core plate at Oyster Creek are of a different design than at the foreign plant where cracking was detected in those components. Specifically, the Oyster Creek core plate is bolted in place. The top guide is restrained vertically by hold-down devices and horizontally by lateral supports. These configurations result in a highly redundant structure, and even if cracking similar to that observed in the foreign plant were to occur, it would not adversely affect the safety of the plant, and these components could still perform their safety-related functions.

NIRS states that the NRC failed to admit that the licensee did not inspect the core plate mentioned above during its most recent outage, even though General Electric (GE) had recently reported an incident of cracking of this component in a foreign reactor. As noted above, the core plate at Oyster Creek is of a different design than the one GE identified as vulnerable to cracking. Additionally, the NRC requires that licensees follow an inservice inspection (ISI) program for safety-related components. As prescribed by the ASME code, this program addresses reactor vessel internal components including the core plate. The recently identified core plate cracking phenomena at the foreign plant is not relevant with respect to Oyster Creek, which has a bolted core plate. Therefore, there was no impetus for the NRC to request the licensee to inspect this component.

With regard to NIRS' statement that the NRC has failed to require GPUN to perform an analysis and evaluation of the synergistic effects of the cracking of multiple reactor internal components on the safe operation of Oyster Creek, we are preparing our final response to the NIRS 10 CFR 2.206 petition dated September 19, 1994, which expressed the same concern. The fact is that the cracks that have been identified in the Oyster Creek top guide are considered to be relatively minor and in no way would jeopardize public health and safety.

In summary, the intent of my December 16, 1994, letters to you was to summarize the most significant results of the most recent Oyster Creek inspection. It was not my intent to provide a comprehensive report on all inspection activities performed during the Oyster Creek outages, nor to imply that inspections had not revealed cracks in other components. I trust that this information will clarify any questions that might have resulted from the NIRS press release of January 10, 1995, and that it addresses NIRS' concerns that appear in its letter of January 6, 1995. If you need further information, I will be glad to provide it to you and your staff.

Sincerely,
Original signed by
James M. Taylor
James M. Taylor
Executive Director
for Operations

Enclosures: As stated

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See next page

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The Honorable Bill Bradley

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AUG 9 1991

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: Inspection Report No. 50-219/91-21

This refers to the inspection conducted by Mr. J. Carrasco of this office on June 24-28, 1991 at the Oyster Creek Nuclear Station. The findings were discussed with you and members of your staff at the conclusion of the inspection.

This inspection focused on engineering disposition for the BWR top-guide structural integrity issue and the follow-up of the safety system functional inspection reported in Inspection Report 50-219/89-80. Areas examined during this inspection are described in the NRC Region I inspection report which is enclosed with this letter. The inspection consisted of selected examinations of procedures and representative records, interviews with personnel, and observations by the inspector.

The inspector found the licensee's disposition for the crack in the top-guide structure was acceptable based on the Electric Power Research Institute guidances and other analyses. The licensee took the proper corrective and preventive actions in response to the findings shown in a NRC Safety System Functional Inspection (SSFI) that were reviewed during this inspection.

No reply to this letter is required. Your cooperation with us is appreciated.

Sincerely,

Jacque P. Durr, Chief
Engineering Branch
Division of Reactor Safety

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W. Ruland, DRP
M. Conner, TSS (SALP Reports Only)
K. Brockman, Regional Coordinator, RI, EDO
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E. Wenzinger, DRP
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Carrasco	Gray	Durr
08/12/91	08/12/91	08/9/91

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U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-219/91-21

AUG 9 1991

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: June 24-28, 1991

Inspectors: J Carrasco 8-9-91
J. E. Carrasco, Reactor Engineer, Materials Section, EB, DRS
date

Approved by: E H Gray 8/12/91
E. H. Gray, Chief, Materials Section, EB, DRS
date

Areas Inspected: An inspection was performed of licensee activities related to BWR Top-Guide Integrity and a follow-up on previous SSFI on ESW/CS inspection findings.

Results: The inspector found the licensee's disposition for the crack in the top-guide structure was acceptable based on the Electric Power Research Institute guidances and other analyses. The licensee took the proper corrective and preventive actions in response to the findings shown in a NRC Safety System Functional Inspection (SSFI) that were reviewed during this inspection.

No violation or deviations were identified.

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DETAILS

1.0 Persons Contacted

GPU Nuclear Corporation

*J. Barton, Director, OCNGS
*T. Dempsey, Manager, Plant Engineering
*C. Lefler, Manager of Technical Functions
*D. Robillard, QA Staff Assistant
*T. Blount, Licensing Engineer
*P. Thompson, Site Audit Manager (Acting)
J. Langenbach, Technical Functions Director (Acting)
D. Ranft, Plant Engineering Director (Acting)
R. Barrett, Plant Operations Director
J. Abramovici, Engineering and Design

US Nuclear Regulatory Commission

*E. Collins, Senior Resident Inspector

* Denotes those who attended the exit meeting.

2.0 Review of the Licensee's Disposition for the Top-Guide Structural Crack Material Nonconformance Report (MNCR) No. 91-0162 (92702)

2.1 Background

Over the years there have been a number of failures of highly irradiated stainless steel boiling water reactor (BWR) internal components. Early failures included fuel cladding, source holders, and control rod absorber tubes. As both neutron irradiation and high stress were believed to be contributing to the problem, this type of cracking was called Irradiation Assisted Stress Corrosion Cracking (IASCC).

Report number NP-4767 by the Electric Power Research Institute (EPRI) entitled "Evaluation of BWR Top-Guide Integrity" evaluated the potential for IASCC in the BWR top-guide structure and if necessary, recommended a program to minimize its impact on BWR owners. EPRI report number NP-6050 entitled "BWR Top-Guide Integrity: Further Evaluation" evaluated the impact of multiple IASCC with reduced fracture toughness properties of BWR top-guide structures.

2.2 Description of BWR-Top-Guide

In the BWR models 2 through 5, the top guide structure is formed by a series of type 304 stainless steel beams joined at right angles by means of vertical slots with beams welded to a peripheral ring. Each opening between beams provides lateral support and guidance for four fuel assemblies. Normal operating loads on the top guide structure are small, but a seismic event would result in relatively large lateral loads to the top guide beams from lateral motion of the fuel assemblies.

2.3 Findings

The inspector reviewed the EPRI documents and discussed the engineering disposition for MNCR 0162 with the licensee's responsible engineer. The disposition for the specific crack found in the upper top guide cell 44-33 was considered "acceptable as is" based on EPRI evaluations, finite element and fracture mechanics analyses. These analyses determined the tolerance of BWR top-guide structures to IASCC in conjunction with a seismic event. The analyses showed that top guide failure during an earthquake would require extensive IASCC, much greater in extent than the present top-guide crack, coupled with significantly degraded fracture toughness properties. Also, the EPRI Report NP4767 evaluated four different type of cracks which are all postulated to be at the joints rather than at the mid span. The cracks analyzed in the EPRI report were found to be acceptable and enveloped the crack found in the upper top guide cell 44-33. The inspector found that the licensee has been conducting its mandatory inspections of the reactor internals and will continue to monitor the top-guide structure for cracks.

In conclusion, the inspector found the licensee's disposition for the MNCR 91-0162 of "Acceptable as is" adequate based on the EPRI report No. NP-4767 and the licensee's continuation of recommended inspections.

3.0 Licensee Action on Previous Inspection Findings

3.1 (Closed) Unresolved Item 50-219/89-80-01 Preparation of Stress Isometric Drawings for Safety Related Piping

3.1.1 Background

In response to the finding shown in a NRC Safety System Functional Inspection (SSFI) Team Report No. 50-219/89-80, the licensee committed to prepare stress isometric drawings for safety related systems in the November 7, 1985 IEB 79-14 response letter to the NRC.

3.1.2 Finding

The licensee revised a previous commitment to prepare stress isometric drawings for safety related piping, for the following reasons: the licensee utilized the piping stress isometric sketches that were consistent with the pipe stress analysis calculations. The licensee stated that these sketches constitute the as-installed configuration of the plant and were based on the quality control walkdowns performed in 1985 and 1986. Discrepancies found between the "as-designed" and the "as-built" configurations were recorded and documented via Material Nonconformance Reports (MNCR's). The inspector verified that the 79-14 Bulletin related documents affecting any particular piping drawing were entered into the "Computer Assisted Records and Information Retrievable System" (CARIRS). The inspector reviewed procedures to update and maintain drawings to ensure that these drawings are the true and accurate representation of the piping system configuration at the plant. Two procedures were used for updating and maintaining drawings. These were: GPU Nuclear Technical Functions Division No. 5000-ADM-7312.02 EP-025 Rev. 1, titled "As-Built Drawings" and GPU Nuclear Technical Functions Division No. 5000-ADM-7312.01 EP-002, Rev. 4, titled "GPUN Drawings". The inspector reviewed these procedures and found them acceptable based on a sample of drawings retrieved from the CARIRS. It appeared that the procedures and the CARIRS system are maintaining an adequate plant configuration for the safety related piping.

In conclusion, the Unresolved Item 89-80-01 is closed.

3.2 (Closed) Potential Violation 50-219/89-80-02 Regarding Possible Inadequate ESW-CS Calculations

3.2.1 Background

From the NRC report No. 50-219/89-80, the licensee's non-conservative calculational assumptions and the failure to effectively evaluate operation of the emergency service water and containment spray ESW/CS systems with the ESW discharge valves throttled, constituted a potential violation.

3.2.2 Finding

On December 5, 1989 during an enforcement conference the licensee made a presentation to the Region I management regarding the ESW/CS engineering calculations to show that:

a) ESW/CS system would have effectively performed within the design basis when operated with the ESW system discharge valves in the throttled position and assuming elevated canal temperatures.

b) GPUN did use conservative calculational assumptions.

c) GPUN did not fail to effectively evaluate operation of the ESW/CS system with ESW discharge valves throttled and elevated canal temperatures.

The licensee presented the inspector with a NRC Region I letter regarding the Enforcement Conference which states: "We have reviewed the additional information presented by the GPUN staff during the December 5, 1989 enforcement conference at the NRC Region I office and the related documentation obtained during the inspection. Based on this information, much of which was not presented to the NRC during the SSFI, we have concluded that plant operation above a canal temperature of 85°F with throttled ESW system flow ocurred with adequate GPUN engineering analyses. Therefore, this issue does not warrant further enforcement action." Based on this NRC letter dated December 20, 1989 and no further questions by the inspector, the item 89-80-002 is closed.

3.3 (Closed) Violation 50-219/89-80-03, FSAR Not Adequately Updated

3.3.1 Background

The NRC SSFI report No. 50-219/89-80 indicated the licensee did not comply with 10 CFR 50.71(e) that required periodic revision of the FSAR to contain the latest material developed.

3.3.2 Findings

The inspector compared the FSAR Revision 3 with the FSAR Revision 5 to verify that the licensee has incorporated specific concerns regarding the failure to maintain the FSAR updated. The following specific concerns were generated during the SSFI inspection:

(a) Proposed new table delineating the peak suppression pool temperatures and the net positive suction head (NPSH) available (core spray pumps) post LOCA with minimum containment spray (CS) system and emergency service water (ESW) System flows at 85° and 90°F intake canal water temperatures, and heat exchanger cleanliness factors of 65% and 90%.

The inspector verified the current revision of the FSAR (revision 5) and the proposed table are shown on Table No. 6.2-15.

(b) Enhanced the description of the automatic start sequence for the containment spray and emergency service water pumps. The inspector verified that the enhanced description is in the current revision of the FSAR. On page 6.2-20, the third paragraph states: " One pump (51A, 51C) in each containment spray system loop is initiated automatically on high drywell pressure and low-low reactor water level, after a 40 second time delay. After the containment spray pumps start, one pump (52A, 52C) in each emergency service water (ESW) system loop is automatically initiated in an additional 45 second delay."

(c) Revised pump flows to reflect current system operation, and corrected heat exchanger capacity.

The inspector verified the current revision of the FSAR section 6.2.2.3.1 the second paragraph which contained the revised pump flows in conjunction with table 6.2-7 which includes the containment spray heat exchangers capacity.

(d) Since table 6.2-14 refers to valve position for drywell purging (ventilation system), the reference is assumed as an error, and the licensee proposed no changes.

The inspector verified that the table in question (table 6.2-14) is for valve position during purging. The inspector concluded that the licensee took proper corrective and preventive action to avoid further violations in this area. The licensee has a program in-place which was initiated by the Licensing Department to obtain Technical reviews for each section of the FSAR. In addition, during 1990 a full time employee was assigned responsibility for the upgrade process. This individual's function is to coordinate the information received from the various technical reviews and existing projects in progress.

The inspector had no further questions and the violation identified as item 50-219/89-80-03 is closed.

3.4 (Closed) Unresolved Item 50-219/89-80-07 Heat Exchanger (HX) Relief Valves, Chlorine Line Boundary Valves not Included in IST Program.

3.4.1 Background

From the NRC report No. 50-219/89-80, the manager of plant engineering committed to include the heat exchanger relief valves in the IST program and to provide an acceptable resolution of the chlorination piping boundary valves in the program.

3.4.2 Finding

The inspector verified revision 6 to the IST program. This revision included testing of the containment spray (CS) heat exchanger relief valves as part of the IST program.

The licensee performed engineering evaluations to determine that required ESW flow is available assuming a chlorine line break due to an earthquake. The analysis concluded that sufficient ESW flow would still be provided to ensure heat removal of the CS system. Therefore, (IST) testing of these valves is not required. The inspector discussed in detail this determination with the engineer responsible for the system and found it acceptable based on the postulated leakage of the chlorine line. The heat removal capability is not impaired due to the postulated pipe rupture. Therefore, Unresolved Item 89-80-07 is closed.

3.5 (Closed) Unresolved Item 50-219/89-80-08 Potential Clogging Of Containment Spray Strainers In Torus

3.5.1 Background

The containment spray system takes suction from the torus through three strainers located inside the torus. Each strainer has about 8.3 square feet of open area with 0.187 inch diameter holes. The SSFI team was concerned that the strainers could be vulnerable to clogging and was interested in the maintenance requirements of these strainers.

3.5.2 Finding

The inspector reviewed the disposition of conduct of plant engineering (PETA) No. 90-138 along with the Safety Evaluation No. SE-000187-001 (DRF 074848). PETA No. 90-138 was prepared to determine whether a preventive maintenance inspection of the torus suction strainers for blockage is necessary.

Sources of blockage during normal plant operations (non-accident conditions) are minimal. Blockage could occur as a result of internal torus coating failure or other debris introduced into the torus. The torus coating has been evaluated by Safety Evaluation SE 000187-001. The inspector reviewed this Safety Evaluation (SE) titled "Evaluation of Blistered Torus Coating". This SE was prepared to assess the impact of the as-left condition of the torus coating on the integrity of the primary containment pressure boundary and the ability to shutdown the reactor and maintain it in a safe shutdown condition during and following a design basis accident (DBA).

The SE indicated that if a Design Basis Accident occurs, the suppression system experiences three phases of blowdown. They are Pool Swell, Condensation Oscillation and Chugging. Pool swell is the most violent and turbulent phase. Therefore, in the unlikely event that paint particles become dislodged, it is most likely that they will be dislodged during the pool swell. However, a compressive force was simulated using a knife point and/or finger pressure. A relatively light load was applied on the blisters which revealed that they were elastic. A heavier load which would compress the blisters against the shell resulted in cracking of the blisters, but produced no spalling or blister growth. In order to establish that the coating adhesion was adequate to withstand DBA, an in-situ adhesion test method was developed. The coating ligaments in between the blisters were tested in different bays using elcometer adhesion test device, it was concluded that the test result has demonstrated adequate adhesion strength of the coating ligaments. Therefore, the evaluation concluded that the coating would not fail in such a way as to cause problems with flow blockage at the suction strainers even during a DBA.

Other debris which may accumulate accidentally were considered to be a minimal threat. Because, access to the torus internals is controlled, an inspection of the torus strainers is not necessary. The inspector had no further questions regarding the licensee's determination. Therefore, the Unresolved Item 50-219/89-80-08 is closed.

3.6 Management Meeting

Licensee management was informed of the scope and purpose of the inspection at the beginning of the inspection. The findings of the inspection were discussed with the licensee representatives during the course of the inspection and presented to the licensee management at the June 28, 1991 exit conference (see paragraph 1 for attendees).

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
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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 MOBILE NDE LABORATORY INSPECTION 50-219/92-22

This letter refers to the inspection conducted by the NRC Mobile NDE Laboratory team during the period December 7-11, 1992, and January 11-15, 1993, at the Oyster Creek Nuclear Power Station in Forked River, New Jersey. During this period, elements of your inservice inspection, erosion/corrosion, feedwater nozzle inspection and drywell liner inspection programs were reviewed. The NRC inspection of these safety-related areas showed that for the specific components reviewed, your programs meet the minimum requirements prescribed for them. The attached report describes the specific areas reviewed.

No reply to the letter is required. Your cooperation with us in this matter is appreciated.

Sincerely,

E H Gray /for JD

Jacque P. Durr, Chief
Engineering Branch
Division of Reactor Safety

Enclosure: NRC Inspection Report No. 50-373/92-022

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Dope

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A. Dromerick, NRR/PD 1-4
F. Young, SRI, Three Mile Island
Docket File 55-60537
G. Johnson, NRR

U.S. NUCLEAR REGULATORY COMMISSION

DOCKET/REPORT NO. 50-219/92-022

LICENSE NO. DPR-016

LICENSEE: GPU Nuclear Corporation
100 Interpace Parkway
Parsippany, NJ 07054

FACILITY NAME: Oyster Creek Unit 1

INSPECTION AT: Forked River, New Jersey

INSPECTION DATES: December 7-11, 1992 and January 11-15, 1993

INSPECTORS: Richard H. Harris for PMP — 3-19-93
P. M. Peterson, Technician, Mobile NDE Laboratory, EB, DRS — Date

Richard H. Harris — 3-19-93
R. H. Harris, Technician, Mobile NDE Laboratory, EB, DRS — Date

Richard H. Harris for RMcM — 3-19-93
R. M. McBrearty, Reactor Engineer, Material Section, EB, DRS — Date

W. M. Mingus; TET, Inc.; Mobile, Alabama
J. Bryant; TET Inc.; Mobile, Alabama

APPROVED BY: Michael Modes — 3/19/93
M. C. Modes, Chief, Mobile NDE Laboratory, Engineering Branch, DRS — Date

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Inspection Summary and Conclusions: An announced inspection was conducted by the NRC's Mobile Nondestructive Evaluation (NDE) Laboratory at Oyster Creek Power Station, Unit 1, during the period December 7-11, 1992, and January 11-15, 1993, (Report No. 50-219/92-022). The purpose of the Mobile Nondestructive Examination (NDE) Laboratory is to perform independent evaluations of components, systems and welds to assure that NDE performed by the licensee is done in compliance with the requirements.

Areas Inspected: Selected areas of the core spray system (CS), containment spray system (CSS and CTS), shutdown cooling system (SDC), main steam system (MS), reactor coolant system (RC) and feedwater system (FW) were examined by the NRC utilizing various NDE methods as listed in the attached table. The licensee's procedures, in conjunction with NRC procedures, were used for nondestructive evaluation. The licensee's final evaluation reports were reviewed and compared with the results obtained by the NRC.

Results: The inservice inspections, evaluated by the NRC, were in compliance with the requirements of the Federal Code and the requirements of the American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, (ASME) Section XI, for inservice inspections (ISI). The program for inspection is manned by professional personnel and the individual inspections performed were conservatively executed.

1.0 INTRODUCTION

The Code of Federal Regulations, 10 CFR 50.55a requires that inservice inspections of safety related equipment be performed to identify any service related degradation of safety systems. These inspections are required to be performed in accordance with ASME Section XI Code. The inspection performed by the NRC at Oyster Creek was made using the NRC Mobile NDE Laboratory. The Mobile NDE Laboratory is capable of independently duplicating the examinations required of the licensee. This provides the NRC with an overview of the licensee's ISI program and tests the adequacy and accuracy of the licensee's inspections.

2.0 INSERVICE INSPECTION PROGRAM REVIEW (73753, 73751, 73752, 73755)

During the period of December 7-11, 1992, and January 11-15, 1993, an on site independent inspection was conducted of Oyster Creek Unit 1. The inspection was performed by NRC inspectors and NDE personnel contracted by the NRC. The objectives of this inspection were to assess the adequacy of the licensee's inservice inspection and flow accelerated corrosion (FAC) inspection program.

These objectives were accomplished by independently performing examinations selected from the Oyster Creek ISI plan and the flow accelerated corrosion program. The ISI program is described in <u>The GPU Nuclear Third Interval Inservice Inspection Program Update for the Oyster Creek Nuclear Generating Station, Jersey Central Power and Light Company</u>, Revision 0, dated February 27, 1992; submitted to the NRC on April 16, 1992. The flow accelerated corrosion program is described in GPU specification SP-1302-12-237, <u>Nuclear Safety Related Pipe Wall Thinning Inspections Specification for Oyster Creek Nuclear Generating Station Erosion/Corrosion Program</u>, Revision 5, dated 1/11/90. The emphasis in selecting components for examination is placed on safety systems.

3.0 NONDESTRUCTIVE EXAMINATION (NDE)

3.1 Visual Examination (57050)

Eighteen (18) safety related pipe weldments and adjacent base material (1/2 inch on either side of the weld) located in the CTS, CS, CSS, SDC, MS, RC and FW systems were visually examined in accordance with NRC procedure NDE-10, Rev. 1, and GPU procedure 6100-QAP-7209.16, Rev. 0, dated 9/16/91. Visual examination was performed of pipe systems and attached components utilizing QC documents, isometric and as-built drawings. The examination was performed specifically to identify any cracks or linear indications, gouges, leakage, arc strikes with craters, or corrosion, which may infringe upon the minimum pipe wall thickness. Mirrors, flash lights and weld gauges were used, as required, to aid in the inspection and evaluation of the weldments.

<u>Results</u>: The visual examinations performed at Oyster Creek were found to be adequate.

3.2 Ultrasonic Examination (73753, 57080)

Thirteen (13) safety-related pipe weldments located in the CS, CSS, CTS, SDC, MS and FW systems were ultrasonically examined using a Stavely Model 136D ultrasonic flaw detector in accordance with NRC procedure NDE-11, Revision 1, and GPU procedure 6100-QAP-7209.08, Manual Ultrasonic Examination of Similar Metal Piping Welds, Revision 0, dated 9/16/91. Three (3) welds in the CS (NZ-1-58, 59 and NZ-3-47), one (1) weld in the CSS (NQZ-1-1-14), one (1) weld in the MS (MS-1-30), six (6) welds in the CTS (NQ-1-64, NQ-2-81, NQ-2-31, 35, 88, and 94), one (1) weld in the FW (RF-2-61) and two (2) welds in the SDC (NU-1-5 and 7) were examined by ultrasonic examination. The Stavely Model 136D was verified for linearity in conformance with NRC procedure NDE-2, Rev 1. To obtain the greatest possible repeatability, the examination was undertaken utilizing transducers and cable that matched, as closely as possible, those used by the licensee. The distance amplitude compensation curves, used for acceptance of the welds, was established utilizing the appropriate Oyster Creek Unit 1 calibration standards.

In addition to a direct comparison of the results of the ultrasonic examination, a number of the welds were profiled utilizing a profile gauge and thickness readings. This data was used to construct a scale model of the weld in order to determine if adequate coverage was obtained in keeping with the requirements of ASME Section XI, Appendix III. These coverage calculations were then compared with the coverage claimed by the subcontractor and accepted by the licensee in the final inspection reports.

Results: The ultrasonic calibration-for-test performed by Oyster Creek is in conformance with the requirements of Appendix III of Section XI. It is the intention of Appendix III of Section XI that the calibration for test be performed using I.D. and O.D. notches. The side drilled holes are included, by ASME, so that the shape of the acceptance curve can be determined. The results of the NRC examinations were essentially the same as those of Oyster Creek .

3.3 Observations (73753)

3.3.1 Reactor Vessel Visual

The remote underwater visual examination of the steam dryer and top guide was selected for inspection to ascertain that the results were clearly recorded and were of sufficient quality to permit proper evaluation.

The examinations were performed by General Electric Company personnel using underwater, remotely operated video equipment and the results were recorded on video tape. Evaluation and disposition of the results were performed by the licensee.

The fillet welds attaching a support bracket to the steam dryer at bank 5 and another bracket between banks 5 and 6 were found to be cracked with the cracks ex ding into the base material in each case. Material Non-Conformance Report (MNCR) No. 920144, dated

12/15/92, was prepared to track and disposition the two cracked areas. The licensee has subsequently reported that the cracks have been repaired by welding. Similar repairs of cracks on the same brackets were performed in 1983 and 1986, respectively.

The remote visual examination of the top guide was performed to monitor a crack which was detected in 1991, during refueling outage 13R at guide blade location 42-31 and which the licensee committed to monitor during the present 14R refueling outage.

The results of the examination verified that the original crack had not propagated, but a second crack was identified on the same member. Visual examination was obstructed of a portion of the member in 1991 and it is possible that the second crack was present at that time. A third, new crack, was identified at fuel cell 20-45. Each of the cracks extended through the affected member. The licensee evaluation of the condition resulted in a use-as-is disposition.

The results of the steam dryer and top guide visual examinations and the related dispositions have been reported to NRR.

During the performance of an air test of core spray system 1, bubbling was identified coming from the area of a fillet weld on the sleeve connecting two sections of the core spray downcomer piping in the reactor vessel annulus between the vessel wall and the core shroud. The sleeve was used to aid pipe fit up during installation of the piping. The bubbling was identified with the use of a remotely operated underwater video camera which, because of its size, could not identify the precise origin of the bubbles. Further examination using a smaller camera identified an approximately 1/8" diameter opening in the fillet weld which the licensee attributed to an original weld defect, not a crack.

At a meeting on January 6, 1993, with the NRC at Rockville, MD, the licensee discussed the core spray system leak and proposals for disposition, including the use of a mechanical clamp to secure the pipe in the event the defect caused a complete failure of the piping. At the conclusion of the meeting, the licensee was requested to provide additional information to the NRC so that a determination could be made regarding the necessity of the clamping device.

A portion of the ultrasonic examination of isolation condenser system, 8" diameter, weld No. NE2-212 was observed to ascertain that procedural and regulatory requirements were complied with. The examination was performed subsequent to the application of the Mechanical Stress Improvement Process (MSIP) using the General Electric Company SMART 2000 automated ultrasonic examination system and was intended to comply with NUREG-0313, Revision 2, and Generic Letter 88-01 requirements. Additionally, the weld was examined to the requirements of the ASME Boiler and Pressure Vessel Code, Section XI.

The inspector determined that applicable requiren.ents were complied with, examination personnel were qualified and certified in accordance with the provisions of SNT-TC-1A and the examiners were listed on the latest edition of the Registry of Qualified Personnel for UT of IGSCC which is published by the Electric Power Research Institute (EPRI) NDE Center at Charlotte, North Carolina.

Conclusions: The video tapes of the underwater visual examinations of the core spray system, steam dryer and top guide clearly show the condition of the various components and provide an excellent means for evaluation of the results. The objects under examination are well lit and in sharp focus which will permit comparison with the results of subsequent visual examination of those components.

The ultrasonic examination was performed in compliance with applicable code and regulatory requirements by properly qualified and certified examiners.

3.3.2 Phased Array Examination

The inspector observed the interpretation of current data taken from angle 57°, at a radius of 338 mm on the "B" nozzle and the data taken in 1988, on nozzle B1, 57°, at an incremental radi of 350 to 400 mm. The inspector also observed the gathering of data from the D nozzle at multiple phased angles at a radius of 498 mm. The calibration curves for each of the current data sets were established on the vessel mockup, burned into an eprom (uniquely identified), and compared against a reference standard before examination commenced. All the examinations were undertaken in conformance with the commitments delineated in Procedure SNPS-AUT-04.01, Revision 7, GPUN/Oyster Creek - Automated Phased Array Ultrasonic Inspection of RPV Nozzles.

Conclusions: The examinations and interpretations were executed in a conservative manner with close attention paid to details. All examinations were in conformance with the requirements and undertaken by well trained professional personnel.

3.4 Flow Accelerated Corrosion (49001)

Concerns regarding flow accelerated corrosion (a. k. a., erosion/corrosion) in balance of plant piping systems has increased as a result of the December 9, 1986, feedwater piping line rupture which occurred at Surry. This event was the subject of the NRC Information Notice 86-106, issued December 16, 198̃, and its supplement issued on February 13, 1987.

The licensee's actions with regard to the detection of erosion/corrosion in plant components were reviewed with respect to NUREG-1344, "Erosion/Corrosion Induced Pipe Wall Thinning in U. S. Nuclear Power Plants," dated April 1989, Generic Letter 88-08 issued May 2, 1989, and NUMARC Technical Subcommittee Working Group on Piping and Erosion/Corrosion Summary Report, dated June 11, 1987.

The Oyster Creek flow assisted corrosion (FAC) inspection program is defined in their procedure SP-1302-12-237, Revision 5, dated 5/29/92. The program is administered by a GPU corporate engineer. It was observed that the following systems are included in the program per 5.7.2 of their procedure: cold reheat (cross-under piping), hot reheat (cross-over piping), high pressure turbine extraction steam, low pressure turbine extraction steam, heater drains and vents, turbine drains (drains to condenser), feedwater (including inside containment), condensate, service water (as an augmented inspection), residual heat removal, and feedwater recirculation. In addition to these systems, other systems may be included based on their susceptibility to FAC due to the following parameters: moisture content, water chemistry, temperature, material composition, and flow path geometry. In addition to this determination, the CHECMATE program is used with the basic heat balance derived by a separate model: PEPSE TRD 153, 8/17/83.

4.0 REVIEW OF SITE NDE PROCEDURES AND MANUALS (73052)

The following ISI procedures were selected for inspection to ascertain that the procedures complied with ASME Code and regulatory requirements and that the procedures are capable of performing their intended function.

General Electric Company Procedures

Procedure GE-UT-208, Revision 1, "Procedure For Automated Ultrasonic Examination of Similar and Dissimilar Piping Welds For IGSCC"

Procedure GE-UT-209, Revision 1, "Procedure For Automated Ultrasonic Examination of Dissimilar Metal Nozzle to Safe End Welds"

Procedure GE-UT-212, Revision 1, "Procedure For Automated Ultrasonic Examination of Weld Overlaid Austenitic Piping"

Siemens Nuclear Power Services, Inc. Procedures

Procedure SNPS-AUT-04.01, Revision 7, "GPUN/Oyster Creek - Automated Phased Array Ultrasonic Inspection of RPV Nozzles"

GPUN Procedures

Procedure 6100-QAP-7209.01, Revision 0, "Magnetic Particle Examination"

Procedure 6100-QAP-7209.02, Revision 0, "Liquid Penetrant Examination"

Procedure 6100-QAP-7209.13, Revision 1, "Manual Ultrasonic Examination of Dissimilar Metal Welds Using Refracted Longitudinal Waves"

Procedure 6150-QAP-7209.29, Revision 0, "Ultrasonic Examination of Weld Overlay Repaired Joints"

The General Electric Company procedures are intended for use with the GE SMART-2000 automated ultrasonic examination system, and the Siemens phased array procedure will be used to control the ultrasonic examination of feedwater nozzles and the control rod drive return nozzle required by NUREG-0619.

The phased array procedure and the GPU procedures were determined to be acceptable and were generally well written. The Siemens procedure describes how applicable NUREG-0619 requirements as interpreted by Siemens and GPUN will be implemented.

Several questions concerning the SMART-2000 procedures arose and as a result, Field Revision Requests (FRR) Nos. OC 14R-01, OC 14R-02 and OC 14R-03 were prepared which resolved all of the questions. All of the inspected procedures, including the FRRs, were approved by the licensee and are acceptable for use at Oyster Creek.

Results: The prompt response by the licensee to resolve questions concerning its vendor's NDE procedures resulted in acceptable procedures and demonstrated good control over inservice inspection vendor activities.

5.0 LICENSEE ACTIONS ON PREVIOUS INSPECTION FINDINGS (92701, 92703)

(Closed) Unresolved item 91-37-03: The adequacy of ultrasonic examinations performed on weld overlay repairs at Oyster Creek.

The Electric Power Research Institute (EPRI) NDE Center at Charlotte, North Carolina, in conjunction with the BWR Owners Group, was instrumental in developing ultrasonic examination techniques which are capable of examining the weld overlay material and the base material directly under the overlay. At the time of NRC Inspection No. 91-37, EPRI recommendations were that the calibration block and production weld should be similar in diameter, wall thickness and overlay thickness. During a telephone conversation on December 12, 1991, with cognizant EPRI personnel, the inspector and a licensee representative discussed the overlay examinations as performed at Oyster Creek. EPRI suggested that the licensee perform an investigation to determine the adequacy of using calibration standards that differ from the production weld with respect to overlay thickness and diameter. With EPRI assistance, the investigation was performed using the Oyster Creek facility. The investigation method, results and conclusions are documented in GPUN Technical Data Report No. 1070, Revision 0. Examination sensitivity, based on original EPRI recommendations, is established from side drilled hole reflectors in a calibration block containing a weld overlay. The GPU investigation concluded that greater sensitivity is attained by establishing a 5% to 20% full screen height noise level from the production weld. The inspector agrees with the conclusions reached by the investigation.

Based on the GPU investigation conclusions, which were concurred with by EPRI, and the fact that the scan sensitivity of the questioned Oyster Creek examinations was based on a 10% to 30% noise level through the production weld, the examinations are considered to be acceptable and this item is closed.

(Closed) Violation 91-18-01: Use of an ultrasonic calibration block not in compliance with the block required by the governing procedure.

The calibration block used for the ultrasonic examination of shutdown cooling system weld NU-3-5 contained weld overlay 0.450" thick and the production weld contained weld overlay 0.29" thick. The governing procedure required that the calibration block be overlaid with weld material of the same thickness range as that of the part to be examined. Licensee corrective actions included instructing its NDE vendor on the importance of following approved procedures and what to do in the event that the ability to follow a procedure is precluded. Licensee Deficiency Report (DR) No. 91-044 was issued to track the item.

The procedural violation resulted in a concern regarding the adequacy of the ensuing ultrasonic examination which is related to item 91-37-03. Based on the licensee's closeout of DR No. 91-044, and the conclusions of TDR 1070, Revision 0, this item is closed.

6.0 MANAGEMENT MEETINGS

Licensee management was informed of the scope and purpose of the inspection at the entrance interview on December 7, 1992. On January 15, 1993, an exit interview was held with members of the licensee's staff listed in Section 7.0. At the meeting, the findings of the inspection were discussed with licensee's management.

7.0 PERSONS CONTACTED

GPU

S. Levin	Director, O&M
J. Kimbel	Director, QA
P. Scallon	Manager, Plant Ops
J. Rogers	Manager, Outage
D. Covill	Manager, NDE/ISI
P. Czaya	Manager, OC Licensing
P. Manning	Manager, NDE/ISI Services
B. Tilton	Manager, Site QA/QC (Acting)
G. Rhedrick	Lead ISI Engineer
R. Nademus	NDE Engineer
M. Heller	OC Licensing Engineer

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D. Vito	Senior Resident Inspector
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