Jersey Central Power & Light Company

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June 16, 1970

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

Dear Dr. Morris:

Re: Oyster Creek Unit No. 1
Docket No. 50-219
Inspection of Reactor Vessel Nozzles

In the summer and fall of 1968, all nozzles in the Oyster Creek reactor vessel, which contained furnace sensitized austenitic stainless steel safe ends, were modified to eliminate the sensitized material or to clad the inside and outside surfaces of the furnace sensitized safe ends with nonsensitized Type 308L stainless steel and/or Inconel weld metal. The sensitized stainless steel reactor vessel head nozzles were replaced with nonsensitized stainless steel.

In April 1970, the Oyster Creek Station was shut down for examination and maintenance of the control rod drive mechanisms. In view of the core spray nozzle cracking problems encountered at the Nine Mile Point Nuclear Station, it was decided to perform visual and liquid penetrant examinations of the outside surfaces of the Oyster Creek core spray nozzles during this outage. Accordingly, during the latter part of April 1970, the thermal insulation was removed from the north and south core spray nozzles to permit access for examination.

The purpose of this letter is to present the results of these examinations and to describe the action taken as a result of several weld defects found in the north core spray nozzle weld overlay cladding. A summary of additional examinations and review of analyses relative to the core spray piping is also presented.

The outside surfaces of the south core spray nozzle were examined visually and by the liquid penetrant technique. The area examined included the weld overlay that was deposited over the remaining sensitized stainless steel safe end and the exposed portion of replacement safe end between the weld overlay and the

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field weld joining the nozzle to the piping. The liquid penetrant examination was performed by a qualified General Public Utilities Corporation examiner using a liquid penetrant procedure that meets the requirements of Section III of the ASME Boiler and Pressure Vessel Code. The liquid penetrant examination showed two indications in the weld overlay cladding that were of questionable relevancy due to the as-welded surface condition of the weld overlay deposit. As a result, these areas were polished by grinding to determine if they were a result of surface condition or were defect indications. Re-examination indicated they were not relevant defect indications and the surfaces examined met ASME Code, Section III, Acceptance Standards.

Subsequent to the inspection of the south core spray nozzle, the north core spray nozzle was prepared for liquid penetrant examination. In order to insure that the surface of the as-welded cladding would be sufficiently smooth to obtain an interpretable penetrant examination, the weld overlay was ground to eliminate surface roughness and crevices between weld beads. The nozzle was then liquid penetrant examined in the same manner and using the same procedure as described above for the south core spray nozzle. The results of this examination showed a number of relatively isolated, acceptable rounded and pinpoint indications of weld porosity (estimated to be less than 1/32" in diameter) and several linear indications. The linear indications were approximately 1/16- to 1/8-inch in length, were randomly oriented and were located primarily in the last 1/2- to 1-inch of weld overlay on the outboard (piping) side of the cladding. All were located in cladding over the nonsensitized stainless steel replacement safe end. No indications were found in wrought material or in the shop or field welds that join the piping, safe end and nozzle. Several of the indications in the lower half of the safe end overlay (at approximately 6 o'clock looking toward the reactor vessel) were ground to a depth of 1/16- to 1/8-inch over an area approximately 1/2-inch wide and 2 inches in circumferential extent. Re-examination by liquid penetrant showed that the linear indications had not been removed. Based on the results of these examinations, further grinding of the north core spray nozzle overlay was deferred pending evaluation to determine the nature of the defects.

On May 1, 1970, a "boat sample" was removed from the north core spray nozzle cladding for metallurgical examination. The sample was approximately 1/8-inch in depth and 1 inch in length and was removed from a location in the outermost edge of the cladding where several linear indications were detected. The sample was mounted, cross sectioned and examined metallurgically by GE-APED. Based on the results of these metallurgical investigations, GE concluded the following:

1. The observed defects are located in Inconel weld metal.
2. The cracks are Inconel weld fissures or hot tears that occur during weld solidification. Many of the fissures do not communicate with the surface.
3. There is no indication of stress corrosion attack.

Results of these examinations were reviewed by Mr. Robert D. Wylie, Southwest Research Institute, and MPR Associates representatives who concurred in the above conclusions.

As a result of the liquid penetrant and metallurgical examinations of the core spray nozzles, it was considered prudent to perform additional examinations of other nozzles having Inconel clad safe ends to determine if there is any indication of a general Inconel weld cracking problem. For this purpose, the control rod drive hydraulic return line nozzle (which is clad with Inconel) and the two isolation condenser nozzles (which are clad with both Inconel and Type 308L stainless steel) were selected. Liquid penetrant examinations performed using the same procedure as used on the core spray nozzles, but without prior surface grinding, revealed no relevant indications on the CRD hydraulic return nozzle or the two isolation condenser nozzles.

Based on the results of the metallurgical and liquid penetrant examinations, which indicate no evidence of stress corrosion attack or general Inconel weld fissure problems, the following action was taken to repair the cladding defects in the north core spray nozzle:

1. A dimensional inspection of the nozzle, cladding and safe end configuration was made prior to additional grinding.

2. Indications of defects were removed by grinding, all cavities blended smoothly into the contour of the surface and the final surface examined by the liquid penetrant technique. This liquid penetrant examination was also performed by a qualified General Public Utilities Corporation examiner using a liquid penetrant procedure that meets the requirements of Section III of the ASME Boiler and Pressure Vessel Code and it showed no relevant indications.

3. Thickness and dimensional measurements were taken to assure that the minimum specified wall thickness remains and that no sensitized stainless steel is exposed.

In addition to the above examinations of the core spray nozzles, the thermal expansion and weight induced stress calculations that were performed by Burns and Roe, Inc., during the piping system design phase of the Oyster Creek Station construction have been checked by the General Electric Company and by GPU Nuclear Power Activities Group engineering personnel, and it has been determined that they are accurate. These stress levels are low (less than 3000 psi) and present no problem in terms of potentially exceeding allowable stress values. Furthermore, a visual inspection has been made of both core spray loop pipe supports and restraints and no irregularities were observed. Stress calculations performed by the reactor vessel manufacturer (Combustion Engineering) for the core spray nozzle safe ends were also reviewed by General Electric to confirm that the as-repaired configuration meets original and current code standards for both internal pressure and piping loads.

It is recognized, as a result of these examinations, that the overlay cladding on the outside of the core spray and CRD hydraulic return nozzles differs from that described in Amendments 40 and 43 of the License Application. These amendments indicate that the cladding over the sensitized portion of the stainless steel safe end would be 308L stainless steel and Inconel; whereas, examination has revealed that the entire clad overlay is Inconel. This change has

been reviewed by the Plant Operations Review Committee and the General Office Review Board. Since the general repair procedures presented in the above amendments do provide for the use of Inconel as a weld overlay on stainless steel and it has been determined this cladding was installed in accordance with a written field instruction using a qualified welding procedure, it has been concluded this change does not involve an unreviewed safety question.

Examination of the reactor coolant system will be continued at the first refueling outage (poison curtain removal) or at an earlier date in the event conditions amenable to such examinations exist.

Very truly yours,

Ivan R. Finfrock Jr.

Ivan R. Finfrock, Jr.
Manager of Nuclear Generating Stations

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