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July 3, 1991

Dr. Thomas E. Murley, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attn: Document Control Desk

Subject: Dresden Nuclear Power Station Units 2 and 3
Quad Cities Nuclear Power Station Units 1 and 2
LaSalle County Nuclear Power Station Units 1 and 2
Reactor Vessel Head Closure Studs
NRC Docket Nos. 50-237/249, 50-254/265 and 50-373/374

References: (a) Conference Call between CECo (M. Richter, et al.) and NRR (L. Olshan, R. Hermann) on January 11, 1991.

(b) Conference Call between CECo (M. Richter, et al.) and NRR (L. Olshan, R. Hermann, et al.) on January 15, 1991.

(c) Conference Call between CECo (M. Richter, et al.) and NRR (B. Siegel, R. Hermann) on January 22, 1991.

(d) Conference Call between CECo (M. Richter, et al.) and NRR (B. Siegel, R. Hermann) on April 19, 1991.

Dr. Murley:

In early 1989 during the eleventh refueling outage for Dresden Unit 2 (D2R11), ultrasonic examinations performed as part of the Section XI Inservice Inspection (ISI) Program identified cracks in the lower threaded portion of two reactor vessel head closure studs (61-198-47 and 61-198-70). The two studs were removed and replaced with spare studs during that refueling outage. Subsequently, a metallurgical evaluation on a sample from stud 61-198-47 determined the cracking was the result of stress corrosion cracking, and that the material toughness of the stud was lower than previously reported in the original Certified Material Test Report (CMTR). The results of the metallurgical evaluation were informally transmitted to your staff and discussed during the Reference (a), (b) and (c) teleconferences. These discussions also addressed the actions taken for Dresden Unit 2 and Quad Cities Unit 1 which were in refueling outages (Fall 1990 refueling outages), and the structural evaluation being performed to ensure safe plant operation. Additionally, Commonwealth Edison Company (CECo) was requested to formally transmit the results of the metallurgical evaluation, the structural evaluation, and those actions planned for future refueling outages at CECo's Boiling Water Reactor (BWR) units. This letter presents that information.

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Background

During D2R11, ultrasonic examinations were scheduled for approximately 33% of the Unit 2 reactor vessel head closure studs as part of the Section XI ISI Program. The inspection technique was a straight beam examination from the upper end of the studs (end-shot examination). On January 24, 1989, the ultrasonic examinations revealed a crack indication in stud 61-198-47. As a result of this indication, the original sample size for these Section XI examinations was expanded to include 100% of the reactor vessel head closure studs (92 studs total). The expanded sample revealed a crack indication in one additional stud (61-198-70). Both of the studs, which were original plant equipment, were removed and replaced with spare studs during D2R11.

Subsequent magnetic particle testing on the outer diameter of the removed studs verified the presence of the cracks, which were located in the lower threaded portion of the studs. Additional ultrasonic testing using an internal diameter (ID) probe (bore probe) estimated the maximum crack depths to be 0.88 inches for stud 61-198-47, and 2.09 inches for stud 61-198-70. In an effort to determine the cause of the cracking, stud 61-198-47 was sectioned to allow for metallurgical evaluation.

Metallurgical Evaluation

The reactor vessel head closure studs measure approximately 65 inches long and 6 inches in diameter, with a one-inch diameter bore hole. The studs were fabricated from material specified as ASME SA320-Gr.L43 (with Code Case 1335), a quench and tempered low alloy steel that is similar to AISI Grade 4340. A 27-inch long section containing the lower threaded region of stud 61-198-47 and 12 inches of the shank was removed for metallurgical evaluation. The evaluation, which consisted of metallographic examinations and mechanical/chemical testing, was performed by Argonne National Laboratory for CECo.

Attachment "A" presents the report which documented the results from the evaluation on stud 61-198-47 (note, these results were previously documented in Licensee Event Report 91-002-0, Docket 050237). Metallographic examinations concluded that the cracking was the result of stress corrosion cracking (SCC) originating from the outer surface of the stud at pitted locations in the root of the threads. The probable cause of the SCC is the exposure of the lower threaded portion of the stud to oxygenated water while in the tensioned condition (following head retensioning) at the conclusion of each refueling outage. Typically, the reactor vessel head is tensioned for approximately three weeks prior to vessel heatup, at which time the moisture contributing to the SCC environment is evaporated. Additionally, the metallographic examinations determined the maximum crack depth for stud 61-198-47 to be 0.70 inches. Based on these results, the detectability limit of the Section XI ultrasonic examinations (end-shot examinations) for Dresden Unit 2 was estimated to be a crack depth of 0.70 inches.

Mechanical testing results on the sectioned stud material indicated a reduction in material toughness and an increase in tensile strength from that reported in the original CMTR. At this time, the cause of this discrepancy is being investigated.

Examinations and Structural Evaluation

At the time the results of the metallurgical evaluation were received, Dresden Unit 2 and Quad Cities Unit 1 were near the conclusion of refueling outages (Fall 1990 refueling outages) in which reactor vessel head closure stud examinations had not been performed. In response to the results from the metallurgical evaluation, CECo developed an enhanced end-shot ultrasonic technique (detectability of a 0.3 inch deep saw cut notch) to examine a sample of the studs at each unit prior to startup. For Dresden Unit 2, ten (10) studs (92 studs total), which had reported Rockwell "C" hardness values between 34 and 39 (converted from BHN), were examined utilizing the enhanced end-shot ultrasonic technique. For Quad Cities Unit 1, six (6) studs (92 studs total), with reported Rockwell "C" hardness values between 35 and 37 (converted from BHN), were examined utilizing the enhanced end-shot ultrasonic technique. The examinations at both units did not reveal any crack indications.

In addition to the examinations, a structural evaluation (Attachment "B" to this letter) was performed to determine the impact that potential stud cracking, combined with low stud material toughness, have on the ASME Code requirements for joint (RPV closure flange) integrity. The evaluation determined that ASME Code margins can still be maintained for Dresden Unit 2 with either 13 fully cracked studs which are evenly distributed around the head flange, or 43 partially cracked studs (crack depth of 1.3 inches) in random distribution around the head flange. The results verify that there is significant margin relative to the Code requirements for joint integrity. Although this evaluation was originally initiated for Dresden Unit 2, it was expanded to include the other CECo BWR units.

A subsequent detensioning-retensioning cycle was experienced by the Dresden Unit 2 reactor vessel head closure studs during a short, mid-cycle maintenance outage in the Spring of 1991. During the maintenance outage, the studs were not exposed to water; therefore, tensioning was not performed with the studs exposed to an environment which could contribute to SCC. Additionally, no unusual stud behavior was observed during the tensioning process.

The only CECo BWR unit with a Spring 1991 refueling outage was LaSalle Unit 1. During that outage, all (100%) of the Unit 1 reactor vessel head closure studs (68 studs total) were examined utilizing the enhanced end-shot ultrasonic technique. The examination did not reveal any crack indications.

On-Going Metallurgical Evaluations

To further investigate the apparent increase in tensile strength and reduction in impact toughness observed during the mechanical testing of stud 61-198-47, CECo will perform an additional evaluation on material from both studs (61-198-47 and 61-198-70). At this time, it is expected that the evaluations will include various quench and temper, and aging heat treatments followed by; mechanical and Charpy V-Notch testing, metallography, scanning electron microscopy, and transmission electron miscroscopy. The results will assist in determining the necessity for further stud mechanical property testing.

In addition to the CECO evaluation, General Electric Company (GE) is performing J-Resistance tests on compact tension specimens machined from stud 61-198-47. Elastic-plastic fracture mechanics properties determined from this testing will be used to: 1) establish the critical flaw size for the stud material; 2) determine allowable flaw sizes based on fatigue and stress corrosion crack growth analyses; and 3) perform an ASME flange bolting analysis. Upon completion of this work, guidelines for acceptable modes of operation with assumed cracked reactor vessel head closure studs will be developed for the CECo BWR units.

Future Examinations/Actions

At the next scheduled refueling outage for each of the six CECo BWR units, all (100%) of the reactor vessel head closure studs will be examined utilizing the enhanced end-shot ultrasonic technique. An interval for future examinations will be established based on the results of these examinations and the currently on-going metallurgical evaluations. As indicated previously, LaSalle Unit 1 has recently completed this examination during the Spring 1991 refueling outage, and found no crack indications.

As discussed with your staff in the Reference (d) teleconference, CECo is considering the removal of a sample of studs from each BWR vessel (during upcoming refueling outages) in order to perform further examinations. CECo will notify your staff if these examinations are pursued.

In addition to the examinations, CECo will be evaluating alternative stud materials, enhancements to existing materials, and techniques for protecting studs from exposure to water.

Please contact this office should further information be required.

Respectfully,

Milton H. Richter

M. H. Richter
Nuclear Licensing Administrator

Attachments: A - CECo Examination Report on Reactor Vessel Head Closure Stud 61-198-47.

B - Structural Evaluation of Commonwealth Edison BWR Reactor Pressure Vessel Head Stud Cracking.

cc: A.B. Davis - Regional Administrator, Region III
B.L. Siegel - NRR Project Manager, Dresden/LaSalle
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