Florida
Power
CORPORATION
Crystal River Unit 3
Docket No. 50-302

April 1, 1997
3F0497-25

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555-0001

Subject: Correction to 90-Day Inservice Inspection (ISI) Report

Reference: FPC to NRC letter, 3F0896-12 dated August 15, 1996

Dear Sir:

This letter is being provided to correct two typographical errors contained in Florida Power Corporation's (FPC) referenced correspondence to the NRC. The reference submitted the 90-day ISI Report after Refuel Outage 10. Pages 2 and 3 and the attached ISI Summary Report, page 5, paragraph two, stated that a random sampling of seventy- five (75) Surveillance Specimen Holder Tube (SSHT) studs was performed. FPC is clarifying that the actual population of SSHT studs is 72 and the number of SSHT studs inspected by Ultrasonic Test (UT), as part of the random sample of SSHT studs, was 21. The attached replacement pages correct the typographical errors.

Please remove these pages in the original correspondence and replace them with the attached revised pages.

Sincerely,

J J Holden

J. J. Holden, Director
Nucle r Engineering and Projects

JJH/LVC

Attachment

xc: Regional Administrator, Region II
Project Manager, NRR
Senior Resident Inspector

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ATTACHMENT

List of replacement pages provided:

1. Revised page two, FPC letter 3F0896-12
2. Revised page three, FPC letter 3F0896-12
3. Revised page five, Attachment to FPC letter 3F0896-12

the case of the fastener MARs, those MARs which addressed fasteners considered most susceptible to failure required that all stud-nut-locking cup assemblies be video inspected and UT inspected during the next ten year refueling outage when the CSA is positioned on the CSA support stand for RV inspection. MARs for fasteners considered less susceptible to failure simply stated that inservice inspection requirements would be determined by the FPC organization responsible for the ASME Section XI Inservice Inspection program (ISI section). As a result of the common CSA fastener failures at B&W designed plants, in parallel with replacement of failed fasteners, the B&W Owners Group formed a task force to study, evaluate, and provide recommendations to preclude future failures. The results of the task force were published in January, 1986, in B&W Topical Report, BAW-1843PA. The topical report, which was reviewed and approved by NRC, recommended that each B&W designed plant evaluate each fastener connection in their respective CSA against specific criteria established by the task force. The report outlined some key parameters to be considered when developing an inspection plan, if so desired. The topical report was reviewed by FPC to ensure the acceptability of the fastener replacements which were performed, however, no specific review for inservice inspection requirements was performed at that time. In 1987 when the ISI section began preparation of the ASME Section XI inspection plan for CR-3's second ten year interval of operation, no review of BAW-1843PA for determination of inservice inspection requirements was performed. As a result, the MAR requirement to perform UT inspection of CSA fasteners considered most susceptible to failure was conservatively, but incorrectly, applied to all CSA fasteners, including those with low probability of failure as determined by BAW-1843PA. Misapplication of the MAR inspection requirement subsequently resulted in the FPC commitment to perform augmented 100% UT inspection of all CSA fasteners made in Reference A and approved by NRC in Reference B.

In 1989, INPO issued SER 30-89 describing damage to reactor system components due to a loose Baffle Assembly Screw. Internal review of SER 30-89 for applicability to CR-3 identified the need for a detailed review of CSA fasteners by design engineering prior to the ten year inspection outage (Refuel Outage 10) to determine specific inspection requirements necessary to assess the condition of each type of CSA fastener. A review was subsequently initiated in December of 1994 with final recommendations for CSA fastener inspections documented in the resolution to Request for Engineering Assistance (REA) 94-1328, dated August 14, 1995. REA 94-1328 recommended the following CSA fastener inspections be performed:

UT inspection of UCBs
Video inspection of UCB locking cups
UT inspection of a random sample of SSHT studs
Video inspection of SSHT locking cups, and
Video inspection of all other CSA fasteners.

These recommended inspections were incorporated into the Refuel Outage 10 NDE inspection schedule and have been completed as described in the attached Summary Report. No defects were found during UT inspection of either the UCBs or the random sample of the SSHT studs. Additionally, all CSA fasteners were confirmed by visual inspection to be in place with no visible degradation.

Since Refuel Outage 10 was the last scheduled refuel outage for CR-3's second ten year inspection interval, a self-assessment of the NDE Inspection plan was initiated in February of 1996. The intent of the self-assessment was to ensure that all ASME Section XI required inspections, as well as regulatory required inspections, would be completed during Refuel Outage 10 to allow close out of the second year inspection

plan. As part of the self assessment, a review of the NDE Inspection plan was performed against Reference B, the NRC Safety Evaluation Report (SER) for the Second 10 year interval Inservice Inspection program plan. This review identified a discrepancy between the NDE Inspection plan (which required no CSA fastener inspections), the Refuel Outage 10 Inspection schedule (which included inspections recommended by REA 94-1328), and Reference B (which acknowledged FPC commitment to perform 100% UT inspection of all CSA fasteners). A historical review of revisions made to the NDE Inspection plan identified a revision on March 2, 1992 which deleted all CSA fastener inspections from the plan. The only basis for this revision was that augmented exams were performed during the first inspection interval to satisfy MAR (i.e. CSA fastener MAR) requirements. Review of associated document records also determined that notification of this revision to the NDE Inspection plan was made via Reference C, however, no technical justification for the revision was provided. Additionally, FPC has not received formal acceptance of this change through NRC issuance of an SER for the program revision.

In order to close this open item, FPC requests NRC review and approval of the technical basis for the UT inspections performed on CSA fasteners during Refuel Outage 10, provided in the attached Summary Report, as justification for modification of the original commitment made in Reference A. Specifically, this change reduces the scope of the commitment from performance of 100% UT inspection of all CSA fasteners to performance of the following:

UT inspection of 100% of Upper Core Barrel Bolts
UT inspection of 21 randomly selected SSHT studs; and
Visual inspection of all CSA fasteners.

Although this change in commitment requires less extensive UT inspection than the original commitment made in Reference A, it is FPC's position that the technical basis for this change is justified and that the inspections performed provide sufficient assurance of the continued integrity of the CSA fasteners.

Crystal River Unit 3's current 10-Year ISI Interval (Second Interval) ends in March 1997. Although Refuel 10 was the last refuel outage of this interval, several augmented examinations, pressure tests and corrective actions to resolve open problem reports initiated during program self assessment remain to be completed. Therefore, an additional Summary Report will be submitted by March 15, 1997 to close out the interval.

Sincerely,

G. L. Boldt
Vice President
Nuclear Production

Attachment

xc: Regional Administrator, Region II
Project Manager, NRR
Senior Resident Inspector

System Pressure Tests

Relief Request 95-020 was issued for use of ASME Section XI Code Case N-498-1, "Alternative Rules for 10-Year System Hydrostatic Pressure Testing for Class 1, 2, and 3 Systems". Pressure testing was conducted on 9 systems to meet the ASME Section XI code requirements. Pressure testing for each Repair/Replacement is documented on the applicable NIS-2 form attached to this report. A summary listing of these tests is included later in this report.

AUGMENTED EXAMINATIONS:

Ultrasonic examination of the Upper Core Barrel bolts (UCB) and the Surveillance Specimen Holder Tube studs (SSHT) was performed during Refuel 10 as an augmented examination. This examination consisted of a 100% (120) UT of the Upper Core Barrel bolts and a random sampling (21) of the 72 SSHT studs. These examinations are not ASME Section XI code required and were performed only to detect any crack type degradation of the fastener material. No defects were found in any of the fastener material.

The basis for performing ultrasonic examination on 100% of the upper core barrel bolts and a random sampling of SSHT studs is based on a comparison of peak operating stress and nominal operating stress for each bolt location to a statistically derived upper limit of stress founded in part from the operating data for other utilities. The comparative data has been extracted from B&W Topical Report BAW-1843PA and CR-3 White Paper, "A Basis for Reactor Vessel Core Support Assembly Bolting Examinations".

1. The upper core barrel bolts (UCB) are fabricated from SA-453 Grade 660, Class A material which is similar to the original bolting material that failed in the 1983 time frame; ASTM A-286. The SA-453 Grade 660, Class A material was selected based on availability and schedule after the failures were found on the original A-286 bolts. The replacement bolts were installed by torquing (940-1010 ft-lbs) with final preload being adjusted using UT. The average peak stress in the bolt was calculated to be 71,700 PSI. The statistical evaluation in BAW-1843PA shows that for greater than 95% confidence, there is a 99.9% probability that Alloy A-286 bolts stressed below 100,000 PSI should not exhibit failure before 6 to 10 years. It is noted that the calculated nominal peak stress for the Oconee-2 original UCB bolts (same as CR-3 originals) was 101,000 PSI with no failures having been reported. Oconee-2 went critical in Nov. 1973. The calculated nominal peak stress for the Arkansas Nuclear One - Unit 1 (ANO-1) original LCB bolts (same as CR-3 originals) is 72,000 PSI and no failures have been reported. ANO-1 went critical in Aug. 1974. This data supports the theory that A-286 bolts with a peak stress less than 100,000 PSI have a high probability of success.

   A 100% UT examination of the UCB joint was performed, even though the stress levels are less than the upper threshold, due to the fact that the replacement bolt material is chemically and physically the same as the original material; and during the testing performed by the B&W Owners Group, a failure of this material could not be produced in the test fixture. This would indicate that there may be additional, and currently unidentified parameters, leading to the failure of this type of material.