Florida Power
CORPORATION
Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72

July 19, 1999
3F0799-09

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

Subject: Request for Additional Information Regarding Second Ten-Year Interval Inservice Inspection Relief Request 98-009-II (TAC No. MA3314)

Reference: FPC to NRC letter, 3F0798-16, dated July 31, 1998, "Second Ten-Year Interval Inservice Inspection Relief Requests 98-009-II, 98-010-II and 98-011-II"

Dear Sir:

The purpose of this submittal is to provide the Florida Power Corporation (FPC) response to a Request for Additional Information by the NRC regarding the use of Relief Request 98-009-II for the Crystal River Unit 3, ASME Section XI, Inservice Inspection Second Interval. The relief request was submitted to the NRC in the referenced letter. The attachment to this letter provides the NRC question, and the corresponding FPC response, as discussed during a telephone conference on June 25, 1999.

This letter establishes no new regulatory commitments. If you have any questions regarding this submittal, please contact Mr. Sid Powell, Manager, Nuclear Licensing at (352) 563-4883.

Sincerely,

Daniel L. Roderick
Director
Nuclear Engineering and Projects

DLR/lvc

A047

Attachment

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

ATTACHMENT

QUESTION

To comply with the augmented reactor vessel examination requirements of 10 CFR 50.55a(g)(6)(ii)(A), licensees must volumetrically examine essentially 100% of each of the Item B1.10 shell welds. The Regulations define essentially 100% as coverage greater than 90% of the examination volume of each weld. As an alternative to the greater than 90% coverage requirement of the Regulations, the licensee proposed that the examination coverage obtained be considered to provide an acceptable level of quality and safety for the Reactor Pressure Vessel (RPV) welds.

At Crystal River, Unit 3 (CR-3), the augmented coverage requirements cannot be met for two shell welds due to core guide lugs, inlet nozzle openings, and the outlet nozzle boss extensions that limit scan coverage. For Welds B1.2.1 and B1.2.3, the physical obstructions limited coverage to 29% and 75%, respectively, of the required volume. To achieve complete coverage for the subject welds, design modifications would be required to increase access from the inside surface (ID).

However, as a result of the augmented volumetric examination rule, licensees must make a reasonable effort to maximize examination coverage of their reactor vessels. In cases where examination coverage from the ID is inadequate, examination from the outside surface (OD) using manual inspection techniques is a potential option. The licensee has not described the efforts taken to maximize examination coverage either by modification of inspection equipment or by performing additional examinations from the OD of the reactor vessel. Therefore, based upon the insufficient information provided by the licensee concerning methods that have been used to maximize examination coverage, the Idaho National Engineering Laboratory (INEL) staff is unable to complete the evaluation of this proposed alternative.

RESPONSE

The response to this question will be discussed from two perspectives. The first will discuss the accessibility to the reactor vessel shell welds from outside the reactor vessel. The second will discuss the effort taken to maximize examination coverage from the inside diameter of the reactor vessel for the shell welds in question.

1. External Examination of the Reactor Vessel

   Examination from the outside surface (OD) is not feasible without dose intensive activities and/or extensive modifications as described below:

The design of the CR-3 Nuclear Steam Supply System (NSSS) locates the reactor vessel inside a concrete cylinder (primary shield wall), with approximately 2-3 feet of clearance between the reactor vessel and the concrete surface. This annular cavity contains the reactor vessel mirror insulation and its support structure. The annular cavity is sealed at the top by the fuel transfer canal seal plate, which is permanently welded to the fuel transfer canal liner and to the reactor vessel closure head flange. The design of the mirror insulation, coupled with limited access, does not permit removal of the mirror insulation panels at the areas of interest only. In order to gain access to these welds, the BARRITE plugs (used for neutron shielding) and the reactor vessel mirror insulation would have to be removed starting with the top ring and working progressively down to the areas of interest, until sufficient access to perform the examination had been provided. The design of the fuel transfer canal and design of the reactor vessel support skirt further restrict access to the annular cavity. Access from the bottom is blocked by the design of the reactor vessel support skirt. The support skirt is a welded steel structure with limited penetrations, none of which is of sufficient size to allow personnel access to the annular cavity.

FPC personnel estimated dose to remove and reinstall the BARRITE plugs, the mirror insulation, and its support structure would be in excess of 25 person-Rem. The estimate is conservatively based on actual doses associated with the Core Flood Nozzle surface examinations.

2. Examination Coverage From Inside Reactor Vessel

<u>Component B1.2.1, Reactor Pressure Vessel Lower Shell-to-Transition Piece Weld</u>

Due to interferences caused by the core support guide lugs and flow stabilizer vanes, the angle beam metal paths were increased using a full-vee technique in an effort to maximize examination coverage of this weld. 40% of the required weld and adjacent base material between each guide lug (360 degrees around the vessel) was examined with 2 axial full-node angle beam scans and 70-degree near-surface beams. No unacceptable indications were detected in the 40% of available weld length between each lug. Any evidence of serious inservice degradation of this weld would have been detected. Since this weld is located outside of the area of highest irradiation in the reactor vessel, and the weld areas inside the belt-line region of the reactor vessel (inspected in March 1996 during the 10-year inspection of the Reactor Vessel) have not experienced any rejectable indications, no inservice degradation is expected in the areas not examined at this weld location.

Component B1.2.3, Reactor Pressure Vessel Nozzle Belt Intermediate Weld

The reactor pressure vessel nozzle belt intermediate weld examination was performed using 2-inch by 2-inch ultrasonic transducers arranged in a two by two array. The size of the transducer head was not the limiting factor in this examination. The inlet nozzle holes, core flood nozzle holes, and the 6-inch tall outlet nozzle bosses were the limiting factors with respect to examination coverage of this weld. No unacceptable indications were detected in the 75% of the weld length examined in March 1996, 10-year inspection of the Reactor Vessel. No unacceptable indications have ever been detected during the preservice, first interval, or second interval examinations of this weld (~19 years of operation). Since this weld is located outside of the area of highest irradiation in the reactor vessel, and the weld areas inside the belt-line region of the reactor vessel have not experienced any rejectable indications, no inservice degradation is expected at the areas not examined at this weld location.