Florida Power
CORPORATION

May 13, 1985
3F0585-09

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
Attn: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72
ASME Section XI, Relief Request

Reference: Letter dated October 31, 1984 from E. C. Simpson, FPC, to H. R. Denton, NRC

Dear Sir:

The referenced letter transmitted relief requests from requirements in the ASME Boiler and Pressure Vessel Code, Section XI, 1974 Edition through Summer 1975 Addenda. In discussions with members of your staff on May 2, 1985, the need to provide an alternate examination for the reactor vessel support skirt weld (Relief Request #140) was identified. Transmitted herewith is the Relief Request #140 which has been revised to incorporate this information.

Should you have any questions, please contact this office.

Sincerely,

G. R. Westafer

G. R. Westafer
Manager, Nuclear Operations
Licensing and Fuel Management

DGG/pjs

Attachment

cc: Dr. J. Nelson Grace
Regional Administrator, Region II
Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission
101 Marietta Street, N.W., Suite 2900
Atlanta, GA 30323

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FLORIDA POWER CORPORATION
INSERVICE INSPECTION
RELIEF REQUEST #140
CRYSTAL RIVER - UNIT 3

Reference Code: ASME Boiler and Pressure Vessel Code, Section XI, 1974 Edition through Summer 1975 Addenda.

I. **Component for which exemption is requested:**

(a) Name and Identification Number:

Reactor Vessel Support Skirt
(FSAR Figure 4-13)

(b) Function:

Supports the Reactor Vessel

(c) ASME Section III Code Class:

Class 1

(d) Valve Category:

N/A

II. **Requirement that has been determined to be impractical:**

ASME Boiler and Pressure Vessel Code, Section XI, Table IWB-2600, Item B1.12, Examination Category B-H, Volumetric Examination.

III. **Basis for Requesting Relief:**

In a letter dated April 14, 1977, Florida Power Corporation informed the NRC that Crystal River Unit 3 (CR-3) did not have integrally welded vessel supports and was, therefore, not required to examine them. This was based on the ASME Code, Section XI, 1974 Edition through Summer 1975 Addenda. It was later determined through examination of the Winter 1976 Addenda to ASME Section XI that the reactor vessel supports at CR-3 are considered to be integrally welded by the Code. This was documented in a letter to the NRC dated July 15, 1977. Florida Power Corporation is, therefore, again requesting relief from the requirement to perform a volumetric examination of the reactor vessel support weld.

The radiation level in the area of the reactor vessel support skirt to vessel weld is estimated to be as high as 1000 R/hr. This radiation field is due to the position of the incore detectors and contamination in the area due to the reactor coolant drain tank rupture that occurred on 2/26/80. This high radiation level combined with the necessity for insulation removal and the amount of time required to obtain acceptable examination results makes it impractical to examine this weld volumetrically.

This weld is not considered part of the Section XI, Class 1 (IWB) boundary under the requirements of the 1980 Edition through Winter 1981 Addenda and, therefore, would be exempt from volumetric examination requirements.

IV. Alternate Examination:

As an alternative, a visual examination will be performed on 10% of the reactor vessel support skirt to vessel weld. The examination will be conducted at three positions along the length of the weld at approximately 120 degree segments. Prior to the inspection, a survey of the area will be taken and provisions will be made to keep worker exposure as low as reasonably achievable (ALARA).

V. Implementation Schedule:

The alternate examination will be performed at or near the end of the first ten-year interval.