September 9, 2008

Mr. Dale E. Young, Vice President
Crystal River Nuclear Plant (NA1B)
ATTN: Supervisor, Licensing & Regulatory Programs
15760 W. Power Line Street
Crystal River, Florida 34428-6708

SUBJECT: CRYSTAL RIVER NUCLEAR PLANT, UNIT NO. 3 - REQUEST FOR RELIEF NO. 07-001-SS REGARDING THE FOURTH 10-YEAR INTERVAL INSERVICE INSPECTION VISUAL EXAMINATION OF THE REACTOR VESSEL SUPPORT SKIRT (TAC NO. MD7737)

Dear Mr. Young,

By letter dated December 21, 2007, as supplemented by letters dated June 5, 2008, and August 14, 2008, the Florida Power Corporation (the licensee), submitted Relief Request (RR) No. 07-001-SS, related to the Fourth 10-Year Interval Inservice Inspection (ISI) Program for the Crystal River Nuclear Plant, Unit 3 (CR3). In RR No. 07-001-SS, the licensee proposed to perform an alternative VT-3 visual examination of the reactor vessel support skirt based on the difficulties of performing the inspection required by the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, and the significant dose that would be absorbed.

Based on the information provided in the application, the U.S. Nuclear Regulatory Commission (NRC) staff concluded that the licensee's compliance with the applicable ISI code would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Therefore, pursuant to Title 10 of the *Code of Federal Regulations*, Part 50.55a (a)(3)(ii), the NRC authorizes the ISI program alternative proposed in RR No. 07-001-SS for the fourth 10-year ISI interval of CR3.

The NRC staff's safety evaluation is enclosed. If you have any questions regarding this matter, please contact Farideh Saba at (301) 415-1447.

Sincerely,

/RA/

Thomas H. Boyce, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-302

Enclosure: Safety Evaluation

cc w/enclosure: See next page

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* by memo dated **OFFICIAL RECORD COPY**

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST NO. 07-001-SS RELATED TO THE FOURTH 10-YEAR

INTERVAL INSERVICE INSPECTION PROGRAM

FLORIDA POWER CORPORATION

CRYSTAL RIVER NUCLEAR PLANT, UNIT 3

DOCKET NO. 50-302

1.0 INTRODUCTION

By letter dated December 21, 2007, as supplemented by letters dated June 5, 2008, and August 14, 2008, the Florida Power Corporation (the licensee), submitted Relief Request (RR) No. 07-001-SS, related to the Fourth 10-Year Interval Inservice Inspection (ISI) Program for the Crystal River Nuclear Plant, Unit 3 (CR3). In RR No. 07-001-SS, the licensee proposed to perform an alternative VT-3 visual examination of the reactor vessel support skirt based on the difficulties of performing the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) required inspection and the significant dose that would be absorbed.

The licensee's request for relief is based on the hardship and unusual difficulty of accessing the entire reactor vessel support skirt for inspection due to restricted physical access, which would require concrete and insulation removal, and thus expose personnel to high radiation levels during the process. The U.S. Nuclear Regulatory Commission (NRC, the Commission) staff has evaluated the licensee's request for relief pursuant to Title 10 to the *Code of Federal Regulations* (10 CFR) Part 50.55a(a)(3)(ii) to determine that compliance with the ASME Code requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

2.0 REGULATORY REQUIREMENTS

10 CFR 50.55a(g) requires that ISI of ASME Code Class 1, 2, and 3 components be performed in accordance with Section XI of the ASME Code and applicable addenda, except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).

In addition, according to 10 CFR 50.55a (a)(3)(ii), alternatives to the requirements of paragraph 50.55a(g) may be used, when authorized by the NRC, if an applicant demonstrates that the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Enclosure

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations further require that ISI of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference into 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The ISI Code of Record for the fourth 10-year interval at CR3 is the 2001 Edition through the 2003 Addenda of the ASME Code, Section XI.

3.0 TECHNICAL EVALUATION

ASME Code Requirements

The ASME Code, Section XI, Examination Category F-A, Item F1.30, requires a 100 percent VT-3 visual examination of weld and mechanical connections at intermediate joints in multi-connected integral and non-integral supports, as defined by Figure IWF-1300-1.

Request for Relief from ASME Code Requirements

The licensee proposed to perform an alternative VT-3 visual examination on 10 percent of the Reactor Pressure Vessel Support Skirt weld B1.12.1. The licensee stated that “VT-3 visual examination will be performed on 10 percent of the interior of the reactor vessel support at three positions along the circumference of the support. The same segments of the support skirt, located 120 degrees apart that were previously examined during the first, second, and third inspection intervals, will be examined in the fourth inspection interval.”

Basis for Requesting Relief

By letter dated June 5, 2008, the licensee responded to the NRC staff’s request for additional information to support the hardship and unusual difficulty of performing the VT-3 visual inspections on the reactor vessel support skirt weld. The licensee’s response stated:

> Even with the improvements in inspection techniques since the last inspection period, there are dose and other limitations, due to the physical configuration of the under vessel area, which prevent a 100 percent VT-3 visual inspection from being performed. The area under the reactor vessel is a high radiation area, and the CR3 commitment to ALARA [as low as is reasonably achievable] requires minimization of the time spent in this area.
>
> The outside surface of the weld is restricted from access by a welded seal plate in the refueling cavity. This plate would have to be removed and a person lowered down between the reactor vessel and the shield wall to remove the insulation and perform the inspection of the support skirt

> weld. An optional access point for the inspection would be to remove the concrete plug, allowing access from below. This option involves the removal of 42 cubic feet of 5000 psi concrete, which is further complicated by the fact that the plug is overhead and there is insufficient room for jackhammers to be used. Therefore, only the inside surface of the support skirt weld is reasonably accessible, which automatically limits the examination to no more than 50 percent.
>
> Additionally, removing the insulation panels on the bottom of the reactor vessel is not easily performed due to the number of incore monitoring tubing interferences in the bottom head. The panels are 30 inches in length and the physical shape of each panel is such that there would be significant difficulty in maneuvering them in the cramped area without potentially impacting one or more of the thin-walled incore monitoring tubing. Since any leakage in one of these thimbles would be un-isolable, any activity that could potentially cause leakage at this location should be avoided.
>
> The areas inspected are reviewed each time to note any trends of degradation. Any mechanism which could damage the vessel would be evident at the three locations examined and not confined to small areas around the vessel. If the entire vessel shifted, the damage to the bolting, support skirt, or associated weld would be visible over a majority of the examination area.

<u>Proposed Alternative</u>

In its letter dated June 5, 2008, the licensee proposed the following alternative to the ASME Code, Section XI, Examination Category F-A, Item F1.30, which requires a 100 percent VT-3 visual examination of weld and mechanical connections at intermediate joints in multi-connected integral and non-integral supports:

> It is expected that more than 10 percent of the inside support skirt weld is inspected during the required VT-3 visual examination, but there is no accurate method of measurement. Therefore, CR3 cannot commit to a larger percentage of the inspection area. The area is too confined to get a measurement device in place. The last time this examination was performed was in conjunction with the bare metal visual inspection of the bottom reactor vessel head during Refueling Outage 13 in 2003.
>
> CR3 will perform a VT-3 examination of the reactor vessel support skirt weld, to the extent practical, when maintenance or other activities remove the insulation covering the support skirt weld. This can be performed in conjunction with the bare metal visual inspection of the bottom reactor vessel head. The inspection is performed with a pole mounted camera and enables a long reach around the circumference of the support skirt.

The licensee also made the following regulatory commitment in relation to the proposed alternative: “CR-3 will revise the Inservice Inspection Plan to assure that the performance of a VT-3 examination of the reactor vessel support skirt weld, to the extent practical, will occur when maintenance or other activities remove the insulation covering the support skirt weld.” This commitment will be completed by September 30, 2008.

4.0 STAFF EVALUATION

The ASME Code requires the licensee to perform an essentially 100 percent surface examination of both the inside and outside surfaces of the reactor vessel support skirt. However, the licensee demonstrated that the area under the reactor vessel is a high radiation area and access to the outside surface of the reactor vessel support skirt weld is physically restricted, as well as being cramped and filled with difficult to maneuver components (i.e. incore monitoring tubing and large insulation panels).

The licensee stated that the 10 percent estimate for VT-3 visual examination of the inside surface of the reactor vessel support skirt weld was a conservative estimate. The licensee also provided figures of the area in question to demonstrate that the area around the reactor vessel skirt is too confined to put a more accurate measurement device in place. Furthermore, visual examinations during the first, second, and third ISI intervals indicated no degradation of the same areas of the support skirt covered by this request.

Considering (1) the amount of radiation exposure that would result from inspection of 100 percent of the support skirt weld, (2) that reactor vessel support skirts have no history of failure, and (3) that a sample of 10 percent of the interior weld surfaces of the support skirt will provide reasonable assurance of structural integrity, the NRC staff finds that performing a VT-3 visual inspection of 100 percent of the reactor vessel support skirt weld presents an unusual difficulty in gaining physical access and would result in a hardship on the licensee without a compensating increase in the level of safety or quality. Thus, the proposed alternative is acceptable pursuant to 10 CFR 50.55a(a)(3)(ii).

5.0 REGULATORY COMMITMENT

The licensee in its letter dated August 14, 2008 has committed to the following with regards to RR No. 07-001-SS:

Commitment	Due Date
CR-3 will revise the Inservice Inspection Plan to assure that the performance of a VT-3 examination of the reactor vessel support skirt weld, to the extent practical, will occur when maintenance or other activities remove the insulation covering the support skirt weld.	09/30/2008

6.0 CONCLUSION

Based on the staff’s evaluation of the request for relief contained in RR No. 07-001-SS, the licensee’s proposed alternative would provide reasonable assurance of structural integrity, and compliance with the ASME Code requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the proposed alternative in RR No. 07-001-SS is authorized for the fourth 10-year ISI interval of CR3. All other requirements of the ASME Code, Section XI for which relief has not been specifically requested remain applicable, including a third party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: Carolyn Fairbanks

Date: September 9, 2008