Progress Energy

Crystal River Nuclear Plant
Docket No. 50-302
Operating License No. DPR-72

Ref: 10 CFR 54

June 21, 2010
3F0610-02

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

Subject: Crystal River Unit 3 – Response to Requests for Additional Information for the Review of the Crystal River Unit 3 Nuclear Generating Plant, License Renewal Application (TAC NO. ME0274) and Amendment #11

References:
(1) CR-3 to NRC letter, 3F1208-01, dated December 16, 2008, "Crystal River Unit 3 – Application for Renewal of Operating License"
(2) NRC to CR-3 letter, dated May 21, 2010, "Request for Additional Information for the Review of the Crystal River Unit 3 Nuclear Generating Plant, License Renewal Application (TAC NO. ME0274)"
(3) NRC to CR-3 letter, dated June 2, 2010, "Request for Additional Information for the Review of the Crystal River Unit 3 Nuclear Generating Plant, License Renewal Application (TAC NO. ME0274)"

Dear Sir:

On December 16, 2008, Florida Power Corporation (FPC), doing business as Progress Energy Florida, Inc. (PEF), requested renewal of the operating license for Crystal River Unit 3 (CR-3) to extend the term of its operating license an additional 20 years beyond the current expiration date (Reference 1). Subsequently, the Nuclear Regulatory Commission (NRC), by letters dated May 21, 2010, and June 2, 2010, provided requests for additional information (RAIs) concerning the CR-3 License Renewal Application (References 2 and 3). Enclosure 1 to this letter provides the response to Reference 2. Enclosure 2 to this letter provides the response to Reference 3. Enclosure 3 to this letter contains Amendment #11 to the License Renewal Application.

No new regulatory commitments are contained in this submittal.

If you have any questions regarding this submittal, please contact Mr. Mike Heath, Supervisor, License Renewal, at (910) 457-3487, e-mail at mike.heath@pgnmail.com.

Sincerely,

Jon A. Franke
Vice President
Crystal River Unit 3

JAF/dwh

Enclosures:
1. Response to Request for Additional Information (Reference 2)
2. Response to Request for Additional Information (Reference 3)
3. Amendment 11 Changes to the License Renewal Application

xc: NRC CR-3 Project Manager
NRC License Renewal Project Manager
NRC Regional Administrator, Region II
Senior Resident Inspector

A140
NRR

Progress Energy Florida, Inc.
Crystal River Nuclear Plant
15760 W. Power Line Street
Crystal River, FL 34428

STATE OF FLORIDA

COUNTY OF CITRUS

Jon A. Franke states that he is the Vice President, Crystal River Nuclear Plant for Florida Power Corporation, doing business as Progress Energy Florida, Inc.; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission the information attached hereto; and that all such statements made and matters set forth therein are true and correct to the best of his knowledge, information, and belief.

Jon A. Franke
Vice President
Crystal River Nuclear Plant

The foregoing document was acknowledged before me this 21 day of June, 2010, by Jon A. Franke.

Carolyn E Portmann

Signature of Notary Public
State of Florida

CAROLYN E. PORTMANN
Commission # DD 937553
Expires March 1, 2014
Bonded Thru Troy Fain Insurance 800-385-7019

(Print, type, or stamp Commissioned Name of Notary Public)

Personally Known ✓ -OR- Produced Identification ______

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50 - 302 / LICENSE NUMBER DPR - 72

ENCLOSURE 1

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (REFERENCE 2)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (REFERENCE 2)

RAI 2.4-1.1

Background:

By letter dated August 31, 2009, the U.S. Nuclear Regulatory Commission (NRC) staff requested additional information related to the Crystal River Unit 3 Nuclear Generating Plant (CR-3), license renewal application (LRA) including request for additional information (RAI) 2.4-1 which questioned the exclusion of fire barrier components from the scope of license renewal that appeared in a Safety Evaluation Report dated July 27, 1979. CR-3 letter dated September 30, 2009, responded to RAI 2.4-1 and stated that:

"As identified in Table 2.4.1-1, there are no fire doors, fire door penetration seals or interior fire hose stations in the Reactor Building. There are fire barrier assemblies which include Thermo-Lag fire barriers on conduits, junction boxes, transmitters, and penetrations encapsulated by stainless steel as discussed in response to RAI 2.3.3.36-3."

Issue:

National Fire Protection Association (NFPA) 14 "Standard for the Installation of Standpipe and Hose Systems," defines a Class III system as "A system that provides 1 ½ in. (40 mm) hose stations to supply water for use by trained personnel and 2 ½ in. (65mm) hose connections to supply a larger volume of water for use by fire departments." The CR-3 Final Safety Analysis Report (FSAR) Section 9.8.7.4, "Manual Fire Suppression Systems, "c. Standpipe and Hose Stations," states that, *"The standpipes and hose station systems installed at CR-3 are Class II with the following exceptions: Reactor Building-Class III..."* The response to RAI 2.4-1 seems to conflict the FSAR description and the NFPA 14 definition in terms of the presence of hose stations in the Reactor Building.

Request:

Verify whether interior hose stations are present in the Reactor Building and if they are in the scope of license renewal in accordance with 10 CFR 54.4(a) and whether they are subject to an aging management review (AMR) in accordance with 10 CFR 54.21(a)(1). If they are excluded from the scope of license renewal and are not subject to an AMR provide justification for the exclusion.

Response

There are no permanent hose stations within the Reactor Building (RB). There is a Fire Service Water System standpipe installed within the RB which provides Fire Service water for manual fire suppression. The system provides 2½ in. hose connections at eight locations. Fire hose for use inside the RB is staged near the RB personnel hatch in the Intermediate Building. The fire hose is considered a short-lived item that is replaced on condition, and is not in the scope of License Renewal.

The RB Standpipe System is in the scope of License Renewal and subject to AMR. The standpipe and standpipe hose connections are included in LRA Table 2.3.3-36 in the

Component/Commodity group, "Piping, piping components, standpipes, hydrants, and tanks," and in LRA Table 3.3.2-36 with aging management by the External Services Monitoring, Boric Acid Corrosion, and Fire Water System Aging Management Programs. LRA Table 2.4.1-1 did not identify Fire Hose Stations as a civil commodity because there are no cabinets, enclosures, houses, racks, or reels which support or provide protection for fire hoses.

The configuration of the RB Standpipe System, with standpipe and standpipe hose connections inside the RB and the fire hose staged near the RB personnel hatch in the Intermediate Building, was reviewed by the NRC during a fire protection inspection and was determined to meet commitments to the NRC. (Refer to the letter from Paul J Kellogg of the USNRC to J.A. Hancock of Florida Power Corporation, Subject: Report No. 50-302/81-25, dated December 8, 1981.)

RAI 2.4-1.2

Background:

CR-3 letter dated September 30, 2009, states on page 6 of 6 of Enclosure 3 that:

"As identified in Table 2.4.2-9, there are no fire barrier assemblies or interior fire hose stations in the Diesel Generator Building."

Issue:

Section 5.4, "Diesel Generator Room," of the NRC Safety Evaluation Report, dated July 27, 1979, page 5-18 states that, "The diesel generator rooms and the control rooms are protected by a preaction automatic sprinkler system actuated by rate-compensating heat detector. In addition, the control room has smoke detectors that alarm in the plant control room. Portable fire extinguishers and interior hose are available for manual suppression."

Request:

Verify whether interior hose mentioned in Section 5.4, "Diesel Generator Room," of the NRC Safety Evaluation Report, dated July 27, 1979, above are in the scope of license renewal in accordance with 10 CFR 54.4(a) and whether they are subject to an AMR in accordance with 10 CFR 54.21(a)(1). If they are excluded from the scope of license renewal and are not subject to an AMR provide justification for the exclusion.

Response

There are no interior fire hose stations located in the Diesel Generator Building. There are interior fire hose stations available in the Auxiliary Building. These fire hose stations are in the scope of License Renewal as identified in LRA Table 2.4.2-1 and were subject to AMR. The fire hose reels are age-managed by the Structures Monitoring Program and the Boric Acid Corrosion Control Program. The fire hose is considered a short-lived item that is replaced on condition, and is not in the scope of License Renewal. The Fire Service Water System standpipes associated with these hose stations are in the scope of License Renewal and were subject to an AMR. The standpipe and standpipe hose connection are included in LRA Table

2.3.3-36 in the Component/Commodity group, "Piping, piping components, standpipes, hydrants, and tanks," and in LRA Table 3.3.2-36 with aging management by the External Services Monitoring, Fire Water System Program, and Boric Acid Corrosion Control Program.

It is noted that the RAI reference to Section 5.4 of the NRC Safety Evaluation Report should be to Section 5.14.4.

RAI 4.3.3-5

Background:

LRA Section 4.3.3 states that a bounding F_{en} factor of 1.49 was used for the Alloy 600 component (incore instrumentation nozzle). It also states that the environmental effects for nickel alloys was obtained from Mehta, H. S., and S. R. Gosselin, "Environmental Factor Approach to Account for Water Effects in Pressure Vessel and Piping Fatigue Evaluations," Nuclear Engineering and Design, 1998. NUREG/CR-6335, which provides the statistical characterizations used to derive this F_{en} factor for Alloy 600, states the fatigue S-N database for Alloy 600 is extremely limited and does not cover an adequate range of material and loading variables that might influence fatigue life. It further states that the data was obtained from relatively few heats of material and are inadequate to establish the effect of strain rate on fatigue life in air or of temperature in a water environment. NUREG/CR-6909 incorporates more recent fatigue data using a larger database for determining the Fen factor of nickel alloys.

Issue:

The reference used in the LRA to determine the environmental effects on nickel alloys may be non-conservative. The F_{en} for nickel alloys based on NUREG/CR-6909 varies based on temperature, strain rate and dissolved oxygen. Based on actual plant operating conditions the F_{en} factor can vary from a value 1.0 to 4.52 based on this methodology. Therefore, the cumulative usage factor (CUF) value for the incore instrumentation nozzle may be as high as 2.61 using the CUF presented in the LRA and the maximum F_{en} derived from NUREG/CR-6909 which would exceed the design limit of 1.0 when considering environmental effects of reactor coolant during the period of extended operation.

Request:

1. Since the F_{en} for nickel alloys can vary from 1.0 to 4.52 based on NUREG/CR-6909 and the CUF value may exceed the design limit of 1.0 for the incore instrumentation nozzle, justify using a value of 1.49 for the F_{en} factor for this nickel alloy component.

2. Describe the current or future planned actions to update the CUF calculation with F_{en} factor for the Alloy 600 component only, consistent with the methodology in NUREG/CR-6909. If there are no current or future planned actions to update the CUF calculation with F_{en} factor for the Alloy 600 component consistent with the methodology in NUREG/CR-6909, provide a justification for not performing the update.

Response

1. *The environmentally-adjusted CUF was calculated by applying a F_{en} of 1.49 to the design CUF of 0.58 (see Table 4.3-2, List No. 9 on LRA page 4.3-17). The CUF of 0.58 is comprised of 0.57 due to design transients and 0.01 due to an overcooling event on February 26, 1980. The original stress report assumed the material was low alloy steel and utilized Figure N-415(a) to obtain the allowable number of cycles for the transients. Using Figure A.3 and Table A.2 from NUREG/CR-6909, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials," to recalculate the design CUF for this location yields an in-air CUF of 0.2055. The F_{en} is then calculated using Equations A.14 through A.17 with a temperature of 579°F, O' set to 0.16 for PWR water, and strain rate selected to maximize the environmental penalty. This yields a F_{en} of 4.13. The cumulative usage factor, U_{en}, considering the effects of reactor coolant environment is then calculated as follows:*

$$U_{en} = U_{in\text{-}air} * F_{en} = 0.2055 * 4.13 = 0.85$$

   Therefore, the design limit of 1.0 is not exceeded using the methodology outlined in NUREG/CR-6909.

2. *There are no current or future plans to update the CUF calculation consistent with the methodology in NUREG/CR-6909 since it has been demonstrated that this location is acceptable.*

Based on this response, a LRA amendment has been prepared; refer to Enclosure 3.

RAI B.2.23-1.1

Background:

By letter dated December 1, 2009, the staff issued RAI B.2.23-1 requesting confirmation that the enhancements proposed for the External Surfaces Monitoring Program will specifically include physical manipulation and other investigative methods designed specifically to detect hardening and loss of strength in elastomers. By letter dated December 30, 2009, the applicant responded to the RAI. In that response, the applicant stated that the External Surfaces Monitoring Program will be used to visually inspect external surfaces and that the Internal Inspection of Miscellaneous Piping and Ducts Program will be used to visually and mechanically inspect internal surfaces.

Issue:

The staff notes that in polymeric materials the aging effect hardening/loss of strength can, depending on the environment to which exposed, initiate on either the internal or external surface of the component. Given that some components may be thick or rigid, it is not clear to the staff how mechanically inspecting a component from the interior surface alone will detect hardening/loss of strength which may initiate on the external surface. Additionally, unless the external surfaces monitoring program contains some requirements for the manual manipulation of polymeric materials, the staff is unsure how the applicant is specifically providing for the manual inspection of polymeric materials which are inspected only from the outside (such as those listed in tables 3.3.2-22, 3.3.2-23, 3.3.2-24, 3.3.2-25, 3.3.2-29, 3.3.2-48, 3.3.2-51, 3.3.252, and 3.3.2-54).

Request:

1. Describe how the Internal Inspection of Miscellaneous Piping and Ducts Program will be used to detect hardening/loss of strength which originates on the external surfaces of polymeric materials or propose an alternate program which will accomplish this purpose.

2. Describe how the manual inspection of polymeric materials which are inspected only from the outside is addressed.

Response

1. *The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program includes physical investigations of polymeric materials (by manipulation, physical testing, etc.) to detect hardening and loss of strength initiating at inside or outside surfaces, as applicable. The program directs these aging management activities during preventive maintenance intervals, when the component(s) can be taken under clearance and both surfaces are available for inspection and testing.*

2. *As described in (1), the visual inspections performed by the External Surfaces Monitoring Program are augmented by the inspections performed by the Inspections of Internal Surfaces of Miscellaneous Piping and Ducting Components Program. These inspections include physical manipulation of the external surface, as appropriate.*

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ENCLOSURE 2

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (REFERENCE 3)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (REFERENCE 3)

RAI 3.2.1.47-1

Background:

Crystal River Unit 3 Nuclear Generating Plant (CR-3) license renewal application (LRA) Table 3.2.1 item 47 states: "This item is not applicable to CR-3. Cast austenitic stainless steel (CASS) valves associated with engineered safety feature (ESF) Systems are inside Class 1 boundaries and evaluated with reactor coolant system components (see Table 3.1.1, Item 3.5.1-55)."

Issue:

Standard Review Plan (SRP) Table 3.2.1 item 47 states that CASS piping, piping components, and piping elements exposed to treated borated water > 250°C are subject to aging effects of loss of fracture toughness due to thermal aging embrittlement and recommends managing the aging effect with the Thermal Aging Embrittlement of CASS Program. LRA Tables 3.2.2-2 "Engineered Safety Features - Summary of Aging Management Evaluation - Core Flood System" and 3.2.2-3 "Engineered Safety Features - Summary of Aging Management Evaluation - Decay Heat Removal System" contain aging management reviews (AMRs) for CASS piping, piping components, and piping elements. The aging effect of loss of fracture toughness due to thermal embrittlement is managed by the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD. This is consistent with SRP Table 3.1.1 item 55, which recommends aging management specifically for CASS Class 1 pump casings and valve bodies and bonnets exposed to reactor coolant >250°C.

Request:

Provide a list of component types covered by the above AMRs in the Core Flood System and Decay Heat Removal System. For components other than Class 1 pump casings and valve bodies and bonnets provide justification for managing the loss of fracture toughness due to thermal embrittlement using the ASME Section XI, Inservice Inspection, Subsection IWB, IWC, and IWD Program in lieu of the Thermal Aging Embrittlement of CASS Program.

Response

Core Flood System

The component type covered by the AMR for CASS components in the Core Flood System indicated in Table 3.2.2-2 on Page 3.2-29 of the LRA are valves (includes valve bodies and bonnets).

Decay Heat System

The component type covered by the AMR for CASS components in the Decay Heat System indicated in Table 3.2.2-3 on Page 3.2-37 of the LRA are valves (includes valve bodies and bonnets).

Since these AMR lines do not contain component types other than valve bodies and bonnets, the ASME Section XI, Inservice Inspection, Subsection IWB, IWC, and IWD Program provides adequate aging management as described in NUREG-1801, "Generic Aging Lessons Learned (GALL) Report."

RAI 3.31.53-1

Background:

The GALL Report Table 3, line items 53 and 54 indicate that steel or stainless steel piping, piping components, and piping elements exposed to condensation should be managed by the Compressed Air Monitoring Program. The GALL AMP XI.M24, "Compressed Air Monitoring Program" includes recommendations for the management of aging effects through visual inspection and for preventive maintenance including air quality checks and performance monitoring. Specifically, the "Monitoring and Trending" element of the GALL Report Compressed Air Monitoring Program recommends testing to verify proper operation of the compressed air system by comparing measured system performance values with specified performance limits as part of the aging management program.

Issue:

The applicant indicates in the LRA that the GALL Report Table 3, line items 53 and 54 are not applicable to CR-3. As a result, the applicant has not developed a Compressed Air Monitoring Program consistent with the GALL guidance to manage steel or stainless steel piping, piping components, and piping elements exposed to condensation. The LRA indicates that the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program is used to manage corrosion of internal surfaces of compressed air systems that *might* be subject to internal condensation. Although the applicant's Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program addresses the detection of age-related degradation products on the internal surfaces of system piping and ducting, it does not include preventive maintenance activities as recommended by the GALL Report for the Compressed Air Monitoring Program.

Request:

1. Clarify which specific table 2 line items are credited to manage loss of material for steel compressed air system piping, piping components, and piping elements exposed to internal condensation.

2. Describe how performance monitoring and air quality considerations recommended by the GALL AMP XI.M24, "Compressed Air Monitoring Program" are included in the selected AMP for managing the materials, environments, and aging effects that are significant to the plant compressed air system.

Response

The CR-3 AMR considered that compressed air system components upstream of system dryers, and throughout the system in compressed air systems without dryers, are in an environment susceptible to moisture and condensation from the compression process. These *components were included in the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program, where they will be subject to periodic inspections. System air receivers and drain traps were also conservatively included in the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program, irrespective as to whether the system design incorporated dryers. The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program was selected on the basis that the monitoring requirements of the GALL aging management program (AMP) XI.M24, "Compressed Air Monitoring Program," are not applicable to portions of the system upstream of system dryers/filters (i.e., air quality monitoring relates to finished air supplied to the distribution system), and that physical inspection prior to and periodically repeated through the period of extended operation provides an effective means of determining whether aging effects are manifest. Systems having corresponding Table 2 line items in the CR-3 LRA are: Control Complex Ventilation System (Table 3.3.2-11), EFP-3 Diesel Air Starting System (Table 3.3.2-26), Emergency Diesel Generator System (Table 3.3.2-33), Instrument Air System (Table 3.3.2-38), Station Air System (Table 3.3.2-50), and Leak Rate Test System (Table 3.3.2-40).*

Components downstream of the compressed air system dryers were considered to be in a dry air environment and, therefore, not expected to exhibit aging effects. A review of CR-3 operating experience was performed, which supported the efficacy of system dryers in reliably supplying dry, oil-free, and particulate-free air, as designed. Accordingly, no AMP was specified for these components.

The CR-3 methodology in this regard is consistent with the guidance of NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 – The License Renewal Rule," as well as the methodology in previous Progress Energy Corporation LRAs. With regard to disqualification of spatial interaction involving air/gas systems, NEI 95-10, Appendix F, states that there are no credible aging mechanisms for air/gas systems with dry internal environments, and notes that a review of site-specific operating experience should be performed to verify this assumption. NUREG-1785, "Safety Evaluation Report Related to the License Renewal of the H. B. Robinson Steam Electric Plant, Unit 2," (Section 3.3.2.4.5.2) and NUREG-1856, "Safety Evaluation Report Related to the License Renewal of the Brunswick Steam Electric Plant, Units 1 and 2," (Section 3.3.2.3) document the staff's acceptance of the applicant's findings of no aging effects for Instrument Air, supported by an operating experience review. Similarly, NUREG-1916, "Safety Evaluation Report Related to the License Renewal of Shearon Harris Nuclear Power Plant, Unit 1," (Section 3.3.2.3.23) documents the staff's acceptance of AMRs that identified no aging effects in a dry air environment.

PROGRESS ENERGY FLORIDA, INC.

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ENCLOSURE 3

AMENDMENT 11 CHANGES TO THE LICENSE RENEWAL APPLICATION

Amendment 11 Changes to the License Renewal Application

Source of Change	License Renewal Application Amendment 11 Changes
RAI 4.3.3-5	Revise LRA Table 4.3-3 on Page 4.3-19 to add Note 4 to the list of Notes to read: 4. A verification of the Incore Instrumentation Nozzle (Ni-Cr-Fe) was performed using the methodology in NUREG/CR-6909 and the Environmentally-Adjusted CUF did not exceed the design limit. Also, change the Component in the second line of the table to read: Incore Instrumentation Nozzle (Ni-Cr-Fe) (Note 4) In addition, add the following sentence to the first full paragraph of LRA Section A.1.2.2.10 on Page A-35: A verification of the Incore Instrumentation Nozzle (Ni-Cr-Fe) was performed using the methodology in NUREG/CR-6909, and the Environmentally-Adjusted CUF did not exceed the design limit.