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

Ref: 10 CFR 54

December 29, 2010
3F1210-09

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

Subject: Crystal River Unit 3 – Response to Request for Additional Information for the Review of the Crystal River Unit 3, Nuclear Generating Plant, License Renewal Application (TAC NO. ME0274) and Amendment #17

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 November 30, 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 letter dated November 30, 2010, provided a request for additional information (RAI) concerning the CR-3 License Renewal Application (LRA) (Reference 2). Enclosure 1 to this letter provides the response to Reference 2. Enclosure 2 to this letter contains Amendment #17 to the CR-3 LRA. No new regulatory commitments are contained in this submittal.

PEF letter to the NRC, 3F1110-03, dated November 23, 2010 (ML103280373), stated that updates to CR-3 LRA Subsection 4.5.1, and the responses to RAI 4.5-1 and RAI B.2.26-1, would be provided later to address any changes associated with containment tendon re-tensioning following repairs. The information regarding the Concrete Containment Tendon Prestress Program provided in PEF letters to the NRC, 3F1210-03, dated December 8, 2010 (ML103470140); 3F1210-06, dated December 16, 2010; and in Enclosure 2 of this letter completes the required updates.

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
2. Amendment 17 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 29 day of December, 2010, by Jon A. Franke.

Tara Barten

Signature of Notary Public
State of Florida

TARA BARTEN
Notary Public - State of Florida
My Comm. Expires Jan 29, 2013
Commission # DD 856189

(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

REQUEST FOR ADDITIONAL INFORMATION (RAI)

RAI B.2.18-1

Background

Generic Aging Lessons Learned (GALL) aging management program (AMP) XI.M32, "One-Time Inspection," states in element 4, "detection of aging effects" that the inspection includes a representative sample of the system population, and, where practical, focuses on the bounding or lead components most susceptible to aging due to time in service, severity of operating conditions, and lowest design margin.

License renewal application (LRA) Section B.2.18, One-Time Inspection, states that the applicant's One-Time Inspection Program is consistent with GALL AMP XI.M32.

Issue

Due to the uncertainty in determining the most susceptible locations and the potential for aging to occur in other locations, the staff noted that large (at least 20%) sample sizes may be required in order to adequately confirm an aging effect is not occurring. The applicant's One-Time Inspection Program did not include specific information regarding how the population of components to be sampled or the sample size will be determined.

Request

Provide specific information regarding how the population of components to be sampled will be determined and the size of the sample of components that will be inspected.

Response:

Consistent with the recommendations of NUREG-1801, draft Revision 2, for components managed by the AMP XI.M2, "Water Chemistry," AMP XI.M30, "Fuel Oil Chemistry," and AMP XI.M39, "Lubricating Oil Analysis" programs, Crystal River Unit 3 (CR-3) will utilize a representative sample size of 20% of the population (defined as components having the same material, environment, and aging effect combination) or a maximum of 25 components. Otherwise, a technical justification of the methodology and sample size used for selecting components for a one-time inspection will be included as part of the program's documentation.

DRAI B.2.19-3

Background

GALL AMP XI.M33, "Selective Leaching of Materials," states in element 1, "scope of program," that the program includes a one-time visual inspection and hardness measurement of a selected set of sample components to determine whether loss of material due to selective leaching is not occurring for the period of extended operation.

LRA Section B.2.19, Selective Leaching, states that a sample population will be selected for the inspections which will be completed prior to commencing the period of extended operation.

Issue

Due to the uncertainty in determining the most susceptible locations and the potential for aging to occur in other locations, the staff noted that large (at least 20%) sample sizes may be required in order to adequately confirm an aging effect is not occurring. The applicant's Selective Leaching Program did not include specific information regarding how the selected set of components to be sampled or the sample size will be determined.

Request

Provide specific information regarding how the selected set of components to be sampled will be determined and the size of the sample of components that will be inspected.

Response:

Consistent with the recommendations of NUREG-1801, draft Revision 2, where practical, the inspection will include a representative sample of the system population and will focus on the bounding, or lead components, most susceptible to aging due to time in service, severity of operating conditions, and lowest design margin. CR-3 will utilize a sample size of 20% of the population, with a maximum sample of 25 components. Otherwise, a technical justification of the methodology and sample size used for selecting components for a one-time inspection will be included as part of the program's documentation. Each group of components with different material/environment combinations is considered a separate population.

RAI B.2.29-1

Background

NRC staff review has determined that masonry walls in the scope of license renewal should be visually examined at least every five years, with provisions for more frequent inspections in areas where significant loss of material or cracking is observed.

Issue

LRA Section B.2.29, under operating experience, noted that a baseline inspection was completed in 1997 and in 2007 a subsequent inspection was completed consistent with the program frequency of at least one inspection every ten years. The LRA did not provide the basis for a ten year inspection frequency.

Request

Explain how the interval will ensure there is no loss of intended function between inspections.

Response:

Prior to the period of extended operation, the Masonry Wall Program will be revised to inspect the masonry walls in the scope of License Renewal every five years.

The Masonry Wall Program already requires a reassessment of the inspection interval after each periodic inspection. The inspection interval may be reduced for more frequent inspection based on the inspection results and the safety significance of the structure.

A five year inspection interval and program requirements for reassessment of the inspection interval after each periodic inspection will ensure there is no loss of intended function between inspections.

This response has resulted in changes to the LRA and a modification to License Renewal Commitment #19. These changes are documented in Enclosure 2 to this letter.

RAI B.2.30-6

Background

NRC staff review has determined that adequate acceptance criteria for the Structures Monitoring Program should include quantitative limits for characterizing degradation. Chapter 5 of ACI 349.3R provides acceptable criteria for concrete structures. If the acceptance criteria in ACI 349.3R is not used, then the plant-specific criteria should be described and a technical basis should be provided for the plant specific criteria.

Issue

Although the LRA discussed ACI 349.3R as a reference for the Structures Monitoring Program, it did not commit to the quantitative acceptance criteria, or clearly identify plant specific quantitative acceptance criteria for Structures Monitoring Program inspections.

Request

a) Provide the quantitative acceptance criteria for the Structures Monitoring Program. If the criteria deviate from those discussed in ACI 349.3R, provide technical justification for proposed acceptance criteria.

b) If quantitative acceptance criteria will be added to the program as an enhancement, provide plans and a schedule to conduct a baseline inspection with the quantitative acceptance criteria prior to the period of extended operation.

Response:

The Structures Monitoring Program follows the guidance of ACI 349.3R for its acceptance criteria of concrete surfaces. However, the Structures Monitoring Program will be enhanced to include the additional quantitative acceptance criteria of ACI 349.3R, Chapter 5. CR-3 will perform a baseline inspection using the quantitative acceptance criteria of ACI 349.3R prior to the period of extended operation.

This response has resulted in changes to the LRA and a modification to License Renewal Commitment #20. These changes are documented in Enclosure 2 to this letter.

RAI 4.3.3-6

Background

In LRA Section 4.3.3, the applicant discussed the methodology to determine the locations that require environmentally assisted fatigue analyses consistent with NUREG/CR-6260 "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components." The staff noted that, in LRA Table 4.3-3, there are ten plant-specific locations listed based on the six generic components identified in NUREG/CR-6260.

Issue

GALL Report AMP X.M1, "Metal Fatigue of Reactor Coolant Pressure Boundary," states that the impact of the reactor coolant environment on a sample of critical components should include the locations identified in NUREG/CR-6260 as a minimum, and that additional locations may be needed. The LRA is unclear whether the applicant verified that the plant-specific locations listed in the LRA Table 4.3-3 per NUREG/CR-6260 were bounding for the generic NUREG/CR-6260 components. Furthermore, the staff noted that the applicant's plant-specific configuration may contain locations that should be analyzed for the effects of the reactor coolant environment other than those identified in NUREG/CR-6260. This may include locations that are limiting or bounding for a particular plant-specific configuration, or that have calculated cumulative usage factor (CUF) values that are greater when compared to the locations identified in NUREG/CR-6260.

Request

a) Confirm and justify that the plant-specific locations listed in LRA Table 4.3-3 are bounding for the generic NUREG/CR-6260 components.

b) Confirm and justify that the locations selected for environmentally-assisted fatigue analyses in LRA Table 4.3-3 consists of the most limiting locations for Crystal River Unit 3 Nuclear Generating Plant (beyond the generic components identified in the NUREG/CR-6260 guidance). If these locations are not bounding, clarify the locations that require an environmentally-assisted fatigue analysis and the actions that will be taken for these additional locations. If the limiting location identified consists of nickel alloy, state whether the methodology used to perform the environmentally-assisted fatigue calculation for nickel alloy is consistent with NUREG/CR-6909. If not, justify the method chosen.

Response:

The two parts of this RAI, i.e., parts a) and b), are answered in turn below. The text of each request item is repeated prior to the associated response.

Request

a) Confirm and justify that the plant-specific locations listed in LRA Table 4.3-3 are bounding for the generic NUREG/CR-6260 components.

Response:

The locations listed in the CR-3 LRA, Table 4.3-3, are consistent with NUREG/CR-6260 generic limiting locations evaluated in Section 5.3 of NUREG/CR-6260 for Babcock & Wilcox (B&W) plants. With respect to limiting locations, NUREG-6260, Section 4.1, states that for both pressurized water reactor (PWR) and boiling water reactor (BWR) plants, these components are not necessarily the locations with the highest design cumulative usage factors (CUFs) in the plant, but were chosen to give a representative overview of components that had higher CUFs and/or were important from a risk perspective. For example, the reactor vessel shell (and lower head) was chosen for its risk importance.

In many instances the design CUFs listed in Table 4.3-2 of the CR-3 LRA for reactor coolant system (RCS) pressure boundary items are very conservative calculations dating back to the preparation of the original stress reports in the 1970s. The methods used to calculate fatigue usage in the 1970s for vessels included interaction analysis and use of enveloping nuclear steam supply system (NSSS) design transients (e.g., selection of 4 or 5 worst-case NSSS transients to bound other NSSS transients including combination of cycles). In general, many of the original design CUFs may be reduced significantly by analyzing the detailed NSSS design transients (i.e., removing enveloping groupings and cycles) and by use of finite element methods. Therefore, selection of bounding locations to evaluate environmentally-assisted fatigue (EAF) for CR-3 license renewal based solely on design CUFs is not an appropriate method to select locations for the evaluation of EAF.

As discussed in the closeout of NRC Generic Safety Issue (GSI)-190, "Fatigue Evaluation of Metal Components for 60-Year Plant Life," the Pacific Northwest National Laboratory (PNNL) performed calculations of the probability of component failure and the Core Damage Frequency (CDF) associated with these failures. PNNL made use of the previous and most recent testing performed to develop fatigue design curves for stainless steel in simulated light water reactor (LWR) environmental conditions. Per Attachment 2 of the closeout letter to GSI-190, the Advisory Committee on Reactor Safety (ACRS) found that the PNNL study showed that some components have cumulative probabilities of crack initiation and through-wall growth that approach unity within the 40- to 60-year period. The maximum failure rate (through-wall cracks per year) was in the range of 10^{-2} per year, and those failures were associated with high cumulative usage factor locations and components with thinner walls, i.e., pipes were more vulnerable to through-wall cracks. There was only a modest increase in the frequency of through-wall cracks in major RCS components having thicker walls. In most cases, the leakage from these through-wall cracks is small and not likely to lead to core damage. Therefore, the projected increased frequency in through-wall cracks between 40- and 60-years of plant life does not significantly increase CDF.

Consistent with the NRC's emphasis on Risk-Informed and Performance-Based Regulation, risk considerations may be used to confirm and justify that the plant-specific locations listed in the CR-3 LRA, Table 4.3-3, are bounding for the generic NUREG/CR-6260 components. A review of the CR-3 RCS components by material type and associated bounding environmental fatigue

penalty factor (F_{en}) relative to a qualitative assessment of risk significance (i.e., consideration of probability of failure and consequence of failure) is provided below.

Low Alloy Steel (LAS) Locations (F_{en} maximum of 2.54 based on NUREG/CR-6583)

The CR-3 RCS pressure boundary components with parts made from low alloy steel include the reactor vessel (RV) (entire vessel), once-through steam generator (OTSG) (upper and lower heads, transition ring, tubesheets, and pressure boundary bolting), pressurizer (heater bundle cover plate and pressure retaining bolting), reactor coolant pump (RCP) (bolting), and RCS attached piping (valve bolting). Only the OTSG and RV have low alloy parts that may be susceptible to EAF should the cladding be breached. The susceptible OTSG parts include Alloy 82/182 clad upper and lower tubesheets and stainless steel clad upper and lower hemispherical heads. The susceptible RV parts include all items that are clad with either austenitic stainless steel or Alloy 82/182.

The most risk-significant component is the RV. In addition, CUFs for the susceptible OTSG parts are all less than 0.13, and when multiplied by a bounding F_{en} of 2.54 for LAS, yields an EAF CUF less than 1.0. The most risk significant RV items include the RV inlet and outlet nozzles, core flood nozzles, and lower head of the RV. These items are all included as NUREG/CR-6260 locations and were all shown to have EAF CUF values below 1.0 in Table 4.3-3 of the CR-3 LRA.

Therefore, the NUREG/CR-6260 LAS items evaluated by CR-3 in LRA Table 4.3-3 represent bounding locations when considering the entire RCS.

Carbon Steel Locations (F_{en} maximum of 1.74 based on NUREG/CR-6583)

CR-3 RCS pressure boundary components with parts made from carbon steel (CS) include the pressurizer (shell, surge, spray and pressure relief nozzles, and manway cover), RCS large bore piping and associated branch connections (i.e., nozzles), OTSG (primary inlet and outlet nozzles, and manway cover). The susceptible pressurizer parts include the stainless steel clad shell and nozzles (surge, spray and pressure relief). The susceptible RCS piping parts include stainless steel clad large bore piping and attached branch connections fabricated from stainless steel clad CS. The susceptible OTSG parts include the stainless steel clad primary inlet and outlet nozzles.

The maximum cumulative usage for the susceptible pressurizer parts are all less than 0.32 (CUF at the inside radius of the pressurizer surge nozzle) and when multiplied by a bounding F_{en} of 1.74 for CS, yields an EAF CUF less than 1.0.

The maximum cumulative usage for the susceptible RCS large bore piping and associated branch connections are all less than 0.49 (inside radius of High Pressure Injection/Make-Up (HPI/MU) nozzle). Multiplying the design CUF of 0.49 by a bounding F_{en} of 1.74 for CS yields an EAF CUF less than 1.0.

The maximum cumulative usage for the susceptible OTSG parts are all less than 0.03 and, when multiplied by a bounding F_{en} of 1.74 for CS, yields an EAF CUF less than 1.0.

Therefore, there are no items made from stainless steel clad CS at CR-3 with EAF CUF values greater than 1.0. The NUREG/CR-6260 locations conservatively include the stainless steel clad

CS HPI/MU nozzle, pressurizer surge nozzle, and the hot leg surge nozzle. Therefore, the NUREG/CR-6260 CS items evaluated by CR-3 in LRA Table 4.3-3 (i.e., HPI/MU nozzle, pressurizer surge nozzle, and hot leg surge nozzle) are conservative and bounding when considering the entire RCS.

Stainless Steel (F_{en} maximum of 15.35 based on NUREG/CR-5704)

CR-3 RCS pressure boundary components with parts made from stainless steel include the 10-inch pressurizer surge line piping and attached branch connection, 2.5-inch pressurizer spray line piping and nozzle, 28-inch transition piping that connects the CS cold leg piping to the RCPs, RCPs, Class 1 portions of ancillary system piping and valves attached to the Class 1 components, and the control rod drive mechanism (CRDM) motor tube housing and extension. CUF evaluations were performed in accordance with USAS B31.7 for the pressurizer surge line, pressurizer spray line, and 28-inch transition piping. CUF evaluations were performed for the RCP in accordance with American Society of Mechanical Engineers (ASME) Code Section III; an exemption from fatigue was justified using ASME III for the CRDM motor tube housing and extension. An EAF need not be considered for this location. The Class 1 portions of ancillary system piping attached to Class 1 components at CR-3 are all designed in accordance with USAS B31.1 and do not have explicit CUF calculations, but consider thermal cycles using a stress range reduction factor.

Due to the conservative maximum environmental penalty for stainless steel, multiplication of design CUFs by the bounding F_{en} of 15.35 will, in nearly all instances, result in an EAF CUF greater than 1.0 for the stainless steel RCS pressure boundary items. With regard to the above items with CUFs, the items that are the most susceptible to EAF are locations with the highest thermal loadings over the life of the plant and the thinnest wall thickness as discussed above in the closeout to GSI-190 (i.e., for CR-3 the pressurizer spray line and the pressurizer surge line). The 10-inch pressurizer surge line and 2.5-inch pressurizer spray line and nozzle are more risk significant than the 28-inch transition piping and the RCPs since the probability of fatigue failure is higher due to thermal stratification. Both the 10-inch pressurizer surge line and 2.5-inch spray line and nozzle are within the scope of the CR-3 ASME Section XI risk-based inspection program.

The NUREG/CR-6260 stainless steel locations include the 10-inch pressurizer surge line and the Decay Heat Removal (DHR) injection tee (Class 1 ancillary piping). The 10-inch pressurizer surge line bounds the 2.5-inch spray line and nozzle relative to risk significance, and the DHR injection tee is one of the highest risk significant lines attached to the RCS. Therefore, the NUREG/CR-6260 stainless steel items evaluated by CR-3 in LRA Table 4.3-3 (i.e., pressurizer surge line and DHR injection tee) represent bounding locations when considering the entire RCS.

Nickel-Based Alloy Locations (F_{en} maximum of 4.52 applied to new design curve from NUREG-6909).

CR-3 RCS pressure boundary components with parts made from nickel-based alloy include the RV, RCS piping, OTSG, and pressurizer. The susceptible RV parts include the bottom mounted instrument nozzles (BMN-instrument are 3/4-inch Schedule 160) and CRDM nozzles. The susceptible RCS piping parts include instrumentation and vent branch connections (Nominal Pipe Size (NPS) ≤ 1-inch), and dissimilar metal welds that connect stainless steel clad RCS piping and branch connections to attached stainless steel piping (e.g., hot leg surge, decay heat

drop line, HPI/MU, letdown, and instrumentation and vent). The susceptible OTSG parts include mechanical sleeves and plugs. The susceptible pressurizer parts include instrumentation and vent nozzles (NPS ≤ 1.5-inch), spray nozzle safe end (4-inch), and the dissimilar metal weld that connects the pressurizer surge nozzle (10-inch) to the stainless steel safe end.

Due to the conservative maximum environmental penalty for nickel-based alloy, multiplication of the design CUFs by the bounding F_{en} of 4.52 will in nearly all instances result in an EAF CUF > 1.0 for nickel-based alloy RCS pressure boundary items. However, for the nickel-based alloy items, the predominant aging effect requiring aging management is primary water stress corrosion cracking (PWSCC); and all of the above items are included in the CR-3 Alloy 600 aging management program. Mitigation of PWSCC for dissimilar metal welds typically includes full structural weld overlay for connections greater than 1-inch NPS, thus moving the pressure boundary from the inside of the pipe to the outside and rendering it not susceptible to EAF. Therefore, the most risk significant items for EAF include the nozzles attached to the RV (i.e., CRDMs and BMN-instrument). The CRDM nozzles were replaced with replacement of the RV closure head in 2003, and the most susceptible and risk significant EAF location for the RV is the BMN-instrument nozzle.

The NUREG/CR-6260 nickel-based alloy locations include the ¾-inch Schedule 160 RV BMN-instrument nozzle, the hot leg surge branch connection dissimilar metal weld, the dissimilar metal weld that connects the HPI/MU branch connection to the stainless steel safe end, and the pressurizer surge nozzle dissimilar metal weld. These locations bound all remaining nickel-based alloy locations for CR-3 as discussed above. Therefore, the NUREG/CR-6260 nickel-based alloy items evaluated by CR-3 in LRA Table 4.3-3 represent bounding locations when considering the entire RCS.

Request

b) Confirm and justify that the locations selected for environmentally-assisted fatigue analyses in LRA Table 4.3-3 consists of the most limiting locations for Crystal River Unit 3 Nuclear Generating Plant (beyond the generic components identified in the NUREG/CR-6260 guidance). If these locations are not bounding, clarify the locations that require an environmentally-assisted fatigue analysis and the actions that will be taken for these additional locations. If the limiting location identified consists of nickel-alloy, state whether the methodology used to perform the environmentally assisted fatigue calculation for nickel alloy is consistent with NUREG/CR-6909. If not, justify the method chosen.

Response:

As discussed in the response to a) above, the generic NUREG-6260 locations represent bounding locations for CR-3 based on consideration of material type and a qualitative assessment of risk. By conservatively assuming that the existing 40-year design CUFs represent a reasonable assessment of probability of failure (defined in this case as fatigue cracking which may result in through-wall leakage), EAF-susceptible locations may be identified for each RCS pressure boundary subcomponent identified in Table 4.3-2 of the CR-3 LRA by multiplying design CUFs by bounding F_{en}s. The use of 40-year design CUFs is appropriate in this evaluation since CR-3 has determined that the NSSS design cycles used to calculate 40-year design CUFs will not be exceeded at 60 years (Reference response to NRC RAI 4.3.2.1-1

in PEF letter to NRC 3F1009-07, dated October 13, 2009, ML092890155). Locations with EAF CUFs > 1.0 (calculated by multiplying design CUFs by bounding F_{en}s) represent locations that warrant additional consideration for the potential for fatigue cracking due to EAF.

Reactor Vessel

RV items with bounding EAF CUFs > 1.0 include the nickel-based alloy CRDM and BMN-instrument nozzles and the stainless steel clad low alloy steel RV outlet nozzle. The BMN-instrument and RV outlet nozzles are included as NUREG/CR-6260 locations in Table 4.3-3 of the CR-3 LRA. The BMN-instrument nozzle was evaluated using NUREG/CR-6909 per CR-3 response to RAI 4.3.3-5 (Reference PEF letter to NRC 3F0610-02, dated June 21, 2010, ML101740057). The CRDM nozzles were replaced in 2003 with the replacement of the RV closure head.

CRDMs

The CRDM subcomponents meet the exemption from fatigue requirements of ASME Section III. EAF is not applicable to the CRDMs.

OTSG

OTSG primary pressure boundary items with bounding EAF CUFs > 1.0 include the nickel-based alloy mechanical sleeves and welded plugs. These items were not evaluated in NUREG/CR-6260.

Pressurizer

The pressurizer item with a bounding EAF CUF > 1.0 includes the nickel-based alloy pressurizer thermowell nozzle. This item was not evaluated in NUREG/CR-6260.

RCPs

RCP items with bounding EAF CUFs > 1.0 include the pump casing and pump cover. These items were not evaluated in NUREG/CR-6260.

RCS Piping (ASME Section XI IWB boundary)

RCS piping within the ASME Section XI IWB inspection boundary with EAF CUFs > 1.0 include the following stainless steel items: pressurizer surge line piping, pressurizer spray line piping, pressurizer spray line cold leg nozzle, HPI/MU safe end and spool piece. The pressurizer surge line piping and HPI/MU nozzle/ safe end were evaluated in NUREG/CR-6260.

Summary of RCS Pressure Boundary Parts with EAF CUFs > 1.0

Based on the discussion above, all RCS pressure boundary parts made from stainless steel clad low alloy steel with EAF CUFs > 1.0 are included as NUREG/CR-6260 locations.

RCS pressure boundary parts made from stainless steel with EAF CUFs > 1.0 that are not included as NUREG/CR-6260 locations include the RCP casing and cover, pressurizer spray line piping, and pressurizer spray line cold leg nozzle. These locations are all bounded by the

NUREG/CR-6260 pressurizer surge line location relative to the probability of leakage due to thermal stratification and fatigue.

RCS pressure boundary parts made from nickel-based alloy with EAF CUFs > 1.0 that are not included as NUREG/CR-6260 locations include the CRDM nozzles, OTSG mechanical plugs, OTSG mechanical sleeves, and the pressurizer thermowell nozzle (1.5-inch outside diameter). The CRDMs were replaced in 2003, are not in a fatigue sensitive location, and are bounded by the BMN-instrument nozzles owing to thermal and mechanical loads on the BMN-instrument nozzles. The OTSG items (mechanical plugs and sleeves) are no longer applicable since the steam generators were replaced during the current refueling outage. The pressurizer thermowell nozzle is located above the pressurizer heater elements and is not in a fatigue sensitive location. In addition, this nozzle is susceptible to PWSCC and is included in the Alloy 600 aging management program. Therefore, the NUREG/CR-6260 BMN-instrument nozzle bounds the other EAF susceptible nickel-based alloy locations for CR-3.

Consistent with the requirements of 10 CFR 54.21(b), CR-3 will provide the NRC with an amendment to the LRA identifying changes to the facility performed during the current refueling outage, including OTSG replacement and Alloy 600 mitigation activities, that materially affect the information in the LRA.

Based on the preceding discussion, the locations selected for environmentally-assisted fatigue analyses in LRA Table 4.3-3 consist of the most limiting locations for CR-3, including locations beyond the generic components identified in the NUREG/CR-6260 guidance.

Supplemental Response to RAI B.2.26-1 provided in PEF letter 3F1110-03, dated November 10, 2010:

Enclosure 1 to PEF letter 3F1110-03, dated November 23, 2010 (ML103280373), stated the percentage of tendon forecast values above the minimum required values will change at the end of the period of extended operation as a result of the repair to the Containment Building wall concrete delamination. Although the percentage of forecast values above the minimum required values at the end of the period of extended operation will change, the Concrete Containment Tendon Prestress Program is in place and will maintain the tendon prestress above the minimum required through the next tendon surveillance and to the end of the period of extended operation. This response completes the update to RAI B.2.26-1.

Based on the above information and the information regarding the Concrete Containment Tendon Prestress Program provided in PEF letters 3F1210-03, dated December 8, 2010 (ML103470140), (provided in response to RAI 4.5.1-1) and 3F1210-06, dated December 16, 2010, the updates to LRA Subsection 4.5.1 and the responses to RAI 4.5-1 and RAI B.2.26-1 indicated in PEF letter 3F1110-03 dated November 23, 2010, have been completed.

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50 - 302 / LICENSE NUMBER DPR - 72

ENCLOSURE 2

AMENDMENT 17 CHANGES TO THE LICENSE RENEWAL APPLICATION

Amendment 17 Changes to the License Renewal Application

Source of Change	License Renewal Application Amendment 17 Changes
RAI B.2.29-1	Revise the second paragraph of LRA Subsection A.1.1.29, on Page A-16 to read: Prior to the period of extended operation, Program administrative controls will be enhanced to (1) identify the structures that have masonry walls in the scope of License Renewal, (2) include inspection of the masonry walls in the Machine Shop in a periodic engineering activity, and (3) require periodic inspection of masonry walls every five years. Add an enhancement to LRA Subsection B.2.29, on Page B-87, by adding a new program element for Detection of Aging Effects as follows: • Detection of Aging Effects Revise program administrative controls to require periodic inspection of masonry walls every five years. Based on this change, License Renewal Commitment #19 has been revised to read: Program administrative controls will be enhanced to (1) identify the structures that have masonry walls in the scope of License Renewal, (2) include inspection of the masonry walls in the Machine Shop in a periodic engineering activity (PMID), and 3) require periodic inspection of masonry walls every five years.
RAI B.2.30-6	Revise the second paragraph of LRA Subsection A.1.1.30, on Page A-17 to add enhancements (13) and (14) to the administrative controls that implement the Program as follows: ~~and~~ (12) require periodic inspection of structures on a frequency of at least once every five years. (13) include the quantitative acceptance criteria of ACI 349.3R Chapter 5, and (14) perform a baseline inspection using the quantitative acceptance criteria of ACI 349.3R prior to the period of extended operation. Add a program element for Acceptance Criteria in LRA Subsection B.2.30, on page B-91 immediately before Operating Experience to read: • Acceptance Criteria 1) Revise Program administrative controls to include the quantitative acceptance criteria of ACI 349.3R, Chapter 5. 2) Perform a baseline inspection using the quantitative acceptance criteria of ACI 349.3R prior to the period of extended operation. Based on this change, revise License Renewal Commitment #20 to add items (13) and (14) as follows: (13) include the quantitative acceptance criteria of ACI 349.3R, Chapter 5, and (14) perform a baseline inspection using the quantitative acceptance criteria of ACI 349.3R prior to the period of extended operation.