Progress Energy

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

Ref: 10 CFR 50.71(e)
10 CFR 50.59(d)(2)
10 CFR 50.4(c)

May 21, 2012
3F0512-06

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

Subject: Crystal River Unit 3 – Final Safety Analysis Report, Revision 33 and 10 CFR 50.59 Report

Reference: CR-3 to NRC Letter, 3F0510-04, dated May 27, 2010, "Crystal River Unit 3 – Final Safety Analysis Report, Revision 32 and 10 CFR 50.59 Report"

Dear Sir:

In accordance with 10 CFR 50.71(e), Florida Power Corporation (FPC), doing business as Progress Energy Florida, Inc., hereby submits Revision 33 to the Crystal River Unit 3 (CR-3) Final Safety Analysis Report (FSAR). One CD-ROM is enclosed for the Document Control Desk and one CD-ROM copy is being sent to the Regional Administrator (Region II). The CR-3 Project Manager will also receive one CD-ROM. The Senior Resident Inspector has received a CD-ROM via FPC's internal controlled document distribution system.

This revision replaces FSAR, Revision 32 (Reference), in its entirety. FSAR text changes are indicated by revision bars on the outside right border of each page.

This FSAR revision includes material which describes the organization, the modifications, flow diagrams, and changes to CR-3 that have been implemented as of April 2, 2012. As required by 10 CFR 50.71(e), a summary of changes made in FSAR, Revision 33, is provided in Attachment A.

Additionally, as required by 10 CFR 50.59(d)(2), Attachment B includes a summary of each 10 CFR 50.59 evaluation completed during this submittal period, with the exception of evaluations associated with changes, tests, or experiments that have not been fully implemented. Many of the completed 10 CFR 50.59 evaluations associated with plant modifications required multiple revisions. The final 10CFR 50.59 evaluation is being reported due to the cumulative nature of the changes made to the modification packages.

There are no new commitments made within this letter.

A053
NRR

If you have any questions regarding this submittal, please contact Mr. Dan Westcott, Superintendent, Licensing and Regulatory Programs, at (352) 563-4796.

Sincerely,

Jon A. Franke
Vice President
Crystal River Nuclear Plant

JAF/par

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

Attachments: A. FSAR Revision 33 Change Summary
B. 10 CFR 50.59 Evaluation Summaries

Enclosure: FSAR, Revision 33 on CD-ROM

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 31 day of May, 2012, 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 ______

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 /LICENSE NUMBER DPR-72

ATTACHMENT A

FSAR REVISION 33 CHANGE SUMMARY

FSAR REVISION 33 CHANGE SUMMARY

This Final Safety Analysis Report (FSAR) revision reflects plant modifications, information, and analyses that constitute changes to the FSAR since the publication of FSAR Revision 32. No changes to the FSAR have been made as a result of License Amendments. This FSAR revision includes changes made to incorporate the following:

A. Changes to plant engineering, programs and revisions to analyses:

- FSAR Change Package 2008-12: This change is based upon the plant modification that installed a digital, intelligent, and addressable fire detection, alarm and control system for the Auxiliary Building, Control Complex, Intermediate Building, Turbine Building and Rusty Building. In addition, monitoring modules reporting to the new fire alarm system will monitor the other plant site fire detection and alarm devices. These include the Reactor Building (RB), the Emergency Feedwater Pump Building, the Technical Support Center, administrative buildings, warehouses, and other panels outside the power block. The Engineering Design Package replaces the existing system in compliance with the National Fire Protection Association (NFPA) 72 - 2007. This change affects Section 9.

- FSAR Change Package 2009-20: This change is based upon an Engineering Change (EC) to the primary metrological tower that is described by the metrological program. The primary metrological tower has been relocated and replaced. The equipment functions remain the same and specifications related to the removed tower were eliminated. New tower information was incorporated into Section 2 and Table 2-16.

- FSAR Change Package 2010-02: This change incorporated information pertaining to main steam line break events and assumptions for the RB pressure analysis of steam line rupture in Section 6. This change was determined necessary from Cycle 17 reload design and safety analysis. Details no longer applicable were removed.

- FSAR Change Package 2010-07: This change included the turbine bypass valve replacements for the Crystal River Unit 3 (CR-3) Extended Power Uprate (EPU) that allow for higher capacity control and steam flow capacity for Cycle 17 startup steam flow and pressure conditions. Details that are no longer applicable were removed with the incorporation of new flow and parameters in Section 10.

- FSAR Change Package 2010-08: This change is based upon the installation of new secondary closed cycle cooling pumps to gain additional flow and total discharge head margin which is required as a part of the CR-3 EPU. This change removed information pertaining to the pumps that are no longer installed and incorporated flow and hydraulic parameters for new pumps in Table 9-12 and Figure 9-9.

- FSAR Change Package 2010-09: This change is based upon the reduction in the maximum emergency feedwater flow rate within the system design bases that is associated with new style Once Through Steam Generators (OTSGs) installed in Refuel Outage 16 (R16). Engineering analysis of information that is no longer applicable has been removed with the incorporation of new flow requirements in Section 10.

- FSAR Change Package 2010-10: This change is based upon the OTSG replacement in R16 to reflect changes in the steam line failure accident method of analysis and the structural integrity description and to remove information no longer applicable. An additional reference was added to Section 14 and the Section 14 description of the steam generator structural integrity was modified by this change.

- FSAR Change Package 2010-11: This change is based upon the OTSG replacement in R16 that resulted in an increase in the low level Emergency Feedwater initiation setpoint by two inches. The change removed the former setpoint value in Table 14-63 and added the current setpoint value.

- FSAR Change Package 2010-12: This change updated the OTSG design parameter and Reactor Coolant System (RCS) parameters within Table 1-1, based upon the OTSG replacement in R16.

- FSAR Change Package 2010-13: This change is based upon the RC hot leg replacement and OTSG replacement in R16. Information related to RCS component codes, steam generator design data, piping design, materials, transient cycle numbers and inspection criteria that is no longer applicable in Section 4 tables have been removed with the incorporation of specifications associated with the new components.

- FSAR Change Package 2010-14: This change is based upon the RC hot leg replacement and OTSG replacement in R16. Information related to the RCS components in Section 4, such as stress evaluation and structural differences, has been modified and information removed that is no longer applicable. The new OTSGs and supporting component characteristics, such as tubes installed, structural supports and piping dimensions, are incorporated into Section 4. Changes to Figures 4-2 and 4-5 that illustrate the RCS and OTSG outline have been included with this change also.

- FSAR Change Package 2010-16: This change was made to correct information in Sections 1 and 8 that identifies a grid line that has been renamed.

- FSAR Change Package 2010-17: This change was made to show a change in the Engineering organization structure based upon Nuclear Generation Group changes. The change modified the position titles and reporting structure under the Vice President of Nuclear Engineering in Section 1 and Figure 1-26.

- FSAR Change Package 2010-18: This change includes an Audit program description change to improve audit schedule consistency in Table 1-3.

- FSAR Change Package 2010-19: This change corrected the transient cycle information pertaining to hydro testing of the RCS and OTSGs based upon a historical error identified in Table 4-8. This change package also modified the 'paint schedule for equipment and structures inside the reactor building' in Table 5-7 based upon OTSG replacement. Information no longer applicable has been removed.

- FSAR Change Package 2010-20: This change was made to reflect a change in the Document Services and Plant Support organizations within Section 1 and Table 1-3, based upon responsibility, reporting structure, and title changes.

- FSAR Change Package 2010-25: This change included a modification to the RCS Zinc Program that allows for zinc addition to the RCS in Modes 3 through 1 within the new OTSGs. This change was made to limit the corrosion rate and reduce the corrosion product release during the passivation process of the Alloy 690TT tube material within the new OTSGs. This change affects Section 9 and Table 4-10.

- FSAR Change Package 2010-28: This change is based upon a modification to the acceptance criteria for the Intake Canal Survey. Criteria was established to meet the minimum water level in the intake canal necessary during probable maximum hurricane blowout conditions. An engineering calculation was completed to assure adequate submergence of Nuclear Services and Decay Heat Sea Water pumps with the application of the criteria changes. This change affects Section 2, Section 9 and Figures 2-61 and 2-62.

- FSAR Change Package 2010-29: This change reduced the minimum service water (SW) pressure value, identified in Sections 6 and 9, to state the SW system pressure must be maintained above the RB peak accident pressure to prevent in leakage during an accident. This description is consistent with assumptions included in the calculation for the minimum SW tank 1 (SWT-1) nitrogen overpressure.

- FSAR Change Package 2011-01: This change incorporated the organization changes through the modification of the Quality Assurance Program description. Various titles were relocated and updated in Training, Support Services, Nuclear Engineering, Nuclear Oversight, Nuclear Upgrades, Operations and Nuclear Protective Services. Financial Services functions were removed from the Support Services department description. This change affects Section 1 and Figure 1-26.

- FSAR Change Package 2011-02: This change added the apex elevation in the dome interior to the RB drawings based upon the observed, as built conditions. This change affects Figures 11-7 and 11-8.

- FSAR Change Package 2011-05: This change implemented new Integrated Control System scales as described in the steam generator control summary for the increased thermal power rating, the increase in main feedwater flow, and the decrease in feedwater temperature based upon EPU. This change allows for removal of scales no longer applicable and incorporated new scales in Section 7 and Table 7-12.

- FSAR Change Package 2011-10: This change included the organizational oversight changes associated with the CR-3 containment repair project management. This change affects Section 1 and Figure 1-26.

- FSAR Change Package 2011-11: This change included the addition of two buildings in the CR-3 site plan. The two Central Alarm Station buildings (CAS and CAS2) contain their own dedicated FM-200 (heptafluoropropane) fire suppression systems. The new fire suppression system and building reference additions are identified in Section 9.

- FSAR Change Package 2011-15: This change is a correction to dimensions shown in a figure that illustrate the delineation of backfill materials placed against in-situ foundation material. This change removed detail errors and added the accurate dimensions in Figure 2-49.

- FSAR Change Package 2012-05: This change is an addition to Quality Program Commitments that affected the quality control testing process for splices in concrete reinforcement used for safety related concrete construction or repair activities. The change allows for the testing of sister splices only, instead of the testing of production and sister splices. The sample frequency for testing of sister splices has been added in Table 1-3.

B. Numerous editorial and clarification changes were made throughout the document. Each change was evaluated for 10 CFR 50.59 applicability utilizing Progress Energy Florida procedures:

- FSAR Change Package 2010-22: This change was made to reflect an editorial change to improve the accuracy of SECY-92-223, "Resolution of Deviations Identified During the Systematic Evaluation Program" applicability to CR-3 principle architectural design criteria based upon the issuance date of the CR-3 construction permit. The change has been included in Section 1.

- FSAR Change Package 2011-16: This change incorporated corrections to numerous typographical errors and inconsistencies identified during the CR-3 License Renewal application process related to accident diffusion estimates, hydrology, seismic hazard evaluation, and the RCS. The editorial corrections and clarifications affect Section 2, Section 4 and Table 4-8.

- FSAR Change Package 2012-01: This change is an editorial correction that improved an organization description with a title change from Document Services to Document Management. This change affects Sections 1 and 12 and Table 1-3.

- FSAR Change Package 2012-03: This change corrected a figure to show the removal of a piping line between the decay heat and core injection header within the RB. The piping line removal is not associated with a physical change. This change affects Figure 6-1.

- FSAR Change Package 2012-04: This change is an editorial correction that modified the Fuel Handling Accident acceptance criteria wording. This change affects Section 14.

C. Some changes are noteworthy because they involve removal of information from the FSAR. These changes were evaluated per the guidance of Nuclear Energy Institute (NEI) 98-03, Revision 1, and were determined to be appropriate:

- FSAR Change Package 2009-08: This change is based upon a modification that eliminated the 'loss of trip power' trip function from the main feedwater pump control scheme. The solenoid valve that trips upon the loss of trip power condition has been replaced with a voltage sensing relay that provides the local indication and control room annunciation in a power failure condition. The new indication and control room annunciation results in the initiation of necessary manual actions. This change affects Section 10.

- FSAR Change Package 2009-24: This change removed a study reference that is no longer relied upon to evaluate tendon redundancy in the RB dome during design accident loading conditions. This study reference is considered historical information that is no longer used for tendon analysis. This change affects Section 5.

- FSAR Change Package 2010-21: This change removed the spent fuel coolant pump air supply fans (AHF-8A and AHF-8B) based upon an Engineering Change package that evaluated the spent fuel cooling pump motor operation requirements and a change in the location of heat generation postulated from steam line breaks since the time of fan installation. Equipment requirements are maintained with the removal of the fans and this change results in a reduction in emergency diesel generator loading following a Loss of Offsite Power event. This change affects Section 9.

- FSAR Change Package 2011-09: This change is based upon a description error in the Reactor Protection criteria considered for the Control Rod Ejection (RCE) Accident that describes RCE over-pressure acceptance criterion. The RCS pressure response to a control rod ejection accident is not part of the CR-3 licensing basis. This change removes RCS pressure from the acceptance criterion and maintains the acceptance criterion related to deformation or rupture. This change affects Section 14.

- FSAR Change Package 2011-13: This change is an editorial correction that removed unnecessary wording within the Quality Program description in Section 1.

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 /LICENSE NUMBER DPR-72

ATTACHMENT B

10 CFR 50.59 EVALUATION SUMMARIES

Table of 10 CFR 50.59 Evaluations

ID Number	Title
AR 368093	Containment Opening for Steam Generator Replacement (EC 63016)
AR 416940	Reactor Building Delamination Repair Phase 4, Concrete Replacement (EC 75220 & 75221)
AR 434859	Intake Canal Evaluation for Alternate Acceptance Criteria (EC 79107)

ID Number: AR 368093

Title: Containment Opening for Steam Generator Replacement (EC 63016)

Summary and Conclusions

Description

A temporary access opening (Opening) will be created through the reactor building containment shell (Shell) during the Steam Generator Replacement (SGR) outage to facilitate transportation of the steam generators into and out of the reactor building (RB).

The Opening will be a penetration through the post-tensioned, reinforced concrete wall and interior steel liner plate of the Shell. Creation of the Opening will commence in Mode 5 (Cold Shutdown) with detensioning and removal of the hoop and vertical tendons within the Opening boundary followed by hydro-demolition of the concrete. After the reactor is defueled (No Mode) the exposed steel liner plate will be cut and removed to complete the Opening penetration.

Creation of the Opening will require the removal of concrete, rebar, tendons, tendon sheathes, and liner plate within the Opening boundary. Creation of the Opening will be accomplished in several phases defined by plant Mode, Opening status, and Shell pre stress level.

EC 63016 will be closed with the containment not restored to its design basis condition. Restoration will proceed to the point where the steel liner plate is welded into place and inspected, but prior to the replacement of reinforcing steel, tendons and concrete. The containment is being left in this condition because it was discovered during creation of the opening that a large section of the containment in the vicinity of the opening was delaminated. The delamination is located in a curved plane at the general location of the horizontal or hoop tendons. The repair of this condition and restoration of the containment to full design basis qualification will be accomplished by the following ECs:

- 75000, Crack Arrest
- 75218, Detensioning
- 75219, Concrete Removal
- 75220, Concrete Replacement, and
- 75221, Re-tensioning

Conclusion

Based on the findings of this evaluation, it is concluded that the proposed activity does not require a License Amendment.

ID Number: AR 416940

Title: Reactor Building Delamination Repair Phase 4, Concrete Replacement (EC 75220 & EC 75221)

Summary and Conclusions

Description

In the process of creating a construction opening in the Crystal River Unit 3 (CR-3) containment building to facilitate replacement of the steam generators, an area of delaminated concrete was discovered in the concrete shell. The delamination was roughly 10 in. from the outer surface of the 42 in. thick reinforced, post-tensioned containment shell. The containment is a cylindrical structure with a flat base and a shallow domed roof. The structure is post-tensioned by a matrix of horizontal (or hoop), vertical, and dome tendons. The cylinder is constructed with six buttresses located at 60° intervals around the outside to provide anchor points for the horizontal tendons. Horizontal tendons span 120° of arc, passing through three buttresses. Buttress 1 is located at 0° which corresponds with compass North and are numbered in the counterclockwise direction from 1 to 6. Azimuth positions around the containment from 0° to 359° also increase in the counterclockwise direction. Vertical tendons run from the top of the ring girder at the top of the containment cylinder to the bottom of the base mat.

Investigations revealed the extent of the delamination was limited to one 60° segment of the cylindrical containment, between buttresses 3 and 4 centered on azimuth 150° and between elevations 158 ft and 240 ft. Impact Response non-destructive examination validated by core samples confirmed that delamination did not exist in other portions of the concrete shell. A series of activities were conducted to remove the delaminated concrete and prepare the newly exposed concrete and steel liner surfaces for restoration.

Engineering Change (EC) 75220 prepared the CR-3 containment shell for return to full qualification by restoring the construction opening and removed delaminated concrete with replacement materials. A Quality Program change was implemented in the containment restoration to test only sister splices in vertical reinforcement at the bottom edge of the concrete removal area. The proximity of the splices to the concrete precluded testing and remaking production splices. This change was made in accordance with 10 CFR 50.54(a)(3)(ii).

EC 75221 completed the qualification of the containment shell by retensioning the replaced and detensioned horizontal tendons and retensioning all vertical tendons in the post-tensioned containment structure. These two ECs combine to restore containment to its design basis capability to meet all normal operating, accident, and event load cases. These two ECs are being evaluated together as a single activity termed Containment Restoration.

The Screen for Containment Restoration yielded 'yes' responses as follows that were evaluated in this 50.59 Evaluation:

- Minor cracks in the existing concrete in Bay 3-4 were left as is when the containment concrete was replaced. This change was evaluated in accordance with Regulatory Issue Summary (RIS) 2005-20, Revision 1, "Operability Determinations & Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality or Safety."
- Hairline cracks in the existing concrete outside Bay 3-4 were left as is. This change is also evaluated in accordance with RIS 2005-20, Revision 1.
- Due to cracks in existing concrete outside Bay 3-4, design calculations did not credit the tensile capacity of the existing concrete. This is a change to an input to the FSAR described calculation methodology for the containment.
- An ANSYS finite element model was used to evaluate the response of the restored containment to the FSAR described design load combinations. Use of this model that is not part of the current Licensing Basis is considered revising an FSAR described evaluation methodology.

Also evaluated herein is increasing the minimum end of life tendon tension and the minimum concrete compressive strength in design analyses. These changes were accepted as not having an adverse effect on the containment but are evaluated here under criteria 50.59(c)(2)(vii) & (viii) as design basis limits for a fission product barrier and as elements of a method of evaluation.

Criterion (i) - more than a minimal increase in the frequency of occurrence of an accident

No accident evaluated in the FSAR is initiated by the containment. The containment is a passive safety barrier that cannot be an initiator of an accident. Therefore, restoration of the containment following creation of a construction opening and replacement of delaminated concrete cannot increase the frequency of occurrence of any previously evaluated accident.

Criterion (ii) - more than a minimal increase in the likelihood of occurrence of a malfunction

Containment restoration has been performed in accordance with approved practices for construction, quality assurance (QA) and quality control (QC). Repair activities have been conducted in accordance with the CR-3 Appendix B Quality Program and with support from contract engineering and construction firms applying their Appendix B Programs. Analyses of the restored containment have been performed using widely accepted analytical tools under software QA requirements. The applicable American Concrete Institute (ACI) Codes specified in the FSAR have been satisfied by the restoration design, installation has been implemented in accordance with approved construction practices, and quality oversight has verified the implementation. Application of these practices to the restoration ensures that the restored containment will perform with not more than a minimal increase in the probability of malfunction. Cracking of the concrete in various areas of the containment shell was evaluated for use as is and was determined to be acceptable and will not result in more than a minimal increase in the probability of malfunction of the containment.

Criterion (iii) - more than a minimal increase in the consequences of an accident

The changes made in restoring the containment concrete shell will not change the consequences of accident analyses that rely on the containment to perform as a barrier to the release of radioactivity. The restored containment will satisfy all applicable code requirements and continue to perform its design function and provide both shielding from radioactivity released to the containment, and to prevent escape of radioactivity in excess of the allowed accident leakage rate. A Structural Integrity Test will be performed to verify the integrity of the containment building. The test will permit verification that the structural response of the principal strength elements is consistent with design. In addition, an integrated leak rate test will be conducted to verify that the accident leak rate limit is not exceeded.

Criterion (iv) - more than a minimal increase in the consequences of a malfunction

Malfunction of the containment was not evaluated in the FSAR as contributing to the consequences of any accident. The containment is a passive barrier that does not contribute to the malfunction of any other structure, system, or component important to safety.

Criterion (v) – possibility of an accident of a different type

The containment will be restored to its design basis functional capability using processes and materials that conform to existing codes and standards. Its structural integrity will be assured by appropriate analyses, calculations, and testing. It is a passive safety barrier that cannot create an accident.

Criterion (vi) – create a possibility for a malfunction of an SSC important to safety with a different result

Malfunction of the containment was not evaluated in the FSAR. The containment is a passive barrier that does not contribute to the malfunction of any other structure, system, or component important to safety. Its structural integrity has been assured by appropriate analyses and calculations. These calculations include consideration for vertical cracks outside of the delamination repair area and their effect on concrete tensile capacity. The reduced tensile capacity of the concrete is being compensated for by increased tendon tension for the remaining plant life to ensure that design basis load cases will continue to be satisfied.

Criterion (vii) – result in a design basis limit for a fission product barrier being exceeded or altered

Increasing the minimum end of life tendon tension and the minimum concrete compressive strength in design analyses were evaluated as possibly being design basis limits for a fission product barrier. In both cases the determination was made that these parameters are not design basis limits for a fission product barrier, but are physical characteristics of containment components and inputs to calculations that determine the repaired containment satisfies the design basis limit of 55 psig internal pressure.

Criterion (viii) - result in a departure from a method of evaluation

In the screening for this activity, the proposed use of an ANSYS finite element model of the containment was assessed as "changing an element..." of the current evaluation methodology. As such the standard for evaluation is if the revised element provides results that are conservative or essentially the same as the existing evaluation methodology. Two benchmark calculations were prepared which successfully demonstrated that the results from the ANSYS model are conservative or essentially the same. On that basis the use of the ANSYS finite element model is not a departure from a method of evaluation described in the FSAR.

The design basis for the CR3 containment is that for factored loads, the containment wall tensile stress is limited to 212 psi. For sections of the containment wall with vertical cracks, the concrete will not carry a tensile stress for design loads. In these areas, the more conservative hoop tensile stress limit of 0 psi will be applied. This is a change to an input to the evaluation methodology, and not a change to the methodology itself.

Increasing the minimum end of life tendon tension and the minimum concrete compressive strength in design analyses were evaluated as possibly being elements of the method of evaluation versus being inputs to the method of evaluation. Based on applying the guidance from NEI 96-07, minimum end of life tendon tension and minimum concrete compressive strength are inputs to the method of evaluation, not elements of the method, and do not represent a departure from a method of evaluation described in the FSAR.

Conclusion

This Evaluation determines that these activities are acceptable for implementation under 10 CFR 50.59, and no License Amendment is required.

ID Number: AR 434859

Title: Intake Canal Evaluation for Alternate Acceptance Criteria (EC 79107)

Summary and Conclusions

Description

EC 79107 evaluates alternate acceptance criteria for the intake canal survey (PT-501). The alternate acceptance criteria will also be added to Commitment System 62177, and applicable sections of the FSAR.

During the extended R16 outage, lower flow rates in the canal resulted in sedimentation near the CR3 extension that did not pass an informational partial PT-501 survey and commitment criteria. After review of the minimum hurricane tide bases and FSAR, it was determined that the condition near the plant was acceptable due to the existence of additional margin in the rest of the canal. Since 1981, the canal has been maintained to a maximum of 68 feet to allow for larger coal barges. Because the setdown slope varies inversely with the canal depth, the condition the canal has been maintained at would result in higher water levels at the intake. EC 79107 evaluated acceptable dimensions east of the barge turning basin based on a canal bottom elevation of 72 feet west of the turning basin. This evaluation reproduced the results of the original methodology as stated in the original analyses and FSAR.

Conclusion

The results of this Evaluation conclude that there is no License Amendment required to implement this change.