DUKE ENERGY.

Crystal River Nuclear Plant
15760 W. Power Line Street
Crystal River, FL 34428

Docket 50-302
Operating License No. DPR-72

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

May 8, 2014
3F0514-02

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

Subject: Crystal River Unit 3 – Final Safety Analysis Report, Revision 34 and 10 CFR 50.59 Report

Reference: CR-3 to NRC letter, 3F0512-06, dated May 21, 2012, "Crystal River Unit 3 – Final Safety Analysis Report, Revision 33 and 10 CFR 50.59 Report," (ADAMS Accession No. ML12151A234)

Dear Sir:

In accordance with 10 CFR 50.71(e), Duke Energy Florida, Inc. (DEF), hereby submits Revision 34 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 I). The CR-3 Project Manager will also receive one CD-ROM. This revision replaces FSAR, Revision 33 (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, modifications, flow diagrams, and changes to CR-3 that have been implemented as of February 11, 2014. As required by10 CFR 50.71(e), a summary of changes made in FSAR, Revision 34, 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. Some of the completed 10 CFR 50.59 evaluations associated with plant modifications required multiple revisions. The final 10 CFR 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.

If you have any questions regarding this submittal, please contact Mr. Dan Westcott, Regulatory Affairs Manager at (352) 563-4796.

A053
NRR

I declare under penalty of perjury that the foregoing is true and correct. Executed on May 8, 2014.

Sincerely,

John Elnitsky, Vice President
Project Management and Construction

JE/pes

Attachments: A. FSAR Revision 34 Change Summary Description
B. 10 CFR 50.59 Evaluation Summaries

Enclosure: FSAR, Revision 34 on CD-ROM

xc: Regional Administrator, Region I
NRR Project Manager

DUKE ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR-72

ATTACHMENT A

FSAR REVISION 34
CHANGE SUMMARY DESCRIPTION

FSAR REVISION 34 CHANGE SUMMARY DESCRIPTION

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 33. No material changes to the FSAR have been made as a result of License Amendments. After the implementation of Amendment No. 243 in December 2013, the operating license for Crystal River Unit 3 (CR-3) changed from Florida Power Corporation to Duke Energy Florida, Inc. As an editorial change, the company logo that was utilized by CR-3 was updated to reflected the name change. This FSAR revision includes changes made to incorporate the following:

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

- FSAR Change Package 2010-24: This change is based upon an Engineering Change (EC) that installed two manual selector switches on the Heating and Ventilating section of the Main Control Board (MCB). This change permits the bypassing of the permissive associated with the Control Complex (CC) Return Duct Radiation Monitor, within the control logic for the CC Normal Duty Supply Fans and the CC Return Air Fans. Since operating procedures define the equipment train(s) available for establishing a safe shutdown following a fire event, selector switches are required within the control logic for fans. The bypass mode for each selector switch will only be used for a safe shutdown fire event that has resulted in the loss of electrical power to the associated Vital Bus Distribution Panel (VBDP-4), where safe shutdown is performed from the Main Control Room. This change affected Section 9.

- FSAR Change Package 2011-14: This change was made to show revisions to the CR-3 organization structure based upon Nuclear Generation Group changes resulting from the merger of Duke Energy and Progress Energy. As a result of the merger, the Organizational Effectiveness Manager, Training Manager, and Station Manager report to the Plant Vice President, and the Site Engineering Director reports directly to the Senior Vice President – Engineering and is matrixed to the Plant Vice President. This change affected Section 1 and Figure 1-26.

- FSAR Change Package 2012-09: This change is based upon an EC that added a new monorail hoist (FHCR-3A) and a manual fuel assembly handling tool to the fuel handling bridge crane (FHCR-3) to enable loading fuel into the peripheral fuel assembly cell locations in the Spent Fuel (SF) pool and into the dry shielded canister (DSC) when placed in the cask pool. The pre-existing hoist was not capable of loading fuel into the peripheral cell locations because of travel limits for the trolley and interference of the mast with the pool liner. The pre-existing grapple for picking up a fuel assembly is larger in size than the fuel assembly cross sectional area and thus did not permit the lowering of a fuel assembly completely down into a DSC storage location. This change affected Section 5 and Section 9.

- FSAR Change Package 2012-13: This change was made as a result of revisions to the CR-3 Nuclear Engineering reporting arrangement based upon Nuclear Generation Group changes resulting from the merger of Duke Energy and Progress Energy. As a result of the merger, the Director/Manager of Plant Engineering now reports to the Nuclear Engineering Senior Vice President. This change affected Section 1.

- FSAR Change Package 2012-14: This change was made to reflect organizational changes in the Nuclear Generation Group that resulted from a number of senior management changes made following the merger of Duke Energy and Progress Energy. The Vice President, Supply Chain and Chief Procurement Officer provides procurement services for each nuclear plant and now reports to the Chairman, President and Chief Executive Officer through the Executive Vice President and Chief Financial Officer. This position supports Nuclear Generation through an interface agreement. Additional organization changes included revising the position title "President and Chief Executive Officer" to "Chairman, President and Chief Executive Officer" and revising the reporting relations of the Vice President, Supply Chain and Chief Procurement Officer from the "Executive Vice President and Chief Administrative Officer" to the "Executive Vice President and Chief Financial Officer." This change affected Section 1.

- FSAR Change Package 2012-16: This change is based upon CR-3's Containment Repair Project re-aligning under the Senior Vice President of Nuclear Operations, the Executive Vice President, and the Chief Nuclear Officer. This change affected Section 1 and Figure 1-26.

- FSAR Change Package 2012-17: This change was necessitated by two EC's that installed a new Ultra High Frequency (UHF) Radio System at CR-3. The previous radio equipment in use at CR-3 operated in the UHF range and faced issues with compliance to the Federal Communications Commission (FCC) Narrowband Technology statutes. The system had also experienced multiple problems due to limited manufacturer support of obsolete critical components. One EC installed the new UHF Radio System. That EC also installed a main antenna on the roof of the Turbine Building in a configuration that allowed the original and new systems to operate simultaneously until the new system could be verified operational. The original system was then physically removed. The original distributed antenna system was also modified to extend coverage into several identified areas where coverage did not previously exist. A separate EC installed the power feed for the new system. This change affected Section 7.

- FSAR Change Package 2013-04: This change is based upon organizational changes in the Nuclear Generation Group that resulted from a reorganization of the corporate executive team. Specifically, the Nuclear Generation Executive Vice President and Chief Nuclear Officer were changed to the Nuclear Generation Executive Vice President and President Duke Energy Nuclear. For CR-3 Quality Assurance functions, two positions will report to the Nuclear Generation Executive Vice President and President Duke Energy Nuclear: the Chief Nuclear Officer and the Vice President Crystal River Nuclear Plant. The Vice President Nuclear Oversight was also given access to the corporate executive team, including the Nuclear Generation Executive Vice President and President Duke Energy Nuclear, to resolve any quality or nuclear safety related concerns. This change affected Section 1 and Figure 1-26.

- FSAR Change Package 2013-06: This change was made as a result of the site transition to the CR-3 Decommissioning Transition Organization (DTO). The Nuclear Regulatory Commission letter (ADAMS Accession No. ML13058A380), dated March 13, 2013, acknowledged CR-3's certification of permanent cessation of power operation and permanent removal of all fuel from the reactor vessel. The letter also acknowledged that pursuant to 10 CFR 50.82(a)(2), the 10 CFR Part 50 license for CR-3 no longer

authorizes operation of the reactor or emplacement or retention of nuclear fuel in the reactor vessel. This change affected Section 1, Table 1-3, and Figure 1-26.

- FSAR Change Package 2013-07: This change was made to reflect CR-3's organizational structure following the site transition to the DTO. The new organizational structure included a number of newly created positions, job title changes, and revisions to various qualification and training requirements. Editorial changes were also made to correct previous errors. This change affected Section 1, Section 12, Table 1-3, and Figure 1-26.

- FSAR Change Package 2013-09: This change is based upon the impact of Duke Energy's decision to retire CR-3 on the design function of the containment building. Pursuant to 10 CFR 50.82(a)(2), the 10 CFR Part 50 license for CR-3 no longer authorizes operation of the reactor or emplacement or retention of nuclear fuel in the reactor vessel. As such, the containment is no longer a fission product barrier and the design function of the containment has been permanently altered. The previous classification and function of the containment for an operating plant are no longer relevant. Changes were made to Section 1, Section 5, Table 5-3, and Figures 5-1 to reflect the new classification and function of the containment for a permanently defueled unit. Specific changes included the physical description, structure classification, design basis load cases, and method of evaluation. Figures 5-5, 5-6, 5-7, 5-8, 5-9, 5-10, 5-11, 5-12, and 5-20 were deleted.

- FSAR Change Package 2013-13: This change is the result of a site decision to fully insert all nuclear incore detector assemblies into the reactor vessel despite the absence of nuclear fuel assemblies and the associated guide tubes the incore detectors normally travel within. The purpose of this decision was essentially to store the highly irradiated portion of the incore detector assemblies within the shielding provided by the reactor vessel and internals materials in order to eliminate a Very High Radiation Area underneath the reactor vessel and thus minimize radiation exposure to plant personnel engaged in the performance of duties within the containment. The incore detectors will remain fully inserted in the reactor vessel until a much later point in the plant decommissioning effort when the time comes to dispose of the reactor vessel and internals. The change revised the incore detector system description and system operation information contained in Section 7.

- FSAR Change Package 2013-16: This change is based upon an EC that replaced the Containment Purge Exhaust Duct airborne radioactivity monitor (RM-A1) and the Auxiliary Building and Fuel Handling Area Exhaust Duct airborne radioactivity monitor (RM-A2) with upgraded monitors that feature digital equipment versus the original analog equipment. Section 11, Figure 11-5, and Figure 11-6 were revised to reflect the differences in equipment design and functional capabilities.

- FSAR Change Package 2013-19: This change substituted Balance of Plant (BOP) sources of makeup inventory to the SF pools with the Fire Service (FS) tanks and FS hose stations. The BOP sources previously identified were the Condensate Demineralizers, the Condensate Storage Tank and the Condenser Hot Well to the SF pools via the Demineralized Water (DW) Supply System. The FS tanks represent a much larger source of makeup inventory for the SF pools than the BOP sources. Additionally, a review of the Condensate system interface with the DW system and SF

pools indicated a complex, circuitous pathway involving equipment in multiple locations, whereas FS hose stations are located on the SF deck immediately adjacent to the SF pools. Section 9 was impacted by this revised strategy for makeup inventory to the SF pools.

- FSAR Change Package 2013-23: This change revised the Fuel Handling Accident, eliminated the Waste Gas Decay Tank (WGDT) Rupture Accident, revised the description of the methodology for calculating atmospheric dispersion factors, and added a new radioactive waste handling event as the limiting radioactive waste event in terms of dose to the public. The Fuel Handling Accident was revised based on the reduced inventory of radioactivity in the nuclear fuel since the plant has been shut down for over 4 years, and atmospheric dispersion factors have subsequently been recalculated. The WGDT Rupture Accident is no longer credible since significant quantities of waste gases can only be produced due to reactor operation. Since permanently shutting down, the inventory in the Waste Gas Decay Tanks has been released per programmatic requirements and the tanks have been permanently reconfigured to preclude repressurization. A limitation was also added by this change regarding the storage locations of radioactive wastes. The new limitation ensures that primary resin storage is consistent with the assumptions of the current radioactive waste handling event analysis. This change affected Section 1, Section 2, Section 11, Section 14, Table 2-8, and Table 14-31. New table 11-16 was also added.

B. Numerous editorial and clarification changes were made throughout the document. Each change was evaluated for 10 CFR 50.59 applicability utilizing Crystal River Unit 3 and Duke Energy procedures:

- FSAR Change Package 2012-12: This change was made to clarify historical radiation zone information for certain areas inside the Reactor Building (RB) and Auxiliary Building. This information is historical and is associated with the design criteria for the original plant shielding. It is not used in any evaluation of actual plant radiological conditions. This change affected Figures 11-7, 11-8, 11-9, and 11-10.

- FSAR Change Package 2013-03: This change was an editorial correction that replaced the word "incident" with the word "accident" in two locations. In each case, accident condition pressure and temperature was referred to as "incident" pressure and temperature. This change affected Section 5.

- FSAR Change Package 2013-12: This change was made to clarify the flood protection function of the RB equipment hatch. The previous wording acknowledged the role of the water-tight door in affording protection against flood levels; however, the flood barrier capability provided by the RB equipment hatch missile shield was not clearly stated. This change affected Section 2.

- FSAR Change Package 2013-14: This change added clarifying statements at various locations to specify that plant accidents previously deleted, based on CR-3's permanent cessation of power operation and permanent removal of nuclear fuel from the reactor vessel, are no longer credible. This change affected Section 1, Section 3, Section 4, Section 5, Section 6, Section 7, Section 9, Section 10, Section 11, Table 3-32, Table 6-6, Table 6-9, Table 7-9, and Table 14-31.

- FSAR Change Package 2013-17: This change added information to clarify that the in-plant sump pumps are supplied from non-safety related power sources. Relevant component tag numbers were also added for various sump pump locations and several corrections were made to erroneous tabular information regarding the endpoint locations of pumped inventories. This change affected Section 2.

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 2013-08: This change deleted reactor core and coolant boundary protection analyses and standby safeguards analyses for plant transients and accidents that are no longer credible at CR-3 based on power operation being permanently ceased and nuclear fuel being permanently removed from the reactor vessel. Since the NRC has acknowledged CR-3's certification of permanent cessation of power operation and permanent removal of all nuclear fuel from the reactor vessel, then pursuant to 10 CFR 50.82(a)(2), the 10 CFR Part 50 license for CR-3 no longer authorizes operation of the reactor or emplacement of nuclear fuel in the reactor vessel. Consequently, all transients and accidents involving energy from or radiological releases associated with nuclear fuel in the reactor vessel are no longer credible. The following accidents/analyses were deleted from Section 14: Uncompensated Operating Reactivity Changes; Startup Accident; Rod Withdrawal at Rated Power Operation Accident; Moderator Dilution Accident; Cold Water Accident; Loss-of-Coolant-Flow Accident; Stuck-Out, Stuck-In, or Dropped Control Rod Accident; Load Rejection Accident; Station Blackout Accident; Steam Line Failure Accident; Steam Generator Tube Rupture Accident; Fuel Handling Accident inside the Containment; Rod Ejection Accident; Loss-of-Coolant Accident; Makeup System Letdown Line Failure Accident; Maximum Hypothetical Accident; Loss of Feedwater; and Main Feedwater Line Break Accident. All Section 14 tables were deleted except Tables 14-31 and 14-59. All Section 14 figures were deleted except Figure 14-37.

- FSAR Change Package 2013-11: This change is based upon CR-3 suspending the implementation of the Regulatory Guide 1.54 committed Safety-Related Coatings Program as a result of Duke Energy's decision to retire the plant. Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants," (Revision 0, June 1973) describes an acceptable method of complying with the NRC's quality assurance requirements with regard to protective coatings and endorses ANSI N101.4-1972, "Quality Assurance for Protective Coatings Applied to Nuclear Facilities." ANSI N101.4 in turn describes the requirements specifically for Service Level I and II safety-related coatings. Since the NRC has acknowledged CR-3's certification of permanent cessation of power operation and permanent removal of all nuclear fuel from the reactor vessel, then pursuant to 10 CFR 50.82(a)(2), the 10 CFR Part 50 license for CR-3 no longer authorizes operation of the reactor or emplacement of nuclear fuel in the reactor vessel. Consequently all transients and accidents involving significant motive forces from the energy released due to reactor core heat with nuclear fuel in the reactor vessel are no longer credible and safety related coatings are no longer required to afford protection to safety significant equipment or structures. Section 1, Section 5, and Table 1.3 have been changed to remove all descriptions of the Safety Related Coatings program and specific coatings used for the program. Tables 5-7 and 5-8 were also deleted.

DUKE ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR-72

ATTACHMENT B

10 CFR 50.59 EVALUATION SUMMARIES

10 CFR 50.59 EVALUATION SUMMARIES

Table of 10 CFR 50.59 Evaluations

ID Number	Title
AR 550699	Perform 50.59 Screen for EC 74407 Rev. 4
AR 599168	Revise 50.59 Screen for EC 76363 Rev. 35 to Include Heat Blanket
AR 605718	REG-10 Evaluation of Changes to FSAR Chapter 14
AR 607446	FSAR Revision to Chapter 5
AR 630583	FSAR Change 2013-0019: CD and SF
AR 642059	FSAR Change 2013-0023

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AR = Action Request

ID Number: AR 550699

Title: Perform 50.59 Screen for EC 74407 Rev. 4

Summary and Conclusions

Description

Engineering Change (EC) 74407 added a new monorail hoist (FHCR-3A) and a manual fuel assembly handling tool to the fuel handling bridge crane (FHCR-3) to enable loading fuel into the peripheral fuel assembly cell locations in the Spent Fuel (SF) pool and into the dry shielded canister (DSC) when placed in the cask pool. The pre-existing hoist was not capable of loading fuel into the peripheral cell locations because of travel limits for the trolley and interference of the mast with the pool liner. The pre-existing grapple for picking up the fuel assembly is larger in size than the fuel assembly cross sectional area, and thus did not permit lowering a fuel assembly completely down into a DSC storage location. The differences between the use of interlocks on FHCR-3 and the lack of extensive interlocks on FHCR-3A was determined to create an adverse effect on the spent fuel handling design function and on the Final Safety Analysis Report (FSAR)-described procedures for the performance of that design function. As such, an evaluation pursuant to 10CFR50.59 was required.

FHCR-3A has been limited to a working load limit of 2500 pounds but has been load tested to 3160 pounds. This illustrates that the brakes, chain, hook and structure are fully capable of lifting the normal fuel assembly and tool load that is less than 2000 pounds. The manual fuel handling tool has a working load limit of 1800 pounds but was load tested to 3550 pounds. Therefore the design capacity of the system is fully adequate for handling fuel elements. If power is lost, the brakes engage and prevent the fuel assembly from lowering any further. In that situation, the fuel assembly would then have to be manually lowered by use of a ratchet and brakes release tool and no fuel damage would result. The new hoist has both an electric brake and a mechanical brake. This redundancy makes a complete braking failure very unlikely. The safety function of FHCR-3A is to enable safe handling of fuel assemblies with minimal possibility of fuel assembly damage being incurred. The sum total of handling operations includes aligning the grapple on top of the intended fuel assembly, grabbing/lifting/holding the fuel assembly, transporting the fuel assembly to the new location, aligning the fuel assembly with the intended new location, and finally, lowering the fuel assembly and releasing the grapple. The safety considerations for each of these steps are as follows.

- Aligning the hoist – Since the operation of the new hoist is manual, the procedure will require the stationing of a spotter and the operator will then have to verify the intended location with the spotter. As an additional safety feature, Crystal River Unit 3 (CR-3) plans to use an under-water camera. The walkway railing will be marked with the storage cell locations. This feature applies to both the initial location as well as the final location of the fuel assembly. Alignment of the hoist, and therefore the tool, will be obvious to the FHCR-3A operator since he/she will be facing it during operation.

- Grabbing the fuel assembly – The new tool will be used to grab the fuel assembly. The new grapple is designed and tested to handle the fuel assembly. Once the assembly is grabbed and lifted, the grapple is designed mechanically such that it cannot release the fuel as long as the assembly is exerting load on the grapple.

- Lifting – The tool and monorail are designed to handle the weight of the fuel assembly. Should the fuel assembly become stuck in the cell, it will exert more load on the hoist than its weight. Exerting a higher force to lift the fuel assembly could damage the fuel assembly and/or the fuel storage racks; therefore, the hoist is equipped with an overload limit above which the hoist will stop lifting. The overload limit is set so as to prevent damaging a fuel assembly or the fuel storage racks. Because of the flexibility the chain hoist offers, the operator can grab the manual tool or chain and move both around to release a bound fuel assembly before lifting further. The load on the hoist is displayed on the hoist control. The hoist is designed to lift vertically with no drift. The hoist is also equipped with a mechanical stop to prevent lifting the fuel assembly beyond an elevation where the fuel water shielding is less than 8 feet. The lift velocity is controlled by the operator and can be set not to exceed a maximum desired velocity.

- Holding – The chain hoist is equipped with brakes to stop the lift when the operator releases the lift button and to hold the assembly at the lifted elevation.

- Transporting – The lateral movement of the hoist is controlled manually by the operator. The procedure requires the operator to confirm by visual observation and with the spotter that the fuel assembly has cleared the racks and any other obstructions prior to commencing lateral movement of the hoist. The lateral movement speed will not exceed a preset limit to prevent excessive drag forces on the hoist and swinging of the fuel assembly. Movement of a suspended fuel element is procedurally limited such that only one motor (FHCR-3 bridge or FHCR-3A hoist) may be operated, thus preventing movement in more than one direction.

- Lowering – The hoist lowering speed is limited to less than 5 feet/minute. Based on visual observation and confirmation with the spotter, the operator will ensure that the fuel assembly has reached the bottom of the storage location and is ready to be released.

- Once the fuel assembly load is off of the grapple, the operator can manually release the fuel assembly and move the hoist away.

The FSAR description of the fuel handling accident states that "There are numerous administrative controls and physical limitations that are imposed to prevent a Fuel handling Accident from occurring..." The manual operation of FHCR-3A contrasts with FHCR-3 operation in that FHCR-3 relies on physical limitations whereas FHCR-3A relies more on administrative controls. Significant additional features of FHCR-3 are summarized below for the contrast.

- The fuel assembly is withdrawn into the mast for protection.

- Lateral and vertical movements of the mast are interlocked such that only one can be used at a time.

- Bridge and trolley jog functions are provided for finer alignment of the hoist over the cell.

- Vertical movement is prevented unless the grapple is fully open or fully closed.

- Mandatory slow zones are provided during insertion of the fuel assembly.

- Position and elevation indications are provided on the control panel.

- Interlocks are provided to prevent the fuel assembly from striking the pool walls or other obstructions.

FHCR-3 presents several safety challenges to the operator due to two operational limitations: 1) the operator is unable to view the fuel assembly as it is withdrawn into the mast since the mast is located directly under the bridge; and 2) the mast is a rigid structure, therefore the only way to release a stuck fuel assembly is by using the finer controls of the bridge and trolley via their jog functions. FHCR-3A eliminates these challenges since the fuel assembly is readily visible to the operator and the spotter, and the chain hoist offers the advantage of the operator being able to grab the manual tool or chain and gently jostle them around in order to release a stuck fuel assembly.

FHCR-3A operation will procedurally require that a minimum of four qualified personnel be utilized to control fuel assembly position when an assembly is being moved to ensure that the alignment is correct and that lateral movement does not result in impacting any pool structures. Once the assembly is grabbed and lifted, the design of the grapple precludes releasing a fuel assembly as long as the assembly is exerting load on the grapple. When there is no load on the grapple, the operator can only release the fuel assembly after visual confirmation is made by both the operator and the spotter that the fuel assembly has reached the bottom of the storage location. The underwater camera will also be used unless it is not functional at the time.

Conclusion

Based on the findings of this evaluation, it was concluded that the implementation of EC 74407 could proceed without a License Amendment because sufficient administrative controls exist to prevent damage to, or misplacement of, a fuel assembly during fuel handling operations being performed with FHCR-3A.

ID Number: AR 599168

Title: Revise 50.59 Screen for EC 76363 Revision 35 to Include Heat Blanket

Summary and Conclusions

Description

EC 76363 replaced the Containment Purge Exhaust Duct (RM-A1) and the Auxiliary Building and Fuel Handling Area Exhaust Duct (RM-A2) airborne radioactivity monitoring equipment. The new sample skids were installed at approximately the same location as the previous skids in the auxiliary building on the 143' elevation. Two new monitor skids were be installed for each of the RM-A1 and RM-A2 systems. The new systems provide continuous real time monitoring and display channels for low (normal) and medium/high (accident) range noble gas activity. The normal range channel is functionally equivalent to the previous low range gas channel and the accident range channel envelopes the ranges of the previous medium and high range gas channels. The display, monitoring, and recording functionalities supported by the new equipment are fully compliant with CR-3 operating and post-accident monitoring licensing bases, although there are digital aspects to these monitors that were evaluated in previous versions of the EC and supporting 50.59 review.

While testing RM-A1 and RM-A2, both accident range detectors repeatedly went into a fault condition. Subsequent testing and analysis by the vendor concluded that humidity inside the detector chamber was the cause of the fault. Droplets or visible condensation is not required to affect the operation of the detector; any filming within the detector will cause a perturbation. Based on a recommendation by the vendor, it was determined that a heat blanket kit is required. The purpose of the heat blanket is to elevate the temperature inside the detector chamber to preclude the formation of any filming. A heat blanket is comprised of heating elements, a Resistance Temperature Detector (RTD), and a temperature control junction box. CR-3 was originally set to order a heat blanket kit for the RM-A1 accident range skid (RM-A1A) and the RM-A2 accident range skid (RM-A2A), however, subsequent to placing the order for the heat kits, the decision was made by Duke Energy to retire CR-3. As a result of the retirement decision, it was eventually decided that only RM-A2A needed to have the heat blanket installed.

EC 76363 R35 was implemented to install the heat blanket kit on RM-A2A. The heat blanket installed on the RM-A2 accident range skid consists of heating elements, RTDs, and a temperature controller. The temperature controller is a digital controller with a digital display which is mounted on the front of the RM-A2 Temperature Control Junction Box. The heat blanket functions to elevate the detector temperature to preclude any condensation or moisture film from forming inside the detector chamber. The temperature controller is set to maintain a temperature of 150°F ± 9°F, based upon recommendation by the vendor. Based on findings at CR-3 as well as vendor recommendation, the heat blanket is required for RM-A2A to operate properly. Because of the potentially adverse effects associated with the addition of the heat blanket kit and the digital temperature controller, the proposed activity required a 10CFR50.59 evaluation.

Conclusion

This evaluation determined that EC 76363 Rev. 35 was acceptable for implementation under 10 CFR 50.59 with no License Amendment required because the replacement airborne radiation monitoring equipment will provide reliable, accurate information comparable to the obsolete units that were replaced.

ID Number: AR 605718

Title: REG-10 Evaluation of Changes to FSAR Chapter 14

Summary and Conclusions

Description

This 10 CFR 50.59 review supported the Chapter 14 "Safety Analysis" of the FSAR revision. The change removed core and coolant boundary protection analyses and standby safeguards analyses for transients and accidents that are no longer credible for a plant that has permanently ceased operation and permanently removed all nuclear fuel from the reactor vessel. This activity therefore revised the FSAR to remove all transient and accident analyses involving energy from or radiological releases associated with nuclear fuel in the reactor vessel.

The following Core and Coolant Boundary Protection Analyses were removed:

- Uncompensated Operating Reactivity Changes
- Startup Accident
- Rod Withdrawal at Rated Power Operation Accident
- Moderator Dilution Accident
- Cold Water Accident
- Loss-of-Coolant-Flow Accident
- Stuck-Out, Stuck-In, or Dropped Control Rod Accident
- Load Rejection Accident, and
- Station Blackout Accident

The following Standby Safeguards Analyses were also removed:

- Steam Line Failure Accident
- Steam Generator Tube Rupture Accident
- Fuel Handling Accident inside the Containment
- Rod Ejection Accident
- Loss-of-Coolant Accident
- Makeup System Letdown Line Failure Accident
- Maximum Hypothetical Accident
- Loss of Feedwater and Main Feedwater Line Break Accident

Since the removal of the majority of accident analyses from the FSAR was essentially an abandonment of the methods of evaluation used in those accident analyses, an evaluation pursuant to 10CFR50.59 was required.

Conclusion

The results of the evaluation concluded that no License Amendment was required to implement the changes to FSAR Chapter 14 because significant releases of energy and/or radioactivity from the reactor core and containment are no longer possible as a result of the permanent removal of nuclear fuel from the reactor vessel. Additionally, the evaluation also determined that the reactor coolant system pressure boundary and the containment should no longer be considered fission product barriers. Consequently, the only remaining fission product barrier at CR-3 is the fuel rod cladding.

ID Number: AR 607446

Title: FSAR Revision to Chapter 5

Summary and Conclusions

Description

This 10 CFR 50.59 review supported the revision of FSAR Chapter 5, "Containment System And Other Special Structures," and specifically Section 5.2, "Reactor Building," to reflect differences between the classification/function of the reactor containment in an operating plant versus CR-3's current status as a permanently defueled plant. AR 605718 had previously determined that CR-3's containment is no longer a fission product boundary since nuclear fuel can never again be installed in the reactor vessel. The changes made to FSAR Section 5.2 included physical description, structure classification, design basis load cases, and methods of evaluation.

This revision to FSAR Chapter 5 removed loading conditions for the containment that were in place to evaluate the containment performance during loss of coolant accident conditions. The revision also removed a description of the method of evaluation used to calculate the effects of containment loading conditions. The changes were thus judged to be a replacement of the then existing FSAR-described method of evaluation with another evaluation method and an evaluation pursuant to 10 CFR 50.59 was warranted.

Conclusion

The results of the evaluation concluded that a License Amendment would not be required to implement the changes to FSAR Chapter 5 because the design function of the containment building to provide a fission product boundary during plant accident conditions is no longer required now that nuclear fuel has been permanently removed from the reactor vessel.

ID Number: AR 630583

Title: FSAR Change 2013-0019: CD and SF

Summary and Conclusions

Description

This 10 CFR 50.59 review supported the FSAR change that replaced several balance of plant (BOP) inventory sources (the condensate (CD) demineralizers, the CD storage tank, and the condenser hot well) with the fire service tanks as an inventory source for spent fuel (SF) pool inventory makeup. Normal operating procedures identify the demineralized water system as the primary source of makeup inventory to the SF pools and the Borated Water Storage Tank (BWST) is identified as the alternate source. The abnormal operating procedure for mitigating a loss of SF cooling describes the emergency actions for SF pool inventory makeup. The Fire Service (FS) system is identified as an alternate makeup source. Additional guidance is also provided to address catastrophic failure of the SF pools and an extended loss of AC power that prevents conventional sources of makeup inventory from being utilized. That guidance identifies various fire hose stations that may be used for SF pool makeup. Water from the intake and discharge canals are also identified as a last resort method of providing makeup inventory to the SF pools in this unlikely situation. Given the procedural guidance, the use of FS water for SF pool makeup inventory would only be during an emergency condition and would not be expected to be a long term condition. The calculated time for the temperature of the SF fuel inventory to increase to boiling following a complete loss of all SF pool cooling was calculated to be approximately 4 days, thus providing an ample amount of time to implement a source of makeup inventory after a complete loss of SF pool cooling assuming no leak exists in the SF pools.

Given the existing procedural guidance, this change to the FSAR substituted the FS Tanks and FS Hose Stations in place of the BOP inventory sources and associated permanent condensate and demineralized water system piping and valves in order to optimize the makeup capability to the SF pools and to establish consistency with the plant normal, abnormal, and emergency operating procedures. Given that the FS Tanks represent a much larger inventory for makeup to the SF pools than the BOP sources and associated piping, this change did not result in any adverse effect.

FSAR section 9.8.7.1 describes the two FS Tanks as "dedicated" to fire service water storage. Since this change is an exception to that statement, it is considered a potential adverse effect. Additionally, the substitution of FS equipment for condensate equipment results in the emergency makeup inventory water source to the SF pools being non-demineralized, which is a potential adverse impact to SF pool water chemistry. For both reasons an evaluation pursuant to 10 CFR 50. 59 was required.

Conclusion

The results of the evaluation concluded that a License Amendment would not be required to implement this FSAR change because the FS Tanks provide a much larger source of SF pool makeup inventory than the BOP inventory sources and associated piping with a less than minimal adverse impact on SF pool water chemistry.

ID Number: AR 642059

Title: FSAR Change 2013-0023

Summary and Conclusions

Description

This 10 CFR 50.59 review supported the FSAR change that revised the Fuel Handling Accident, eliminated the Waste Gas Decay Tank (WGDT) Rupture Accident, revised the description of the methodology for calculating atmospheric dispersion factors, and added a new radioactive waste handling event as the limiting radioactive waste event in terms of dose to the public. The Fuel Handling Accident was revised based on the reduced inventory of radioactivity in the nuclear fuel since the plant has been shut down for over 4 years, and atmospheric dispersion factors have subsequently been recalculated. The WGDT Rupture Accident is no longer credible since significant quantities of waste gases can only be produced due to reactor operation, which is no longer allowed by CR-3's 10 CFR Part 50 license. Since permanently shutting down, the inventory in the Waste Gas Decay Tanks has been released and the tanks have been permanently reconfigured for continuous venting to preclude re-pressurization. A limitation was also added by this change regarding the storage locations of radioactive wastes. The new limitation ensures that primary resin storage is consistent with the assumptions of the current radioactive waste handling event analysis.

The addition of a new radioactive waste handling event is not considered to be revising or replacing an FSAR-described evaluation methodology. However, since this FSAR change replaced the method of calculating atmospheric dispersion factors for accidents, an evaluation pursuant to 10 CFR 50.59 was determined to be appropriate.

Conclusion

The results of the evaluation concluded that a License Amendment would not be required to implement this FSAR change because the previous methodology for calculating atmospheric dispersion factors was replaced with a method developed by the NRC and approved for this application by Regulatory Guide 1.183.

DUKE ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR-72

ENCLOSURE

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