Progress Energy

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

Ref: 10 CFR 50.9

May 25, 2010
3F0510-01

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

Subject: Crystal River Unit 3 - NRC Commitment Change Report – May 2010

Dear Sir:

The purpose of this letter is to provide notification of changes to regulatory commitments contained in previously docketed correspondence from Florida Power Corporation, now doing business as Progress Energy Florida, Inc., to the NRC. The attached report contains the Crystal River Unit 3 (CR-3) Nuclear Operations Commitment System (NOCS) reference numbers, source of the original commitment, statement of the original commitment, statement of the revised commitment, if revised, and justification for the change. This report is being submitted in accordance with Nuclear Energy Institute (NEI) document NEI 99-04, Revision 0, "Guidelines for Managing NRC Commitment Changes," dated July 1999.

Of the fifteen (15) CR-3 regulatory commitments that were modified or inactivated between November 23 2007 and January 5, 2010, seven (7) modified or inactivated regulatory commitments meet the NEI 99-04 criteria for NRC notification.

No new regulatory commitments are made in this letter.

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

Sincerely,

J. W. Holt

James W. Holt
Plant General Manager
Crystal River Nuclear Plant

JWH/dwh

Attachment

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

A001
LIRA

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

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR-72

ATTACHMENT

NRC COMMITMENT CHANGE REPORT – MAY 2010

Nuclear Operations Commitment System (NOCS) Number: None (Security-Related)

Source Document:

Crystal River Unit 3 (CR-3) to NRC letter, 3F0207-02, dated February 12, 2007.

Original Commitment:

Table A.4-4 states, in part: "In addition to the ADVs [Atmospheric Dump Valves], sufficient Main Steam safety valves (MSSVs) will be opened to maintain OTSG [Once-Through Steam Generator] pressure within the discharge capacity of the portable power-independent pump. The MSSVs will be opened using the manual lifting device with installed gags."

Revised Commitment:

Replace the above excerpt with the following sentence: "The valve stroke of the ADVs can be adjusted to allow full opening of the ADVs, which will reduce OTSG pressure to as low as is achievable."

Justification for Change:

The methodology from using the MSSVs to the using the ADVs full open was revised because the MSSVs could not be manually opened with the OTSG depressurized. Manually opening the MSSVs requires OTSG pressure to assist opening the valve. With the OTSG depressurized, the amount of lift pressure to get the MSSV open would have required a hydraulic lift, along with changes in the design of the valve to support the lift point. Opening of the ADVs accomplishes the same requirements, but eliminates the need to modify the MSSVs. Also, using the ADVs simplifies the action and does not require special devices to open the valve.

The change affects plant operation in conditions which exist only in beyond-design-basis events. The intended safety function of the ADVs and/or MSSVs during design basis events is not affected.

Nuclear Operations Commitment System (NOCS) Number: None

Source Document:

CR-3 to NRC letter, 3F0981-02, dated September 2, 1981.

Original Commitment:

In July 1980, the NRC issued NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants-Resolution of Generic Technical Activity A-36." By letter of December 22, 1980, the NRC requested licensees to evaluate load handling operations at their plants against the criteria in NUREG-0612, to identify areas of nonconformance, and to describe actions that would be taken to satisfy the criteria of NUREG-0612. This letter required that the evaluations be provided in two reports; the first report (Phase I) addressing compliance with general criteria such as definition of safe load paths, development of load handling procedures, procedures pertaining to the inspection and maintenance of cranes, crane operator training and qualification, and the evaluation of crane and special handling devices. The second report (Phase 2) is to address the potential safety consequences of load handling accidents and preventive or mitigative features that would be implemented so that consequences would be within defined safety criteria.

CR-3 responded to this request in various reports and other correspondences during the earlier 1980 timeframe. The actual Phase I report was transmitted via letter 3F0981-02. The NRC issued a Safety Evaluation Report (dated July 13, 1984) on the Phase 1 response that documents acceptance of CR-3's methodology of meeting the above guidelines. These documents commit that the various components that make up the "special lifting device" for the Reactor Vessel (RV) head meet the requirements of ANSI N14.6-1978, "American National Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials."

Section 3 of ANSI N14.6-1978 requires that the design of the lifting devices meet certain factors-of-safety against the allowable yield stress and to meet certain load test requirements. ANSI N14.6 requires a factor-of-safety of 3 for the engagement pins inside latch boxes.

Revised Commitment:

Take exception to ANSI N14.6-1978, Section 3, by stating the following: "Use of the special lifting device (i.e., tripod) involves using the original plant lifting latch boxes. These devices were load tested to approximately 150% of lifted load per Section 5 of ANSI N14.6-1978. However, the design of the latching pins inside the latch boxes includes a factor-of-safety of 2 instead of the 3, as required by Section 3 of ANSI N14.6-1978."

Justification for Change:

During the 1980 timeframe, the original lifting devices for the RV head consisted of wire rope and pins that attached to the tripod. These components met the requirements of NUREG-0612 (i.e., ANSI N14.6-1978). In the early 1990s, a plant modification was implemented that replaced the wire rope with fixed rod pendants. These fixed pendants were designed to remain on the RV head to save outage time. The fixed pendants attached to the tripod using heavy pins. The design of the fixed pendants and other lifting components met the requirements of NUREG-0612 (i.e., ANSI N14.6-1978).

In 2005, another plant modification replaced the top part of the fixed lifting pendants from a clevis and pin connection to a T-lug connection, while at the same time allowing the internal fixture latch boxes to be used for lifting both the internal fixture and the replacement RV head. The latch boxes and T-lugs were load tested in accordance with the requirements of NUREG-0612 and ANSI N14.6-1978, Section 5.

Section 3 of ANSI N14.6-1978 also requires that the design of the lifting devices meet certain factors-of-safety against the allowable yield stress, and to meet certain load test requirements. However, the factors-of-safety of the engagement pins inside the latch boxes do not meet the requirements of ANSI N14.6-1978. ANSI N14.6 requires a factor-of-safety of 3, while the calculated factor-of-safety is 2. Credit was taken for the actual load test of the latch boxes as meeting the requirements of ANSI N14.6-1978.

CR-3 management has decided to pursue a commitment change for this specific difference in meeting the requirements of NUREG-0612, rather than modifying the latch boxes.

This issue involves the lifting of the RV head during shutdown conditions. In the unlikely event that the latch boxes were to fail, a load drop analysis of the RV head has been prepared, concluding that both trains of Decay Heat Removal will not be lost and the fuel can be cooled. This issue does not involve any Structures, Systems or Components that affect safe shutdown.

Nuclear Operations Commitment System (NOCS) Number: 9834

Source Document:

CR-3 to NRC letter, 3F1282-02, dated December 1, 1982.

Original Commitment:

Lifting Devices (Not Specifically Designed): The results of an FPC evaluation of dynamic loads revealed that if a compensatory factor of 1/2% of the static load/ffm crane hook speed is incorporated into the sling rating, a sufficient margin of safety will be attained. The hook speeds for non-exempt cranes at Crystal River Unit 3 are as follows:

Spent Fuel Pool Missile Shield Crane (HCR-7) speed = 16.2 ft/min.
Fuel Handling Area Crane (FHCR-5) speed = 4.8 ft/min.
Intake Gantry Crane (CWCR-1) speed = 14.3 ft/min.
Reactor Building Crane (RCCR-1) speed = 4 ft/min.
Reactor Vessel Tool Handling Jib Crane (RCCR-2) speed = 27 or 9 ft/min.
Spent Fuel Pool Gate (SFHT-7) speed = 16.6 ft/min.

RCCR-2 has a high speed switch for its 27 ft/min. speed. The high speed switch will be disconnected to minimize the effects of dynamic loading during lifts performed by RCCR-2. Based on this data and the FPC evaluation, all slings and other special lifting devices not specifically designed have been derated by 10% to compensate for dynamic loading.

Revised Commitment:

Add the following NOTE at the end of the Statement of Commitment: "When determining rigging requirements for non-Engineered lifts, over-rig by 15% additional capacity instead of derating by 10% to compensate for dynamic loading. This is more conservative than the derating by 10% and meets the intent of NOCS 9834."

Justification for Change:

For non-Engineered lifts, over-rigging by 15% additional capacity to compensate for dynamic loading is more conservative than the derating by 10%, and meets the intent of the regulatory commitment made in CR-3 letter to the NRC dated December 1, 1982. The margin of safety for slings and other special lifting devices, not specifically designed, will be increased. No negative impact will be caused by this change in commitment.

Nuclear Operations Commitment System (NOCS) Number: 40774

Source Document:

CR-3 to NRC letter, 3F1183-21, dated November 23, 1983.

Original Commitment:

In 3F1183-21, Florida Power Corporation indicated that the maximum lift height for the Reactor Vessel head assembly when it is above the Reactor Vessel would be five feet. In order to comply with this maximum lift height, both fuel handling bridges must be stored in the deep end of the fuel transfer canal.

Procedures will be developed to ensure that the Reactor Vessel head will be moved horizontally away from the Reactor Vessel at or before the maximum height is reached.

Revised Commitment:

Insert a clarifying statement that the maximum lift height limit is applicable while the reactor is in a fueled condition.

Justification for Change:

CR-3's Reactor Vessel head load drop analysis is based on fuel in the vessel and the Decay Heat Removal System remaining operable. When there is no fuel in the vessel, the Decay Heat Removal System is not required to be operable.

Nuclear Operations Commitment System (NOCS) Number: 62363

Source Document:

Babcock & Wilcox Owner's Group (BWOG) Safety and Performance Improvement Program (SPIP) Recommendation Tracking System Report (RTS) #47-1163743-27, TR-174-MSS

Original Commitment:

Improve the response of the modulating Turbine Bypass Valves [TBVs] (Atmospheric and Condenser Dump Valves). Previous experience indicates a stroke open time of 3 seconds or less provides acceptable response. For some plants, this may require hardware modifications such as the addition of volume boosters and larger actuators.

All plants should establish surveillance and maintenance criteria to maintain rapid valve response. It is suggested that stroke time be measured from the step increase in demand signal on the operator until the valve is fully open

Revised Commitment:

Revise the open stroke time of the TBVs from 3 seconds or less, to 5 seconds or less.

Justification for Change:

The B&W SPIP studied the need for improved TBVs response to help prevent repetitive lifts of the Main Steam Isolation Valves (MSSVs). The expected benefit from Recommendation TR-174-MSS [improving the response of the modulating TBVs (Atmospheric and Condenser Dump Valves) is reduced complexity of post-trip steam pressure control and reduced MSSV failures by reducing the number of challenges. Previous experience indicated a full stroke open time of 3 seconds or less provided acceptable response.

Allowing for a full stroke open time of 5 seconds or less for the TBVs is requested based on replacing the current TBVs with larger capacity TBVs to support Extended Power Uprate (EPU). The existing valves are 6" Fisher valves and are being replaced with 12" Masoneilan valves. The replacement TBVs capacity is more than double the capacity of the existing TBVs, (i.e., current 418,500 lbm/hr, new 1,045,000 lbm/hr full open) as documented in Engineering Change 71757. This larger capacity provides for sufficient flow capabilities to accommodate power uprate and for additional margin.

If the existing TBV is rated to flow 418,500 lbm/hr at full open, a new TBV will exceed this flow rate when it is only 60% open. Working backwards, a new TBV will only have to reach about 42% = (418,500/1,000,000) x 100 of the rated Cv to equal the flow rate of an existing TBV at full open. This 42% Cv occurs at approximately 47% of valve travel. This will be reached in about 2.4 seconds. Therefore, the effective stroke speed of the new TBVs will be faster than the existing TBVs.

Nuclear Operations Commitment System (NOCS) Number: 62733

Source Document:

CR-3 to NRC letter, 3F1097-07, dated October 8, 1997.

Original Commitment:

MP-110A will be revised to add steps ensuring that either one manway is still open or the inspection cover purge fans are secured prior to manway closure.

Revised Commitment:

Revise this commitment to read: "MP-110A will be revised to add steps ensuring that either one manway is still open or a ventilation offset adapter is installed on the inspection handhole prior to manway closure."

Justification for Change:

Originally, the maintenance procedure (MP) to install manway covers on the Once-Through Steam Generator did not provide provisions to secure the handhole ventilation fans prior to the installation. The intent of this regulatory commitment was to ensure that the ventilation does not pull a vacuum on the Reactor Coolant System (RCS), which will cause erroneous RCS level indications. To support proper ventilation of the RCS pressure boundary, in a loss of decay heat cooling event, the handhole vent path needs to be offset from the flange to provide an open, unrestricted flow path. This will allow the ventilation fans to remain on to capture gas releases from the area just above the inspection handhole. The ventilation offset will allow for an open vent path as well as a suction path that does not pull a vacuum on the RCS, thereby agreeing with the original intent of this commitment.

Nuclear Operations Commitment System (NOCS) Number: 100156

Source Document:

CR-3 to NRC letter, 3F0586-19, dated May 14, 1986.

Original Commitment:

FPC will select individuals to perform STA duties who have bachelor's degrees in a scientific or engineering discipline with at least 4 years nuclear power experience.

Revised Commitment:

Revise wording to the following: "Progress Energy will select individuals to perform STA duties who have bachelor's degrees in a scientific or engineering discipline with a minimum of one year nuclear power plant experience, with at least six months experience onsite."

Justification for Change:

In a letter dated October 1, 1985 (3F1085-01), Crystal River 3 made a final revision to its commitment regarding the qualifications and selection criteria for Shift Technical Advisors (STAs). This letter to the NRC committed to at least four years of nuclear power experience. At the time, the commitment met the requirements of NUREG-0737 and the Crystal River 3 Technical Specifications.

In Generic Letter 86-04, "Policy Statement on Engineering Expertise on Shift," the Nuclear Regulatory Commission requested each licensee send a response detailing the status of and anticipated changes to the STA program for their utility. FPC replied on May 14, 1986 (3F0586-19), restating the commitment made in the October 1, 1985 letter.

Currently, ANSI/ANS-3.1-1981, "American Nuclear Standard for Selection, Qualification and Training of Personnel for Nuclear Power Plants," lists the experience requirements for the STA position, specifically requiring the STA to have one year of nuclear power plant experience, with six months on site. This standard was endorsed by the NRC in Regulatory Guide 1.8, "Qualification and Training of Personnel for Nuclear Power Plants," and is the standard recognized throughout the industry. NUREG-0737, the CR-3 Improved Technical Specifications, and the CR-3 Final Safety Analysis Report require compliance with ANSI/ANS-3.1.