UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
245 PEACHTREE CENTER AVENUE NE, SUITE 1200
ATLANTA, GEORGIA 30303-1257

October 12, 2010

Mr. Jon A. Franke
Vice President, Crystal River Nuclear Plant
Crystal River Nuclear Plant (NA2C)
15760 W. Power Line Street
Crystal River, FL 34428-6708

SUBJECT: CRYSTAL RIVER NUCLEAR PLANT - SPECIAL INSPECTION REPORT 05000302/2009007

Dear Mr. Franke:

On September 2, 2010, the U.S. Nuclear Regulatory Commission (NRC) completed a Special Inspection at your Crystal River Unit 3 Nuclear Plant. The enclosed report documents the inspection results, which were discussed on September 2, 2010, with you and members of your staff in an exit meeting open for public observation at the Crystal River Nuclear Plant EOF/Training Center.

The purpose of this inspection was to examine activities associated with the delamination identified in the Unit 3 containment. The inspection was conducted in accordance with the Special Inspection charter issued by the Region II Administrator on October 13, 2009. As part of the inspection, the NRC examined your evaluation of extent of condition, your evaluation of root cause, and your planned corrective action for continued operability. The inspectors interviewed personnel and conducted plant walk downs, including visual examination of accessible portions of the outside and inside surfaces of the containment.

This inspection was conducted prior to the completion of your planned corrective actions. Follow-on inspections of your corrective actions to repair the containment and conduct post maintenance testing are being conducted in accordance with NRC inspection procedure IP 50001, Steam Generator Replacement Inspection.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of

NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Christopher G. Miller, Acting Director
Division of Reactor Safety

Docket No.: 50-302
License No.: DRP-72

Enclosures:
1. Inspection Report 05000302/2009007
   W/Attachment: Supplemental Information
2. Special Inspection Charter
3. Crystal River Root Cause Report

cc w/encl.: (See page 3)

cc w/encl:

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Letter to Jon A. Franke from Christopher G. Miller dated October 12, 2010

SUBJECT: CRYSTAL RIVER NUCLEAR PLANT - SPECIAL INSPECTION REPORT 05000302/2009007

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DATE	10/9 /2010	10/6/2010	10/4/2010	10/6/2010	10/1/2010	10/1/2010	10/6/2010	10/5/2010
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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No:	50-302
License No:	DRP 72
Report No:	05000302/2009007
Licensee:	Progress Energy Company
Facility:	Crystal River Unit 3
Location:	15760 West Power Line Street Crystal River, Florida
Dates:	October 13, 2009 through September 2, 2010
Inspectors:	L. Lake, Senior Reactor Inspector and Inspection Team Leader R. Carrion, Senior Reactor Inspector F. Farzam, Senior Structural Engineer A. Masters, Senior Construction Inspector D. Naus, Consultant G. Thomas, Structural Engineer
Approved by:	M. Franke, Chief Engineering Branch 3 Division of Reactor Safety

Enclosure 1

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EXECUTIVE SUMMARY
Crystal River, Unit 3
NRC Inspection Report 05000302/2009007

Crystal River shut down for a planned refueling outage on September 26, 2009. One of the major work activities planned for this outage was a steam generator replacement. In order to take the old steam generators out and put the new steam generators in, the licensee created a construction opening in the side of the containment building. On October 2, 2009, while creating this opening, workers saw that there was a gap, or separation, affecting the outer layer of concrete of the building wall. The gap or separation in the concrete has been commonly referred to as a delamination.

The discovery of the delamination did not represent an immediate safety concern because the plant was shut down. However, the discovery was important because the containment building is one of the three main barriers that protect public safety. The concrete delamination was not expected and had not been seen before during steam generator replacement activities at other nuclear plants. Because this issue involved possible adverse generic implications, because the structural integrity of the containment was not fully known and the concrete separation was not well understood, and because of the potential safety implications, Region II began a Special Inspection to better understand the issue. The inspection started on October 13, 2009, and was completed on September 2, 2010.

The Special Inspection Team (SIT) found that the delamination did not represent an increase in risk to the public. There were no radiological releases associated with the event, and no Technical Specification Safety Limits were approached or exceeded. The SIT found that the licensee's investigation was thorough and supported their conclusions on the delamination cause, extent of condition, and safety significance. The SIT did not identify immediate generic safety concerns associated with the delamination.

The licensee's investigation concluded that the delamination was caused during the creation of the opening in containment. As part of preparing the containment building for making the opening, tendons in the containment building wall were detensioned. The main cause of the delamination was attributed to the scope and sequence of this tendon detensioning. Tendon detensioning began after the plant was shut down in Operating Mode 5, when containment operability was not required.

The licensee found that the delamination was centered on the steam generator opening and formed the shape of an hour-glass. The delamination was limited to the containment bay between buttresses 3 and 4, and did not affect other bays of containment.

The licensee's repair plan to remove and replace the delaminated condition included: (1) additional detensioning of containment; (2) removal of delaminated concrete; (3) installation of reinforcement, including radial reinforcement through the delamination plane; (4) placing of new concrete; (5) retensioning containment; and (6) post-repair confirmatory system pressure testing. The licensee developed new finite element analysis models to predict stresses in the repaired containment wall under design basis loads. Using these and other supporting new analysis models, the licensee planned and implemented additional containment detensioning without causing further delamination. Subsequent to removal of the delaminated concrete, vertical cracks were observed along the vertical tendon lines in the concrete surface between buttresses 3 and 4. Following an engineering evaluation, the licensee planned to excavate and fill in cracks above elevation (El) 176'. The licensee's evaluation determined that cracks below 176 feet were acceptable as-is. The licensee's investigation also identified additional tight vertical cracks in other locations around the circumference of the containment wall. The licensee's evaluation of these vertical cracks remained ongoing at the conclusion of the Special Inspection.

Crystal River Unit 3 (CR3) is unique in that no other plants that have cut openings in their containments have experienced similar delamination. As discussed above, the root cause analysis determined that the delamination was caused by scope and sequence of this tendon detensioning in preparation for making the opening. The licensee developed new analytical methods to adequately identify the redistribution of stresses in the containment wall and identify an acceptable expanded detensioning scheme to perform the repair. The licensee has been communicating to other plants, through INPO and other industry organizations, lessons learned from this event.

REPORT DETAILS

INTRODUCTION AND CHARTER

In accordance with Management Directive 8.3, “NRC Incident Investigation Program,” the deterministic and conditional risk criteria were used to evaluate the level of NRC response for this event.

Based on the deterministic criteria that this issue involved possible adverse generic implications, the event was evaluated for risk in accordance with Management Directive 8.3. Due to lack of information on the structural integrity of the concrete containment and how it would interact with the steel liner during a seismic event in its current condition, a risk analysis was not able to be performed at the time of the event. However, if the concrete containment reacted adversely with the steel liner during a seismic event, the Large Early Release Frequency (LERF) would be adversely affected and a Special Inspection would be warranted. Therefore, Region II has determined that the appropriate level of NRC response was to conduct a Special Inspection. Enclosure 2 of this report contains the Special Inspection Team (SIT) charter.

BACKGROUND

The purpose of this section is to provide information about the design of the Crystal River Unit 3 (CR3) containment structure (containment), the maintenance history of the structural integrity of the containment, and the condition of the containment leading up to the delamination discovered during the recent plant shutdown for refueling outage #16 (RFO 16). Included is information related to a delamination that was identified in the containment dome during the final stages of containment construction and before initial plant startup.

Crystal River Nuclear Generating Plant Unit 3 (CR3) is a pressurized water reactor. It is has been in operation for over 30 years, with the current license due to expire December 3, 2016. Crystal River has submitted an application to the NRC to request a license renewal to extend the operation of the plant for another 20 years.

Design

In a pressurized water reactor plant there are three main barriers that protect the public from the radiation hazards associated with nuclear operations. One of those barriers is known as the containment, which houses the fuel, the reactor, and the reactor cooling system. The Crystal River containment structure is a steel lined post-tensioned cylindrical concrete structure of about 157 feet in height with an outside diameter of about 138 feet. The containment has 42- inch thick concrete walls and has a flat foundation mat and a shallow torispherical dome. Post-tensioning is achieved by utilizing an outer array of horizontal tendons immediately adjacent to an inner array of vertical tendons that are embedded in the walls about 15 inches from the outside surface. Tendons are also provided in the dome. In addition, steel rebar is embedded in the concrete walls at the outside surface and at other locations.

The containment is lined with a continuous 3/8-inch-thick carbon steel liner (that acts as a vapor barrier for leak-tightness and also as an inner form for the concrete. A series of equally spaced vertical steel angle irons are attached to the concrete side of the liner and serve as anchors and stiffeners. The dome is post-tensioned by 123 tendons that are arranged in a three-way (layer) configuration and are anchored to a ring girder. The containment walls include 282 horizontal and 144 vertical tendons that are anchored to 6 vertical buttresses equally spaced circumferentially around the containment. Each tendon consists of numerous small diameter wires, which are greased and housed inside a conduit. The conduit for each tendon is about five inches in diameter and is made of galvanized steel. The concrete has a minimum 28-day compressive strength of 5,000 pounds per square inch (psi).

Dome Delamination

On April 14, 1976, about two years after completion of concrete placement of the containment dome and one year after tensioning the tendons, electricians were attempting to secure drilled-in anchors to the top surface of the dome and certain anchors would not hold. Upon further investigation a delamination in the containment dome was discovered. The area of the delaminated concrete was approximately circular in shape with a 105- foot diameter. The approximate thickness of the delamination was 15 inches, with a maximum gap of approximately two inches between layers. No cracks appeared on the dome surface and, except for springiness when walking on the dome, there were no other indications of any problems.

Analysis of the dome delamination included engineering investigations that took into consideration all factors that were believed to be potentially contributory to the delaminated condition. Factors considered included: (1) properties of concrete and its constituents; (2) radial tension due to prestress; (3) compression-tension interaction; (4) thermal effects; (5) tendon alignment; (6) heavy construction loads; (7) coastal location; (8) location adjacent to fossil units; (9) construction methods; (10) impact loads; and (11) shrinkage effects. The dome investigation team concluded that a compression-tension interaction failure had occurred. The dome delamination was caused by the effects of radial tensile stresses combined with biaxial compressive stresses and lower-than-normal concrete tensile strength and aggregate strength. Tensile tests of the concrete indicated that the concrete had low resistance to crack propagation thus permitting local cracking to propagate. These factors and the conclusion were included as part of the 75 failure modes considered in Progress Energy's root cause investigation into the delamination identified during RFO 16. For more information on the root cause see Section 5 of this report.

The dome repair process included: removal of the delaminated dome cap; meridional, hoop, and radial reinforcement; and placement of a new dome cap. Instrumentation was installed to monitor the dome during tendon detensioning, retensioning and during a structural integrity test. The structural integrity test subjected the repaired containment to 115 percent of design pressure.

<u>Maintenance history of the containment</u>

The structural integrity of the containment structure was initially confirmed by the structural integrity test prior to the plant's initial startup. Baseline crack mapping of the outside surface of the containment wall was conducted at this time. Subsequent periodic inspections and tests were conducted to confirm containment integrity. This included dome monitoring, leak rate tests, and visual examinations.

Since 1976, the containment passed six integrated leak rate tests. The most recent integrated leak rate test was conducted in December 2005. These tests pressurized the containment to accident pressure to measure for leakage. They also included internal and external visual examinations.

Other periodic examinations and testing included:

> <u>Examination of containment surfaces</u> – The containment was periodically visually examined in accordance with the requirements of Title 10 of the Code of Federal Regulations (CFR) Part 50.55a and Subsections IWL and IWE of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code). These examinations were conducted on a frequency of every 5 years and no unacceptable conditions were identified.
>
> <u>Tendon Testing</u> – Tendon testing was performed in accordance with the requirements of 10 CFR Part 50.55a and Subsection IWL of the code. These tests required measurement of tendon lift-off force and visual examination of the tendons for grease leakage and tendon wire condition. Tendon testing was performed during the first 3 refueling outages and at a frequency of every 5 years (during outages). The 30th year testing was performed during refueling outage #15 between October and November

2007. Results of the examinations included minor anomalies and corrective action was taken. No unacceptable conditions were identified.

Metal Surfaces - Metal surfaces of the containment have been visually examined in accordance with the requirements of the requirements of 10 CFR Part 50.55a and Subsection IWE of the code. These visual examinations cover all accessible surfaces and require the documentation of any anomalies such as corrosion, cracking and bulges, in the containment liner. Bulges in the containment liner were identified by visual examination conducted in 1996 and have been monitored for acceptability in accordance with regulatory requirements. No unacceptable conditions have been identified.

The inspectors reviewed the containment building maintenance and test records, from 1976 to present, to determine whether the containment was monitored in accordance with regulatory requirements. No issues of significance were identified. Furthermore, the inspectors did not identify any potential signs in the test and maintenance history between 1976 and 2009 that a concrete separation occurred, before the wall delamination was discovered in October 2009.

1.0 Description of the Problem

Crystal River scheduled the replacement of the steam generators during the CR3 Refueling Outage 16. Replacing steam generators is a major activity and in preparation for the steam generator replacement the licensee planned to make a construction opening in the containment wall, approximately forty feet directly above the equipment hatch. This opening would facilitate removing the existing steam generators and the transport of the replacement steam generators into the Reactor Building. Cutting the opening in the containment included preparing the containment by detensioning tendons, then removing the concrete with high pressure water, a process known as hydrolazing, and cutting the containment liner plate.

The licensee began Refueling Outage 16 on September 25, 2009 with a reduction in power from Mode 1. The unit continued down in power through Modes 2, 3, 4, and 5 on September 26. Upon reaching Mode 5 at about 5:00 p.m. on September 26, tendon work activities began. Two vertical tendons (34V12 and 34V13) were detensioned simultaneously. These tendons were detensioned prior to cutting the button heads. From September 26 through October 1, eight additional vertical tendons and seventeen hoop tendons were detensioned by plasma cutting the button heads as part of the process to make the required construction opening. The licensee began the removal of concrete by hydrolazing (hydro-demolition) on September 30, 2009, as the first step in making the construction opening. This process was accomplished by using water under pressure (as great as 25000 psi) to “cut” the concrete. On October 3, during hydro-demolition work to expose the first layer of tendon sheaths, water from the work was observed leaking from the exterior surface of the containment at various locations below the elevation of the bottom of the construction opening. The leaking water was not limited to the construction opening, but was observed at the edges of the construction opening extending into undisturbed concrete for an indeterminate distance but at least as far as the post-tensioning buttresses (Buttresses #3 and #4). As the work continued, some of the concrete rubble unexpectedly broke off into large pieces. Licensee personnel inspected the construction opening and discovered a concrete separation condition. It was located approximately in the cylindrical plane of the centerline of the hoop tendons, approximately nine to twelve inches from the exterior surface of the containment building. Approximately 30 inches of concrete remained in apparent good condition all the way to the liner plate. The hydro-demolition of the concrete continued through October 7, when the containment building liner was exposed and all concrete had been removed down to the liner.

The licensee assembled a technical analysis group dedicated to analyze the separation (called a delamination) to: (1) determine the extent of condition; (2) determine the root cause; (3) perform a design basis analysis; and (4) perform a repair alternatives analysis. Each area was enhanced by support from external consultants. These consultants included one with expertise in non-destructive testing of large concrete structures to help determine the extent of the delamination, and another recognized for its expertise in root cause analyses to determine the root cause for this event. The licensee also included a consultant with expertise in computer analysis and design basis calculations to participate on the team to review the effect of the delamination on the design bases and design margin of the containment structure. This effort was placed under the responsibility of the Containment Project Manager. The Containment Project Manager was also responsible for interfaces with INPO, NEI, the NRC, media, etc. as well as project controls dealing with scheduling, contract administration, and financial issues.

2.0 Operability and Reportability

a. Scope

The inspectors reviewed the licensee's Reportability Evaluation to determine whether the containment de-lamination was reportable in accordance with the CFR, specifically 10 CFR Part 50.72 and 10 CFR Part 50.73. The review included: (1) any operation or condition which was prohibited by the plants technical specifications; (2) any event or condition that resulted in the condition of the nuclear plant, including its principal safety barriers, being seriously degraded; and (3) any event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to shutdown the reactor and maintain it in a safe condition, remove residual heat, control the release of radioactive material or mitigate the consequences of an accident.

b. Observations

The licensee determined that the extent of condition of the delamination was limited to the location of the steam generator replacement opening made in the containment between buttresses 3 and 4 to support the replacement of steam generators during RFO16. The licensee's root cause analysis determined that the delamination occurred during the creation of the steam generator replacement opening and was caused by the pattern and scope of the detensioning of tendons within the area of the opening. These activities were performed after the plant was shut down for RFO16 and when the plant was in Operating Mode 5. In accordance with Technical Specification (TS) 3.6.1, the containment is required to be operable in Operating Modes 1 through 4. The containment is not required to be operable to meet containment integrity requirements for Modes 5, 6, and NO Mode, but is required to perform the key safety function of containment closure. All fuel had been transferred out of containment before the liner was opened.

Therefore, based on the above information, the inspectors concluded that none of the reporting criteria under 10 CFR Part 50.72 or Part 50.73 applied and this condition was not reportable to the NRC.

3.0 Structural Integrity

a. Scope

The SIT reviewed the licensee's program that implements regulatory requirements required to maintain the structural integrity of the containment. These requirements included the structural integrity test conducted prior to plant initial operation, containment leakage tests required by TS 5.6.2.20 and 10 CFR Part 50, Appendix J, Option B, the inspections required by 10 CFR Part 50.55a and the requirements of ASME Section XI, Article IWL. Inspectors also reviewed the root cause and repairs associated with the containment dome delamination identified in 1976.

b. Observations

The SIT reviewed the licensee's Structural Integrity Test report conducted prior to plant initial operation. The structural integrity test was conducted in accordance with the intent of NRC Regulatory Guide 1.18, Structural Acceptance Test for Concrete primary Reactor Containment. The inspectors also reviewed the procedure used by the licensee to conduct the Containment Integrated Leak Rate Tests (ILRT). The ILRT incorporated the methods and provisions of the total-time method using the provisions of BN-TOP-1 or the mass-point method as specified in ANSI/ANS-56.8-1987 or ANSI/ANS-56.8-1994. Six ILRTs have been conducted on the Crystal River Unit 3 containment building, including the pre-service ILRT in October 1976. The other required ILRTs were performed in June 1980, July 1983, November 1987, November 1991, and December 2005. The inspectors reviewed each of the referenced ILRT results to verify that the acceptance criteria were satisfied. The inspectors also interviewed the engineer responsible for the licensee's program to evaluate his knowledge of the program and its requirements. The inspector determined that each ILRT satisfied the acceptance criteria and that the program engineer was knowledgeable and well versed in the requirements of the ILRT program.

The SIT also reviewed Progress Energy Procedure NAP-02, Preparation and Control of CR3 Site Specific Special Process Specifications and Guidelines. NAP-02, Appendix 4, Attachment Q, Visual Examination of ASME Section XI, Subsection IWL Components, is used as the specification for performing visual examinations of the concrete portions of the containment building. Procedure SP-182, Reactor Building Structural Integrity Tendon Surveillance Program, is used as the procedure for conducting examinations of the tendons specified in ASME Section XI, Subsection IWL. Procedure NDEP-0620, VT-1 and VT-3 Visual Examination of ASME Section XI, Subsection IWE Components of Nuclear Power Plants, defines the methods and requirements for performing the examinations for the steel liner specified in ASME Section XI, Subsection IWE. Procedure NDEP-A, Nuclear NDE Program and Personnel Process, establishes the nuclear nondestructive examination (NDE) program criteria and process for certification of NDE personnel, such as the inspectors performing the visual examinations (VT) for the IWL and IWE examinations specified in the ASME Section XI.

The SIT reviewed a sample of inspection results documentation for VT-1 and VT-3 examinations from various years. VT-3C and VT-1C reports were sampled and reviewed for the IWL visual inspections of the concrete associated with Refueling Outage 15 completed on December 3, 2007, which was completed and documented in connection with Work Order (WO) 681043-08. The Final Report for the 30th Year Containment IWL Inspection was reviewed to evaluate the adequacy of the tendon examinations conducted between October and November 2007. Visual examination (VT-3) reports were sampled and reviewed for the IWE visual inspections of the steel liner associated with the Refueling Outage 13, completed on August 21, 2003, and which was completed and documented in connection with WO 321405-01. Examiner qualifications were reviewed for inspectors that conducted visual examinations in connection with the Refueling Outage 15 IWL and IWE examinations.

Based on the review, the inspectors concluded that the structural integrity of the CR3 containment was maintained consistent with regulations and the requirements of ASME Code. No findings of significance were identified.

4.0 Adequacy of Maintenance and Inspection Programs

a. Scope

The SIT reviewed plant Maintenance Programs including plant administrative and implementing procedures associated with the plant's IWL and IWE programs. The IWL program requires visual examination of the containment concrete surface, and performing periodic testing of tendons. The IWE program also requires visual examination of the containment liner. In addition the SIT reviewed the results of these inspections, engineering evaluations, and related corrective actions.

The inspectors reviewed procedures and specifications for the in-service inspections (ISI) required by ASME Section XI, and reviewed the historical documentation regarding the dome delamination issue and associated repairs. The inspectors also reviewed a sample of ISI-related documentation, including various reports of results from examinations and examiner qualifications.

The SIT reviewed various licensee procedures, specifications, and documentation of inspection results to determine if surveillance programs for monitoring the concrete structure, tendons, and steel liner were in accordance with ASME Boiler and Pressure Vessel Code Section XI, Rules for ISI of Nuclear Power Plant Components. The inspectors reviewed Progress Energy Procedure NAP-02, Preparation and Control of CR3 Site Specific Special Process Specifications and Guidelines. The inspectors reviewed NAP-02, Appendix 4, Attachment Q, Visual Examination of ASME Section XI, Subsection IWL Components, which is used for performing visual examinations of the concrete portions of the containment building specified in ASME Section XI, Subsection IWL. Also reviewed was Procedure SP-182, Reactor Building Structural Integrity Tendon Surveillance Program, which is used as the procedure for conducting examinations of the tendons specified in ASME Section XI, Subsection IWL. The inspectors reviewed

Procedure NDEP-0620, VT-1 and VT-3 Visual Examination of ASME Section XI, Subsection IWE Components of Nuclear Power Plants, that defines the methods and requirements for performing the examinations for the steel liner specified in ASME Section XI, Subsection IWE, and reviewed Procedure NDEP-A, Nuclear NDE Program and Personnel Process, which establishes the nuclear NDE program criteria and process for certification of NDE personnel, such as the inspectors performing the visual examinations (VT) for the IWL and IWE examinations specified in the ASME Section XI.

The SIT reviewed a sampling of inspection results for scheduled VT-1 and VT-3 examinations. The inspectors reviewed Work Order 681043-08 that documented the VT-3C and VT-1C reports for the IWL visual inspections of the concrete associated with the Refueling Outage 15. The Final Report for the 30th Year Containment IWL Inspection of the tendon examinations conducted between October and November 2007 was also reviewed. VT-3 reports were sampled and reviewed for the IWE visual inspections of the steel liner conducted using WO 321405-01 during Refueling Outage 13, completed on August 21, 2003. Qualifications were reviewed for three inspectors who conducted IWL and IWE visual examinations during that Refueling Outage 15.

The inspectors verified the licensee was adequately examining, monitoring, and documenting ISI requirements in accordance with ASME Section XI. The inspectors verified that the NDE performance was adequate and personnel performing the examinations were appropriately qualified, trained, and performed examinations in accordance with documented procedures and specifications.

b. Observations

The inspectors verified that the rough and uneven surface condition of the dome has existed since the 1976 repair. The licensee has periodically completed numerous surface patches to attempt to address the surface spalls. Following additional reviews of the dome tendon stresses and monitoring, the uneven surface also appeared to be a result of concrete installation and finishing from the 1976 repair and not related to settlement of the dome or the concrete wall delamination issue.

The SIT concluded that the licensee's ISI program was adequate. No findings of significance were identified.

5.0 Extent of Condition and Root Cause

5.1 Extent of Condition

a. Scope

The SIT reviewed Progress Energy's (PE) determination of the extent of condition of the delamination, including results of the NDE and the testing of core bores taken from Bay 3-4, and other areas of the containment. The SIT also conducted a thorough review of the licensee's in-process information associated with the root cause analysis and the root cause report that determined the root cause is the scope and sequence of detensioning the tendons in preparation for making the Steam Generator Replacement (SGR) opening. The licensee's activities associated with its investigation into the extent of the delamination and its effect on the containment was also reviewed by the SIT inspectors.

b. Observations

The licensee contracted CTLGroup to perform the non-destructive testing (NDT) in support of the analysis to determine the extent of the condition associated with the delamination. CTLGroup's main objective was to characterize the extent of the delamination around the opening, and assess whether similar delamination existed elsewhere within the wall structure. CTLGroup performed initial trial testing to evaluate suitability of several NDT techniques in detection of the delamination in the containment wall structure. A test procedure/program was developed for the selected test methods. Impulse Response (IR) was selected as the most suitable method for detecting a delamination consistent with the delamination in Bay 3-4. Documentation was provided related to quality control, safety training, qualification requirements, and equipment calibration.

The SIT reviewed the licensee's information on the NDT that was performed at all accessible areas, including nearly all of the exposed exterior containment building wall surface, portions of wall areas accessed inside adjacent buildings, and a portion of the containment dome. Bays 1-2, 2-3, and 6-1 were tested over the entire exterior surface of each bay. Bay 3-4 was tested from the top of the equipment hatch enclosure to ~7' below the ring girder with the exception of the SGR opening and panels within the inner boundary of the delamination (~4,100 ft^2). Bay 4-5 was tested ~90% of the exterior surface (~5,200 ft^2). Bay 5-6 was tested over ~80% of the exterior surface (~2,500 ft^2). Portions of each bay were tested below the roof lines of the adjacent structures. The areas tested were limited by numerous factors such as, 1) high radiation and contamination areas, 2) physical restrictions due to plant equipment located near the walls or components attached to the walls, 3) areas in the fuel transfer area restricted by energized equipment and fuel transfer pool foreign material concerns. The expansive scope of this IR testing over the containment surface provided assurance that sufficient testing has been performed to detect delamination similar to what has been observed between Buttresses 3 and 4.

The SIT reviewed the CTLGroup project report that presented the test program and findings. IR data files with graphical imagery were provided to corroborate the IR Log Sheets that were used to document the IR results.

The SIT also reviewed concrete cores that were drilled to confirm the NDT results. Visual examination of core bored holes was performed using a boroscope. Cores were also taken to provide test specimens in support of the root cause analysis. Material test results were retained with the root cause investigation. ASME Section XI, IWL, visual examination of the affected areas was performed to identify non-conforming conditions observed since the last surveillance, which may have been caused by the delamination event.

The SIT reviewed licensee information in conjunction with NDT and core boroscope inspections performed on the dome, and the survey that was performed that replicated the dome surveillance established after the dome delamination event during construction. The dome survey was performed to identify if there were significant changes on the surface of the dome by comparing the current survey data to the final dome survey performed in

1981. A survey of accessible surfaces of the buttresses was performed to determine the relative position of the buttress corners. A survey was also performed inside containment.

Condition Assessment Results - IR and Confirmatory Core Bore Boroscope Examinations

The SIT reviewed the following licensee information on the methodology and results of the extent of condition analysis.

The IR test method employs a low strain transient impact, generated by a hammer, to send a stress wave into the concrete structure. The resulting bending behavior of the structure is analyzed to characterize the integrity of the structure. The IR analysis produces an average mobility, which is the principal parameter. Presence of significant voiding or an internally delaminated or unbounded layer will result in an increased average mobility value. On the other hand, a sound concrete element without distress will produce a relatively low average mobility value. The presence of delamination will effectively reduce the thickness of wall or slab responding to the impact, which results in a drastically increased average mobility value.

The IR method is utilized on a comparative basis, which allows the engineer to compare the difference in dynamic responses between test areas within the structure. The measured response data are correlated to condition via intrusive sampling such as core drilling or chipping. For those test areas that do not warrant intrusive sampling, as determined by the CTLGroup IR Team Leader/Senior Engineer, test results are evaluated and a technical basis for acceptance is provided. The basis for acceptance was reviewed and approved by the Progress Energy Civil Engineer.

Bay 3-4 was found to have large delamination with an hour glass shape centered at the SGR opening. The delamination was concluded to be within an area of approximately 80 ft by 60 ft, extending between the edges of the two buttresses in horizontal direction, and from top of the equipment hatch opening to approximately ten feet below the ring girder in the vertical direction. Average mobility values exceeded the potential damage threshold in the delaminated area. Based on core sampling and boroscope examination of core holes, the depth of delamination ranged from three to ten inches, with an average delamination depth between seven and eight inches from the exterior face. The delamination appeared to be in the plane of the hoop tendons. The extent of the delamination was confirmed by the core samples and boroscope examination.

The remaining five bays were found to have sound concrete, based on the IR test results. Significant delamination similar to that noted in Bay 3-4 was not found in the other areas of the containment wall structure. A small portion of the tested areas showed elevated average mobility values. Very isolated points had average mobility values slightly above the potential damage threshold. These areas required further investigation to determine the higher average mobility values, which include visual inspections, ground penetrating radar testing, hammer sounding, and core sampling. IR testing was also performed on selected areas of the dome, which is discussed later in this section.

The small portion of tested areas showing elevated average mobility values were evaluated on a case-by-case basis. A large number of elevated average mobility values were resolved due to a lack of a significant number of elevated values. Localized test locations with elevated average mobility values do not represent a condition similar to the observed delamination in Bay 3-4. A conservative correction factor was applied to normalize the average mobility values used for a specified population of panels. The normalization value determination produced a correction factor of 1.77; however, a correction factor of 2.0 was applied conservatively. This produced elevated average mobility values that were evaluated as acceptable based on the actual normalization value. A number of elevated average mobility values were resolved by determining the proximity of the test location to tendons. A tendon having less cover than normal will produce a higher average mobility value. Test locations between two closely spaced tendons may produce higher values without the presence of a defect. Core boroscope inspections confirmed this condition. Variations in material properties, such as less-than-normal coarse aggregate in the concrete, can influence IR testing. In all cases, the case-by-case justification of acceptance was provided by CTLGroup and reviewed and approved by Progress Energy as documented in WO 1636782-03.

The very isolated points above the potential damage threshold required additional investigation to determine the condition. Three panels exhibited some type of isolated internal defect. Bay designated RBCN-0003B, below the equipment hatch, had an elevated average mobility value. Core Number 54, 2-inch diameter x 12-inches deep, was drilled near the highest mobility value. The boroscope inspection detected an indication but a determination could not be made. It was observed that this location appeared to contain less large aggregate than normal. This condition would cause elevated average mobility values. An additional core, 4-inch diameter x 12-inches deep, was drilled directly over the two-inch diameter core. After removing the core sample and performing the boroscope inspection, it was clear that there was a crack. The core bore confirmed that the IR test detected the existence of a crack. The area was excavated (WO 1686388), the depth of the core exposing rebar and the top surface of a tendon sheath. The concrete was removed to a diameter of approximately eight inches at which point the crack could no longer be seen. With the defect removed, it was believed that the relatively large area of high mobility as shown on the IR Mobility Plot, for an isolated condition, was attributed to the less-than-normal amount of large aggregate.

Panel W of Bay RBCN-0012 had an elevated average mobility value that required investigation. The bottom left corner of the panel exhibited an isolated internal defect. Core Number 85 was near the location but didn’t reveal any indication of a crack. It was likely that this defect was a shallow spall similar to that of other panels that had experienced spalling around the perimeter of a panel at the edge of the feature strip surrounding the panels. (The feature strip is a ¾-inch deep by 4-inch wide horizontal and vertical recess located in the horizontal plane at the ten-foot placement joints and in the vertical plane at 20-foot intervals.) Concrete had fallen off in these areas or was removed for personnel safety concerns during the IR testing. The corner was sounded using a hammer and an isolated intact shallow spall was determined to be present with an area of slightly larger than one square foot. The shallow spall is to be removed by WO 1686388. Any unanticipated results found after the spall is removed will be addressed in a planned revision to this engineering change (EC).

The remaining defect was in RBCN-0011; Panel Q, in the upper right hand corner. Core Number 42 was drilled near the elevated average mobility value with no defect detected. The licensee was determined that another core would be necessary to characterize this suspect area after the area was sounded and nothing abnormal was detected. The IR Mobility Plot shows an area of elevated mobility with an area of approximately one square foot. The core was worked under WO 1686388. The results from the boroscope inspection indentified an anomaly approximately 8¾" deep. This anomaly was being investigated further and is assumed to be an isolated condition with no adverse structural affect, similar to that identified by IR at Core Number 54. Any unanticipated results found after the spall is removed were to be addressed by the licensee.

An area on the Reactor Building dome between Buttresses 3 and 4 was IR tested to assist with determining core bore locations to allow for examination of the dome after tendon detensioning. The core boroscope inspections assessed if changes occurred in the dome that would have been caused by tendon detensioning via EC 75218. These inspections also provided information regarding the condition of the interface between the original concrete and the repair material used to address the dome delamination that occurred in 1976. A total of seven cores were bored, two in the area of the dome that was not repaired after the dome delamination which occurred during construction and five in the repaired area. The mobility scale used for the walls in the six bays was not used due to the differences in various aspects such as thinness, orientation, materials, amount and configuration of tendons and reinforcement. Two relatively high average mobility values were identified in the repaired area. Cores were located at these locations to assess the condition. Boroscope inspection of each of the seven core locations did not detect evidence of cracking or delamination. The apparent bonding adhesive used in the repair could be seen and there was no appearance of a separation between the original concrete and the repair material. Although the core samples did separate upon removal when removed during boring operations, the licensee assumed that the boring process exerted forces at the interface of the original concrete and the repair material.

The scope of the dome IR testing expanded to the area of the dome between Buttresses 1 and 6 due to the planned tendon detensioning of vertical tendons between Buttresses 1 and 6 as part of EC 75218. The IR test data will be used to compare IR test results obtained after tendon detensioning, which was part of the testing section in EC 75218, the scope of which is not part of the Condition Assessment. Results from a core (Number 130) and boroscope inspection at a relatively high average mobility value indicate that an anomaly exists approximately seven inches deep in the repair area that has a step transition from the outer bounds of the delamination to the full excavation depth of approximately twelve inches. The licensee plans to investigate this anomaly further under NCR 388332. Any addition cores and boroscope results are planned to be incorporated into a revision to EC 75218. In addition to the IR testing and core bore drilling on the dome, a condition assessment was made on a depression in the concrete surface of the dome. This assessment was made in response to a NRC SIT request. The request asked if the depression is evidence of repeat delamination damage. The assessment concluded that the cracking was in the cosmetic patches used to smooth the surface during construction where a depression was present. The cosmetic patches were determined to have no correlation to the structural integrity of the Reactor Building dome. The licensee concluded that the thin concrete patches may be removed and not replaced without adversely affecting the operational function of the dome structure.

Reactor Building Containment Surveys

Surveys of the exterior and interior surfaces of the containment were performed at the request of the Containment Root Cause Team. An exterior survey of the containment dome and buttresses was performed to identify if there were significant changes to the surface of the dome compared to the final dome survey performed in 1981. The dome survey data did not indicate the presence of a new delamination of the dome surface. The buttress survey was performed to determine the relative position of the buttress corners. The buttress survey data showed that the vertical alignment of the buttresses was within the vertical alignment tolerance provided by American Concrete Institute (ACI) 117-90. A survey was also performed inside containment. Its purpose was to duplicate a survey performed as part of an original structural integrity test (SIT) measurement data set. The dome and buttress survey data was used by the Root Cause Team to address multiple Failure Mode (FM) Analyses. The survey performed of the internal diameter of containment was not used in the FM analysis but was used in the finite element model developed by Performance Improvement International (PII).

SGR Opening - Top and Bottom Wall Crack Assessment

An assessment was performed on cracks that were documented in Quality Control (QC) inspection report dated December 7, 2009. NCR 370853 documented this condition and was tracking the repair of the identified condition. The assessment concluded that the cracks in the top and bottom of the SGR opening were not the result of the delamination event, based on observations made after the initial event. The SGR opening and the overall repair surface area are to be mapped for cracks in accordance with EC 75219. The licensee planned to incorporate the assessment of any cracks and determine the necessary repairs in EC 75220.

Condition Assessment Conclusion

Delamination similar to that noted between Buttresses 3 and 4 was not found in other areas of the containment wall structure or dome. IR testing with confirmatory core bores identified delamination only between Buttresses 3 and 4 above the equipment hatch, as accurately predicted by IR. The few isolated areas found with small defects are being repaired by the Containment Repair Team. The remaining condition assessment information has been accepted by the Containment Root Cause Team and is being used as input into the root cause analysis as the team deems necessary

5.2 Root Cause Analysis

a. Scope

Based on the scope, significance, and complexity of the issue, the licensee’s overall Root Cause Investigation Team was broadly organized into four separate branches: Root Cause Analysis, Condition Assessment, Design Basis Analysis, and Repair Analysis. Each branch focused on a particular strategic area, with appropriate information sharing and cross-checking, such that each group’s efforts fed into the common goals of: (1) understanding the cause of the event, (2) determining the extent of the problem, and (3) identifying the repairs necessary to satisfy the design basis requirements of the containment structure. The inspection review documented in this section is limited to the technical root cause analysis work performed by the licensee’s Root Cause Analysis branch.

The SIT inspectors reviewed the licensee’s problem investigation activities related to determining the technical root cause(s) of the containment wall delamination event and corrective actions to prevent recurrence and/or propagation of the delamination. The inspectors reviewed the licensee's root cause analysis activities to determine whether the licensee's causal analysis activities were performed in accordance with applicable licensee procedures and standards. The objectives of the inspection were to determine: (1) whether the licensee's efforts and the determination of the technical root cause(s) and contributing causes were comprehensive and reasonable, and whether the time-line of its containment wall delamination event was reasonable. In addition, the SIT inspectors reviewed the licensee's responses to the SIT questions associated with the root cause analysis.

The inspectors reviewed documentation of all identified failure modes and contributing causes for significance. The inspectors also reviewed the licensee's new computer modeling approach and input material properties used to simulate the delamination in support of the root cause determination. The inspectors reviewed the corrective actions to prevent recurrence and for betterment identified in the licensee's root cause evaluation report to determine whether they addressed the causal factors. Note: The review and discussion documented by this inspection review relates specifically to the delamination failure.

The SIT review of the root cause analysis included 75 potential failure modes in the areas of: (1) Containment Design and Analysis; (2) Concrete Construction; (3) Use of Concrete Materials; (4) Concrete Shrinkage, Creep, and Settlement; (5) Chemically or Environmentally Induced Distress; (6) Concrete-Tendon-Liner Interactions; (7) Containment Cutting; (8) Operational Events; and (9) External Events. A total of 67 failure modes were refuted. The remaining eight FMs were combined to determine that the root cause was a combination of inadequate detensioning scope and detensioning sequence. The number and order of the detensioned tendons resulted in redistribution of stresses in the containment wall which led to the delamination. A much larger scope with a detailed sequence would be required to maintain acceptable tensile stress levels. The licensee determined that new modeling and evaluation techniques were required to successfully determine or predict margin to delamination.

b. Observations

Criterion XVI, "Corrective Action," of Appendix B of 10 CFR Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," requires that… "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition. The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action taken shall be documented and reported to appropriate levels of management." The delaminated condition of the CR3 containment wall was identified to be a significant condition adverse to quality and, therefore, this regulation required the licensee to perform and document a root cause evaluation to determine the cause(s) and implement corrective action(s) not only to correct the condition but also to prevent recurrence.

The CR3 reactor containment building provides the third and final fission product barrier against the uncontrolled release of radioactivity to the environment. The delamination event of October 2009 during RFO16 resulted in a degraded condition of the containment wall that reduced its structural capability and could challenge the ability of the structure to perform its safety function of providing a leak-tight barrier against the release of radiation to the environment and the public in the event of a design basis accident. The licensee classified the unprecedented containment wall delamination event as a "Significant Adverse Condition" in accordance with its procedure CAP-NGGC-0200, "Corrective Action Program," which implements the requirements of 10 CFR Part 50, Appendix B, Criterion XVI. Accordingly, the licensee initiated Nuclear Condition Report (NCR) Action Request 358724 (Assignment 1) and conducted a technical root cause evaluation (RCE) to determine cause and take corrective action(s) to prevent recurrence (CAPR).

The licensee conducted and documented a structured root cause evaluation of the "Significant Adverse Condition Investigation" of the delamination event in accordance with the guidance in its supplemental procedure CAP-NGGC-0205, Revision 11, "Significant Adverse Condition Investigations and Adverse Condition Investigations – Increased Rigor." The resulting report of the investigation documented the technical root cause evaluation and identified the root cause(s) and the contributing causes, including a failure mode timeline, of the CR3 containment wall delamination event and corrective actions to prevent recurrence (CAPRs) and betterment actions.

In accordance with procedure CAP-NGGC-0205, the licensee convened a root cause review team that performed an informal review of specified elements of the containment wall delamination root cause investigation to ensure that the procedural and investigative process requirements were met and to improve the quality of the report to the "highest degree possible" prior to the Quality Review Board (QRB) review. The technical root cause report was reviewed and approved by the licensee's QRB panel and the Plant Nuclear Safety Committee (PNSC). The inspectors observed the proceedings of the QRB and PNSC meetings for the root cause.

The licensee's investigation team for the root cause analysis comprised of individuals from across the Progress Energy fleet, industry peers, and an external "subject matter expert" root cause contractor. The contractor with whom Progress Energy partnered for the root cause analysis was PII, a consulting firm specializing and experienced in structured root cause investigations. The PII team consisted of over 15 engineering professionals and analysts with expertise in root cause investigations, concrete materials, behavior and testing, containment structural analysis and design, and advanced finite element analyses and computer modeling. PII's root cause evaluation work of the CR3 containment wall delamination event was performed under CR3's 10 CFR 50 Appendix B Program.

<u>Technical Root Cause Determination</u>

During RFO16, the delamination gap was observed in the vertical plane of the horizontal tendons, approximately ten inches from the outer surface, of the CR3 containment wall. The delamination was first observed while concrete removal (by hydro demolition) was in progress for the creation of a 25 ft wide by 27 ft high construction opening in the 42-inch thick containment wall to facilitate the steam generator replacement (SGR). The SGR opening was centered on azimuth 150^{o} between elevation (El) 183 ft and El 210 ft. The creation of the opening involved detensioning and removal of 17 horizontal (or hoop) tendons and 10 vertical tendons that traversed the foot print of the SGR construction opening. The licensee's condition assessment of the delamination event, using the impulse response non-destructive testing technique with confirmatory core boring and boroscopic examinations, determined that the extent of the delamination was limited to Bay 3-4 corresponding to an approximately 60 ft wide by 82 ft high hourglass-shaped area, which included the SGR construction opening, between Buttresses 3 and 4, extending from above the equipment hatch to approximately ten ft below the bottom of the ring girder. The tendon pattern at the SGR opening and the hourglass delaminated boundary is shown in Figure 5.2.1.

The root cause analysis team identified 75 FMs that could potentially cause or contribute to the delamination of Bay 3-4 of the CR3. These failure modes were binned into nine categories, as indicated below, and subsequently evaluated using a “Support/Refute” methodology to conclude whether or not it was supported as a causal factor.

1. Containment Design and Analysis
2. Concrete Construction
3. Use of Concrete Materials
4. Concrete Shrinkage, Creep, and Settlement
5. Chemically or Environmentally Induced Distress
6. Concrete-Tendon-Liner Interactions
7. Containment Cutting (SGR-related construction activities)
8. Operational Events
9. External Events

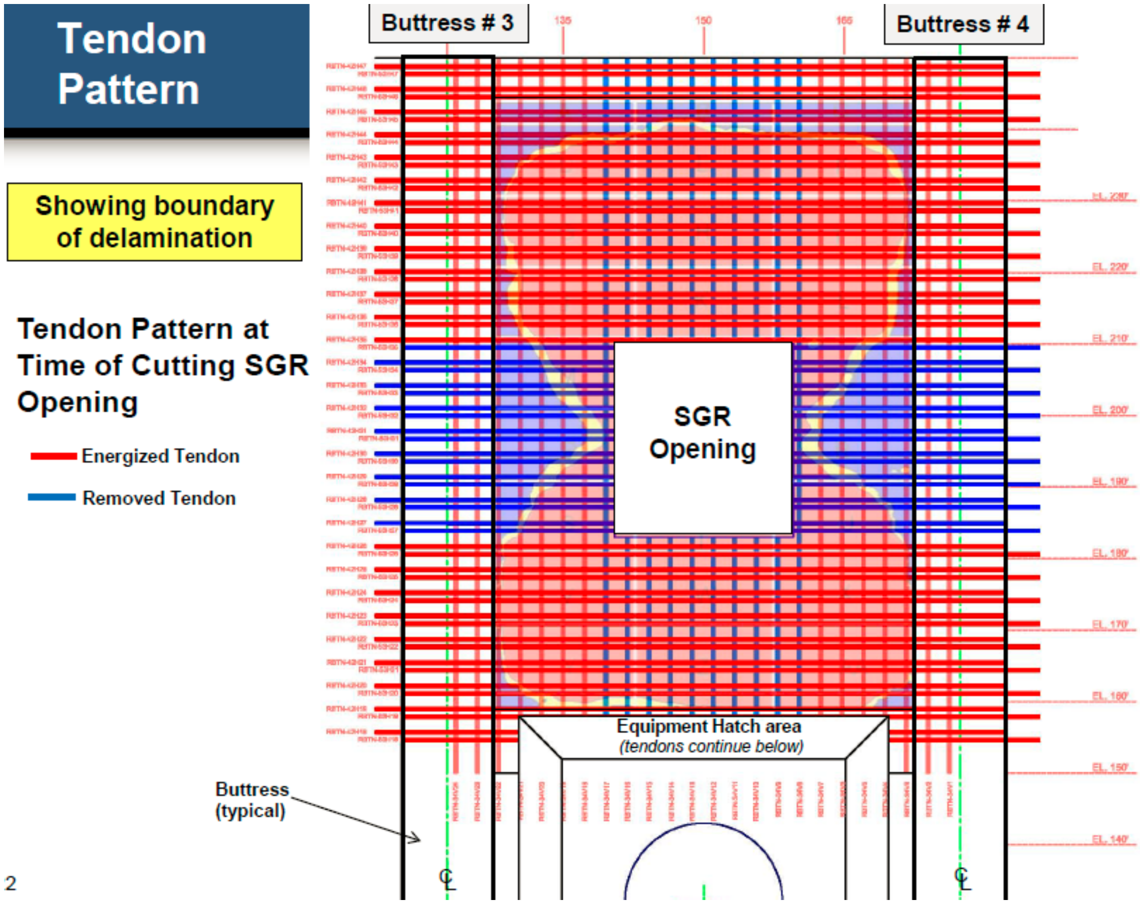

Figure 5.2.1 Tendon Pattern at SGR Opening and Delamination Profile

The support/refute methodology involved collection and analyses of information and evidence for each failure mode that allowed a conclusion to be drawn regarding its involvement (or lack of) in the event. The evidence used to inform the investigation included historical plant records of design, construction and operation of the CR3 containment, related industry operating experience, field observations and surveys, interviews of cognizant personnel, specialized technical literature, laboratory tests, and specialized analyses, including new state-of-the-art concrete fracture-based finite element computer models for simulation of the delamination event. These models allowed PII to start with information now known about how the CR3 containment structure behaves and work backwards in time to determine what factors contributed to the delamination event. The material property inputs (such as concrete elastic modulus, creep coefficient, tensile strength, fracture energy) used in the model were based on measured properties obtained following the delamination event.

The root cause team also reviewed and considered the operating experience and causal factors from the delamination events of the containment dome, all during original construction, at Turkey Point Unit 3 (June 1970), Crystal River Unit 3 (April 1976), and Kaiga Atomic Power Project Unit 2 (India, May 1994), as well as containment design data, from some of the other operating plants with similar post-tensioned containments that had successfully executed SGR construction openings. As each failure mode was investigated, the resulting conclusions were reviewed within PII, by Progress Energy, and also by third party reviewers. The licensee, based on the evidence and modeling, refuted 67 of the 75 failure modes and determined that they did not contribute to this event. The remaining 8 supported failure modes (associated with Categories 1, 2, 3, and 7) were confirmed by the licensee as contributors. The failure modes that were supported as contributors to the delamination event are discussed below. The licensee's root cause analysis determined that the delamination was not caused by any one of these factors acting alone, but was the result of the combined interplay of these contributing factors acting together.

- FM 1.1 Excessive Vertical and Hoop Prestress / Compressive-Tensile Interaction: The CR3 containment design uses larger diameter (163-wire) prestressing tendons, which results in higher vertical and hoop compression stresses and higher peak stresses, when compared to the design of other 6-buttress containments. The hoop prestressing system at CR3 is designed to resist 1.5 times the design accident pressure, which is in the high end among post-tensioned containment designs. Although the high level of prestress did not alone cause the delamination, the licensee determined that it likely reduced the margin to delamination by augmenting inherent weakness in the potential delamination plane.

- FM 1.2 Excessive Radial Tensile Stresses/No Radial Reinforcement: The high as-designed vertical and hoop prestress translates to relatively higher radial tensile stresses in the outer concrete region. The CR3 design did not provide radial reinforcement in these locations at the elevations of the delamination, which is not uncommon for post-tensioned containments of this vintage. The licensee determined that the high level of radial tensile stress did not alone cause delamination but reduced the margin to delamination.

- FM 1.15 Inadequate Design Analysis Methods for Local Stress Concentrations: The presence of the tendon sleeves in the concrete creates local stress concentration effects adjacent to the sleeves under prestressing force, which result in high peak tensile stresses, especially around the hoop tendon sleeves, that potentially caused small localized cracks during original tendon tensioning. These small cracks are potential sites for delamination initiation and propagation during modification activities such as tendon detensioning and concrete removal for SGR openings. These stress concentrations were not explicitly considered in the original containment design. The licensee determined that the predicted radial tension in the original design using standard industry calculation tools, based on only hoop prestressing force, is smaller than the actual.

- FM 2.12 Inadequate Strength Properties: The test results of concrete cores from the delaminated Bay 3-4 show that the tensile strength is variable and occasionally low, primarily due to the soft aggregate used in the concrete. Even though the average strength properties met design criteria for CR3, the tensile strength and fracture energy of the concrete was not adequate for radial stresses produced during the activities of tendon detensioning and of cutting the opening for the SGR project. Therefore, the licensee determined that the lower-than-normal concrete tensile strength was a contributor to the delamination.

- FM 3.4 Inadequate Aggregate: The Florida (Brookville) calcareous limestone coarse aggregate used in the CR3 containment concrete was relatively soft, porous, and gap graded. This aggregate was not appreciably stronger than the hardened cement paste and allowed cracks to propagate directly through the aggregate. Typically, the aggregate provides a dense stable medium that helps arrest and slow the propagation of cracks. While its properties were sufficient to produce concrete of acceptable specifications and design requirements, the tensile strength, elastic modulus and ability to arrest cracks were lower than normally found at other nuclear containment structures. Thus, the licensee determined that the strength properties of the coarse aggregate contributed to the delamination at CR3.

  It is noted that FMs 1.1, 1.2, 1.15, 2.12, and 3.4, discussed above, are inherent in the original containment design and construction, and the containment structure met the requirements for all design basis conditions.

- FM 7.3 and FM 7.4 Inadequate Scope and Sequence of Detensioning of Tendons (used for creation of SGR opening) (ROOT CAUSE): The number and order of detensioned tendons resulted in the redistribution of stresses in the containment wall. The redistributed stresses exceeded the tensile capacity (fracture energy) of the CR3 concrete and, therefore, this was the driver that caused the delamination. In an area as large as Bay 3-4 with large stress concentration effects on a detailed scale, it is probable that cracking began in isolated points of highest stress. As the stresses increased and shifted (due to the progression of growing number of tendons being detensioned), small cracks grew and joined until they eventually covered the entire delaminated area. Delamination occurs where radial tensile stress is high. Delamination occurs at sharp bends and points of reversal of curvature in the wall. The scope and sequence of detensioning tendons for the SGR created a bulge and curvature reversal in the wall and the stress created by the curvature of the bay wall precipitated the cracking and delamination of Bay 3-4. Figure 5.2.2 shows, to an exaggerated scale, the profile of radial displacements following original tendon tensioning, creep deformation, and progressive detensioning of tendons for the SGR opening.

  It is noted that an expanded scope and sequence of tendon detensioning was developed in March 2010 for the delamination repair and it successfully detensioned the containment without delamination in any of the other bays.

- FM 7.5 Added Stress Due to Removing Concrete at the SGR Opening: The licensee found that although the evidence seems to indicate that much of the delamination occurred prior to large scale removal of concrete, the removal of concrete at the construction opening increased the stresses in the remaining concrete which contributed to the final extent of the delaminated condition.

The inspection team noted that several of the causal factors of the CR3 containment wall delamination, i.e., biaxial compression-tension interaction, such as high radial tension; stress concentrations; concrete tensile strength; and material properties of the coarse aggregates, were consistent contributors as in the CR3 containment dome delamination event of 1976.

Following the analysis of failure modes, the licensee determined the timing of the contribution of each causal failure mode to the wall delamination event in the Failure Modes Timeline, as shown in Figure 5.2.3. The licensee's root cause analysis and computer simulation determined that the large-scale delamination observed did not exist prior to the SGR opening operations but occurred during construction activities for creation of the SGR construction opening, while the unit was shut down. This determination was supported by the following evidence: (1) the condition assessment confirmed that the delamination was present only in Bay 3-4, the panel where the SGR opening operations were performed, and not in any of the other five bays; (2) the shape of the delaminated area of Bay 3-4 corresponds exactly to the region of detensioned horizontal and vertical tendons (see Figure 5.2.1), indicating that the detensioning process influenced the delamination process, which could not happen if the delamination occurred prior to the detensioning; and (3) traditional petrographic analyses did not detect the presence of pre-existing defects (general lack of significant carbonation and mineral growth on the fracture surface).

In summary, the delamination was determined to have been caused by the combination and interplay of (1) certain design features of the CR3 containment structure; (2) the type of concrete used (material characteristics) in the CR3 containment structure; and (3) the acts of de-tensioning and cutting of the CR3 containment structure for creating the SGR opening. Through state-of-the-art nonlinear fracture-based computer models that PII developed using CR3 specific forensic information that was available after the delamination had taken place, the licensee determined that none of the individual contributing factors, on its own, would have caused a delamination. Rather, the complex interplay between all the contributing causes led to the delamination, with the tendon detensioning being the driver.

The licensee stated that, of particular importance in this analysis is that typical industry containment structure analysis tools that were used to calculate the radial tension and assess the delamination potential, when creating containment openings in other similar projects, have consistently shown large margins for stress tolerance to delamination by using average tensile stress values. The licensee stated that its computer simulation, however, had shown that the CR3 containment structure had a lower margin of tolerance to delamination cracking than other plants which have used common industry calculations.

In conclusion, the licensee's root cause evaluation determined that the immediate technical root cause of the delamination event was the redistribution of stresses, as a result of the SGR containment opening activities, resulting in additional stress beyond original containment design. The condition exceeded the fracture capacity/tensile strength of the concrete resulting in cracking along the high stress plane connecting the horizontal tendons. As the cracks propagated and joined, delamination occurred over a wide area. Evaluation of the contributing failure modes identified only one practical technical root cause subject to licensee's control, and that was inadequate scope/sequence of tendon detensioning used in the SGR outage. The licensee stated that the fracture-based computer simulation effort and post-event research by its contractor highlighted the inability (or limitation) of standard industry accepted analysis tools to predict delamination.

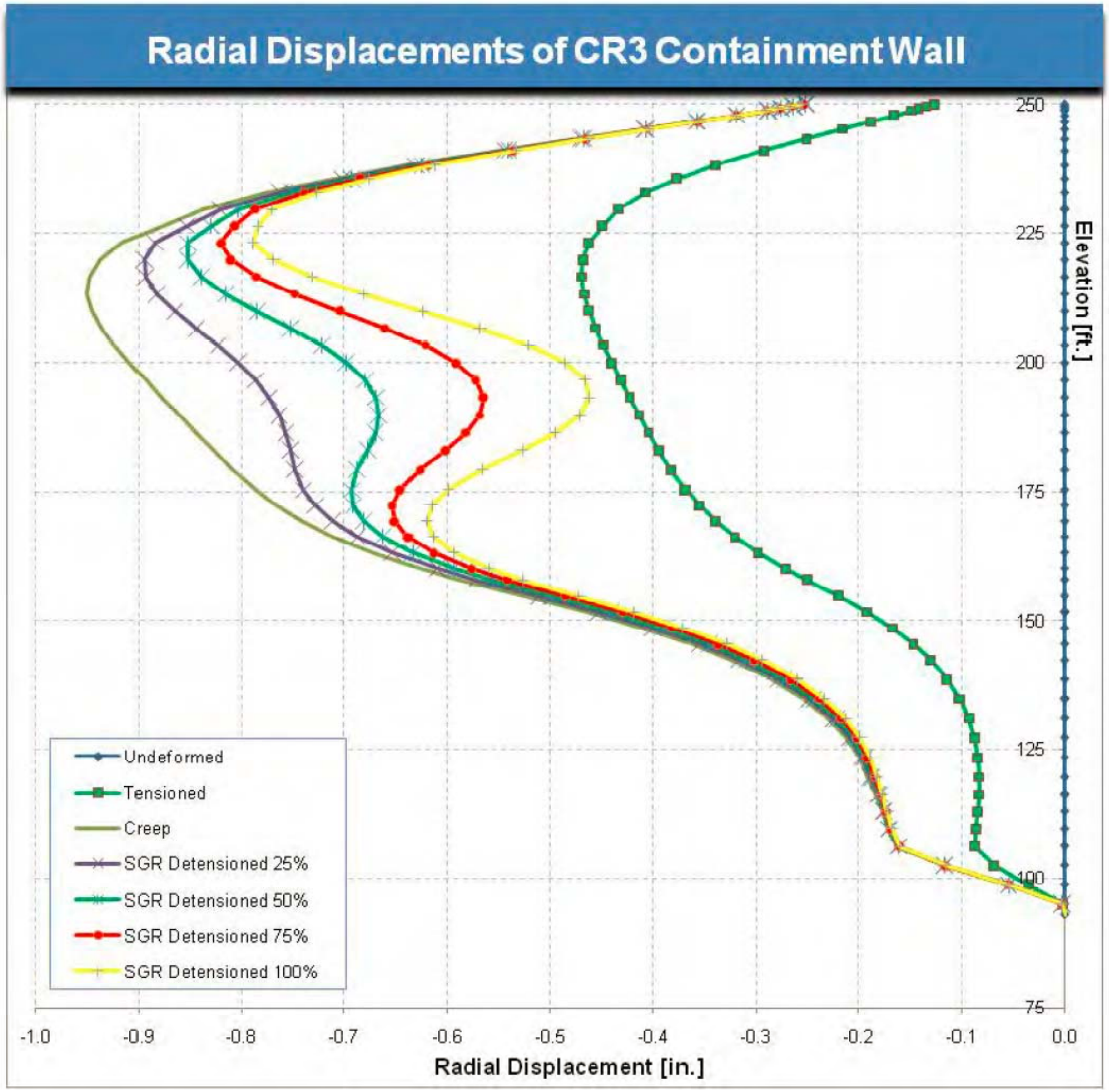

Figure 5.2.2 Bay 3-4 midline vertical profile of radial displacements during progressive SGR detensioning (Note that displacements are plotted to an exaggerated scale)

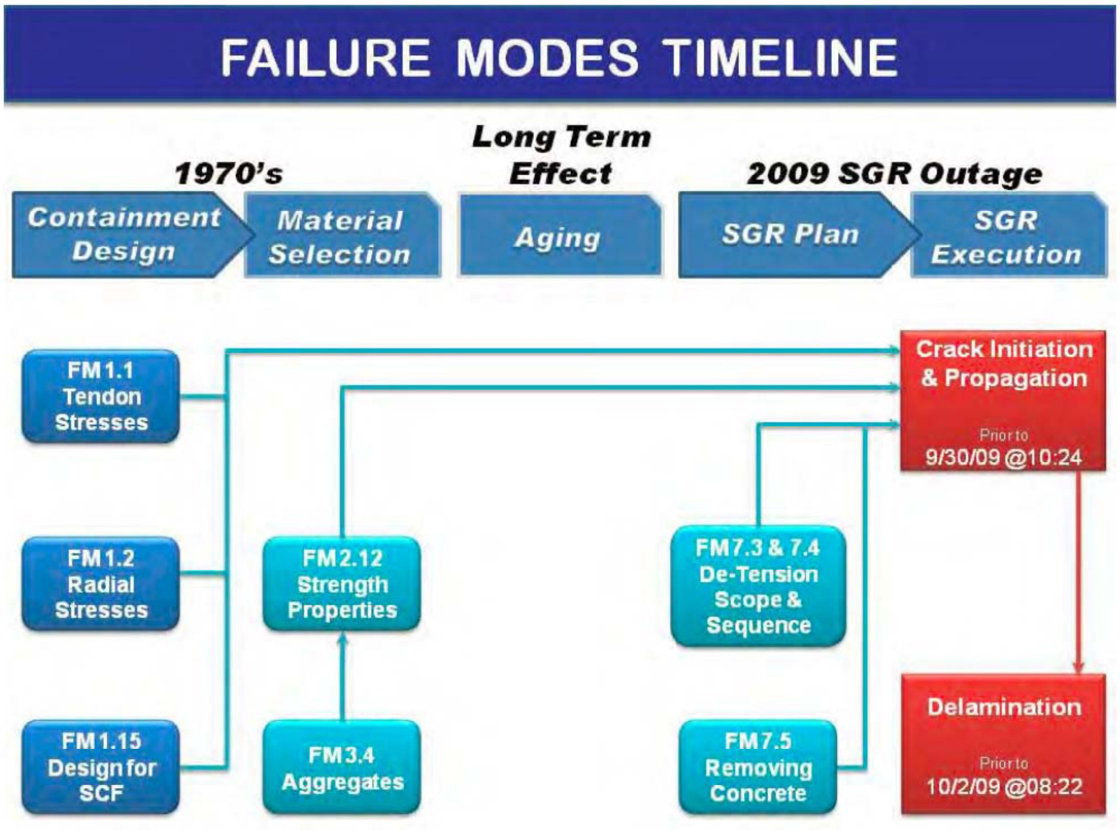

Figure 5.2.3 Failure Modes Timeline

<u>Role of Prestressing Force in Vertical Tendons in Producing Radial Tensile Stresses and Stress Concentrations in the Plane of the Hoop Tendons</u>

The licensee's root cause analysis report identified excessive radial tensile stress in the plane of the hoop tendons without radial reinforcement (FM 1.2) and peak tensile stress concentrations stresses in the concrete around the hoop tendon sleeve holes (FM 1.15) as contributing failure modes to the CR3 delamination event. These high radial tensile stresses and peak stress concentration effects, that are potential localized crack initiators around the periphery of the hoop tendon sleeves, are caused by the prestressing force in the hoop tendons, and apparently also the prestressing force in the vertical tendons, following initial tensioning. This was also physically indicated in the root cause investigation by the presence of small (approximately 5mm long) pre-existing cracks that were observed at the intersection of the fracture surface and tendon sleeves using phenolphthalein experiments on concrete specimens taken from the delaminated area. A paper by Acharya and Menon in Nuclear Engineering and Design (2003) highlights that significant local tensile stresses are developed at the periphery of the duct hole that can contribute to the problem of delamination, and that these localized effects are difficult to compute theoretically.

The common theoretical approach for calculating radial forces and stresses is a cylindrical shell analysis (such as in CR3 Calculation S09-0054, and reviewed in the paper in Nuclear Engineering and Design (2003) that gives an average radial tensile stress value in the plane of the hoop tendons and is based on the prestressing force in the hoop tendons only. This approach does not account for stress concentration effects around the tendon ducts, nor does it include the effect of prestressing forces in the vertical tendons in producing radial tensile stresses/concentrations in the plane of the hoop tendons. The licensee observed that standard industry analysis of this type (that predict average radial tensile stresses due to hoop tendon forces only) cannot adequately predict the radial tensile stresses that would lead to delamination, such as that observed at CR3. Therefore, in support of the root cause determination, the licensee implemented a three-dimensional fracture-based computer analysis methodology in order to be able to adequately simulate the delamination observed at CR3.

From the CR3 root cause analysis report, it is noted that the standard industry analysis for hoop pre-stressing performed for CR3 predicted the average radial tensile stress in the vertical plane of the hoop tendons to be of the order of 31 psi. The PII computer model calculated actual combined stress conditions at a given location due to all the contributing factors, whether they be from horizontal tendons, vertical tendons, or stress concentrating factors. For the PII linear elastic analysis case assuming no concrete cracking, the tensile stress peaks at a very high value at the edge of the horizontal tendon hole, at the intersection with a vertical tendon, due to the vertical and horizontal tendons. It reaches 1630 psi at the hole and declines rapidly by a factor of two for approximately every inch away. More generally, tensile stress peaks at about 510 psi on the edge of a horizontal tendon hole away from the vertical tendons and declines by a factor of two for every inch away. Note that the 1630 psi peak stress is of such a large magnitude that it would likely exceed the tensile strength of all concrete materials and initiate a small localized crack.

The SIT inspectors questioned the licensee about whether the contribution of vertical tendon prestress was adequately considered. The licensee agreed that the vertical tendon prestressing contributed to the concrete radial tensile stresses and stated that the PII simulation of the delamination event was achieved based upon modeling the combined effect of the entire set of conditions which produced radial tensile stresses and stress concentrations in the plane of the hoop tendons, without a separate quantitative assessment of the contribution of each individual factor. The licensee stated that the purpose of the analysis was to draw overall root cause conclusions versus separating the contribution of the horizontal and vertical tendons.

The SIT inspectors conducted a review by comparing the peak radial tensile stresses from PII's uncracked analysis with the average predicted radial tensile stress from the standard industry analysis, along with some calculations. The results indicated that the prestressing force in the vertical tendons could have a more significant effect than the hoop tendons in producing radial tensile stresses in the vertical plane (delamination plane) of the hoop tendons and peak tensile stress concentrations around the hoop tendon holes. These peak stresses can drive the formation of small localized cracks around the hoop tendon holes during tendon tensioning that become potential sites for delamination initiation and propagation during later modification activities that involve tendon detensioning. The peak tensile stresses can also be influenced by the level of bond between the tendon sleeve and the adjacent concrete. Typical industry calculations consider only radial tensile stresses caused by prestress from the hoop tendons (Reference CR3 calculation S09-0054, and paper in Nuclear Engineering and Design (2003)).

<u>Root Cause - Corrective Actions to Prevent Recurrence (CAPRs)</u>

The repair option selected by the licensee to correct the delaminated condition for continued operability was to “remove and replace the delaminated concrete.” Although the delamination was limited to Bay 3-4, all six bays and the dome of the containment building are susceptible to a similar event. Therefore, CAPRs developed by the licensee’s root cause investigation considered the entire containment building. The selected repair of the containment building requires additional detensioning and subsequent retensioning of tendons, and will therefore subject the building to additional stress redistribution events. Corrective actions specified in the root cause investigation are designed to minimize the risk of delamination in any areas of the containment building, either during the repair process or during modifications involving tendon detensioning and retensioning in the future.

The CAPRs developed by the licensee’s root cause analysis were:

1. Perform a detailed analysis of the tendon detensioning plan in support of the containment repair effort. Modify the plan as necessary and ensure that the stresses show positive margin as validated using CR3 delamination data. This had already been completed at the time of completion of this inspection and the licensee was successful in performing the expanded detensioning without propagating the delamination and without causing causing additional delamination.

2. Perform a detailed analysis of the tendon re-tensioning plan in support of the containment repair effort. Modify the plan as necessary and ensure that the stresses show positive margin as validated using CR3 delamination data.

3. Establish programmatic controls to prevent de-tensioning more than one tendon/group without a validated detailed analysis using CR3 delamination experience.

The corrective action plan from the root cause analysis also included additional “betterment” corrective actions, considering life extension and aging characteristics, with an appropriate tracking plan:

- Monitor displacement of the containment walls during re-tensioning to confirm the building response relative to computer prediction.

- Monitor the containment wall with strain gauges and acoustic instruments during re-tensioning to ensure that responses are within established limits per the repair design documents.

- Perform a detailed analysis of the stress consequences of typical activities such as heating up and cooling down of containment during outages or solar heating of an entire bay. Ensure that there is no cumulative impact with time.

- Establish an inspection plan to periodically monitor the containment concrete condition to ensure that there are no unexpected changes. The inspection should use NDE, such as Impulse Response mapping of the area and selective core drilling, in areas identified as suspect by NDE.

- Establish a monitoring program that evaluates the response of the installed containment monitoring sensors to ensure the two types of concrete in Bay 3-4 are behaving consistently as an indication of good coupling/bonding.

In accordance with its corrective action procedures, the licensee had a designated action item to perform an effectiveness review after the CAPRs and designated corrective actions have been implemented and sufficient time has elapsed to determine if the action(s) taken were effective in preventing recurrence of the condition and effective in resolving the investigated problem. This effectiveness review was to be performed using the following criteria: (1) monitoring of the containment building during Refuel Outage 17 using installed instrumentation and NDE as described in the monitoring programs established as part of the root cause investigation, to demonstrate that there is no delamination of concrete in the repaired wall or other bays; and (2) verifying that the tendon surveillance program has been revised to include the appropriate limitation on detensioning multiple tendons and the program owner understands the limitations.

Concluding Observations on the Root Cause Evaluation

The NRC SIT inspectors determined that the licensee performed its root cause investigation of the CR3 containment delamination event in accordance with its standard implementing procedures, CAP-NGGC-0200 and CAP-NGGC-0205, for its corrective action program. The inspectors determined that the licensee's problem investigation activities related to determining the technical root cause and contributing causes of the CR3 containment wall delamination event were comprehensive and thorough.

The inspectors determined that the technical root cause and contributing causes, and failure mode timeline, identified by the licensee were reasonable, and adequately supported by appropriate evidence. The inspectors determined that the licensee's investigation results reasonably supported their conclusion that the technical root cause of the delamination was attributed to the scope/sequence of tendon detensioning used for the creation of the SGR construction opening, in combination with other contributing factors related to certain design features of the CR3 containment structure, the materials used in the containment concrete, and the activities related to the cutting of the SGR opening. The inspectors determined that the approach and inputs used in developing fracture-based computer models used to simulate the delamination were reasonable.

The inspectors also determined that the corrective actions developed/taken by the licensee to prevent recurrence and/or propagation of the containment wall delamination, as well as, other related betterment corrective actions developed from the root cause investigation were appropriate and addressed the causal factors.

The NRC inspectors noted that the contribution of the prestressing force in the vertical tendons in causing radial tensile stresses and stress concentrations in the delamination plane of the hoop tendons may be significant and may require research to determine its significance and impact, if any, in design of post-tensioned containments. This is important since vertical tendons are typically tensioned before hoop tendons during tendon detensioning. Typical industry calculations to assess radial tension are based on hoop prestressing force only. The PII modeling appropriately considered the combined effect of all contributing factors, including vertical tendons, in producing radial tension in the delamination plane, but the relative significance of each individual contributing factor was not separated in the assessment.

The licensee presented their root cause to the NRC Special Inspection Team and staff of NRC Region II and Office of Nuclear Reactor Regulation during a meeting held at the CR3 site on June 14, 2010. Results of the root cause analysis were also presented to the NRC staff during a public meeting held on June 30, 2010 at NRC Headquarters in Rockville, MD. Information and data presented at the meetings were consistent with the NRC's independent review and on-site verification. The licensee's root cause analysis report was used for the inspection team's review for potential generic implications addressed in SIT Charter Item 8.

6.0 Corrective Actions

a. Scope

To repair the containment for continued operability, the licensee elected to remove and replace the delaminated part of the containment wall. The licensee performed design calculations in support of implementation of the selected repair such that the restored containment is in compliance with the plant's licensing design basis described in the FSAR. SIT inspectors conducted a review of selected evaluations relative to the licensee's new finite element analysis and evaluations of the interim and restored configuration of the containment under design basis loads and load combinations described in the FSAR.

b. Observations

The licensee reviewed several repair alternatives. Their review of repair options included consideration for restoration of design margin, extended life, and root cause findings. Some of the rejected repair options included use-as-is, anchorage only, cementious grout, and epoxy resin. The final repair decision was to remove and replace the delaminated part of the containment wall. The licensee's repair plans included: (1) additional detensioning of containment; (2) removal of delaminated concrete; (3) installation of reinforcement, including radial reinforcement through the delamination plane; (4) placing of new concrete; (5) retensioning containment; and (6) post-repair confirmatory system pressure testing.

After the tendon detensioning and removal, the delaminated concrete was removed. Upon removal of the delamination, the licensee observed vertical cracking in the concrete surface between Buttresses 3 and 4. The cracks were evaluated for size and through-wall penetration. These vertical cracks essentially lined up with the location of the vertical tendons and in most cases went through-wall to the containment liner. The cracks were excavated and were filled in with new concrete. The licensee evaluated the cracks below El 176 and determined them to be acceptable as-is. The licensee also identified tight vertical cracks in other locations around the circumference of the containment. The evaluation of these was on-going at the conclusion of this inspection.

Plant Design Basis Analysis

The SIT inspectors also conducted a review of selected licensee's design basis analysis calculations and evaluations that were in progress and ongoing. The inspectors reviewed containment structural drawings and the applicable portions of the FSAR in order to determine the appropriateness of the design parameters used in these calculations. At the conclusion of this inspection, the licensee was continuing to develop the design basis calculations to bring the containment structure into compliance with the CR3 design basis as described in the FSAR, and to respond to questions that arose as the result of inspectors' review of licensee engineering calculations. The NRC continued to review the available calculations related to the structural modeling, loads and load combinations, design and analysis methodology, structural acceptance criteria, and design of reinforcing steel for the repair of the delaminated bay.

The SIT inspectors began a review of the licensee's evaluations to establish: (1) correlation between the new finite element analysis of the containment structure and the original design basis computer analysis; (2) calculations to demonstrate compliance with the structural acceptance criteria specified in the FSAR and the CR3 design code of record, ACI 318-63; and (3) the design of the required reinforcing steel for the repair of the delaminated bay, all of which to bring the containment structure into compliance with the CR3 design basis as described in the FSAR. The SIT identified errors in data used in calculation S10-0030. These errors were incorporated into the plant's corrective action program for further evaluation to ensure that they did not impact the final results of this calculation. The licensee was conducting additional evaluations on other calculations to determine the extent of this condition.

The NRC planned continued review of selected licensee engineering calculations related to the design basis of the containment structure and its ability to function under all applicable loading conditions. The continued review will be conducted as part of inspections under NRC Inspection Procedure (IP) 50001, "Steam Generator Replacement Inspection."

At the conclusion of the Special Inspection, the licensee had further detensioned the containment and had removed the delaminated concrete. The licensee's repair activities and design engineering evaluations remained ongoing.

7.0 Safety Significance

The licensee's root cause report determined that the delamination was caused by the scope and pattern of detensioning the tendons in preparation of cutting the construction opening for replacement of the steam generators. The inspectors found that the licensee's investigation was thorough and supported its conclusion that the delamination was caused by tendon detensioning. Therefore, the assessment of the safety significance related to the formation of the delamination was limited to the time period after the beginning of RFO16 and when the plant was shut down in Operating Mode 5.

The following information is from the plant operator logs.

- On September 25, 2009, at 7:03 p.m., the plant commenced power reduction for refueling outage 16.
- On September 26, 2009, at 4:51 p.m., the plant was shut down in Operating Mode 5.

- On September 26, 2009, at 4:58 p.m., authorization was given to begin work on tendon activities.
- On October 1, 2009, at 5:44 p.m., the plant was in Mode 6
- On October 9, 2009 the plant was in No Mode with all fuel removed from the reactor and placed temporally in the spent fuel pool.

After reaching Mode 5 on September 26, 2009, two vertical tendons (34V12 and 34V13) were detensioned simultaneously. From September 26 through October 1, additional vertical tendons and 16 hoop tendons were detensioned as part of the process to make the required construction opening to support the steam generator replacement. Hydrolazing activities began on September 30, 2009. The hydrolazing activities continued through October 7. Work to create an opening in the containment steel liner began on October 15.

According to plant TS, the containment is not required to be operable to meet containment integrity requirements for Modes 5, 6, and "No Mode," but is required to perform the key pressure retaining safety function of containment closure. Because the containment liner was intact and functional until October 15, 2009, when the liner was opened, the containment was able to perform this function.

Based on the above time-line, and plant technical specifications, all activities associated with the work on the steam generator construction opening related to the delamination were conducted either when containment operability was not required or when all fuel had been transferred to the spent fuel pool. The inspectors determined that the delamination did not represent an increase in risk to the public.

No findings of significance were identified.

8.0 Generic Issues

The SIT did not identify immediate generic safety concerns.

The CR3 containment wall delamination was an unprecedented event. The SIT determined that some aspects of the issue may warrant consideration for further industry review and information sharing.

Some of the contributing causes (such as high radial tension without radial reinforcement, stress concentrations around the tendon sleeves, etc.) may be inherent in the containment designs of many post-tensioned plants, and therefore, licensees should be aware of the potential adverse effects of these conditions when evaluating potential containment modifications that involve detensioning of tendons. From the licensee's root cause analysis, it appears that standard industry analysis tools typically used for predicting radial tension may be limited in their ability to predict the potential for delamination failures for major modification activities, such as the creation of SGR construction openings that involve detensioning of tendons.

As contributing causes to the CR3 delamination event, the root cause analysis identified high radial tensile stresses in the delamination plane of the hoop tendons and peak tensile stress concentrations around the hoop tendons sleeves. These high tensile stresses could initiate one or more small localized cracks around the hoop tendon sleeves during original tensioning (or subsequent retensioning, if performed) which then become potential sites for delamination initiation and propagation under additional drivers such as detensioning during later modification activities. The licensee's root cause analysis acknowledges that the prestressing force in the vertical tendons contributes to causing radial tensile stress and stress concentrations in the vertical plane of the hoop tendons, in addition to prestressing from the hoop tendons. The inspectors noted that this finding maybe a departure from a common belief that radial tension in the plane of the hoop tendons is caused by the prestressing force in the hoop tendons only. Typical industry calculations only consider radial tensile stresses caused by prestressing from the hoop tendons.

It appears that the contribution from the vertical tendons in producing radial tensile stress may be significant and possibly even more significant than the contribution from hoop tendons. Although the PII computer simulation included the combined effect of all conditions in producing radial tension that causes delamination, it did not provide a quantitative assessment of the relative contribution and significance of individual factors (primarily the prestressing force in hoop tendons and vertical tendons) in producing the high radial tensile stresses, which are potential local crack initiators. This issue may warrant further review to understand the role and relative significance of prestress in vertical tendons in causing local tensile stress around tendon conduit and address its impact, if any, on design of post-tensioned containments. Exploratory analytical research may be warranted to positively quantify the relative contributions and significance of the prestressing force in the vertical tendons and hoop tendons in producing radial tensile stresses and peak tensile stresses due to stress concentration effects, which could result in small local cracks at the tendon holes. This is important because the vertical tendons are typically tensioned before the hoop tendons. Any research in this area should also factor in the role of the bond between the tendon sleeves and adjacent concrete in producing tensile stress concentration effects.

Meetings, Including Exit

On September 2, 2010, the inspectors presented the inspection results to Mr. J. Franke and other members of the licensee staff in an exit meeting open for public observation at the Crystal River Nuclear Plant EOF/Training Center, 8200 West Venable Street, Crystal River, Florida.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

B. Komara, National Inspection and Consultants (NIC)
Charles Williams, HNP Technical Services Superintendent
Craig Miller CR3 Lead Mechanical Engineer
Dennis Herrin CR3 Licensing
Don Dyksterhouse NPC Supervisor
Ed Avella CR3 Major Projects Manager
Emin Ortalan CR3 Design Engineering Supervisor
G. Pugh, Progress Energy
Garry Miller, Vice President, Progress Energy
Glenn Pugh CR3 Senior Civil/Structural Engineer
Howard Hill CR3 Contractor
J. Franke, Vice President, CR3
Joe Lese CR3 Lead Engineer
John Holliday CR3 Contractor
Martin Souther BNP Senior Engineer
Paul Fagan RNP Technical Services Superintendent
R. Pepin CR3 Mechanical Maintenance Superintendent
R. Portmann CR3 IWE/IWL Program Manager
R. Knott NPC Lead Engineer
R. Tyrie CR3 Operations Supervisor
S. Cahill, Engineering Manager, CR3
W. Rogers HNP Senior Engineer
W. Worthington CES Lead Engineer

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

None

Sequence of Events
Time Line

Plant Operations

9/25/2009 7:03:00 p.m., commenced power reduction for R16 planned outage per OP-209A.

9/26/2009 12:29:00 a.m., CR3 is in Mode 2.

9/26/2009 12:44:00 a.m., CR3 is in Mode 3.

9/26/2009 8:39:00 a.m., CR3 has entered Mode 4.

9/26/2009 4:51:00 p.m., CR3 has entered Mode 5.

9/26/2009 4:58:00 p.m., authorized to commence tendon work activities.

10/1/2009 5:44:31 p.m., Crystal River Unit 3 is in Mode 6.

Steam Generator Opening Activity Time Line

At approximately 5:00 p.m. on September 26, 2009, following entry into Mode 5, the tendon crews began detensioning and removing the tendons in the area of the construction opening. Vertical tendons 34V12 and 34V13 were detensioned first. These tendons were detensioned prior to cutting the button heads in case they needed to be reused. The button heads on the remaining tendons were plasma cut under full tension. Following 34V12 and 34V13, the crews started on vertical tendon 34V8 and progressed to 34V17 while simultaneously detensioning horizontal tendons 42H27 and 53H27 through 42H34 and 53H35. The horizontal tendons were detensioned from the bottom to the top. The detensioning and tendon removal was completed on October 1, 2009.

On September 30, 2009, the hydro-excavation crew performed a test demonstration on an 8 feet wide by 6 feet high area in the lower right corner of the construction opening. The depth of the concrete removal generally varied from 4-1/2 inches to 8 inches, and exposed some horizontal and vertical tendon sleeves.

Full scale hydro-excavation commenced at approximately 4:30 a.m. on October 1, 2009 and removed the outer layer of concrete down to the rebar by 1:00 p.m. that afternoon. Hydro-excavation restarted at about 4:00 a.m. on October 2, 2009, following removal of the rebar, but was stopped an hour later due to obstructions in the debris chute that caused water to spill onto the berm. During this period of hydro-excavation operation, workers identified a stream of water flowing from a crack below and to the right of the construction opening. A subsequent inspection of the construction opening revealed the presence of the separation (delamination).

LIST OF DOCUMENTS REVIEWED

GAI Report No. 1930, Crystal River Unit 3 Nuclear Generating Plant Reactor Containment Building Structural Integrity Test, dated December 7, 1976.

Crystal River Unit 3 Nuclear Power Plant Reactor Containment Building Integrated Leakage Rate Test Report, dated June 30, 1980

Crystal River Unit 3 Nuclear Power Plant Reactor Containment Building Integrated Leakage Rate Test Report, dated July 11, 1983

Crystal River Unit 3 Nuclear Power Plant Reactor Containment Building Integrated Leakage Rate Test Report, dated November 15, 1987

Crystal River Unit 3 Nuclear Power Plant Reactor Containment Building Integrated Leakage Rate Test Report, dated November 7, 1991

Progress Energy Crystal River Unit 3 Plant Operating Manual SP-178 Containment Leakage Test – Type “A” including Liner Plate. Revision 29C (used for 2005 ILRT)

Florida Power Letter 3F0301-05 to NRC, dated March 7, 2001; License Amendment Request #267, Revision 0; Revision to Improved Technical Specification 5.6.2.20, "Containment Leakage Rate Testing Program"

NRC Letter to Florida Power responding to Florida Power Letter 3F0301-05, issuing Amendment No. 197 to Facility Operating License No. DPR-72 for Crystal River Unit 3.

Analysis/Calculation S09-0045, Integrity Evaluation of Cracked Containment Shell for LODHR (Loss of Decay Heat Removal), Revision 0, dated October 6, 2009

Progress Energy Crystal River Unit 3 Plant Operating Manual SP-178 Containment Leakage Test – Type “A” including Liner Plate. Revision 30

Licensee Event Report (LER) 97-014, Report Date: July 11, 1997

Final Safety Analysis Report, Revision 31.3, Section 5.6.3, Initial Integrated Leak Rate Test.

Final Safety Analysis Report, Revision 31.3, Section 5.6.4, Operational Leak Rate Tests.

NAP-02, Preparation and Control of CR3 Site Specific Special Process Specifications and Guidelines

SP-182, Reactor Building Structural Integrity Tendon Surveillance Program

NDEP-0620, VT-1 and VT-3 Visual Examination of ASME Section XI, Subsection IWE Components of Nuclear Power Plants

NDEP-A, Nuclear NDE Program and Personnel Process

Work Order 1636782-03

Work Order 681043-08

Work Order 321405-01

Work Order 1686388

Final Report for the 30th Year Containment IWL Inspection, Dated 02/04/2009

AR 368389

NAP-02, Preparation and Control of CR3 Site Specific Special Process Specifications and Guidelines

SP-182, Reactor Building Structural Integrity Tendon Surveillance Program

NDEP-0620, VT-1 and VT-3 Visual Examination of ASME Section XI, Subsection IWE Components of Nuclear Power Plants

NDEP-A, Nuclear NDE Program and Personnel Process

Work Order 681043-08

Work Order 321405-01

Final Report for the 30th Year Containment IWL Inspection, Dated 02/04/2009

AR 370875

Calculation S10-0002, Revision 0; Finite Element Model Description

Calculation S10-0032, Revision 0; Containment Repair Project – Limiting Load Cases

Calculation S10-0014, Revision 0; Containment Repair Project - Finite Element Model Benchmarking

Calculation S10-0040, Revision 0; Containment Repair Project – Comparison of ANSYS Results to Kalinin's SIT Calculation and SIT measurements

Calculation S10-0012, Revision 1; Containment Repair Project – Stresses Around the SGR Opening due to Design Basis Load Cases

Calculation S10-0015, Revision 0; Containment Repair Project – Refined Containment Finite Element Model Evaluation

Calculation S10-0021, Revision 4; Containment Repair Project – Concrete Radial Reinforcement (Grouted Reinforcement Option)

Calculation S10-0030, Revision 0; Containment Repair Project – Reinforcement Design for Delaminated Containment Wall

Significant Adverse Condition Investigation Report, Action Request Number 358724 (Attachment 1) for Event Date 10/2/09, Revision 0 approved by PNSC on June 7, 2010. Attachment 1 of this is the report "Root Cause Assessment – Crystal River Unit 3 Containment Concrete Delamination," dated June 7, 2010, by Performance Improvement International

Engineering Change Package EC 75218, "Reactor Building Delamination Repair Phase 2 – Detensioning Tendons," Attachment Z40R2 - PII Evaluation of Detensioning Plan and Concrete Removal which includes Margin to Delamination Analysis of Proposed CR3 Detensioning Sequence Option 10F.

Crystal River Unit 3 Nuclear Generating Station, Reactor Building Dome Delamination Final Report, December 10, 1976, Prepared for Florida Power Corporation by Gilbert/Commonwealth Associates Inc., Reading, PA.

Progress Energy Nuclear Generation Group Standard Procedure CAP-NGGC-0205, Significant Adverse Conditions Investigations and Adverse Condition Investigations – Increased Rigor, Revision 11.

Progress Energy Nuclear Generation Group Standard Procedure CAP-NGGC-0200, Corrective Action Program, Revision 28.

Progress Energy CTL Group Project No. 059169, Nondestructive Evaluation of Delamination in a Containment Wall Structure, Crystal River Nuclear Plant Unit 3, January 22, 2010, Submitted by CTL Group, Skokie, IL

Calculation S09-0054, Rev. 0, Containment Repair Project – Radial Pressure at Hoop Tendons

Acharya, S., and Menon, D., Prediction of radial stresses due to prestressing in PSC shells, Nuclear Engineering and Design 225 (2003) 109-125, Elsevier.

LIST OF ACRONYMS USED

ACI	American Concrete Institute
AREVA	AREVA NP (formerly Framatome ANP) a supplier of nuclear services
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing Materials
CAP	Corrective Action Program
CAPR	Corrective Action to Prevent Recurrence
CFR	Code of Federal Regulations
CLB	Current Licensing Basis
CR3	Crystal River Unit 3
EC	Engineering Change
EFPY	Effective Full Power Years
El	Elevation
EPRI	Electric Power Research Institute
FM	Failure Mode
FSAR	Final Safety Analysis Report
Ft	foot (feet)
ILRT	Integrated Leak Rate Test
INPO	Institute of Nuclear Power Operations
IP	Inspection Procedure
IR	Impulse Response
ISI	Inservice Inspection
LERF	Large Early Release Frequency
LLRT	Local Leak Rate Test
LR	License Renewal
NCR	Nuclear Condition Report
NDE	Non-Destructive Examination
NDT	Non-Destructive Testing
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NUREG	Publications Prepared By The NRC Staff
OE	Operating Experience
PdM	Predictive Maintenance
PE	Progress Energy
PII	Performance Improvement International
PNSC	Plant Nuclear Safety Committee
Psi	pounds per square inch
PWR	Pressurized Water Reactor
QC	Quality Control
QRB	Quality Review Board
RAI	Request for Additional Information
RCE	Root Cause Evaluation
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RFO	Refueling Outage
SGR	Steam Generator Replacement
SIT	Special Inspection Team and Structural Integrity Test
UT	Ultrasonic Examination
VT	Visual Examination
WO	Work Order

October 13, 2009

MEMORANDUM TO: Louis F. Lake, Team Leader
Special Inspection

FROM: Luis A. Reyes ***/RA/***
Regional Administrator

SUBJECT: SPECIAL INSPECTION CHARTER TO EVALUATE CRYSTAL RIVER CONTAINMENT BUILDING

You have been selected to lead a Special Inspection to assess the circumstances associated with a delamination discovered in the concrete of the containment building at Crystal River Unit 3. Your onsite inspection should begin on October 13, 2009. Robert Carrion, Senior Reactor Inspector, George Thomas of NRR, Anthony Masters, Senior Construction Inspector, and an independent Structural Engineering Contractor (TBD) will assist you in this inspection.

A. <u>Basis</u>

Recently, the plant was shut down for a planned refueling outage as well as to replace the steam generators inside containment. In order to move the steam generators into containment, workers began removing concrete to create the necessary opening. During that work, a small gap was found while making approximately a 25 foot x 25 foot concrete cut (liner is still intact). The gap is about one half inch wide and 10 inches inward from the outside edge of the concrete and located just at the layer of horizontal tendons. The Crystal River containment is about 42 inches thick, contains both horizontal and vertical tensioned steel tendons, and is lined with steel plate.

The licensee is evaluating the extent of the condition. The discovery of this gap in the concrete does not represent an immediate safety concern because the plant is shut down.

In accordance with Management Directive 8.3, “NRC Incident Investigation Program,” deterministic and conditional risk criteria were used to evaluate the level of NRC response for this event.

Based on the deterministic criteria that this issue involved possible adverse generic implications, the event was evaluated for risk in accordance with Management Directive 8.3. Due to lack of information on the structural integrity of the concrete containment and how it would interact with the free-standing steel liner during a seismic event in its current condition, a risk analysis was not able to be performed at this time. However, if

CONTACT: Mark Franke, RII/DRS
404-562-6349

the concrete containment reacted adversely with the free-standing steel liner during a seismic event, LERF would be adversely affected and a Special Inspection would be warranted. Therefore, Region II has determined that the appropriate level of NRC response is to conduct a Special Inspection.

B. Scope

The inspection is expected to perform data gathering and fact-finding in order to address the following:

1. Develop a complete description of the problems and circumstances surrounding the gap in the containment building.

2. Verify that the licensee has appropriately evaluated Operability and Reportability.

3. Review structural integrity testing data of the containment.

4. Assess the adequacy of the licensee's maintenance and inspection programs related to this event.

5. Assess the licensee's activities related to the problem investigation (e.g., root cause analysis, extent of condition, etc).

6. Assess the licensee's corrective action/"fix" in addressing the containment delamination issue.

7. Collect data necessary to develop and assess the safety significance of any findings in accordance with IMC 0609, "Significance Determination Process."

8. Determine potential generic issues or any design and construction inadequacies and make recommendations for appropriate follow-up actions (e.g., Information Notices, Generic Letters, and Bulletins).

C. Guidance

Inspection Procedure 93812, "Special Inspection," provides additional guidance to be used during the conduct of the Special Inspection. Your duties will be as described in Inspection Procedure 93812. The inspection should emphasize fact-finding in its review of the circumstances surrounding the event. Safety or security concerns identified that are not directly related to the event should be reported to the Region II office for appropriate action.

Your team will report to the site, conduct an entrance, and begin inspection no later than October 13, 2009. In accordance with IP 93812, you should promptly recommend a change in inspection scope or escalation if information indicates that the assumptions utilized in the MD 8.3 risk analysis were not accurate. A report documenting the results of the inspection should be issued within 45 days of the completion of the inspection. A copy of the inspection report shall be forwarded to the Crystal River Unit 3 License Renewal Inspection Team. The report should address all applicable areas specified in Section 3.02 of Inspection Procedure 93812. At the completion of the inspection, you should provide recommendations for improving the Reactor Oversight Process baseline inspection procedures and the Special Inspection process based on any lessons learned.

This charter may be modified should you develop significant new information that warrants review. Should you have any question concerning this charter, contact Mark Franke at (404) 562-6349.

Docket No.: 50-302
License No.: DPR-72

References:

Inspection Procedure 93812, Special Inspection
Management Directive 8.3, NRC Incident Investigation Program
Inspection Manual Chapter 0609, Significance Determination Process
Inspection Manual Chapter 0612, Power Reactor Inspection Reports

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PROGRESS ENERGY ROOT CAUSE REPORT

INSERT ROOT CAUSE REPORT