BEFORE THE FLORIDA PUBLIC SERVICE COMMISSION

DOCKET NO. 100437-EI
PROGRESS ENERGY FLORIDA, INC

Submitted for Filing:
October 10, 2011

IN RE: EXAMINATION OF THE OUTAGE AND REPLACEMENT FUEL/POWER COSTS ASSOCIATED WITH THE CR3 STEAM GENERATOR REPLACEMENT PROJECT, BY PROGRESS ENERGY FLORIDA, INC.

TESTIMONY & EXHIBITS OF:

JON FRANKE

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I. INTRODUCTION AND QUALIFICATIONS.

Q. Please state your name and address.

A. My name is Jon Franke. My business address is 15760 W. Powerline Street, Crystal River, Florida 34442.

Q. By whom are you employment and in what capacity?

A. I am employed by Progress Energy Florida, Inc. ("PEF" or the "Company") in the Nuclear Generation Group and serve as Vice President – Crystal River Nuclear Plant ("CR3").

Q. Please describe your duties and job responsibilities in that position.

A. As Vice President I am responsible for the safe operation of the Crystal River nuclear generating station. The Plant General Manager, Site Support Services and training sections report to me. Additionally, I have indirect responsibilities in oversight of major project and engineering activities at the station. Through my management team, I have about 420 employees that perform the daily work required to operate and maintain the station.

Q. Please summarize your educational background and work experience.

A. I have a Bachelor's degree in Mechanical Engineering from the United States Naval Academy in Annapolis, MD. I have a graduate degree in the same field from the University of Maryland and Masters of Business Administration from the University of North Carolina at Wilmington.

I have over 20 years of experience in nuclear operations. I received training by the U.S. Navy as a nuclear officer and oversaw the operation and maintenance of a nuclear aircraft carrier propulsion plant during my service. Following my service in the Navy, I was hired by Carolina Power & Light and have been with the Company through the formation of Progress Energy. My early assignments involved engineering and operations, including oversight of the daily operation of the Brunswick Nuclear Plant as a U.S. Nuclear Regulatory Commission ("NRC") licensed Senior Reactor Operator. I was the Engineering Manager of that station for three years prior to assignment to Crystal River as the Plant General Manager in 2002. I was promoted to my current position in April 2009.

II. PURPOSE AND SUMMARY OF TESTIMONY.

A. <u>PURPOSE:</u>

Q. What is the purpose of your direct testimony?

A. The first and foremost purpose of my direct testimony is to explain that the delamination in the CR3 containment structure that took place in October, 2009 resulted from unprecedented and unpredictable circumstances beyond PEF's control and in spite of PEF's prudent management of the steam generator replacement

("SGR") project. I will also explain that in all other aspects, PEF managed and executed the SGR project in a reasonable and prudent manner through the October, 2009 delamination. It is my understanding that other issues associated with the extended outage at CR3 beyond October, 2009 will be addressed in a subsequent hearing before the Commission.

Q. Can you briefly explain the reasons for the extended outage at CR3 during the SGR project?

A. Yes. During the SGR project there was a delamination in October, 2009 in the outer layer of concrete in the bay in the CR3 containment building where the opening was cut to remove the old once through steam generators ("OTSGs" or "steam generators") and replace them with new OTSGs. A delamination is a separation in the concrete, in this case a separation varying from about 1/64 inch to about 2 inches in size that appeared approximately nine inches into the 42 inch thick concrete containment wall from the outside of the containment building around the opening cut to move the OTSGs out and into the building. The delamination did not impact the successful replacement of the OTSGs. PEF successfully removed the old OTSGs and replaced them with new OTSGs that will enable PEF to continue to operate efficiently CR3 through the end of its current operating license period and through its planned life extension through 2036.

B. SUMMARY:

Q. Mr. Franke, please provide a summary of your testimony.

A. The October, 2009 concrete delamination that PEF experienced at CR3 during the SGR project was unprecedented and unpredictable. Utilities at several other similar nuclear plants had also cut openings in their containment structures for OTSGs or reactor vessel head removals before PEF cut a containment opening for the OTSGs for its SGR project in 2009. These utilities used the same industry standard engineering analyses and construction methods for their containment structure opening that PEF used on the SGR project and none of them experienced a delamination. PEF employed experienced engineers and contractors on the SGR project who applied these industry standard engineering analyses and construction methods to plan for and implement the replacement of CR3's OTSGs through an opening cut in the containment wall at CR3. Application of these industry standard engineering and construction analyses and methods did not predict or reveal the delamination that occurred at CR3 before it occurred. In fact, the industry specialist in root cause investigations PEF hired to determine the root cause of the delamination concluded that the engineering calculations supporting the CR3 containment opening work during the SGR project were performed in accordance with industry standards. This specialist -- Performance Improvement International ("PII") -- further found no organizational or human performance root cause for the delamination. Instead, PII concluded from the root cause investigation that the inability of these industry standard engineering analyses and modeling to predict the delamination was the programmatic root cause of the delamination. Because the existing industry standard engineering calculations and modeling methods for concrete structures like the CR3 containment building were inadequate to predict the delamination, PII had to develop state-of-the-art improvements to the standard industry engineering analyses and models to accurately determine how the delamination occurred at CR3. PII's computer model analysis was only possible because of the information obtained from the delamination that was learned during the root cause investigation. This model ultimately was able to replicate the October 2009 delamination, and identify and confirm the delamination cause and contributing factors with this information by simulating the delamination event in the computer model. Due to the inherent limitations in the industry standard analytical and modeling methodologies, PEF did not and could not foresee the October, 2009 delamination that occurred during the SGR project.

Prior to the October, 2009 delamination, PEF also prudently managed the SGR project in all other aspects. PEF implemented prudent project management policies, procedures, and controls on the SGR project. PEF adhered to those procedures, policies, and controls, and effectively and aggressively managed its contractors.

Q. Do you have any exhibits to your testimony?

A. Yes, I am sponsoring the following exhibits to my testimony:

- Exhibit No. ___ (JF-1), photos of the CR3 reactor building during initial plant construction;
- Exhibit No. ___ (JF-2), an aerial photo of the CR3 plant site;
- Exhibit No. ___ (JF-3), a photo of the CR3 reactor building;

- Exhibit No. ___ (JF-4), a cross-section of the CR3 containment structure showing the location of the OTSG's in the CR3 containment building;
- Exhibit No. ___ (JF-5), a photo of a CR3 containment structure tendon without the buttonhead;
- Exhibit No. ___ (JF-6), photos of vertical and horizontal hoop tendons;
- Exhibit No. ___ (JF-7), a schematic drawing illustrating the location of the horizontal and vertical tendons in the CR3 containment building;
- Exhibit No. ___ (JF-8), a photograph of a typical ram device that pulls the tendon cables, effectively tightening or tensioning the tendons in the CR3 containment wall;
- Exhibit No. ___ (JF-9), a photograph of the process of placing shims inserted near the end of the cables, which holds the tendon in place with the help of button heads on the end of the cables, in the tensioning process for the tendons in the CR3 containment wall;
- Exhibit No. ___ (JF-10), a cutaway diagram of a steam generator showing the steam generator tubes;
- Exhibit No. ___ (JF-11), a photo of the new CR3 steam generators;
- Exhibit No. ___ (JF-12), a list of eleven (11) pressurized water reactor ("PWR") nuclear power plants that are post-tensioned concrete containment buildings with steel liners like CR3 that successfully created and restored temporary construction openings in their containment walls to replace their steam generators or reactor vessel heads prior to the CR3 SGR project;

- Exhibit No. ___ (JF-13), a diagram looking down on the CR3 containment building from above the building showing the six bays in the CR3 containment building;
- Exhibit No. ___ (JF-14), a photograph of Bay 3-4 in the CR3 containment building showing the construction opening;
- Exhibit No. ___ (JF-15), a diagram that shows the vertical and horizontal tendons that were de-tensioned and removed from the construction opening in the CR3 containment building;
- Exhibit No. ___ (JF-16), a picture of Bay 3-4 in the CR3 containment building with the construction opening and the outline of the delamination around the construction opening illustrated;
- Exhibit No. ___ (JF-17), is an exhibit of photographs of how the delamination looked viewed from inside the construction opening in Bay 3-4 of the CR3 containment structure;
- Exhibit No. ___ (JF-18), is a cross section cut away view of the delamination in Bay 3-4 of the CR3 containment structure;
- Exhibit No. ___ (JF-19), Sargent & Lundy, ("S&L") Calculation No. S06-0002, rev. 1 for the SGR project;
- Exhibit No. ___ (JF-20), Calculation S09-0054 for the SGR project;
- Exhibit No. ___ (JF-21), S&L Calculation S06-0003, the evaluation of the CR3 containment shell for the applicable conditions and loads associated with the creation and restoration of the construction access opening in the containment wall for the SGR project;.

- Exhibit No. ___ (JF-22), S&L Calculation S06-0005, rev. 2, Containment Shell Analysis for Steam Generator Replacement – Shell Evaluation during Replacement Activities;
- Exhibit No. ___ (JF-23), S&L Calculation S06-0004, the calculations and supporting engineering study for the S&L de-tensioning plan for the SGR project;
- Exhibit No. ___(JF-24), Methodology Study – Generator Transport Through Containment Engineering Change ("EC") – ED 61170;
- Exhibit No. ___ (JF-25), a composite exhibit of pictures of the original installation of the OTSGs at CR3;
- Exhibit No. ___ (JF-26), MPR Associates ("MPR") computer animation of equipment hatch option for replacement of the OTSGs for the SGR project;
- Exhibit No. ___ (JF-27), a composite exhibit of pictures of equipment and material that must be moved or removed and replaced to use the CR3 equipment hatch for the SGR project;
- Exhibit No. ___ (JF-28), a chart of the steam generator replacement projects demonstrating that generator replacement transportation method decisions are based on the location of the equipment hatch;
- Exhibit No. ___ (JF-29), May 9, 2006 Business Analysis Package ("BAP") for the SGR project;
- Exhibit No. ___ (JF-30), a composite exhibit of CR3 cable tray structures;
- Exhibit No. ___ (JF-31), the CR3 SGR project Monthly Report, November 2004;

- Exhibit No. ___ (JF-32), the CR3 Steam Generator Design and Implementation Strategy, February 22, 2005; and
- Exhibit No. ___ (JF-33), a composite exhibit of the Integrated Project Plans ("IPPs") for the SGR project.

These exhibits were prepared under my direction or they are documents routinely relied upon by me and others in the Company in the usual course of our business and they are true and correct.

Q. How is your testimony organized?

A. My testimony is organized into four major sections. First, I provide an overview of the CR3 plant design, including the reactor containment building construction and design, and the SGR project. Second, I explain that the October, 2009 delamination was both unprecedented and unpredictable and why PEF's management of the containment opening work where that delamination occurred was prudent. Similarly, I will next explain the detailed analysis that PEF performed to decide the best way to replace the OTSGs and explain why that analysis was prudent based on the information available to PEF at that time, given the unprecedented and unpredictable nature of the delamination. Finally, I will describe PEF's management of the SGR project, explaining that PEF prudently managed the project, successfully implementing PEF's project management policies, procedures, and controls on the SGR project.

III. THE CR3 PLANT DESIGN AND THE REASONS FOR THE CR3 EXTENDED OUTAGE.

A. BACKGROUND.

Q. Please provide a description of the CR3 nuclear plant and how it works.

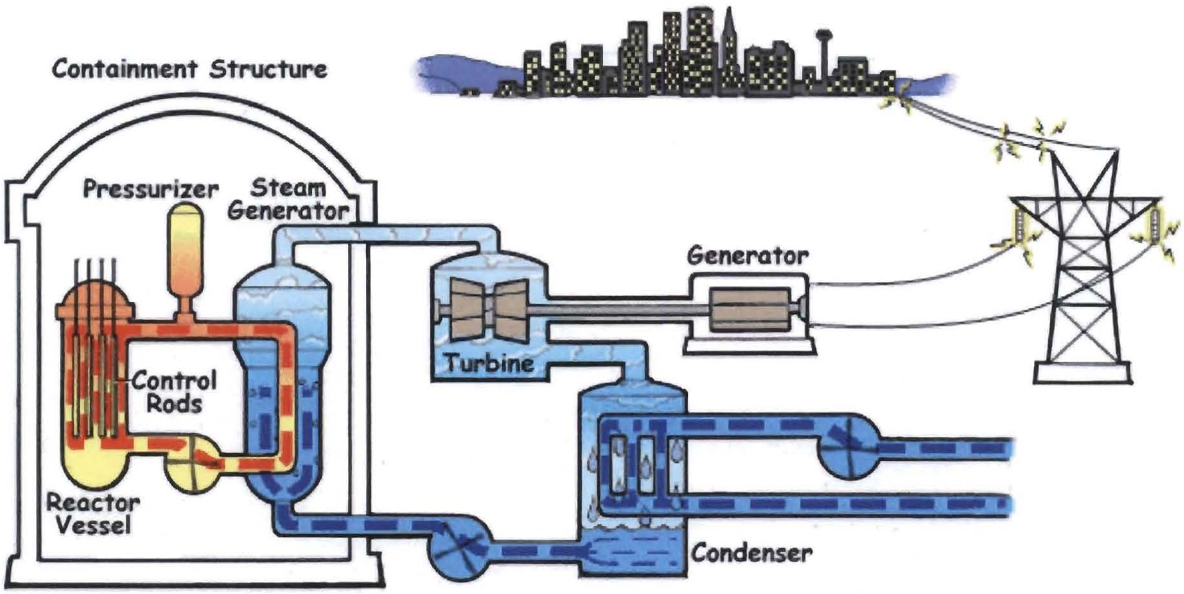
A. The CR3 nuclear plant was constructed between 1968 and 1976 and placed in commercial service in 1977. CR3 is a Babcock & Wilcox ("B&W") designed pressurized water reactor ("PWR") nuclear plant. JA Jones Construction Company built the plant and Gilbert Associates (now Worley Parsons) designed it. Exhibit No. ___ (JF-1) to my testimony is a composite exhibit of photographs of CR3 during initial plant construction. Exhibit No. __ (JF-2) is an aerial photo of the CR3 plant site, with the completed CR3 building located roughly in the middle of the photograph. Exhibit No. ___ (JF-3) is a photograph of the completed CR3 reactor building itself.

There are 69 pressurized water reactors operating in the United States today. Each reactor is unique; however, CR3 has several B&W "sister" units that are similar in design: Oconee Units 1, 2, and 3 operated by Duke Energy; Three Mile Island ("TMI") owned by Exelon; Davis-Besse owned by FirstEnergy; and Arkansas Nuclear One, Unit 1 ("ANO 1") operated by Entergy.

A PWR nuclear plant includes a Primary and Secondary System. The Primary System is located within the containment building and includes the reactor vessel, pressurizer, OTSGs, primary coolant system, and related equipment. Figure 1 below is a simplified illustration of the major components of a PWR nuclear plant.

Figure 1

The Pressurized Water Reactor (PWR)



The Primary System is a closed loop system. The nuclear reactor produces heat that eventually is turned into steam then into electricity. The heat is removed from the reactor by water in the primary coolant system (the yellow and orange portion of Figure 1) that is continuously pumped around the Primary System. This process of the continuous flow of water around the Primary System is what is meant by the term "closed loop system." Within the Primary System, heat transfers from the fuel cells located in the reactor vessel to the surrounding metal fuel cladding, which in turn heats the water flowing between and around the fuel rods in the reactor vessel. The heated water then travels from the core or reactor vessel through pipes to the OTSGs (only one OTSG is shown in the simplified illustration in Figure 1). In the steam generators, heat is transferred from the reactor primary coolant system to the physically separated secondary coolant system (the blue portion in Figure 1) producing steam in the secondary system. The Primary System operates at about 600 degrees F and 2155 pounds per square inch ("psi"). The high pressure prevents the water in the primary system from turning to steam.

The secondary water coolant system is under less pressure, operating at over 450 degrees F and 850 psi, and when the water in the secondary coolant system is heated it turns to steam. The steam turns the turbine that powers the generator. The steam exiting the turbine is condensed to water. The water is pumped back to the steam generators by a series of pumps and heat exchangers where it is once again converted to steam, thereby completing the cycle.

In the diagram above in Figure 1, the Primary System, including the OTSGs, is located within the containment structure. At CR3 the containment structure is the CR3 containment building.

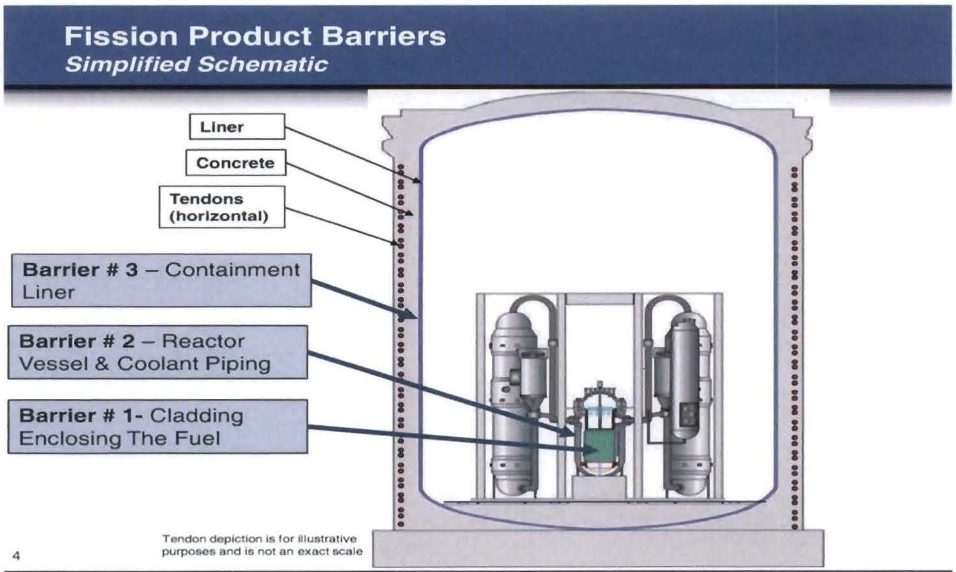
Q. What is the purpose of the containment building?

A. The containment building is a safety-related structure under NRC guidelines and regulations. This means in lay terms that the containment building helps protect the public and the environment from the potential release of radiation in the unlikely event of a reactor core accident. The CR3 containment building is designed to, among other things, withstand hurricanes, tornados, tornado missiles, earthquakes, significant temperature changes, and internal pressure changes under accident scenarios. The design basis – that is, the engineering analyses and design information upon which the NRC issued the CR3 Operating License – is set forth in the Final Safety Analysis Report or "FSAR" for CR3.

There are three key independent nuclear fission product barriers within the containment structure: (1) the cladding encasing the nuclear fuel; (2) the reactor vessel, in which the fuel sits in rods encased with the carbon-steel cladding, and coolant piping system; and (3) the 3/8 inch carbon steel containment liner, which lines the inside of the containment building walls and which is anchored to the 12-foot 6-inch reinforced concrete base mat (there is also a carbon steel liner below the concrete dome roof). Figure 2 shows these redundant barriers in the containment structure.

The concrete wall provides the structural support for the steel containment liner. The steel reinforcements in the wall, the tensioned steel tendons, and the 42-inch thick concrete provide the compressive strength behind the structural support for the steel liner. For example, in the unlikely event that a major loss of coolant accident ("LOCA") occurred in the building, much of the water in the reactor system would flash to steam and the pressure would rise in the building. The post-tensioned, pre-stressed containment wall would withstand that build-up of pressure within the building and ensure the steel containment liner performs its function.

<u>Figure 2</u>



Q. Can you describe the containment building?

A. The CR3 containment building is a steel-lined, post-tensioned, pre-stressed concrete structure. As noted in the table and cross-section of the CR3 containment building in Figure 3 below, the CR3 containment building is a circular or cylindrical building with an outside diameter of about 138 feet. It is about 157 feet high. It has a 42-inch thick concrete wall around the cylindrical 3/8 inch steel liner, and a 36-inch thick concrete dome. The cross-section in Figure 3 below and in Exhibit No. __ (JF-4) to my testimony also show the key components in the CR3 containment building. The OTSGs are installed in a vertical position on either side of the reactor vessel and connected to the reactor vessel in the containment building by way of the reactor coolant piping. The photographs included in Exhibit No. ___ (JF-1) include photographs of the original installation of the OTSGs. As the photographs show, the OTSGs were installed when there was no dome on the plant, no concrete on the building, and little additional construction of walls and other fixtures and equipment inside the reactor building. Removing and replacing the OTSGs, after the construction of CR3 has been complete and the plant has been in commercial operation for over thirty (30) years, was a major project for the Company.

The CR3 containment building concrete wall is reinforced by an outer layer of steel rebar matting and steel tendons. One tendon consists of 163 parallel 7 millimeter (mm) wires, greased and enclosed in a 5 ¼ inch sheath. (Exhibit No. ___ (JF-5) is a photo of the tendon without the buttonheads). The tendons are contained throughout the CR 3 containment building wall, running vertically and horizontally. The horizontal tendons are located about 10 inches inside the building wall concrete from the outer surface and the vertical tendons are just inside the horizontal tendons.

Typical photos of the wall tendons are included in Exhibit No. ___ (JF-6) to my testimony. As shown in Figure 4 below, the containment building has six buttresses, and the walls between the buttresses are called bays. PEF made the construction opening in the containment building for the SGR project in "Bay 3-4" – the bay between buttresses 3 and 4.

There are also tendons across the dome of the containment building. The dome tendons are also made up of a bundle of smaller steel tendon wires, greased, and encapsulated in the 5.25-inch diameter metal conduits. The containment building dome has 123 tendons that span the dome in a convex geometric design within the dome concrete.

The CR3 containment building wall has 144 vertical tendons. The horizontal tendons are arranged in 94 hoops that span the circumference of the containment building. Each hoop consists of 3 tendons that cover a third of the building's perimeter (i.e., each horizontal tendon runs the span of 2 of the 6 bays of the containment building). Thus, the CR3 containment building contains a total of 282 horizontal tendons. The horizontal tendon hoops are located adjacent to the vertical tendons in the concrete wall. The horizontal tendons are anchored to the six vertical concrete buttresses that form the six bays of the containment building. The vertical tendons are anchored to the foundation and the dome. Exhibit No.__ (JF-7) is a simplified schematic illustrating the location of the horizontal and vertical tendons in the containment building.

The purpose of the tendons is to strengthen the containment building and improve its ability to handle load, including loads associated with events such as tornados, earthquakes and loss of coolant accidents. This is accomplished by tightening the tendons, a process called tensioning. This tensioning occurred in 1976 following the initial construction of the CR3 containment building. In order to tension the tendons, a specialized piece of equipment, called a hydraulic ram, pulls the tendon cables, effectively stretching the tendons. *See* Exhibit No. ___ (JF-8).) Shims are then inserted near the end of the cables, which holds the tendon in place with the help of button heads on the end of the cables. *See* Exhibit No.___ (JF-9).) This keeps the tendons under controlled tension and effectively compresses the containment building, adding pre-stress which increases its ability to handle the loads I briefly described above. The presence of these tensioned vertical and horizontal tendons in the containment wall is the reason the CR3 containment building is referred to as a "post-tensioned, pre-stressed" containment system or building. CR3 is one of several post-tensioned, pre-stressed nuclear containment facilities in the country. Thirty-two (32) of the 69 PWRs in the United States are post-tensioned, pre-stressed nuclear containment facilities.

Figure 3

CR3 Containment Dimensions

Dimension	Value
Containment Outside Dimension (OD)	137 ft 0.75 in
Dome Thickness	36 in
Basemat Thickness	12 ft 6 in
Liner Thickness	0.375 in
Wall Thickness	42 in
Buttress Wall Thickness	5 ft 10 in
Vertical & Hoop Conduit OD	5.25 in
# of Vertical Tendons	144
# of Tendon Hoops	94
# of Tendons per Hoop	3
# of Prestressed Dome Tendons	123

25 NGG

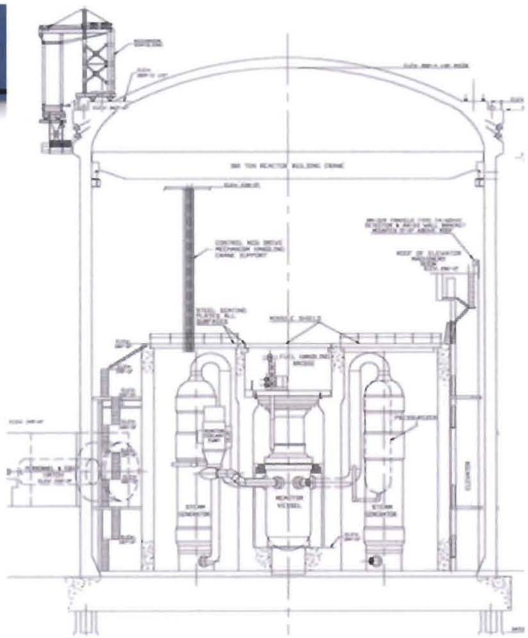
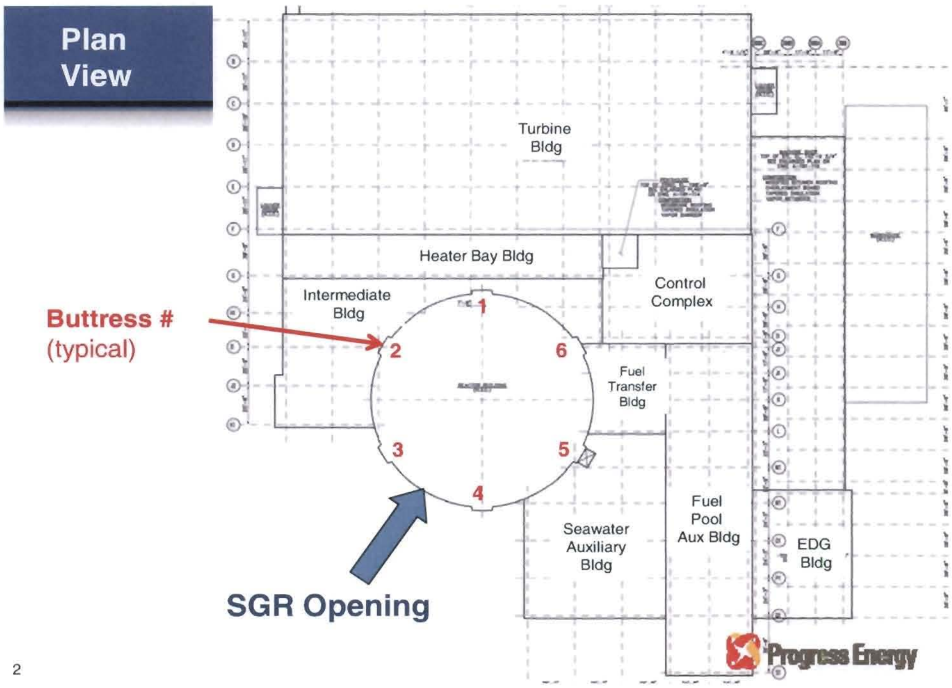


Figure 4



Q. What was the SGR project?

A. The SGR project involved the replacement of the OTSGs at CR3. The CR3 unit was placed in service in 1977 with B&W OTSGs that were initially designed to last the life of the original forty (40)-year life of the plant under its NRC operating license. The same or similar OTSGs were installed in many other nuclear power plants. Like every other nuclear plant using these steam generators, PEF experienced stress corrosion and cracking in the OTSG tubes as they aged that required an increase in tube inspection and repair activities. In addition to increasing operation and maintenance ("O&M") costs, these phenomena shorten the useful life of the steam generators such that a license extension beyond 2016 would be impractical with the original OTSGs.

In mid-2002, the Company began a study to evaluate replacing the OTSGs and obtaining a license extension for CR3 or decommissioning CR3 at the end of its original license in 2016. This study demonstrated that replacement of the steam generators would provide more savings versus decommissioning CR3 in 2016 and building new capacity. The study also showed that it was more cost-effective to replace the OTSGs as soon as possible (2009) rather than as late as possible (2016). Based on the results of this study, Company management decided to replace the OTSGs at CR3 in 2009.

In 2004, the Company initiated a multi-year project to replace the OTSGs during the 2009 refueling outage (the "R16" refueling outage) with new steam generator components manufactured with improved, corrosion-resistant materials. Exhibit No. ___ (JF-10) to my testimony is a cross section of the steam generators showing the steam generator tubes. Exhibit No. ___ (JF-11) is a photograph of the new CR3 OTSGs. These steam generators are essentially the same size as the old OTSGs. They are over 73 feet tall, over 12 feet in diameter (15 feet with the shop installed piping), and weigh approximately 500 tons. As a result, they can only be moved out of and into the containment building through use of heavy duty crane and rigging systems.

Q. What was the scope of work for the SGR project?

A. The SGR project first involved creating a containment opening to establish a transport path for the removal and replacement of the OTSGs. The project also involved cutting the piping connections of the existing OTSGs and removal of the OTSGs from their constraints inside the "D-ring" structures in which they were installed in the containment building. The next step was to remove the existing OTSGs from the containment building using a temporary crane and rigging transport system and then move the new OTSGs into the containment building using the same transport system. Finally, the new OTSGs had to be installed into the D-rings and the piping connections welded. The final step was to close the containment opening. The project also necessarily included design, engineering, procurement, fabrication, oversight of replacement components, planning and implementation of the work scope, workforce support and project management. At the time, CR3 was the fifth of a group of very similar B&W plants to replace steam generators and one of dozens of other non-B&W plants to have replaced its steam generators.

Q. What was the Company's plan for removing the old OTSGs and installing the new ones in the CR3 containment building?

A. To remove the old OTSGs and replace them, PEF planned to cut an access opening in the containment building above the OTSGs, cut the connections to the old OTSGs and remove them with a heavy duty rigging and crane system, move the new OTSGs into the building using the same heavy duty rigging and crane system, weld the connections for the new OTSGs, and close the access opening. This was a common practice in the industry to replace steam generators or for other major projects, and had been implemented at 11 other facilities before the SGR project.

The opening in the containment building was 25 feet by 27 feet. The access opening provided enough space to remove and replace the OTSGs, with shop attached piping, by moving them out and in the access opening horizontally on a crane system. Inside the containment building, the old OTSGs were disconnected and raised vertically before being turned and placed horizontally on the crane system to be removed from the building. The reverse arrangement was used to install the new OTSGs, they were brought into the containment building through the access opening horizontally on a crane system, then turned vertically to be lowered into position in the D rings in the containment building.

The creation of the opening in the containment building involved de-tensioning and removal of the vertical and horizontal tendons in the proposed containment opening area. De-tensioning the tendons involved relaxing the steel wires within the tendon conduits that make up the tendons. The containment opening in the concrete was cut with high pressure water nozzles. The steel rebar was removed, all tendons and tendon conduits in the containment opening were removed, and concrete removal continued down to the steel liner. The steel liner was then cut to create the opening into the building. After removal of the old OTSGs and placement of the new OTSGs in the building, the steel liner was replaced and welded in place, new concrete was placed in the containment opening along with new or replaced horizontal and vertical tendons, and steel rebar reinforcement. The tendons were then re-tensioned to restore the compressive strength of the containment wall.

Q. Why did the tendons have to be de-tensioned?

A. To remove and replace the concrete in the construction opening the tendons had to be removed, replaced, and re-tensioned. The tendons in the immediate area of the opening had to be removed to eliminate an interference to transporting the OTSGs through the opening. To remove the tendons they had to be de-tensioned. Additionally, de-tensioning other tendons around the construction opening in the containment building was necessary to restore the concrete to its pre-stressed condition after placement of new concrete and tendons in the opening. De-tensioning the tendons in and around the construction opening in the containment structure reflected the industry understanding at the time of the SGR project that de-tensioning reduced and did not increase the stresses in the concrete in and around the construction opening.

Q. How are the tendons de-tensioned?

A. There are two primary methods for de-tensioning the tendons. As with the initial tensioning process, a ram can be used to pull the tendon wires so the shims can be removed. This method is used when the tendons are being relaxed, but left in place for future re-tensioning. A second method is to cut the button heads with the tendon still under tension. This method is generally used when the tendon is to be removed and replaced. For this project, the tendons within the immediate opening were de-tensioned by cutting the button heads off so that the tendons could be removed from the opening. Tendons outside the immediate opening were to be de-tensioned by using the ram and removing the shims.

Q. Was this a complex project?

A. Yes it was, but other utilities had replaced their steam generators at their nuclear facilities in the same manner, so PEF was not proceeding with a project that had never been successfully completed. One of the primary challenges to replacing the steam generators, though, is the sheer size and weight of the steam generators that must be moved out and into the nuclear reactor containment building. The new steam generators are essentially the same size as the old steam generators. They are massive pieces of equipment. As I explained generally above, each of the steam generators measures about 15 feet in diameter including the shop installed piping, is 73 feet 3 inches tall, and weighs 500 tons. They can only be moved out and into the containment building through a carefully planned use of heavy duty cranes and riggings. To accomplish this task, the Company spent years planning the project and obtaining lessons learned and best practice information from other, prior steam generator replacement projects and other nuclear power plant construction projects that involved the creation of temporary construction openings in nuclear containment structures. PEF further retained experienced engineers and contractors to assist the Company in carrying out the complex engineering and construction methods required to successfully remove the old OTSGs and install the new OTSGs.

Q. What utilities had successfully completed construction projects by using a temporary construction opening through the containment structure?

A. As I discussed above, CR3 is a PWR nuclear plant with a post-tensioned, pre-stressed concrete containment building with a steel liner. Eleven (11) similar PWR plants had steam generator replacement or reactor vessel head replacement projects and they successfully created and restored temporary construction openings in their containment walls to replace their steam generators or reactor vessel heads prior to the CR3 SGR project. These plants have equipment hatches which either were too low or too small to accommodate the transportation of the steam generators or reactor vessel heads through them. Exhibit No. ___ (JF-12) to my testimony contains a list of these 11 PWR plants.

This list excludes Exelon who successfully replaced its steam generators at Three Mile Island Unit 1 ("TMI 1") by creating a construction opening through the containment wall in late October 2009 shortly after the SGR project work started at CR3. However, the list includes Arkansas Nuclear Unit 1 ("ANO 1"), who also successfully completed a steam generator replacement project in 2005 by removing and replacing the steam generators through a construction opening cut in the containment wall. Other utilities on the list successfully completed steam generator replacement or reactor vessel head removal projects at ANO 2, Byron 1, Braidwood 1, Oconee 1, 2, and 3, Palisades, San Onofre, and Turkey Point Units 3 and 4 by creating temporary construction openings in their containment walls. As illustrated by Exhibit No. __(JF-12), the ability to cut into a nuclear containment structure to perform necessary construction work within the nuclear power plant and restore the containment structure to its required strength to perform its design function was well established in the nuclear industry.

Q. What happened on the CR3 SGR project?

A. PEF commenced work on the SGR project according to its project plan to remove and replace the OTSGs through a temporary containment opening in the CR3 containment building. The temporary containment opening was located in Bay 3-4 above the equipment hatch. Bay 3-4 refers to the section of the containment building between the Nos. 3 and 4 buttresses. The containment building has 6 buttresses and therefore 6 bays, Bays 1-2, 2-3, 3-4, 4-5, 5-6, and 6-1. Exhibit No. __ (JF-13) is a diagram looking down on the CR3 containment building from above the building showing the 6 bays. Bay 3-4 is the section of the containment building in which the construction opening was made in support of the SGR project. Exhibit No. __(JF-14) to my testimony is a photograph of Bay 3-4 showing the construction opening.

Seventeen (17) horizontal tendons and ten (10) vertical tendons that traversed the opening in Bay 3-4 were de-tensioned and removed. The vertical and horizontal tendons were relaxed or de-tensioned over a six-day period from September 26, 2009 through October 1, 2009. The two vertical tendons in the middle of the containment opening (34V12 and 34V13) were de-tensioned and removed first, then the vertical tendons across bay 3-4 were de-tensioned in order from number 34V8 through number 34V17. The horizontal tendons were de-tensioned sequentially from the bottom of the access opening to the top of the opening in the containment building. Exhibit No. ___ (JF-15) to my testimony shows the vertical and horizontal tendons that were de-tensioned and removed.

The access opening in the containment building was created by hydro-demolition. Hydro-demolition uses water under very high pressure to cut through the concrete. A test demonstration was performed successfully in the lower right corner of the construction opening to demonstrate adequate control of the hydro-demolition equipment. Full scale hydro-demolition started on October 1 and removed the outer layer of concrete down to the rebar. Once the concrete was removed down to the steel rebar, the rebar was cut, and then hydro-demolition was used to remove the concrete to the tendon conduits. Once the tendon conduits were removed, the remainder of the concrete was removed down to the carbon steel liner. The carbon steel liner was then cut to complete the access opening.

During the period of hydro-demolition down to the tendons the workers observed a stream of water flowing from a crack below and to the right of the construction opening. An inspection of the construction opening was initiated at this time and during this inspection the delamination was discovered.

Q. Can you describe in more detail the delamination that PEF discovered in the containment wall?

A. Yes. The delamination was a separation of the concrete in the wall of the CR3 containment building in Bay 3-4. The delamination was a separation of an outer layer of concrete along the plane of the horizontal tendon conduits (about 10" deep from the outer surface) from the remainder of the containment building wall in Bay 3-4 by as little as 1/64 inch and up to about 2 inches. The delaminated layer of concrete remained connected to the concrete wall, however, there was a crack between the outer layer of concrete and the remaining 30+ inches of concrete in this area of the containment building wall. The delaminated layer of concrete was centered on the construction opening and formed a large hour glass shape that was approximately 50 feet wide and 85 feet high. Exhibit No. ___ (JF-16) to my testimony shows the outline of the delamination.

Exhibit No. ___ (JF-17) are photographs of how the delamination looked viewed from inside the construction opening. As can be seen in those photographs, the delamination appeared as a long, narrow crack running along the plane of the horizontal tendons. Exhibit No. ___ (JF-18) is a graphic illustration of the delamination in a cut away view that gives a better perspective of the delamination. Although inspection of the cracks at the surface inside the construction opening did not reveal the full extent of the delamination, further investigation confirmed that the delamination was in effect a separation of a large portion of the outer portion of concrete from the rest of the containment building.

B. <u>THE DELAMINATION WAS UNPRECEDENTED AND UNPREDICTABLE.</u>

1. The Contributing Factors to the Delamination.

Q. What caused the delamination in the CR3 containment building?

A. The Company's root cause investigation determined that the immediate cause of the delamination was the redistribution of tensile stresses along a high stress plane connecting the horizontal tendons in and around the planned containment opening that exceeded the concrete's fracture (cracking) capacity. Tensile stress was created between tensioned and de-tensioned tendons and increased with the number of de-tensioned tendons involved until the fracture capacity was exceeded between the horizontal tendons, the concrete between these tendons cracked, and the cracks spread forming the hour glass delamination pattern in Bay 3-4 in the CR3 containment building. *See* Exhibit No. __ (JF-16) to my testimony. In sum, the delamination occurred when concrete above and below the horizontal tendons cracked and rapidly spread out when stresses were redistributed during de-tensioning activities leading up to the creation of the containment opening. De-tensioning alone, however, was insufficient to cause the delamination. Other factors contributed to the delamination.

Q. What are the other factors that contributed to the delamination?

A. The root cause investigation determined that there were seven factors that contributed to the delamination. All seven factors jointly contributed and were necessary but not sufficient for the delamination to occur in the CR3 containment wall. In other words, none of the seven factors, existing alone, would have caused the delamination, and all seven factors contributed to the delamination. These seven factors are discussed in detail in Mr. Miller's testimony and in the root cause assessment report attached as Exhibit No. ___ (GM-4) to Mr. Miller's testimony. These factors combined to cause the delamination during the containment opening activities in a complex interaction that was unprecedented and unpredictable.

Q. You mentioned that there was a redistribution of tensile stresses in the CR3 containment wall. What are tensile stresses?

A. Tensile stress, in lay terms, is the resistance of a material to a force tending to pull it apart. The tensile strength of a material is measured as the maximum stress the material can withstand without being pulled apart or cracking. In engineering terms, tensile strength is considered as one part of a complex calculation or set of calculations that require the consideration of the calculated interplay of the concrete material properties, including compressive strength, the Young's Modulus, Poisson's Ratio, and creep coefficient. Creep in lay terms is the tendency of a solid material to slowly move or change form under the influences of a constant load over time. The interplay of these different physical factors in a solid material like concrete is complex and can be offsetting. Industry standard engineering analyses, calculations, and models were used on the SGR project to determine the interplay of these concrete material properties under the de-tensioning and re-tensioning activities associated with creation and closure of the containment opening for the SGR project.

Q. You testified that the delamination was unprecedented. Can you explain why the delamination at CR3 was unprecedented?

A. Yes. There were eleven previous projects involving construction openings through post-tensioned, pre-stressed containment buildings at the time PEF commenced the SGR project. All 11 projects used the same, or similar, engineering analyses and construction methods that PEF used on the CR3 SGR project, and not one of these projects experienced delamination. In addition, there were only three prior delamination events in the nuclear industry around the world. All three occurred during the initial tensioning process involving the tightening of the post-tensioned tendons during the construction of the domes for the nuclear containment buildings. No delamination had ever occurred during the de-tensioning or relaxation of the tendons during a construction project subsequent to the building's original construction, and no delamination had ever occurred on a containment building wall during tendon tensioning or de-tensioning activities. There certainly was no prior delamination involving concrete in the containment wall in a post-tensioned, pre-stressed containment building where the concrete was not new but was approximately thirty years old like the CR3 containment building. Finally, the delamination was unprecedented because the most experienced contractors in the industry in construction projects involving construction openings in containment buildings – who responded to the request for proposals ("RFP") for the SGR project – did not identify or consider delamination as a potential risk for the construction opening for the SGR project.

Q. You also testified that the delamination was unpredictable. Why was the delamination unpredictable?

A. The primary reason the delamination at the CR3 containment building was unpredictable was that the application of industry standard engineering analyses, techniques, and models to containment opening activities on the SGR project by expert, experienced engineers and contractors could not predict the delamination or even identify delamination as a potential risk on the project. PEF employed experienced engineers and contractors on the SGR project. These engineers and contractors were provided the information regarding CR3's design characteristics and concrete material properties. They developed the de-tensioning scope and sequencing that was used at the SGR project applying industry standard engineering analyses and models. This work was reviewed by experienced contractors who had completed construction projects at other nuclear facilities involving construction openings through the containment building at those facilities using the same, or similar, engineering analyses and construction methods PEF used. None of this work by experienced engineers and contractors applying industry standard engineering and construction methods on the SGR project disclosed the risk of delamination on the project.

The absence of any prior delamination during the containment opening activities at other utility steam generator and reactor head replacement projects prior to the SGR project further provided no reason to predict the delamination during the SGR project. PEF and its engineers and contractors, specifically Sargent and Lundy ("S&L"), considered the prior CR3 dome delamination event, but the circumstances and conditions were so different they provided no reason to predict the 2009 wall delamination at CR3. The only other site in the United States that had a similar dome delamination in its history had previously made a similar decision to use a construction opening for similar work and performed that work without causing a delamination.

The lack of any precedent to predict the CR3 delamination was confirmed during the root cause investigation and subsequent development of the repair plan at CR3. PII could not simulate the delamination and thus predict it using any of the information from the prior industry containment opening projects or delaminations using the industry standard engineering analyses and models. PII needed information learned only as a result of the delamination in the root cause investigation and had to develop new engineering analyses and modeling to simulate the delamination. The industry standard engineering analyses and models at the time the SGR project was planned and implemented were therefore incapable of predicting and preventing the delamination prior to it occurring at CR3.

2. Industry Experience with Similar Projects to the SGR Project.

Q. Can you explain what the industry experience with similar projects was prior to the CR3 SGR project?

A. Yes. At the time PEF planned the SGR project, as I noted before, CR3 was the fifth of a group of very similar B&W nuclear power plants that replaced steam generators and was one of dozens of other non-B&W plants that replaced steam generators. None of these plants experienced delamination in their containment buildings.

PEF studied those plants with post-tensioned containment buildings that replaced their steam generators in planning the SGR project. PEF found that the plants similar in design to CR3 created containment openings through the containment building to replace their steam generators using the same industry standard engineering analyses and construction methods that PEF used and none of them experienced delamination.

For example, Duke Energy replaced its steam generators at Oconee several years prior to CR3 using a similar construction opening. Progress Energy deployed three employees to work full time at Oconee during the replacement of the steam generators on Oconee units 1 and 3. Experience and lessons learned from that assignment were factored into the planning for the CR3 SGR project. SGR project team members also benchmarked the CR3 SGR project against the steam generator replacement project at ANO 1. The steam generators were replaced at ANO 1 in 2005 by removing and replacing the steam generators through a construction opening cut in the containment wall. The SGR project team visited the ANO 1 project in August 2005, immediately prior to the de-tensioning work associated with the containment opening work began in the fall of 2005. The team also was on-site during the time that the containment opening was cut at ANO 1. SGR project team members incorporated lessons learned and best practices dealing with lay-down areas, temporary facilities, and industrial engineering techniques to develop an optimal site layout plan, among other lessons learned and best practices. There was no delamination at any of these other stations during the steam generator replacement projects which preceded the CR3 SGR project.

Exelon replaced its steam generators at TMI 1 (often referred to as CR3's "sister plant" because of its similar design) by creating a construction opening through the containment wall in late October 2009 shortly after the SGR project work started at CR3. TMI 1 is a post-tensioned, pre-stressed containment building with a similar design to the CR3 unit. Members of PEF's SGR project team visited TMI 1 in September 2006, and PEF SGR project team members continued to coordinate with the TMI 1 steam generator project team over the next three years to share and incorporate lessons learned and best practices.

As I mentioned before, there were also a number of other construction projects at nuclear facilities involving the creation of a construction opening in the containment building prior to the SGR project at CR3. These included steam generator replacement or reactor vessel head removal projects at Byron 1, Braidwood 1, Oconee 1, 2, and 3, Palisades, and Turkey Point Units 3 and 4. The projects at Turkey Point Units 3 and 4 involved concrete with similar material properties to the concrete at CR3 including the same aggregate. The same, or similar, industry engineering analyses and construction methods that were used to create the containment opening for the SGR project were used on these projects and none of these projects experienced delamination during the work to create and close the construction opening in the containment wall at these nuclear facilities.

Q. What were these industry engineering analyses and construction methods?

A. The standard construction method for cutting through the post-tensioned, pre-stressed concrete containment structure was to de-tension the post-tensioned tendons in the concrete containment wall, remove the tendons in the immediate area of the opening, and then remove the concrete down to the liner plate. Two methods were employed to remove the concrete. Some plants removed the concrete by mechanical means, i.e. chipping. Other plants used a hydro-demolition process known as hydro-lazing, or the use of high pressure water jets to cut away the concrete. Of the 13 plants that have made construction openings, 8 (including CR3 and TMI 1) used hydro-demolition, and 5 used mechanical means.

PEF hired Precision Surveillance Corporation ("PSC") to perform the de-tensioning (and re-tensioning when the concrete cut out was replaced and the opening closed) work on the SGR project. PSC had performed the same or similar work on virtually every steam generator project in the country. As noted above, PEF used the hydro-lazing process to cut the concrete. PEF hired Mac and Mac, an experienced hydro-demolition contractor, to perform the concrete removal work.

The industry standard engineering analyses were associated primarily with the containment opening and the de-tensioning and re-tensioning work necessary for creation and closure of the containment opening. This engineering work involved the determination of the concrete material properties that I previously mentioned and the calculation of stresses on the containment structure when performing the de-tensioning, removing the old concrete, and re-tensioning of the containment following the placement of new concrete in the opening. These industry standard engineering calculations are performed by a computer based modeling analysis known as Finite Element Analysis ("FEA"). FEA is a computer-based solution of simultaneous equations that can be used to analyze and demonstrate the structural adequacy of structures like the containment building wall. The FEA model created for CR3 uses thin "shell" (two-dimensional element) elements to create a three dimensional model that has the ability to include local geometry variations such as the equipment hatch in the CR3 containment structure. More specifically, this 3-D finite element model is generated using the industry accepted GTSTRUDL structural design and analysis software program. Element dimensions and material properties are specified by the engineering analyst. The FEA model for CR3 is discussed in more detail in Mr. Miller's testimony and in the root cause assessment report attached as Exhibit No. __ (GM-4) to Mr. Miller's testimony.

Q. Who performed the engineering analyses, calculations, and models for the containment opening work on the SGR project?

A. S&L performed this work on the SGR project. S&L is one of the largest, most experienced engineering contractors in the world, and they have considerable experience in the nuclear industry. PEF also retained Enercon Services to perform a third party engineering review of S&L's design criteria document (Calculation S06-

0002) for the CR3 containment analyses. Enercon's responsible engineer for this work had been responsible for the performance of the FEA analyses on prior nuclear projects involving temporary openings in post-tensioned, pre-stressed containment structures for steam generator replacement or reactor head removal projects.

Q. Did the PEF SGR project team benchmark against other prior steam generator replacement and reactor head removal projects?

A. Yes. The PEF SGR project team followed the work at several other recent and relevant projects to determine applicable best practices and lessons learned for the SGR project at CR3. In addition to benchmarking efforts previously discussed, the PEF SGR project team also sent team members to observe steam generator replacement project activities at Comanche Peak and St. Lucie in 2007. These projects were at plants different from CR3, since they did not have post-tensioned, pre-stressed containment buildings, but the SGR project team members still captured best practices and lessons learned that related to CR3 operations and training.

PEF SGR project team members also participated in industry associations that provided information on best practices associated with construction projects like the SGR project. For example, SGR project team members participated in the Black Diamond Services annual roundtable for steam generator replacement projects at nuclear facilities. The roundtable involved presentations on challenges, successes, and lessons learned from utilities that recently completed their steam generator replacements. The format allowed for an open exchange of ideas between participants. This forum also provided PEF with benchmarking reports, lessons learned, and memorialized recommendations on steam generator replacement project improvements. Team members also found it beneficial to contact organizations and personnel associated with SGR projects to ask questions and gain information on their projects.

PEF's participation in these benchmarking and industry best practices activities took place over the course of four years leading up to the SGR project work in the fall of 2009. Industry participants and the SGR project team were aware of prior dome delamination events during the initial construction of nuclear facilities. There was, however, no industry experience with delamination at a nuclear facility during any construction project following the initial construction of the facility, including all the steam generator and reactor vessel head replacement projects that the SGR project team followed in the four years leading up to the SGR project work in the fall of 2009. At no time during the course of the SGR project team benchmarking and lessons learned activities was the potential for delamination during the creation of a construction opening in the containment building wall to remove and replace steam generators identified as a project risk. The industry experience was that delaminations did not occur in construction projects involving the creation of openings in the containment wall in long established nuclear containment structures.

Q. Was there further industry experience associated with the SGR project that indicated the CR3 delamination was an unprecedented and unpredictable event?

A. Yes. In the course of planning the SGR project PEF issued an RFP for engineering, construction, and related work for the SGR project. The RFP was sent to the Steam

Generator Team ("SGT") and Bechtel. SGT (a consortium of companies) and Bechtel were the two vendors that had the most experience in the industry with steam generator replacement work, including steam generator replacements that used construction openings through post-tensioned, pre-stressed containment structures. The PEF RFP stated that the SGR project work scope included constructing and restoring a temporary opening in the CR3 containment building to accommodate the removal of the old OTSGs and the installation of the new OTSGs at CR3.

Both SGT and Bechtel conducted site visits and were familiar with the site conditions including the design and construction of the CR3 containment building prior to submitting their responses to the PEF RFP. SGT and Bechtel were further aware of the industry experience including the prior, limited delamination events during the initial construction of certain nuclear plant domes. Indeed, Bechtel was the contractor on the construction project where one of the prior dome delamination events occurred during initial construction of the nuclear facility. Both SGT and Bechtel proposed to use the temporary containment opening in the CR3 containment building to remove and replace the OTSGs at CR3. Neither SGT nor Bechtel objected to use of a temporary containment opening for the CR3 SGR project and neither SGT nor Bechtel mentioned the potential for delamination that occurred at CR3 in their RFP responses if a temporary containment opening was used to replace the OTSGs at CR3.

Q. Are you saying that no one expected any cracks to occur when concrete containment structures are cut?

A. No, cutting an opening in concrete of any type can create some small cracking or voids along the perimeter of the opening where the cutting or hydro-demolition removes concrete adjacent to the remaining concrete. In analyzing the equipment hatch versus the containment opening options for the SGR project that I discuss later in my testimony, PEF in fact recognized the potential risk that voids or small cracks may be found after hydro-demolition in existing concrete along the perimeter of the cut where the containment opening is created. PEF noted in its analysis documentation that, in the event this risk occurred and such voids or cracks were discovered, the risk mitigation steps included chipping back to sound concrete and then filling any voids or cracks with grout or concrete. SGT pointed out the same risk of perimeter cracking in its response to PEF's RFP. The potential presence of some perimeter cracks and voids is far different from the delamination or separation of one layer of the concrete wall from another over a wide area that occurred at CR3. Indeed, SGT identified perimeter cracking as a risk, but not delamination, because SGT still proposed to create a temporary construction opening to replace the OTSGs in its RFP response. A delamination like the one that occurred at CR3 had never occurred before on any project involving the creation of a temporary construction opening in an established concrete containment building wall, and no one identified it as a risk or predicted its occurrence on the SGR project.

3. Prior Dome Delaminations Involving Tensioning, not De-tensioning, in Initial Construction Projects.

Q. What were the prior industry dome delamination events that you mentioned?

A. Turkey Point Unit 3 and CR3 both experienced dome concrete delamination during their construction in the 1970's. Also, there was a dome delamination event during the construction of the Kaigi Atomic Power Project, Unit 1, in 1994.

Q. What caused these dome delamination events?

A. Each of these dome delamination events occurred during the initial construction and tensioning of the nuclear containment structures. For example, in June 1970 at Turkey Point Unit 3, during the tensioning of 110 of the 165 containment dome tendons during the initial construction of the facility, tendon conduit grease was observed leaking from a crack in the dome surface. The subsequent investigation revealed multiple delamination planes running parallel with but below the concrete surface in a semi-circular pattern. The root cause investigation determined the cause was unbalanced tensioning loads and insufficient adhesion contact areas along construction joints during the initial dome construction.

Later, in April 1976 during the construction of the CR3 containment building, a void was discovered beneath the concrete dome surface during the placement of anchors in the concrete dome. Subsequent investigation revealed a circular delamination approximately 105 feet in diameter in the dome. The delaminated concrete was removed and new concrete was poured with additional radial reinforcement steel rebar added. The causes of that delamination were: (1) high radial tension due to the biaxial compressive stresses from the overlapping of the stressed tendons in the particular geometric (convex) design of the dome; (2) tendon alignment in the construction of the dome with tendon placement being higher near the periphery and lower at the apex than designed, increasing the curvature and loads and radial stresses in localized areas of the overlapping tendons in the dome; (3) impact loads during construction; (4) tendon sequencing which while designed to be balanced could not be completely balanced; (5) thermal effects from tendon greasing and the exposure to the sun; and (6) concrete with lower resistance to tensile stresses and limited ability to arrest cracking due to the concrete aggregate composition. The CR3 dome delamination was not discovered during initial tensioning of the dome tendons, but the dome tendons were tensioned prior to the dome delamination discovery, and a subsequent review of the contemporaneous construction records revealed entries indicating sounds consistent with the delamination after 70 percent of the dome tendons were tensioned.

Finally, the Kaigi Atomic Power Project, Unit 1, was constructed with a double containment structural system with a gap between the inner and outer containment system. Delamination of the inner containment dome took place during construction in 1994 after tensioning the 66th of 183 dome tendons, causing the inner surface of the dome's center to collapse (the design did not include an inner steel liner like the CR3 unit). The cause of the delamination was the radial tension induced by tensioning the dome tendons coupled with the effect of the dome membrane compression, which was higher than the tensile load capability of the dome.

All three prior delaminations occurred during initial construction of nuclear power plant domes involving the initial tensioning of tendons in new or relatively new concrete that exceeded the tensile strength of the concrete under the unique stresses created by the geometric (convex) design of the overlapping tendons in the dome. None of these nuclear power plants, including CR3, experienced delamination during initial construction and tensioning of the tendons in the containment wall structure with the same design for the concrete material for the walls as the domes.

Q. Was PEF aware of these prior dome delamination events when it was planning the SGR project?

A. Yes. PEF's SGR project team, and its engineers and contractors on the SGR project, were aware of the prior industry dome delaminations in Florida. These dome delaminations, however, did not negatively impact the decision to use a construction opening through the containment building wall to replace the OTSGs at CR3. The circumstances and conditions of these prior dome delaminations were different from the circumstances and conditions of the CR3 containment wall opening work and, therefore, there was no reason to suspect a delamination during the CR3 containment wall opening work because of these prior dome delaminations.

Q. Why were the conditions and circumstances of the prior dome delamination considered distinguishable from the conditions and circumstances of the containment wall opening for the SGR project?

A. There were several reasons why PEF, and its engineers and contractors, determined that the prior dome delaminations did not indicate the potential for a delamination during creation of a construction opening in the CR3 containment wall. First, the CR3 containment wall opening work involved de-tensioning, not tensioning, of the tendons in the concrete. The industry understanding at the time of the SGR project was that de-tensioning tendons reduces the radial tensile stresses on the concrete while tensioning the tendons increases this stress. Each of the three prior dome delamination events occurred during the initial construction of the nuclear facilities when the concrete structure was being fully tensioned. Indeed, two of the three prior dome delamination events were discovered at the time the tendons were actually being tensioned and the concrete stressed during the initial construction of the facilities. Issues associated with the balancing of loads during tendon sequencing were also identified as a cause of the prior CR3 dome delamination. Because the CR3 containment wall opening work involved de-tensioning the tendons, and thus reducing not increasing stresses on the concrete material, the industry understanding at the time was that the same challenges involved with stresses on the concrete material during initial construction tensioning were not involved.

Another reason why the dome delaminations were considered distinguishable from the containment wall opening was the different geometric designs of the domes and the containment walls. The physics associated with the convex, overlapping tendon design in the dome were different from the more uniform, less complex design of the hoop and vertical tendons in the containment building wall. The stresses, therefore, were significantly different in the dome than the stresses in the wall. For example, increased curvature due to tendon alignment in the overlapping, convex geometric tendon structure in the CR3 dome was identified as a cause of increased loads and radial tensile stresses contributing to the CR3 dome delamination. Experience indicated these differences were material, while there were three prior dome delaminations no wall delamination was experienced during initial construction of the nuclear plants where the dome delaminations occurred, and no wall delaminations occurred during the similar steam generator and reactor head vessel replacement projects that used temporary construction openings in the containment building walls.

Additionally, the prior dome delaminations occurred during initial construction with construction impact loads on new or relatively new concrete. Age in concrete is an important factor. Concrete hardens and strengthens over time and reacts differently when it is thirty-plus years old from when it was new or relatively new. New or relatively new concrete experiences high creep under load. Thirty-year old concrete will have very little creep potential left and its compressive and tensile strength will have increased over time. Also, the thickness of the concrete in the dome was less than the thickness of the concrete in the wall and this along with the exposure of the dome directly to the sun made the dome concrete more susceptible to thermal effects. The circumstances and conditions of the prior dome delaminations were therefore significantly different from the circumstances and conditions involved with the containment wall on the CR3 SGR project.

Q. Were the prior dome delaminations analyzed in PEF's root cause investigation of the CR3 wall delamination?

A. Yes. The Company's root cause investigation of the CR3 containment wall delamination confirmed that that the circumstances and conditions of the prior dome delaminations were sufficiently different such that the current delamination could not be predicted based on these prior events. Both the CR3 and Turkey Point Unit 3 dome delaminations were distinguishable from the SGR project work on several grounds. Specifically, the benchmarking performed as part of the root cause investigation determined that the radial stresses and biaxial compression from the geometric (convex) tendon design in the dome were different from the stresses in the wall; the CR3 containment wall delamination involved limited de-tensioning (in and around the construction opening) not tensioning of all of the tendons as was done in the dome delaminations; and the concrete in the wall was over 33 years old, not new or relatively new concrete, as was the case for the domes. This root cause assessment is discussed in more detail in Mr. Miller's testimony.

Q. Did PEF nevertheless take into account the prior CR3 dome delamination in its engineering analyses for the SGR project prior to cutting the containment wall opening in the CR3 containment building?

A. Yes. PEF provided S&L the CR3 dome delamination report and supporting documents from the dome delamination root cause investigation. S&L analyzed this information when it performed its engineering calculations for the containment shell analysis for the SGR project. The CR3 reactor building dome delamination report is specifically identified by S&L as a reference material at page 3 of S&L Calculation No. S06-0002, rev. 1 included as Exhibit No. ___ (JF-19) to my testimony. Calculation No. S06-0002, rev. 1, established the design criteria used for the required analyses of the containment shell contained in calculations S06-0003 through S06-0006 and S07-0003 to demonstrate that (1) the CR3 containment shell was structurally adequate for all conditions that existed during the SGR project and (2) the restored temporary construction opening would return the containment to a condition that was equal to or better than the original design basis containment.

S&L concluded the dome delamination had no impact on its calculations for the containment shell analysis for the SGR project. In particular, S&L concluded that the conditions and circumstances of the prior dome delamination were distinguishable and, therefore, inapplicable to the SGR project.

S&L understood, however, that the concrete material properties of the dome and wall were based on a similar mix design and aggregate even though they would not be homogenous due to differences in the batches for and timing of the concrete pours for the wall and dome. Because the CR3 dome delamination root cause investigation identified the strength of the aggregate as one factor in the dome delamination, S&L considered the compressive and tensile strength of the dome concrete and aggregate from the CR3 dome delamination root cause investigation report in its analysis for the CR3 SGR project. S&L concluded that even conservatively adjusting for the concrete compressive and tensile strength readings from the dome delamination root cause investigation report there was no reason to depart from standard industry practice in the engineering calculations that S&L

performed for the SGR project. This conclusion was supported by the empirical observation that, despite a prior dome delamination during initial tensioning at Turkey Point Unit 3, the containment wall there was successfully opened and closed to replace the Turkey Point Unit 3 steam generators without a delamination in the unit's containment wall.

Q. What was the standard industry practice for addressing radial tensile stresses in engineering calculations for nuclear containment structures at the time of the SGR project?

A. As a matter of industry standard practice, radial tensile stresses at the time of the SGR project were ignored in engineering evaluations due to their small magnitude. The GTSTRUDL two-dimensional thin plate elements of that model assume out-of-plane (tensile and compressive) stresses are zero. The reason is that radial tensile stress is typically small and is not consequential in a containment analysis. Radial tension stresses in the containment shell are maximum during the initial tensioning of the tendons and are thus within the scope of the original design basis calculations. Also, de-tensioning the tendons should lower, not increase, radial tensile stresses in the concrete. Consequently, based on industry experience and the industry standard engineering models at the time of the CR3 SGR project, de-tensioning tendons, creating the access opening in the CR3 containment wall, and re-tensioning tendons in the concrete placed in the opening would not have been expected to come close to let alone exceed the stresses that occur during initial tensioning of the tendons. For all these reasons, industry standard practice was not to factor radial concrete stresses in the containment analyses. S&L correctly followed the industry standard practice in the CR3 containment engineering analyses for the SGR project.

Q. How do you know that S&L was correct in following the industry standard practice on the SGR project despite the prior CR3 dome delamination?

A. After the initial delamination event, and during the course of the root cause investigation, PEF determined that if radial tension stresses were included in the CR3 SGR containment analysis the results would have shown that the estimated radial tensile stresses in the containment wall would not have had any meaningful impact on the containment structure during the SGR project based on the information known and the industry standard engineering calculation model methods at the time of the SGR project. A calculation (S09-0054) was performed to calculate the radial stresses in the containment wall using the standard industry engineering calculation model methods at the time of the SGR project. That calculation demonstrated that the average radial tensile stress was 28.1 psi not accounting for the presence of the hoop and vertical tendon conduits. If both the presence of the hoop and vertical tendons were accounted for, reductions in the cross-sectional areas were made thus increasing the radial tensile stresses from 28.1 psi to 45.5 psi. These tensile stresses were still considerably less (by a factor of nearly 16) than the average splitting tensile strength of the CR3 concrete (708 psi) as reported in the CR3 dome delamination root cause investigation report. Since the maximum radial tensile stresses (45.5 psi) from this industry standard calculation was only a few percent of the concrete tensile strength (708 psi), the radial tensile stresses would not have had an impact on the SGR analysis performed by S&L if they had been included in that analysis. Calculation S09-0054 is attached as Exhibit No. __ (JF-20) to my testimony.

4. The SGR Project Engineering Accounted for the CR3 Specific Information under Industry Standard Engineering Calculation Methods.

Q. Did the engineering analyses for the CR3 containment opening work account for the CR3 containment building design features and concrete characteristics identified as contributing factors to the delamination in the root cause investigation report?

A. Yes. S&L was provided information regarding the design and construction of the CR3 containment building. At the start of the SGR project, for example, PEF gathered the original Gilbert Associates design documentation and calculations that included the calculations for the distribution of stresses in the containment building and provided these design documentation and calculations to S&L. PEF provided S&L the Structural Integrity Test report conducted prior to CR3's initial operation. This test was conducted in accordance with NRC requirements for the structural acceptance of reactors prior to operation and the report indicated that CR3 passed the Structural Integrity Test.

PEF also provided S&L the results of routine Integrated Leak Rate Tests ("ILRT") conducted prior to operation in October 1976 and in 1980, 1983, 1987, 1991, and 2005. Each of these test results indicated that the acceptance criteria were satisfied. Finally, PEF provided S&L the historical IWL and IWE data for the CR3 containment building. The IWL and IWE data was derived from tests pursuant to

ASME Boiler and Pressure Vessel Code, Section XI, Subsections IWL and IWE, which provides for the testing of structures such as the CR3 containment building. This data include the results of routine inspections of the concrete, tendons, and steel liner in accordance with industry standard inspection requirements. This data through the 30th year for CR3 demonstrated that the CR3 containment building had not experienced abnormal degradation and was projected to meet its minimum design requirements until the end of its then forty-year life.

For the SGR project, as I discussed above, S&L prepared detailed engineering calculations that analyzed the stresses in the containment structure when the temporary opening was cut in the containment wall for the steam generator transport using the industry standard engineering analyses and models for such calculations. S&L prepared detailed engineering calculation documents for the containment shell analysis for the containment opening and for the number of hoop and vertical tendons to be de-tensioned and re-tensioned for the SGR project. These engineering calculation documents included Project Calculation documents S06-0002, S06-0003, S06-0004, S06-0005, S06-0007, and S07-0003.

These calculations were prepared based on a PEF contract work order that established a scoping document for the containment shell and opening engineering analyses. This work order was issued pursuant to an Engineering Change that defined the engineering requirements for the containment opening. The work order and Engineering Change provided S&L with the information regarding the construction of the CR3 containment structure and its component characteristics that were used in the S&L engineering calculations.

The information PEF provided S&L specifically included information regarding the CR3 containment reinforcement design features, including the size and location of the horizontal and vertical tendons and the location of the steel rebar in the containment building, which were found to be contributing factors to the delamination in the root cause investigation. Calculation S06-0002 rev. 1 includes a description of the original construction of the containment structure. Calculation S06-0002 rev. 1 entitled Containment Shell Analysis for Steam Generator Replacement – Design Criteria is attached as Exhibit No. __ (JF-19) to my testimony.

The information provided to S&L also included information on the concrete characteristics and material properties, including the type of concrete and aggregate used in the construction of CR3, which was also found to be a contributing cause to the delamination in the root cause investigation. Calculation S06-0003 developed a FEA model used for the evaluation of the CR3 containment shell for the applicable conditions and loads associated with the creation and restoration of the construction access opening in the containment wall for the SGR project. This calculation contains the details of the FEA model including geometry, material properties, including the concrete and aggregate, foundation spring values, mesh generation and one load case that were used in a limited benchmarking of the model against similar results that were available from the design basis calculations. The calculation contains a verification of the accuracy of results obtained from the GTSTRUDL FEA model for loads due to hoop and vertical pre-stress and accident pressure. Calculation S06-0003 is attached as Exhibit No. __ (JF-21) to my testimony.

All of these engineering calculation documents were prepared by S&L in accordance with industry standard engineering calculations and modeling analyses that have been successfully used previously on projects involving the creation of a construction opening in a post-tensioned, pre-stressed concrete containment building.

Q. Can you explain how this information was used by Sargent & Lundy?

A. Yes. S&L used this information to develop the industry standard FEA computer model of the containment building. The current structural analysis methods in the FEA model were used to evaluate the structural impacts of the creation of the containment opening and the loads the containment building wall was supporting during implementation of the SGR project. S&L used the design basis concrete properties as inputs into the various calculations as appropriate. Finally, S&L prepared industry standard engineering calculations to verify the structural integrity of the post-tensioned concrete containment wall for all design basis loading conditions during the creation and restoration of the temporary construction opening, and to the end of plant life in 2036, consistent with the design basis acceptance criteria for CR3 in Section 5.2.3 of the FSAR. *See* S&L Calculation S06-0005 rev. 2 attached as Exhibit No. ___ (JF-22) to my testimony.

S&L's detailed stress analyses considered both the presence of the temporary construction containment opening and the restoration of that opening subsequent to removal and replacement of the OTSGs. In performing these analyses, S&L created several FEA models to model the geometry and properties of the containment building shell and foundation with and without the containment opening. Design basis loads and construction loads expected during the replacement of the OTSGs were applied in the models, evaluated for compliance with FSAR specifications, and the design margins were documented. These loads included the engineering calculations of stresses due to de-tensioning the tendons to create the containment opening and re-tensioning the tendons to restore the wall when new concrete was placed in the opening to close it at the end of the project. All containment building component and element stresses were within allowable code limits during the SGR project and there was no reduction in the design margins after the containment opening was restored. S&L's analyses confirmed that the SGR project containment opening and restoration conditions complied with the design basis acceptance criteria in Section 5.2.3 of the FSAR.

As a result, S&L concluded that the activities involved in the creation of the containment opening and the restoration of the opening -- including the de-tensioning and re-tensioning of the vertical and horizontal tendons -- had no effect on the overall structural integrity of the containment building. S&L's engineering calculations and analyses did not reveal or predict the possibility of delamination in the CR3 containment building.

Q. Were Sargent & Lundy's engineering calculations and conclusions related to the containment opening and restoration reviewed as part of PEF's project quality assurance?

A. Yes. Quality assurance for engineering and construction work is an integral part of every PEF project including the SGR project. All safety related calculations developed for the SGR project by S&L were performed under the S&L NQA-1 compliant QA program. The S&L QA program meets all the requirements of 10 C.F.R. part 50 Appendix B. In addition, the S&L calculations were reviewed and accepted by PEF in accordance with PGN standard procedure EGR-NGGC-003 Design Review Requirements. The GTSTRUDL software used for the industry standard FEA engineering modeling calculations on the SGR project also satisfies NRC 10 C.F.R. Part 50 Appendix B requirements. S&L licensed the GTSTRUDL software and verified that the installed software on the S&L computers yields the same results as those obtained by the master GTSTRUDL program.

Calculations preformed by S&L were also subjected to independent reviews. Once S&L completed its work on an engineering calculation this work was also reviewed by PEF's engineering group for acceptance and analytical quality review. In some cases, Progress Energy also obtained reviews by independent third-party experts. As previously noted, Progress Energy used an expert from Enercon to review S&L's design criteria calculation S06-0002. This calculation established the assessment criteria used in the subsequent S&L calculations. Any comments or concerns identified via the internal S&L reviews, the Progress Energy reviews, or third party reviews, were documented and resolved as part of the project quality program prior to acceptance of the work by Progress Energy.

This process of development, review, and approval of the S&L calculations occurred over a period of several months and involved multiple analyses, meetings, question & answer sessions, and follow up items to verify S&L's work. As noted above, comments or concerns were documented and resolved by S&L. As a normal part of this process, S&L refined and adjusted its work as needed, and S&L performed additional confirmatory analyses when requested. PEF accepted S&L's final work product only after this comprehensive process had fully run its course.

Q. Did PEF's review of S&L's work indicate that S&L was not applying appropriate industry engineering standards to its engineering calculations and modeling analyses for the SGR project?

A. No. On the contrary, PEF's quality assessment review of S&L's work indicated that S&L was applying industry standard engineering calculations and modeling analyses to its work for the CR3 containment opening and restoration. PII in its root cause investigation further confirmed that S&L used typical industry containment structure engineering analyses, calculations, and tools when S&L prepared its engineering calculations and analyses for the SGR project.

The industry standard engineering calculations and modeling analyses were simply inadequate to predict the delamination. At no time during the performance and review of this work did S&L or any reviewer ever indicate that the CR3 delamination was possible based on the standard engineering calculations and analyses employed for such projects. PII performed research and modeling based on the industry standard engineering modeling and calculations during its root cause investigation and determined that application of typical industry engineering modeling tools and calculations would not have been able to accurately predict the margin to delamination at CR3. In fact, PII was unable to reproduce the delamination after it happened using the industry standard engineering modeling techniques and calculations. PII concluded that application of the industry standard engineering modeling tools and calculations that S&L used on the SGR project were insufficient to prevent the delamination. *See* Exhibit No. ___ (GM-4) to Mr. Miller's testimony.

Finally, the NRC sent a Special Inspection Team ("SIT") to CR3 to review the delamination conditions, the root cause investigation, and the planned corrective action. This NRC team issued a Special Investigation Report dated October 12, 2010 with the SIT's findings. The SIT is discussed in more detail in Mr. Miller's testimony and the Special Investigation Report is included as Exhibit No. __ (GM-12) to Mr. Miller's testimony. The SIT agreed that the industry standard engineering approach to calculating the stresses associated with the tendons during the creation of the containment opening did not account for the stress concentration effects that actually occurred during the de-tensioning and creation of the CR3 containment building opening. As a result, and specifically because of the limitations of the industry standard calculations, the SIT noted that future research may be required to determine the significance and impact, if any, of these forces in the design of post-tensioned containments. *See* Exhibit No. __ (GM-12) to Mr. Miller's testimony.

Q. Was there any indication that the industry standard engineering calculations and modeling analyses that S&L used on the SGR project were inadequate to predict and prevent the delamination before it occurred?

A. No. The same industry engineering calculations and modeling analyses were used successfully in projects involving the temporary openings in post-tensioned, pre-stressed containment structures similar to CR3 like ANO-1, ONS 1, 2 and 3, and TMI-

1, among other projects that PEF benchmarked against that I have previously identified. There was no industry experience indicating that the stress margins produced by these industry standard engineering calculations and models were not adequate for the SGR project.

The root cause investigation did reveal that certain design features and concrete properties unique to CR3 contributed to causing the delamination. These included the size and location of the tendons at CR3, the absence of radial reinforcement in the containment wall, and the concrete and aggregate properties. As I previously indicated, however, all of these unique design features and concrete properties were included in the information provided to S&L and factored into the standard engineering calculations and modeling analyses used for the SGR project. With this information, the standard engineering calculations and models still did not indicate the possibility of delamination at CR3. Further, none of the experienced contractors involved in the engineering calculations and review for the SGR project identified any deficiency in these industry standards when applied to the CR3 SGR project.

Q. What were the conclusions about the industry standard engineering calculations and modeling analyses in the root cause assessment?

A. PII did find in its root cause investigation that the standard engineering calculations and models did not adequately account for the combination of stresses created by the unique CR3 design features and concrete properties during the de-tensioning processes. This does not mean, however, that S&L did not consider these contributing factors associated with the design and material properties at CR3 in the application of the standard engineering calculations and models on the SGR project. To the contrary, these factors were considered. The standard engineering calculations and models used by S&L produced stress values that simply indicated that there were sufficient stress margins associated with the containment opening activities at CR3 to account for all these factors. Importantly, there was no information available prior to the delamination to suggest how the values that created the stress margins in the standard engineering calculations and models used by S&L should be changed to more accurately account for the combined effect of the factors contributing to the delamination at CR3.

PII was able to determine the appropriate stress and other engineering material property inputs only by starting with information now known about the delamination. Material property inputs such as concrete elastic modulus, creep coefficient, tensile strength, and fracture energy were modified or developed based on measured properties obtained following the delamination event. These material property inputs were obtained through visual inspection and calculations of the nature and extent of the delamination, core borings, Impact Echo equipment, strain gauges, and other condition assessment techniques and tools described more fully in Mr. Miller's testimony and the root cause assessment report attached as Exhibit No. __ (GM-4) to Mr. Miller's testimony. As a result, it was impossible for PEF and its engineers and contractors to recognize that the stress margins and material property values for the CR3 containment building obtained by applying standard engineering calculations and model analyses should be different and what those different values should be prior to the delamination.

Additionally, PII was able to determine what factors contributed to the delamination and how they contributed only by starting with the material property value information and other information about how the CR3 containment structure behaves learned during the root cause investigation and working backwards in time. This process involved engineering analysis and modeling improvements to eventually simulate the delamination event and thereafter develop a repair plan. These engineering analysis and modeling improvements are also discussed more fully in Mr. Miller's testimony and in the root cause assessment report attached as Exhibit No. ___ (GM-4) to Mr. Miller's testimony. Because additional engineering modeling and calculation improvements were necessary to simulate the delamination, however, it was impossible for PEF or its contractors and engineers to recognize that the engineering standard calculations and models used in planning the SGR project were inadequate to predict the delamination prior to the delamination and subsequent root cause investigation.

Q. Was there any reason to question CR3's design features or concrete properties in planning the SGR project?

A. No. All of CR3's containment building design features and material properties, including the concrete and aggregate, were approved for use in the building at the time CR3 was built and the structural integrity of the CR3 containment structure was confirmed by the structural integrity test under NRC guidelines for operating requirements prior to the initial CR3 start-up. Subsequent periodic inspections and tests were conducted and the containment integrity was confirmed. These inspections and tests are described in the NRC SIT report attached as Exhibit No. ___ (GM-12) to Mr. Miller's testimony. The CR3 containment building was performing the job it was designed and constructed to perform and there was no reason to question otherwise at that time.

Q. Was there any reason to question CR3's concrete aggregate?

A. No. CR3 used Type II cement and Florida limestone aggregate to create the CR3 concrete containment walls. The concrete met all design strength requirements for CR3. Indeed, the limestone aggregate used at CR3 can produce high quality, compressive strength concrete. But the limestone aggregate is a more coarse, soft and porous aggregate and, therefore, it had lower tensile strength and crack arresting capabilities than harder, less porous aggregates found elsewhere in the country and used in other containment structures. Considered alone however, the stresses involved in the CR3 containment design were and are well within the capability of the concrete material used. Also, as I discussed previously, calculation (S09-0054) was performed to calculate the radial stresses in the containment wall using the standard industry engineering calculation model methods at the time of the SGR project. That calculation demonstrated that the average radial tensile stress was 28.1 psi not accounting for the presence of the hoop and vertical tendon conduits. If both the presence of the hoop and vertical tendons were accounted for, reductions in the cross-sectional areas were made thus increasing the radial tensile stresses from 28.1 psi to

45.5 psi. These tensile stresses were still considerably less than the average splitting tensile strength of the CR3 concrete (708 psi) as reported in the CR3 dome delamination root cause investigation report. Since the maximum radial tensile stresses (45.5 psi) from this industry standard calculation was only a few percent of the concrete tensile strength (708 psi), the radial tensile stresses would not have had an impact on the SGR analysis performed by S&L if they had been included in that analysis.

5. The Scope and Sequencing of De-tensioning for the SGR Project.

Q. You testified that the root cause investigation indicated that the scope and sequencing of de-tensioning was also a cause of the delamination. Did S&L's engineering work produce the plan for the CR3 de-tensioning work?

A. Yes. The scope or number of the vertical and horizontal tendons to be de-tensioned to create the containment opening was central to S&L's engineering work. As I have explained, S&L employed industry standard engineering models and calculations to determine the scope of de-tensioning for the SGR project. Although the limitations of the industry standard calculations and models to produce the stresses that caused the delamination are discussed in more detail in the testimony of Mr. Miller and in the root cause assessment report attached as Exhibit No. __ (GM-4) to Mr. Miller's testimony, generally the industry standard calculations and models predicted larger stress tolerance margins than actually existed. S&L used these stress margins from the industry standard calculations and models to determine the scope of de-tensioning the tendons to create the containment opening at CR3.

Q. What was the de-tensioning plan scope for the SGR project?

A. S&L's initial de-tensioning plan for the CR3 containment building involved de-tensioning a total of 30 vertical tendons and 35 horizontal hoop tendons. This plan included the 10 vertical tendons and 17 horizontal tendons located within the planned containment wall opening and 10 vertical tendons on either side of the planned containment opening and 9 horizontal tendons above and below the opening. Calculation S06-0004 attached as Exhibit No. ___ (JF-23) to my testimony contains the calculations and supporting engineering study for this de-tensioning plan.

Ultimately, the plan to de-tension 30 vertical and 35 horizontal hoop tendons was divided into two phases. The first phase involved de-tensioning and removing the 10 vertical and 17 horizontal tendons that traversed the planned containment opening. Phase two involved de-tensioning the remaining 20 vertical and 18 horizontal tendons around the containment opening after the OTSGs were moved out of the containment building and the new OTSGs were moved into the building through the crane and rigging system, prior to installation and welding of the liner insert in the containment opening and placement of the new concrete, vertical and horizontal tendons, and steel reinforcement in the containment opening.

Q. Why was the scope of the de-tensioning plan divided into two phases?

A. Mammoet, the heavy rigging contractor on the SGR project, planned to support its bridge through the containment opening on the containment wall. Mammoet had determined this was the most efficient, cost-effective plan for use of the crane and rigging system to move the OTSGs in and out of the containment building. Before allowing Mammoet to proceed with this plan PEF had S&L analyze the strength of this portion of the containment wall under the construction conditions. S&L responded that the containment wall was sufficient to support Mammoet's planned operation of the crane and rigging system if the horizontal tendons below and the vertical tendons adjacent to the containment opening remained in tension. This resulted in the two phase de-tensioning plan involving de-tensioning and removal of only the vertical and horizontal tendons in the construction opening in phase one prior to removal and replacement of the OTSGs. This two-phase de-tensioning plan was confirmed by S&L's engineering calculations and reviewed and approved by PEF. *See* Exhibit No. ___ (JF-23) to my testimony.

Q. If S&L initially planned to de-tension tendons in and around the containment opening prior to creating the opening why did S&L revise its plan?

A. S&L's initial de-tensioning scope was created based on the number of tendons that needed to be de-tensioned to successfully restore the pre-stress in the replacement concrete in the containment opening. The reason is that the replacement concrete in the containment opening is obviously new concrete and new concrete even of the same type and mixture of cement and aggregate as the old concrete still has different physical properties than 30-year old concrete because it has not yet aged and it has higher initial creep characteristics. Detailed engineering calculations and analyses were conducted to determine the number of tendons that needed to be de-tensioned to ensure that sufficient pre-stress was restored to the new concrete taking into account the effects of aging. S&L's initial de-tensioning scope was derived from the engineering calculations and modeling analyses necessary to ensure that sufficient pre-stress is restored to the new concrete in the containment opening. The focus of the initial de-tensioning scope that led to the plan to de-tension more vertical and horizontal tendons than were de-tensioned in phase one was not, therefore, the number of tendons that needed to be de-tensioned to successfully create the containment opening. This calculation and a description of it and the necessary material for the replacement concrete are described in Exhibit No. ___ (JF-23) to my testimony.

The reason utilities and their engineers on projects involving containment structure openings take the approach of initially de-tensioning all tendons necessary to restore adequate pre-stress to the repaired containment opening is because this plan reduces engineering and mobilization and demobilization costs. The number of tendons that need to be de-tensioned to create the containment opening is generally less than the number of tendons that need to be de-tensioned in order to restore the required level of pre-stress to the containment after the opening is repaired. Initially de-tensioning all the tendons necessary to create the opening and to restore the required pre-stress means that mobilization for the de-tensioning needs to be done once not twice. This reduces the cost of sending de-tensioning crews out twice to perform the de-tensioning work. For example, if a plant needs to de-tension 20 tendons to create the containment opening and 40 tendons to repair it, most plants will go ahead and de-tension all 40 tendons on the front end to keep from having to de-tension twice. The driver for the initial S&L de-tensioning scope was therefore construction efficiencies and not the engineering margins to safely and successfully create the temporary containment opening.

On the SGR project, however, the most efficient and cost-effective way to move the OTSGs in and out of the opening was the Mammoet proposal for setting up the crane and rigging system. This was the same crane and rigging system design that was used successfully at the three Oconee units (ONS 1, 2, and 3) and at ANO 1, all of which had containment designs similar to CR3. Using the proven Mammoet design resulted in lower risk to the overall project as the lifting and transport of the OTSGs was considered a high-risk evolution for the project. As a result, S&L confirmed that the Mammoet proposal could be successfully employed at CR3 if the initial de-tensioning scope was divided into two phases thus providing for the de-tensioning of the tendons necessary for successful installation of the replacement concrete in the opening after the OTSGs had been moved out of and into the building.

Q. Did S&L identify a sequence to de-tensioning the tendons for the containment opening?

A. S&L's engineering design documents did not contain a specific sequence for de-tensioning the tendons for the containment opening because S&L determined, based on its engineering calculations and analyses, that the reduced stresses from de-tensioning the vertical and hoop tendons in the containment opening were sufficient to de-tension them in any order. With the exception of the two vertical tendons in the center of the opening, which were de-tensioned first, PEF had PSC de-tension the tendons in phase one sequentially, counterclockwise for the vertical tendons and from bottom to top for the horizontal tendons. The sequential de-tensioning sequence PEF employed for the CR3 containment opening was consistent with the de-tensioning sequence for six out of the eleven nuclear unit construction projects that involved the creation of containment openings in their containment buildings prior to the CR3 project. On these projects, the tendons in and around the construction opening were de-tensioned sequentially. On most of the remaining projects the de-tensioning sequence involved de-tensioning every other tendon in and around the containment opening instead of the tendons in sequential order. PEF confirmed in its root cause investigation that the project plans involving de-tensioning every other tendon rather than each tendon sequentially were not supported by any engineering analyses or other analytical basis. Rather, these de-tensioning sequencing plans were simply based on what was preferred by the contractors involved.

In sum, then, PEF's sequencing for the de-tensioning of the tendons on the SGR project was supported by engineering analyses and calculations using the standard engineering methods at the time. PEF's de-tensioning sequence was also consistent with the de-tensioning sequences used on other projects that involved de-tensioning work associated with the creation of temporary construction openings in containment buildings.

Q. If PEF had implemented the initial de-tensioning scope, or changed its de-tensioning sequence on the SGR project to correspond with other sequences used in the industry, would these actions prevent the delamination?

A. No, they would not. In fact, PII determined in the root cause investigation that if PEF had implemented the initial S&L de-tensioning scope involving the de-tensioning of the tendons in and around the containment opening consistent with industry practice the delamination would likely have been worse. The reason for this is explained in more detail in PII's root cause assessment report and in Mr. Miller's testimony, but basically implementation of the de-tensioning scope originally planned by S&L would have increased the localized stresses in Bay 3-4, adding to the asymmetrical forces in the containment structure, and likely increasing the delamination in Bay 3-4 or resulting in delamination or cracking in other Bays in the CR3 containment building. Similarly, changing the tendon de-tensioning sequence from the sequence primarily used in the industry to other industry de-tensioning sequences would not have reduced or prevented the delamination. The same stresses and combinations of factors identified as the contributing causes in the root cause investigation would have been present regardless of the sequence of the de-tensioning in and around the containment opening.

IV. THE DECISION TO REPLACE THE OTSGs THROUGH A TEMPORARY CONSTRUCTION OPENING IN THE CONTAINMENT BUILDING WAS PRUDENT BASED ON THE INFORMATION AVAILABLE AT THE TIME.

Q. How did PEF decide the best way to remove the old steam generators and put the new ones into the CR3 containment building?

A. PEF started by selecting a project team to perform a comprehensive analysis of the options to move the old and new steam generators out and into the building. There were only two available options: moving the steam generators out and in through the existing equipment hatch to the building or through a temporary construction opening in the containment building wall. There was no preconceived choice. The evaluation for this work started with the concept of "what is the best way to remove and reinstall the steam generators in the reactor containment building." *See* Exhibit No. ___ (JF-24) to my testimony. The decision making process involved a study to evaluate both options to arrive at the best choice solution.

Q. Was the existing CR3 equipment hatch designed to handle replacement of the steam generators?

A. No. At CR3, the existing equipment hatch was designed to transfer some items in and out of the containment building, but there was not a designed transport path to accommodate the movement of large pieces of equipment once the equipment was through the hatch. As can be seen from the photographs in my Exhibit No. ____ (JF-25), the original OTSGs were placed inside the plant when there was no roof on the plant, no wall where the equipment hatch now resides, and no concrete on the building. The original OTSGs were initially expected to last 40 years or about the same period as the initial operating license for CR3. The physical process of removing the old steam generators and installing the new ones through the existing equipment hatch, therefore, presented many challenges because of the numerous interferences that would have to be removed if this path was followed.

Q. What were those challenges?

A. The location and size of the equipment hatch presented a significant obstacle to safely transporting the components through the hatch. The equipment hatch opening is only 22 feet 4 inches in diameter. As I explained earlier, each of the old and new steam generators measure 12 feet 4 inches in diameter (over 15 feet with the shop installed piping), and they are 73 feet 3 inches tall from the bottom of the skirt to the end prep of the hot leg nozzle. To remove and replace the OTSGs through the equipment hatch, each steam generator must be carefully lifted by heavy-duty lifting cranes from its position inside the D-ring. With a gentle fluid motion, each steam generator must then be tilted and angled out of the narrow equipment hatch with only inches to spare, given the location of the hatch.

To understand the constraints associated with the hatch location better it is helpful to understand the size of the interior and the placement of floors, walls, and major pieces of equipment and material inside the building. The inner diameter of the containment building is 130 feet and the interior height is 187 feet. Inside the containment building, there is a poured concrete floor inside the liner. The top of this floor is at the 95 foot elevation, which is the first level of the D-rings and is the floor on which the steam generators are installed. From this floor, there are two additional levels with floors at the 119 foot elevation and the 160 foot elevation. The centerline of the equipment hatch is located at the 132' elevation, which opens into the space between floors at the 119' and 160' elevations, below the operating deck where a vast amount of equipment, structures, cables, conduit, and piping are installed.

Before manipulating the steam generators out of the containment building through the equipment hatch, then, these permanent fixtures and concrete must be removed to allow enough room for the steam generators to be lowered from their elevated position inside the D-rings and carefully angled to exit through the equipment hatch. Once the old steam generators are removed, the new ones must be attached to the lifting device and gently and angularly hoisted into position. Vast amounts of equipment, cables, piping, and flooring must be reinstalled and meticulously inspected prior to taking the plant online in order to use the existing equipment hatch.

Q. How did PEF proceed with the evaluation of the equipment hatch and containment wall opening options for removal and replacement of the steam generators?

A. PEF retained the services of MPR Associates ("MPR"), a leading power industry engineering company, to evaluate the feasibility of using the existing equipment hatch to transfer the steam generators in and out of the containment building. The feasibility of using the existing equipment hatch was evaluated using three-dimensional computer models of the replacement steam generators and containment building.

Q. What information was used to develop the three-dimensional computer models?

A. The models of the major structural components including the containment building, liner, equipment hatch, biological shield wall and floor elevations were developed based on design and construction drawings. MPR and AREVA, using Photogrammetry and laser scanning techniques, completed a walk-down of the reactor building crane bay at the 95 foot, 119 foot, and 160 foot elevation. This was used as input for the engineering assessment of a rigging path through containment for bringing the steam generators in through the equipment hatch. Information from three-dimensional laser scanning of the containment was used to add the smaller, more detailed, field routed components to the model. Members of the project team made four entries into the containment building at power and obtained additional photographs to obtain a better understanding of the configuration, specific potential interferences, and additional design information associated with these potential interferences. Animations were then developed to show the concept of the replacement steam generators routed through the equipment hatch onto the 119 foot elevation and up through the 160 foot elevation floor.

Q. What were the results of MPR's computer models?

A. Using this information in its models, MPR developed three options to route the steam generators through the equipment hatch. MPR's objective was to find a routing option that minimized interferences with existing containment structures, piping, cable trays, and equipment. The modeling results of all three equipment hatch options demonstrated that it was not possible to pass the steam generators through the equipment hatch, through the containment, and into final position without encountering interference from considerable containment internal structures, piping and equipment. Two of the three route options (options 2 and 3) were not preferred because those routes involved multiple interferences whose removal was relatively more difficult than option 1. Although option 1 still involved significant interferences, it was the preferred routing method if the hatch was to be used because it involved less interference with cable trays, structural steel columns, or the platform on the 127 foot elevation. I have attached a copy of MPR's computer animation as Exhibit No. ___ (JF-26) to my testimony.

Since the MPR computer animation shows the steam generators "floating" through the air without support from heavy lifting equipment and also shows equipment moving and disappearing as the generators follow the entry path on the preferred route through the equipment hatch, I have also included static photographs in my Exhibit No. __ (JF-27) to provide a real world perspective on what sort of equipment would be needed to move these generators out of and in the hatch, and what the moving and disappearing equipment inside the containment structure actually looks like.

MPR also evaluated the containment opening option and concluded that due to the location of surrounding buildings, the opening would be located directly above the existing equipment hatch at approximately 195 feet to allow the generators to enter the containment building above the operating deck. This was the option that other plants that are similar in design to CR3 had used to move steam generators out of and into nuclear containment structures, including Three Mile Island, FPL's Turkey Point Facility, and the three units at Duke's Oconee Nuclear Station.

Q. Did PEF consider what other utilities had done in evaluating the equipment hatch and containment opening options?

A. Yes. As I testified earlier, PEF benchmarked against plants that were comparable to CR3 that had completed a SGR project. One of our primary benchmarking locations was Duke's Oconee Nuclear Station. As I previously noted, three of our employees participated in the SGR project for Oconee units 1 and 3. The Oconee units used a construction opening to remove and replace their steam generators. Members of the CR3 SGR Project team also gained invaluable knowledge from benchmarking trips to various other nuclear facilities undergoing steam generator replacements. These included ANO and St. Lucie. By looking at construction opening risks from past experiences of other utilities versus perceived risks expected from the equipment hatch opening, the SGR team gained substantial insight and knowledge of rigging schemes, lessons learned, and the risks and benefits associated with each option.

PEF studied those plants with post-tensioned containment buildings that utilized the existing equipment hatch to remove and replace steam generators as well as those plants choosing to create a construction opening. In doing so, PEF found that the plants that are similar to CR3 in design had created containment openings for their SGR projects rather than using existing equipment hatches.

No plant is exactly the same as CR3, although many nuclear containment buildings are similarly constructed, each plant has different floor configurations and each plant has different interferences. Most importantly, however, some plants have equipment hatches that enter the containment building above the operating deck. Others, like CR3, have an equipment hatch located below the operating deck. As I explained earlier, the containment building houses large and complex pieces of equipment, cables, fans, and piping. If the equipment hatch is above the operating deck, such as the case with Progress Energy's Harris plant, the steam generators can sometimes be positioned in an upend-down-end motion and through the equipment hatch without major interference removal. However, when the equipment hatch is below the operating deck, such as the case with CR3 where the equipment hatch is one floor below the operating deck, there is no easy means of upending the old steam generators to get them out of their vertical resting position and maneuvered through the hatch. By coming in on the operating deck, the generators come in above most of the cables, fans, and piping that interfere below the operating deck. In fact, due to these difficulties from using a lower equipment hatch to replace OTSGs, two sister plants have chosen to continue using a construction opening even after the CR3 delamination experience. In only one case was a lower equipment hatch used in the industry to replace steam generators and that was only because the design of those steam generators allowed the generators to be placed into the containment building in two pieces and assembled inside. *See* Exhibit No. ___ (JF-28) to my testimony.

Exhibit No. ___ (JF-28) is a chart of the nuclear power plants that replaced steam generators through the equipment hatch and through a construction opening in the containment building. This chart demonstrates that OTSGs were replaced through the construction opening at nuclear power plants with a similar containment building design to CR3. To put it simply, if the equipment hatch is located at the operating floor or deck, the steam generators are replaced through the equipment hatch (if the hatch is large enough to accommodate the steam generators). If the equipment hatch is below the operating floor or deck (like CR3), the steam generators are replaced through a construction opening in the containment building.

Q. What was the conclusion of MPR's evaluation of the equipment hatch option?

A. MPR concluded that use of the equipment hatch was technically possible, but further evaluation was needed. Specifically, MPR recommended that more detailed costs estimates be prepared for removing and reinstalling interferences, for more complex rigging and manipulation of the steam generators, and for engineering the equipment hatch option. MPR also recommended that PEF consult a rigging vendor to confirm the feasibility of the proposed path and to estimate the corresponding cost and schedule to implement the rigging for the equipment hatch option. A copy of MPR's March 2005 Report is included in Exhibit No. __ (JF-24) to my testimony.

Q. What action did PEF take with respect to MPR's conclusions and recommendations?

A. To make a more informed decision as to which option to use in moving the steam generators into and out of the containment building, PEF requested MPR to conduct an analysis of the costs associated with creating and restoring a new opening in the containment building. While MPR prepared the containment opening estimate, PEF's SGR Team Major Projects Section prepared an estimate for costs associated with the equipment hatch option. A cross section of industry experts assisted PEF with estimates on crew size, duration, and type of craft required. MPR also contacted industry experts for assistance in calculating tendon, hydro-demolition, and concrete costs. By preparing each of the estimates concurrently, both completed estimates were reviewed and compared side-by-side to ensure that assumptions within each estimate were similar. Adjustments were made to each estimate to allow them to be compared fairly. The comparison concluded that there was construction cost savings of $2.64 million associated with the containment opening primarily due to the equipment hatch interferences. More specifically, total construction costs for the new opening were estimated at $6.27 million, whereas construction costs for the equipment hatch option totaled $8.92 million.

PEF also studied the impact to engineering scope related to both options. The study determined costs associated with only those engineering products specific to each option. Detailed lists were developed itemizing all engineering change ("EC") packages required for each option. Upon comparison, the equipment hatch option increased the number of inside containment modifications significantly. This increase in in-containment Engineering Change packages results in an increase in in-containment Work Orders and an increase in difficulty. Some reasons for the increased scope of in-containment work are the increase in scope for such things as Main Steam, Feed Water, and Emergency Feed Water pipe and pipe support removal/replacement and reinforced concrete wall and floor removal and replacement. The number of ECs increases from 22 for the Construction Opening option to 29 for the Equipment Hatch option. However, the scope of some of the 22 ECs for the Construction opening is increased for the Equipment Hatch Opening option (i.e., large bore piping). The results of the study indicated that engineering costs associated specifically with the equipment hatch option were estimated at $4.31 million, whereas engineering costs associated specifically with the construction opening option were estimated at $1.34 million.

Q. Were there other considerations associated with the equipment hatch option?

A. Yes. Two other considerations involved radiological waste disposal or "Radwaste" and the level of radiation exposure for the workers. Radwaste is waste product containing radioactive material. Radwaste is carefully monitored, segregated and stored depending on the level of radioactivity. Proper handling and disposal of this waste must comply with stringent state and federal regulations. Naturally the cost associated with collecting, handling, and disposing of radioactive waste is significantly greater than for non-radioactive waste. Relative to the construction opening option, the Equipment Hatch option would have resulted in significantly more radioactive waste because of the increased amount of work to be done inside the containment building.

At a nuclear plant, minimizing the level of radiation exposure to the workers is a primary objective for any work activity. Numerous programs are in place to protect employees from exposure to radiation and to record any radiation dose received by the person. The industry approach to achieving this objective is known as "As Low As Reasonably Achievable" or ALARA. Radiation dose is carefully monitored under the Code of Federal Regulations Environmental Radiation Protection Standards for Nuclear Power Operations. At CR3, the inside of the containment building is a Radiation Control Area ("RCA") and those workers who work inside the containment building are exposed to varying levels of radiation.

In comparing the construction opening with the equipment hatch option, PEF determined that if PEF used the existing equipment hatch the additional work inside the containment building to remove and then reinstall interferences would result in approximately 15 times the level of radiation exposure that would be incurred with the construction opening option.

Q. How did PEF evaluate the technical feasibility, cost, risk, and other considerations associated with the equipment hatch and containment opening options?

A. After determining the full scope, project schedule, cost, dose, risk, and other relevant items for the cumulative scope of each option, PEF obtained sufficient input for an objective Kepner Tregoe ("KT") analysis to assist in arriving at the best decision. A KT analysis is a decision assessment methodology that provides a structured and systematic process for analyzing and determining the optimum choice between two or more competing options. The process is based on the establishment of specific decision criteria for the selection, weighting the criteria based on their importance to the overall decision, an analysis, using subject matter experts, of the extent to which each option meets those criteria, and then combining that analysis into an overall weighted score that allows the decision maker to rank the options relative to each other.

The KT analysis approach also includes a risk assessment of the various options. This part of the analysis involves the identification of potential threats and opportunities for each option. Each opportunity and each threat is assigned a high, medium, or low probability of occurrence and a consequence should it occur. Risk is generally defined as the probability of an occurrence times the consequences of that occurrence. The opportunities and threats can then be plotted on a matrix that provides a visual comparison of the relative risks and opportunities of each option.

A copy of PEF's KT analysis for the two SGR replacement path options is included in Exhibit No.__ (JF-24) to my testimony. As one can see, the KT analysis results show that the creation of a containment opening had a higher weighted ranking than the equipment hatch option. The opportunity assessment was more favorable for the containment opening, but, more importantly, the threat assessment was significantly more favorable for the containment opening option. Thus, from a project risk perspective, the creation of a containment opening was vastly superior to the existing equipment hatch option.

Q. Was the potential for delamination in the containment wall at CR3 identified as a potential risk for the containment opening option?

A. No, there was no reason for PEF's SGR project team or MPR to identify the potential for delamination in the containment wall as a risk for the containment opening option on the SGR project and they did not in their evaluations of these two options. As I testified earlier, numerous steam generator and reactor head vessel replacement projects were successfully completed using construction openings through post-tensioned, pre-stressed concrete containment structures without any delaminations. As a result of this prior experience, and based on the existing engineering and construction knowledge at the time PEF was planning the SGR project, no experienced utility, contractor, or engineer in the industry had ever identified delamination as a potential risk for the containment opening option and none of these industry participants made such a risk known to PEF when it was evaluating these options or later planning the SGR project.

Q. Once the SGR project team reached its conclusion about the equipment hatch and construction opening options what were PEF's next steps?

A. Once the CR3 SGR project team determined that the construction opening represented the best option for the steam generator replacement, that option was presented to, discussed with, and approved by management personnel from the CR3 Site Vice President through the Corporate Senior Vice President responsible for PGN's nuclear program (the chief nuclear officer), and eventually to the President of Progress Energy. Final approval to proceed with the containment opening option was obtained in December, 2005. Following that approval, the decision was formally included in a revised business analysis package ("BAP"), which was submitted on May 9, 2006, and approved in October, 2006. The BAP is attached as Exhibit No. __ (JF-29) to my testimony. The cost estimates in the BAP were based on the use of the containment opening option.

Q. Why was creating a containment opening superior to other choices for removing and replacing the steam generators?

A. In summary, the containment opening option presented far less risk, exposed workers to almost fifteen (15) times less radiation, and cost less to perform. Furthermore, while the schedule duration for both options were virtually identical in PEF's analysis based on known activity that would have to be done, for the equipment hatch option PEF had identified multiple factors that could not be quantified with any degree of certainty, but that had the potential to significantly extend the duration and thus the cost of the SGR project.

Q. What were the factors associated with using the existing equipment hatch that you mentioned that could have substantially increased the duration of the SGR?

A. First, at least two cable trays would have to be raised up over their current location to allow steam generator passage through the existing equipment hatch. This would be accomplished by installing temporary supports with turnbuckles that would allow cable trays to be raised, and then lower portions of permanent supports would be cut and removed. There was also a significant risk that the 1970s vintage cable jackets could be damaged during the raising and lowering of the cable trays. The NRC and other applicable codes do not allow splicing of cable inside the cable trays. A splice box would have to be used outside of the cable tray to splice the cables. If all cable had to be pulled back to termination points, the equipment hatch option would not be a feasible option because this activity would require a virtual rebuild or wholesale replacement of the cable tray structures. Photographs of these cable trays are included in my Exhibit No. ___ (JF-30).

Second, the equipment hatch enclosure performs both a missile shield and flood protection function. During the time the equipment hatch enclosure is removed, the flood protection barrier is compromised. A flood barrier redesign would be required to address this issue. This was particularly important since this work would be accomplished during hurricane season. Because of this, PEF would have to determine if a licensing basis change is required to be submitted to the NRC prior to performing any modification for this redesign. If a license change were required, the equipment hatch option schedule duration would be substantially extended and the equipment hatch option could possibly be rendered infeasible.

Third, while PEF provided its best estimate for the duration of the work for the equipment hatch option, there was not a high a degree of certainty regarding PEF's piping and pipe support work duration assumptions. To be clear, choosing the equipment hatch option would add a significant volume of piping and pipe support work to the SGR project. This type of work has been difficult historically to schedule accurately based on Progress Energy's and other utility experience. Further, the addition of concrete floor cuts, concrete wall cuts, reactor cavity fan removal, and removal and rerouting of miscellaneous piping and conduits added additional schedule variability and uncertainty. Thus, PEF's assumptions for the duration of this work in the schedule for the equipment hatch option were conservatively low in favor of the equipment hatch option.

Fourth, in estimating the cost and schedule for the equipment hatch option work, PEF was not able to identify all potential interferences until the unit was offline and open and a thorough walk-down of the core flood tank room walls and the D-ring wall could be performed. Further, the underside of the 160' elevated floor slab cut envelope would need to be performed with the plant offline for all interferences in that area to be identified. These issues naturally had a real potential to add significant time to the duration of the equipment hatch option.

Finally, for the equipment hatch option, PEF was not able to predict with any degree of certainty the amount of schedule time that would be needed for necessary equipment testing prior to the return of CR3 to commercial operation. PEF did not and could not know what work associated with all interferences was required for the equipment hatch option. As a result, PEF did not know all the equipment that must be moved or removed and replaced that would have to be tested prior to commercial operation of the unit. Accordingly, even more time would have to be added to the equipment hatch option schedule if pre-restart testing was required for any of this equipment. PEF documented its comprehensive evaluation of these issues in the documents that I have attached as Exhibit No. __ (JF-24) to my testimony.

For all of these reasons, PEF concluded that creation of a temporary construction opening in the containment building to move the old OTSGs out and the new OTSGs into the building was the best option. This decision matched the decision made by all other utilities for the same or similar projects to the CR3 SGR project with similarly designed equipment hatches. *See* Exhibit No. ___ (JF-28) to my testimony.

V. PEF PRUDENTLY MANAGED THE SGR PROJECT.

Q. When did PEF commence the steam generator replacement project?

A. A steam generator replacement project requires considerable up front planning. The typical replacement project begins approximately five to seven years prior to the actual replacement, encompassing two refueling outages prior to replacement to complete the design and planning effort. Accordingly, a core project team for the replacement steam generator project was formed in November, 2003 and in 2004, PEF initiated the design phase of the SGR project.

Q. How did PEF manage the SGR project?

A. The initial design phase of the project assumed the use of a prime contractor for the project. In 2004, PEF issued a RFP and a Request for Qualifications ("RFQ") seeking qualified firms to manage the SGR project. PEF received proposals from Bechtel and the SGT, and both Bechtel and SGT had prior steam generator replacement project experience using temporary construction openings through nuclear containment buildings. PEF selected SGT as the initial vendor of choice due to the indicative pricing in their RFP response and based on their recent experience and awarded work with OTSG replacements at Oconee and ANO 1. PEF entered into contract negotiations with SGT, but after months of negotiating the commercial terms of the contract, PEF and SGT were not able to reach an agreement on a contract price structure.

Because the SGR project team had prior experience with and knowledge about what was required to perform the SGR project from the SGR project conducted at Progress Energy's Harris plant in North Carolina, PEF also evaluated a self-managed project at the same time. Under this approach, PEF would perform the duties of the prime contractor and manage the SGR specialty contracts directly and a construction management company would be selected to provide construction practices and methods as well as to provide, manage, and supervise the craft labor. In the self-managed approach, PEF develops the engineering and design products using a blend of permanent and secondary engineering personnel supported by a third-party engineering firm. The SGR subcontractors, including heavy lifter, reactor cooling system cutting and welding, construction, and demolition, report directly to Progress Energy without an intervening prime contractor and PEF would also engage third-parties with diverse industry experience to review engineering and design work prior to it being implemented in the field.

Q. Why did PEF consider a self-managed project management approach beneficial for the SGR project?

A. The self-managed SGR project cost less than a prime contractor-managed SGR project because the self-managed approach eliminates the SGR project prime contractor fee; eliminates SGR prime contractor markup on SGR specialty contractors; eliminates SGR prime contractor parent company overheads and indirect costs; and reduces the number of duplicate positions to oversee the efforts of a SGR prime contractor. Additionally, this approach also allows the Company to negotiate more favorable labor rates with both engineering and craft organizations outside of non-negotiable rates from a prime contractor, and to leverage the same subcontractors that would be used in a prime contractor arrangement without a cost markup for prime contractor profit.

Q. How did PEF decide between the project management alternatives?

A. PEF evaluated the costs and risks associated with self-management versus use of a prime contractor. The results of the evaluation are summarized in Exhibits Nos. __ (JF-31) and ___ (JF-32) to my testimony. While there were various risks, costs, and benefits associated with each of the options, PEF found that the self-managed project option was superior to a prime contractor option for two major reasons. First, the self-managed option allowed more independence and objectivity on the project because unlike a prime-contractor project, critical resource groups such as engineering were separate from construction craft labor groups and were directly accountable to PEF. Additionally, each subcontractor was subject to cross review and third-party expert review at PEF's direction, which provided an additional layer of checks and controls to the project. Further, PEF was able to retain all the same contractor experts that would have been used in a prime contractor arrangement and negotiate favorable pricing with them directly rather than having to accept a marked up price for those same services from a prime contractor.

PEF ultimately selected the self-managed approach and took responsibility for obtaining the resources needed for the planning, design, material, and procedures necessary to implement the physical scope of the SGR project. The procurement phase of the project included all activities necessary to award major contracts for engineering, construction management, heavy lifting, hydro-demolition, liner plate removal and restoration, cutting and welding, and replacement steam generator manufacturing. Toward this end, PEF engaged Bechtel to provide craft labor and to manage the construction phases of the project. This allowed PEF to leverage Bechtel's considerable experience and resources that were used in several other SGR projects throughout the country. Through a competitive bidding and evaluation process, PEF engaged S&L to provide engineering services for the project, and PEF also provided internal and external third-party engineering review. PEF selected PSC to perform tendon services for the project given PSC's vast experience in virtually every SGR project that has taken place in the country. PEF also selected Mac & Mac for hydro-demolition services and Mammoet, a proven SGR company, for heavy lifting services. These selections, coupled with PEF resources with prior SGR and major project experience, provided PEF the talent and skill it needed to perform the SGR project.

Q. Prior to starting the SGR project, did PEF employ any processes, procedures, or measures to govern the project work?

A. Absolutely. Prior to any work taking place on the project PEF ensured that its employees and all the contract work taking place on the project were governed by processes, procedures, and risk mitigation measures in six major areas. Those areas are project development, project approval, project controls, project management, project execution, and project review. I discuss each of these major areas below.

Q. What sort of processes, procedures, and risk mitigation measures did PEF use for project development?

A. Various corporate and plant procedures and processes provide guidance for the development, planning, and execution of a major project such as the SGR project. At the plant level, NGGM-PM-0018 (Project Management Manual) outlines the fundamental project management processes to be followed in the initiation, planning execution, control and closeout of projects in the Nuclear Generation Group. The provisions of NGGM-PM-0018 were followed for the SGR project at CR3.

A specific provision of PM-0018 is the development of a risk management plan that describes how a project is going to address risk in the execution of the project. The Risk Management Plan involves the development of a Risk Matrix summary to identify, quantify, and communicate risks associated with a project and to assist in the development of contingency plans where appropriate for dealing with those risks. In short, the Risk Management Plan and the Risk Matrix Summary is a proactive approach to anticipating and preparing in advance for the risks that can reasonably be postulated for a given project. The Risk Matrix Summary for a project is initially developed by a core team consisting of project management, engineering, and project controls personnel along with other subject matter experts as appropriate. Once developed, the Risk Management Plan and the Risk Matrix Summary also provide a tool for periodic risk assessments and updates during the course of a project to reflect changing conditions, scope revisions, and lessons learned from other projects, among other factors.

Other procedures ranged from guidance on engineering rigor, engineering change management, preparation and control of design analysis calculations to guidance on plant walkdowns. For the SGR project, a specific set of SGR Guidance Documents were developed. The documents covered a variety of issues including quality assurance for fabrication and installation, detailed task plan development, staffing management plan, and communications.

Q. What sort of processes, procedures, and risk mitigation measures did PEF use for project approvals?

A. At the corporate level, projects of this magnitude receive approval by executive management via a phased project authorization process. This process is defined in Company procedure ACT-SUBS-00261. At the time the SGR project was initiated, this procedure required a three-phase approach for major projects consisting of a "Study Phase", a "Design Phase," and an "Implementation Phase." A Business Analysis Package ("BAP") was used to describe and provide justification for the project. A Project Authorization Form ("PAF") was the vehicle used to formalize management approval of the project. Approval began at the "Study" phase and further management reviews and approvals were required before a project could advance to the Design Phase or Implementation Phase. The procedure also contained provisions for additional approvals between phases if cost estimates exceeded certain thresholds. During the course of the SGR project, this process evolved to a process that was based on an Integrated Project Plan ("IPP"), which was an enhanced version of the BAP. This revised process was detailed in procedure ADM-SUBS-00080. The phased approval approach was changed to provide for Executive review and approval of the project at specified project milestones and decision points as defined in the IPP. The transition to the new procedure was completed with the approval of the first IPP for the SGR project in March 2008. This natural evolution of the process provided for even closer executive management oversight of a major project like the SGR project.

Q. What sort of processes, procedures, and risk mitigation measures did PEF use for project controls?

A. The Project Management Manual previously discussed (NGGM-PM-0018) outlined specific project management and control requirements, including the development and maintenance of a Risk Management Plan. Project budgets and schedules were developed and tracked in monthly reports. Project site performance was tracked by key performance indicators ("KPI"). Project activities were monitored by schedules and graphs. A self-assessment plan was developed and put in place in January 2005. Self-assessment team members were selected and tasked to prepare self assessments for the purpose of determining whether: (1) SGR project controls were adequate; (2)

corrective actions were effective; (3) activity timing was correct; (4) progress reporting was accurate; and (5) SGR project activities were in agreement with the project plan. The four cornerstones of self-evaluations were used for program compliance and commitment: (1) corrective action program; (2) self-assessment, (3) benchmarking; and (4) operating experience. The self-assessments were based on direct observations, document reviews, and personnel interviews. The self-assessment reports outlined weaknesses, identified items for management consideration, listed acceptable areas, identified key personnel contacted for the assessment, listed consulting references, and set forth follow-up items. In addition to the above, project status was updated and reported via periodic project management meetings.

Q. What sort of processes, procedures, and risk mitigation measures did PEF use for project management?

A. The project was staffed by experienced and qualified personnel. Selected members of the SGR project team completed an Advanced Project Management training program to enhance their overall knowledge of project management principles and best practices. Several members of the project staff were certified as Project Management Professionals by the Project Management Institute. Certification to Progress Energy Standards for Management of Major Projects was also completed.

The Company's project management policies and procedures provide for the management of outside contractors and vendors to ensure that the work performed by the contractors and vendors is consistent with the contract, of sufficient quality to comply with PEF and industry standards, and the costs are reasonable and necessary to perform the contract work. These management policies and procedures with respect to contractors and vendors were employed on the SGR project. These policies and procedures involved quality assessments of contractor and vendor work on the SGR project.

In addition to a well qualified staff, oversight of the project was maintained via periodic project reports (financial and schedule) and management level project review meetings.

Q. What sort of processes, procedures, and risk mitigation measures did PEF use for project execution?

A. Again, the primary guidance came from the Project Management Manual for management of the project. Other guidance was provided through various plant procedures such as Engineering Change Work Management (EGR-NGGC-0009) and Engineering Product Quality (EGR-NGGC-0011). The various aspects of the project were detailed in formal Engineering Change Packages ("ECs"), which received detailed formal review and approval, including in some cases independent third party reviews. Once ECs were approved and released for execution, the requirements of the ECs were translated to a detailed Work Order, which formed the instructions and controls by which craft personnel would implement the work.

All of the work was subject to the Company's Quality Assurance and Nuclear Oversight programs, which provided an independent determination that the work was being done in accordance with the established procedures and the strict quality requirements necessary for work in a nuclear plant.

An Action Item Management System ("AIMS") database was launched in 2005 to provide individuals a mechanism to document actions, concerns, risks, or other important decisions. The shared database stored action items for SGR project weekly staff meetings. Assignees were allowed to update and complete tasks, but tasks could only be closed at management direction. A Risk Management process was also incorporated in the AIMS database.

Q. What sort of processes, procedures, and risk mitigation measures did PEF use for project review?

A. I have previously described the process that was used for approval of a major project at various stages and the use of periodic reports and project review meetings to provide status updates on the project. In addition, the CR3 plant's annual business plan and budget reviews provided an opportunity to ensure that a project was proceeding as expected and meeting its established project milestones.

Also, project scope activities and cost estimates for each activity were developed by the SGR Project team. Project scope activity proposals were evaluated by an Executive Oversight Subcommittee. Proposals were either approved or turned back to the project team for justification and further evaluation. Decisions were then documented in monthly reports.

Q. Did PEF benchmark against other SGR projects for lessons learned and best practices prior to undertaking the CR3 SGR project?

A. Yes, as I testified earlier, PEF extensively benchmarked against other utility experiences. PEF identified twenty-six plants that had performed steam generator or reactor head replacement in post-tensioned containment buildings similar to CR3. Thirteen (13) of those plants created a construction opening for the replacement project. With this information, PEF endeavored to do two things. First, PEF set out to identify the plants most like CR3 so that PEF could get the most "apples-to-apples" comparison for benchmarking purposes. Next, where plants both similar and unlike CR3 had taken actions that were different than those planned for CR3's SGR project, PEF sought to understand why those actions were taken and whether or not PEF should modify or change any of its SGR project plans.

For instance, in addition to the benchmarked projects I referenced earlier in my testimony, in March and April 2007, SGR Project team members traveled to Comanche Peak Unit 1 in Somervelle County, Texas. Comanche Peak was undergoing a SGR at that time and was chosen because many of the same project aspects and outside vendors applicable to Comanche Peak were applicable to the CR3 SGR project, including Bechtel who was managing the Comanche Peak steam generator replacement. The benchmarking team reviewed implementation of the document review process, risk management process, budgeting, scheduling, and reporting processes. Personnel were also made available to discuss containment challenges and requirements along with other miscellaneous topics. The CR3 project team captured best practices, perspectives, and lessons learned that related to CR3 operations and operations training.

Also, in November and December, 2007, the SGR team visited FP&L's St. Lucie Plant. The St. Lucie Plant was performing its SGR outage during that time. The team assessed security aspects as well as quality oversight and control processes. From this assessment, they identified recommendations to be implemented at CR3 and developed quality processes for the CR3 SGR Project implementation. In October 2006, SGR Project team members visited Omaha Public Power District's Fort Calhoun Station to observe replacement steam generator movement, fit-up and welding of large bore reactor coolant system piping. This was one of the largest scope outages in commercial nuclear power history. They were about 30 days into the outage and running about a week and a half ahead of schedule due to extensive planning and preparation. Observations, streamlined processes, and comments were documented and discussed with the CR3 SGR Project team.

In Spring 2005, Turkey Point performed a Reactor Vessel Head Replacement that required use of a construction opening through its containment building. Turkey Point had problems with liner plate buckling during hydro-demolition. Despite these problems, the liner plate was fixed with minimal impact to outage schedule. PEF obtained information on what had caused this incident to happen and incorporated that information into its hydro-demolition operating procedures. These are just some of the examples of PEF's benchmarking efforts against other utilities on the SGR project. Also, as I further testified to earlier, to gain additional knowledge about other steam generator replacement projects, PEF participated in annual steam generator replacement team roundtables hosted by Black Diamond Services. PEF's SGR Project team stayed abreast of all comparable utility projects and industry expertise for the duration of the SGR project, incorporating lessons learned and best practices as they became available for use on the SGR project.

Q. What was the plan to remove and replace the steam generators and how was the plan developed?

A. Members of Progress Energy's Nuclear Projects and Construction Group developed an Integrated Project Plan ("IPP") that set forth detailed information about the SGR project including funding requirements, schedules, economic evaluation, risk management, safety and environmental plans, and various other details. The plan was shared through presentations before project authorization stakeholders and senior management. Approval by all sixteen stakeholders was required before moving forward with the project plan. The scope of the IPP was to create a reactor building opening to establish a transport path for the existing OTSGs, severing the attached piping, rigging the existing OTSGs out of the reactor building, transporting them to an on-site storage facility, transporting and rigging the new OTSGs into the reactor building, welding of attached piping, and closure of the reactor building opening. The scope of the project also included design and engineering, procurement and fabrication oversight of replacement components, planning and implementation of the work scope, as well as mobilization and demobilization of the workforce required to support all aspects of the project. The IPP containing this scope was presented and executed by senior management in March 2008.

Subcontractors were selected and contracts were processed for large bore piping metrology, cutting, machining & welding (AREVA); heavy lifting and transport into containment (Mammoet); construction management (Bechtel); engineering specialty services (Sargent & Lundy); containment hydro-demolition (Mac & Mac); containment tendon support services (PSC); insulation and insulation support services (Transco); and containment liner plate removal and installation (CBI-Nuclear). Two contracts had been executed for fabrication and delivery of replacement OTSGs (B&W – Canada) and OTSG tube fabrication and delivery (Sumitomo Metals, Inc.). Other miscellaneous components such as hot leg nozzles, hot leg piping, and insulation were ordered under purchase orders.

The IPP was revised in March 2009 and again in September 2009 to reflect firm contracts and contract prices. Total project cost expectations were also adjusted to $314M including a $4M contingency fund including sunk costs. Copies of the aforementioned IPPs are included as Exhibit No. __ (JF-33) to my testimony.

Q. Were the SGR Project policies, procedures, risk mitigation measures, and other measures that you have described unique to the SGR Project?

A. No. PEF implemented its project management and cost control oversight mechanisms for the SGR project consistent with its Project Management Manual and Progress Energy Project Governance Policy. Likewise, because the SGR project is a major capital project for PEF, the project had to comply with the Company's policies and procedures in its Major Capital Projects – Integrated Project Plan. These project management policies and procedures are the same ones employed on other major PEF capital projects -- like the Levy Nuclear Project and the CR3 Uprate project -- that have been reviewed and determined to be reasonable and prudent project management, contracting, and controls by the Commission in the 2009 and 2010 nuclear cost recovery clause dockets.

Q. Did PEF prudently manage the SGR Project work from a process and procedure standpoint?

A. Yes. The project plan for removal and replacement of the steam generators set forth in the SGR Project IPPs was implemented as planned. The existing steam generators were successfully removed and the new steam generators were successfully installed. The delamination in the containment wall did extend the SGR project schedule and cost, but as I have explained above, this delamination was an unprecedented and unpredictable event that was beyond PEF's control.

VI. CONCLUSION.

Q. Were PEF's decisions and actions prudent on the SGR project?

A. Yes. PEF prudently managed the SGR project throughout the project. The delamination events at CR3 do not mean that PEF was imprudent. PEF employed experienced engineers and contractors on the SGR project and applied lessons learned from benchmarking other SGR projects around the world, including very similar projects where a construction opening through a pre-stressed, post-tensioned nuclear containment wall was used to remove and replace steam generators. PEF further employed industry standard engineering analyses and construction methods to plan for and implement the replacement of the CR3 steam generators through the opening cut in the CR3 containment wall. In all respects, PEF's management of the SGR project was consistent with industry leading project management decisions and practices for projects like the SGR project. In fact, had PEF decided to use the lower equipment hatch to transport the OTSGs it would have represented the only such project in the industry to attempt to accomplish the task in this manner.

There were no indication that delamination was a risk on the SGR project much less that it would in fact occur. This was demonstrated by the enhancements in the standard, industry-leading engineering modeling analyses and calculations at the time to determine the causes of the CR3 delamination and replicate them in the analyses. The programmatic root cause of the initial delamination in the CR3 containment wall was the inability of industry standard engineering analyses and calculations to predict the delamination. Simply put, the initial delamination in the CR3 containment wall was an unprecedented and unpredictable industry event that was beyond PEF's control.

Q. Does this conclude your testimony?

A. Yes.

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