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Florida Power
CORPORATION

BEFORE THE FLORIDA PUBLIC SERVICE COMMISSION

DOCKET NO. 970261-EI

In Re: Review of Nuclear Outage
at Florida Power Corporation's
Crystal River Unit No. 3

DIRECT TESTIMONY OF

PAUL F. McKEE

For Filing April 14, 1997

FLORIDA POWER CORPORATION
DOCKET NO. 970261-EI

DIRECT TESTIMONY OF
PAUL F. MCKEE

I. BACKGROUND

Q. WHAT IS YOUR NAME, ADDRESS, AND OCCUPATION?

A. I am Paul F. McKee. My business address is 15760 West Powerline Street, Crystal River, Florida. I am employed by Florida Power Corporation ("Florida Power" or "the Company") as the Manager of Plant Operations Support at the Crystal River Unit 3 nuclear power plant ("CR-3").

Q. WHAT ARE YOUR DUTIES AND RESPONSIBILITIES AS MANAGER OF PLANT OPERATIONS SUPPORT AT CR-3 AND HOW LONG HAVE YOU HELD THAT POSITION?

A. I am responsible for coordinating and supporting interface functions at CR-3 between Systems Engineering, Design Engineering, and Operations. Systems Engineering is responsible for the maintenance of 123 systems at the plant; Design Engineering is responsible for designing and planning modifications to the safety, electrical, mechanical, structural, instrumentation, and other systems at the plant; and Operations has the responsibility for the day-to-day operation of CR-3. I have held this position since October 1996.

Q. PLEASE DESCRIBE OTHER POSITIONS THAT YOU HAVE HELD AT FLORIDA POWER AND THE DATES THAT YOU HAVE HELD THEM.

A. From September 1992 through September 1996, I served as Director of Quality Programs at CR-3. In that capacity, I was responsible for conducting regular Quality Control Inspections at the plant of all physical work (including modifications), all equipment, and all operations. We conducted these inspections using personnel who were independent from the work or functions being inspected. I was also responsible for our Quality Auditing program. In this connection, I oversaw a diverse group of about twenty auditors -- including engineers -- who were responsible for conducting reviews to determine how all programs at CR-3 were operating, including, for example, the work of design engineers associated with plant modifications.

I served as Plant Manager of CR-3 from August 1983 through September 1992. In this position, I was responsible for the day-to-day operation and maintenance of the plant. I participated in developing and implementing the plant's strategy for meeting all applicable safety requirements and for operating the plant in a prudent and cost-effective manner. In this connection, I reviewed and approved all safety-related modifications to the plant's equipment and systems.

I served as the Operations Manager of CR-3 from 1979 until August 1983. In this capacity, I was responsible for the day-to-day operation of the plant. At the same time, I sat on (and occasionally chaired) the plant's Plant Review Committee (sometimes referred to as the "PRC"). The Plant Review Committee conducted reviews of modifications carried out at the plant in order to ensure that the modifications were performed in accordance with applicable regulatory procedures and plant policies and to ensure that the modifications met applicable safety requirements.

I served as Technical Support Manager of CR-3 in 1978. In this capacity, I acted as the manager of the systems engineers and was responsible for the engineering support of plant operations.

I held the position of Assistant Plant Manager at CR-3 from 1974 through 1978, as the plant prepared to operate and then started up. I was responsible for operations, maintenance, and licensed operator training.

From 1968 through 1974, I served as the Operations Manager of CR-3 and was involved in the plant design, concept review, operator training, and the training of personnel to obtain a license from the Atomic Energy Commission (later succeeded by the Nuclear Regulatory Commission or "NRC").

Prior to working at CR-3, I worked as a plant engineer at Florida Power's Turner power plant from 1966 through 1968. From 1965 through September 1966, I worked as a Results Engineer at Florida Power, travelling throughout the system, testing power plants.

Q. WHAT IS YOUR FORMAL EDUCATION?

A. I received a Bachelor's degree in Mechanical Engineering from the University of Florida in 1964, after studying math and physics at Hobart College. I received a Bachelor's degree in Nuclear Engineering Science from the University of Florida in 1969.

II. <u>PURPOSE AND SUMMARY OF TESTIMONY</u>

Q. WHAT IS THE PURPOSE OF YOUR TESTIMONY?

A. I will explain the background for the current shutdown of CR-3 and the reasons why it is reasonably necessary to incur the costs associated with that shutdown. Because the plant was taken off line to perform modifications necessary to restore compliance with Florida Power's operating license for CR-3, it is helpful to understand the basic configuration of CR-3, basic regulatory requirements that the Company must meet in operating the plant, and the process by which modifications are made. Accordingly, I describe these matters in my testimony. Against that background, I explain historical modifications and regulatory requirements leading up to the Company's decision to take the plant out of service. Finally, I explain the Company's decision to enter and then to extend the current outage, and the reasonable necessity of incurring the expenses of this outage.

Q. WOULD YOU PLEASE SUMMARIZE YOUR TESTIMONY?

A. Yes. In my testimony, I provide a general description of the configuration and operation of CR-3. To summarize, CR-3 is a pressurized water reactor that produces heat through a process of fission. That heated, pressurized water is carried to a pair of steam generators where the heat is used to boil water and to produce steam. The steam is used to drive turbines, which in turn generate electricity.

The plant was designed and built with certain safety squipment and systems called the "Engineered Safeguards system," which has as its fundamental purpose maintaining the temperature of the fuel below a certain specified limit. This is necessary to ensure that the fuel does not cause damage to the facility and, in turn, to ensure that radioactive material does not escape from the core.

In order to obtain its license to operate CR-3, Florida Power had to demonstrate to the Atomic Energy Commission ("AEC," now the Nuclear Regulatory Commission or "NRC") that the Company had equipment, systems, policies, and procedures in place that were adequate to provide a reasonable assurance of safety. This showing was incorporated in a document called a "Final Safety Analysis Report" or "FSAR" and in Technical Specifications for the plant, which include design limitations for equipment, testing and surveillance requirements, and other information relevant to how the plant will cope with possible emergency situations. The AEC reviewed and approved the Company's Final Safety Analysis Report and Technical Specifications in issuing CR-3's operating license. The operating license incorporates the plant's Technical Specifications. The Company is required to operate the plant in accordance with the operating license.

CR-3 is currently in an extended shut down so that we may make modifications that are necessary to restore compliance with the license for the plant. These modifications arise out of requirements that were imposed on the industry after CR-3 was built to treat the auxiliary feedwater system of a plant as a "safety-related system," now called the "Emergency Feedwater system" (or "EFW" system). The Emergency Feedwater system is used to carry water to the steam generators. The

NRC first required that this system be deemed safety-related after the well-publicized accident at Three Mile Island in 1979, which implicated the need for this system to cope with certain safety risks. This resulted in the adoption of a series of regulatory requirements and efforts by the industry to retrofit plants ever since.

Florida Power made a number of modifications over the years to respond to these requirements. The Company follows detailed procedures in connection with safety-related plant modifications. This procedure is mandated, in part, by NRC regulations. The procedure includes a careful process of evaluation of modifications by design engineers to determine potential safety issues, verification, management approval, selected reviews by a plant committee and independent review committee, and documentation of modifications.

With the benefit of input from industry groups and its plant designer, Babcock & Wilcox ("B&W," now Framatome Technologies), the Company analyzed the configuration of CR-3 and developed plans to modify the plant to conform to the NRC's post-Three Mile Island requirements. Following the procedures outlined above, and with the concurrence of the NRC, the Company upgraded its auxiliary feedwater system to a safety-related system. The Emergency Feedwater system includes an electric-powered pump that requires an emergency power source for the system in the event of a loss of the normally available offsite power. Although the electric-powered feedwater pump had previously been provided with an emergency power source -- namely, Emergency Diesel Generators -- to supply electricity in the event of a loss of offsite power, before 1980 this source had to be actuated manually by plant operators. After 1979, the Company was required to ensure that an Emergency Diesel Generator would be actuated automatically to power the Emergency Feedwater system.

Over the years, the Company made a number of modifications to the plant and to its control systems and procedures to accommodate this mandatory load, on the Emergency Diesel Generator. Among other things, in 1987, the Company sought to reduce the hydraulic load of the electric-powered Emergency Feedwater pump by providing for the simultaneous operation, in the event of an emergency, by the second, steam-driven pump that did not require back up by the Emergency Diesel Generators.

The Company was also required to engage in hypothetical emergency planning regarding the performance of the Emergency Feedwater pumps themselves. In particular, the Company took steps to guard against malfunction by the pumps in extreme, hypothetical emergency operating conditions.

Each plant made by the same manufacturer pursued other solutions that the plant deemed reasonable and suitable for its particular configuration. Certain of these plants encountered difficulties or significant costs in making these modifications. Florida Power made the judgment at the time to pursue what appeared to be suitable, cost-effective solutions for CR-3.

During the refueling outage commenced in February 1996, CR-3 engineers added more accurate, state-of-the-art instrumentation to the Emergency Diesel Generators. In the course of analyzing this modification, the engineers became concerned that the 1987 modification providing for simultaneous operation of both Emergency Feedwater pumps might result in a malfunction caused by a problem called "cavitation" in a certain hypothetical situation. "Cavitation" occurs when there is an excessive flow of feedwater through the pumps. Accordingly, the Company determined to reverse the 1987 modification based on the fact that the Emergency Diesel Generators had been upgraded, and the manufacturer of the diesels assured the Company that the generators could support the electric-powered Emergency Feedwater pump operating alone (without the assistance of the steam-driven pump). With the benefit of this solution to the recently identified design problem, we were able to restart the plant and end the refueling outage.

In September 1996, a lube oil pipe in the main turbine unexpectedly ruptured. The Company had to shut down the plant to diagnose and remedy this problem. The pipe ruptured due to vibration in the operation of the turbine combined with latent imperfections in the pipe. The pipe was not observable while the plant was in operation.

In the meantime, CR-3 engineers and the NRC came to focus on a concern that the 1996 modification to the Emergency Feedwater system had introduced inadvertently other safety issues that the Company and the NRC had not fully considered. Upon further analysis, it became apparent that the Company would have to make extensive modifications to the plant -- particularly concerning the Emergency Diesel Generators and the Emergency Feedwater system -- to restore compliance with the conditions of the CR-3 operating license.

Once this situation was discovered, the Company had to redress it within 72 hours or keep the plant shut down. Because the Company could not restore compliance within that time period, the Company determined to enter an extended outage to permit a thorough investigation and the completion of necessary modifications.

Even if this discovery had been made sooner, however, the same kind of outage would have been required to deal with the situation. This situation involved a recognition that the Company was out of compliance with conditions of its license (the ability to respond to certain remote, hypothetical emergency situations) and, upon discovery, the Company had to redress the problem or shut the plant down in 72 hours. We had exhausted every practicable alternative to making the extensive modifications that now must be made. There was no alternative to entering such an extended shutdown.

In addition, if the need for these modifications had been recognized earlier, the necessary work could not have been done within the timeframe of any prior outage. The modifications are too extensive, and the Company was fully occupied with other exigencies during prior outages. Accordingly, the current outage could not have been reasonably avoided.

III. <u>DESCRIPTION OF PLANT DESIGN AND LICENSING BASIS</u>

A. <u>DESCRIPTION OF CR-3 AND ENGINEERED SAFEGUARDS SYSTEM</u>

Q. SO THAT WE MAY UNDERSTAND THE CONTEXT OF THE MODIFICATIONS BEING MADE DURING THE CURRENT OUTAGE, WILL YOU DESCRIBE THE BASIC CONFIGURATION AND OPERATION OF CR-3?

A. Yes. CR-3 is a pressurized water reactor. The reactor fuel, through fission, heats pressurized water surrounding the fuel. That heated, pressurized water then travels from the reactor through large diameter pipes to tubes housed within two steam generators. (The circulation of water to and from the core is called the "primary system.") The heat given off from the water forced through the steam generator tubes is used to boil feedwater circulating outside the tubes that carry the pressurized water from the reactor. (The circulation of the feedwater through the steam generators is called the "secondary system.") This produces steam, which is used to drive a turbine-generator. The turbine-generator in turn produces electricity. The pressurized water that had been heated by the reactor fuel is cooled by this process and is returned to the reactor core to repeat the cycle.

Likewise, the steam utilized to drive the turbine-generator is condensed into water and recycled through the steam generators.

Plants like CR-3 have an Engineered Safeguards system that is designed to protect the nuclear reactor core in the event of emergency conditions. The principal safety system is designed to cool the reactor fuel within limits specified in CR-3's license.

To meet the critical objective of cooling the reactor fuel, CR-3 is equipped with two identical (and redundant) systems (which include the High Pressure Injection pump, pressurized core flood tank, and the Low Pressure Injection pump) that stand ready to pump water into the reactor vessel in the event of an accidental loss of coolant. The pumps are normally powered by electricity generated outside CR-3 (for example, by other Florida Power generating plants), referred to as "offsite" power. Florida Power is required to maintain a backup power source for CR-3's safety system. Accordingly, each set of pumps is backed up by an Emergency Diesel Generator that can generate electricity in the event that offsite power is interrupted. Please see PFM Ex. 1 for a basic schematic diagram of CR-3.

Babcock & Wilcox (now known as Framatome Technologies Inc.) designed the nuclear steam supply system of CR-3 in the late 1960s and early 1970s. The Engineered Safeguards system of the plant is provided to cope with a set of hypothetical "design basis" accidents. A design basis accident is a chain of events prescribed by the NRC that a plant must be specifically engineered to withstand. At the time that CR-3 was built, the most severe accident that engineers anticipated was an accidental break of large coolant pipe (e.g., a 36-inch pipe) -- resulting in a significant loss of coolant -- combined with a loss of offsite power and the active failure of any other single piece of equipment. Plants like CR-3 were designed to cope with this combination of events. This type of safety planning is called "single failure-proof" design, and the particular situation described is referred to as a "Large Break Loss of Coolant

Accident." At the time CR-3 was built, engineers and the NRC believed that if a plant were capable of dealing both with this situation and with a one inch instrument line break, it could deal with all sizes of loss of coolant accidents in between the two bounding analyses.

B. EXPLANATION OF CR-3 LICENSE REQUIREMENTS

Q. CAN YOU SUMMARIZE THE REGULATORY PROCESS ASSOCIATED WITH THE INITIAL LICENSING OF CR-3 BY THE NRC?

A. Yes. CR-3 was subjected to a rigorous licensing process prior to beginning operation in 1977. First, a multi-volume application for a construction permit was prepared and submitted for review. The NRC Staff examined the application to determine whether the CR-3 design was safe and consistent with NRC rules and regulations; whether valid methods of calculation were employed and accurately carried out; and whether analyses and evaluations were of sufficient depth and breadth to support NRC approval with respect to safety. When the NRC Staff was satisfied that acceptance criteria were met, it prepared a Safety Evaluation Report ("SER") summarizing the results of the NRC Staff review.

Following publication of the Safety Evaluation Report, a review was performed by the independent, statutory Advisory Committee on Reactor Safeguards. The Advisory Committee then prepared a report presenting the results of its evaluation and recommending that a construction permit be granted. A public hearing was then held. At the completion of the hearing, a construction permit was issued.

Prior to the completion of construction, a second application was prepared and filed with the NRC, this time for an operating license. As a part of this application, Florida Power conducted and submitted a Final Safety Analysis Report (or "FSAR"), which described the operating characteristics of the plant equipment and how Florida Power proposed to meet all applicable safety requirements, including testing and surveillance procedures. Among other requirements, Florida Power is required by NRC regulations to ensure that the temperature of CR-3's reactor fuel does not exceed an established limit, which is set to provide a margin of safety against damage to the fuel and a possible exposure to the public of radioactive materials. Florida Power was required to demonstrate in its application how the Company proposed to fulfill this safety requirement.

In this connection, Florida Power had to demonstrate in its Final Safety Analysis Report that the Company could satisfactorily cope with all "design basis" accidents. The Final Safety Analysis Report presents numerous examples of hypothetical, design basis accidents that might result in damage to a reactor core. But the NRC does not prescribe how any particular plant must deal with such accidents. Every licensee must demonstrate that it is able to respond to the occurrence of all design basis accidents.

As I described above, in this connection, Florida Power was required to demonstrate that it could satisfactorily respond to a hypothetical design basis accident (e.g., a large break loss of coolant accident) at the same time that it might incur one condition failure (e.g., a loss of offsite power) and one "single" failure (e.g., the failure of a High Pressure Injection pump).

In licensing CR-3, the NRC followed a process similar to that employed for approving the construction permit. Specifically, the NRC Staff and Advisory Committee on Reactor Safeguards reviewed Florida Power's submission. In the course of licensing and regulating a licensee, the NRC may raise questions about the licensee's Final Safety Analysis Report, and the licensee must respond until the NRC is satisfied. The NRC subsequently prepared a Safety Evaluation Report, approving the Company's submission. Following successful completion of the review, the NRC issued an operating license.

The license incorporated the representations that Florida Power had made about the operating characteristics of the plant. The license incorporates two (three-inch thick) volumes of Technical Specifications for CR-3, which describe, among other things, safety limits, the characteristics of its Engineered Safeguards system, physical operating limits established for safety equipment, testing and surveillance requirements, and administrative controls.

In order for a piece of equipment or system to be considered operable according to the Technical Specifications, it must not only be capable of performing its intended function, but all necessary instrumentation, controls, electrical power, cooling or sealing water, lubrication, or other auxiliary equipment required for the equipment or system to perform its function must also be capable of performing their related support functions. Therefore, a piece of equipment or system can be functioning (i.e., a pump is pumping as required) when it is determined to be inoperable according to the Technical Specifications because of a problem in one of the supporting systems. The problem can consist of a degradation of equipment or system condition, but it can also be simply new questions about the conformance of the equipment to codes and standards or the discovery of an existing but previously unanalyzed condition or postulated accident scenario.

C. DESCRIPTION OF PLANT MODIFICATION PROCESS AND OVERSIGHT

Q. PLEASE DESCRIBE THE PROCEDURES THAT FLORIDA POWER FOLLOWS IN MAKING MODIFICATIONS TO ITS ENGINEERED SAFEGUARDS SYSTEM?

A. NRC regulations -- codified at 10 C.F.R. § 50.59 -- specify the requirements that must be followed to modify a safety-related system at a nuclear power plant. These procedures have been in force from the inception of the operation of CR-3. These procedures require that the licensee evaluate the modification against a set of specific criteria that address the requirements of the plant's Technical Specifications and Final Safety Analysis Report. As I have described, the Final Safety Analysis Report documents the results of the engineering analyses of the plant, which have been reviewed and approved by the NRC as a condition of granting the license.

As I have also described, the Technical Specifications for CR-3 comprise detailed specifications for how various items of equipment and various systems at CR-3 must function, including, among other things, surveillance procedures for testing equipment and systems. For example, the Technical Specifications pertaining to the Emergency Feedwater system require, among other things, that plant operators verify (usually through "walk downs") that there are no leaks and that temperature, pressures, and valve positions meet specifications. Operators must start and operate the Emergency Feedwater system every 45 days to ensure that all systems and equipment function properly.

Before any modification is made, the licensee must specifically analyze and respond to three questions: (1) could the modification increase the probability of occurrence or consequences of a malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report? (2) could the modification create a possibility for an accident or malfunction of a diffe ent type than any evaluated previously in the Final Safety Analysis Report? and (3) will the modification cause a reduction in the margin of safety as defined in the basis for any of the plant's Technical Specifications? If the answer to any of these questions is "yes," the modification will create an "Unreviewed Safety Question" or "USQ," namely, a condition that the NRC did not review and accept in approving the Company's operating license. In such circumstances, the licensee must obtain approval from the NRC before making the modification.

At Florida Power, the analysis required under Section 50.59 (sometimes called a "50.59" analysis) is conducted in the first instance by Design Engineers. The Company maintains its own Design Engineering group, which has consisted of approximately 50 design engineers from 1979 through 1996. These engineers are divided into various units, including, among others, electrical, mechanical, structural, instrumentation, and safety analysis. Each group has a manager or supervisor.

In making a modification, the Design Engineers assigned to the systems that will be affected by the change prepare conceptual designs, in consultation with other plant groups. Alternatives to the design and the assumptions underlying the design are considered and discussed. (On occasion, the Company has engaged an Architect-Engineer, Gilbert Commonwealth Inc. (now Parson's Power), to design plant modifications.)

Once the concept for a modification is developed, the Design Engineers must prepare a Modification Approval Record (or "MAR"). Florida Power has specific written procedures that govern the documentation of the Modification Approval Record. Under these procedures, Design Engineers must review plant drawings and procedures to ensure that any proposed modification is compatible with other equipment and systems. Further, as part of the documentation process, the Design Engineer must specifically address in writing the criteria set forth in 10 C.F.R. § 50.59.

In order to conduct the 50.59 analysis, the Design Engineers are required to review the plant's Final Safety Analysis Report and to consider specifically the impact of any proposed modification on the plant's Technical Specifications. As I have explained, in conducting the 50.59 analysis, Design Engineers must also consider whether the proposed modification may otherwise increase the risk of any accident occurring, the consequences of any accident, or the possibility of a malfunction of equipment.

Once a modification is approved and implemented, the plant's design records must be modified to reflect the modification made.

A Modification Approval Record may not be "closed out" until required steps are documented. The Company has procedures in place to monitor the completion of the various documentation requirements.

Although these procedures have remained in force throughout the time the plant has been in operation, the procedures have been improved over time, as the nuclear industry has become more sophisticated in its evaluation, implementation, and documentation of plant modifications. In 1992-1993, Florida Power initiated a Business Process Improvement ("BPI") study of the Modification Approval Record process. As a result of this study, the Company implemented a process called the Plant Equipment Equivalency Record, to account for the fact that the Company is forced at times to purchase equivalent replacement parts for old equipment that is no longer manufactured. The Company also determined to exclude from the Modification Approval Record process less significant, non-safety related modifications, in order to make the process more efficient and effective.

At Florida Power, once the Design Engineers have completed the Modification Approval Record process, the proposed modification with supporting documentation is reviewed independently by a Verification Engineer.

If the Verification Engineer approves the proposed modification, it must be reviewed and approved by the Supervisor for the relevant Design Engineering group or groups involved in developing the design. Again, the Supervisor must review the modification package independently to determine whether the modification is appropriate.

Next, the proposed modification must be reviewed and approved by a plant committee comprising persons who are independent from the persons whose work is being reviewed. This committee is called the Plant Review Committee. NRC regulations require the establishment of this committee, govern the qualifications and expertise of its members, and prescribe how it must operate. The members consist of top level managers from CR-3, including managers from Operations, Maintenance, Health Physics, Quality, and Engineering.

The Plant Review Committee is charged with the responsibility, among others, to ensure that a proposed modification meets applicable NRC safety requirements. Finally, if the Plant Review Committee approves the modification, the Plant Manager at CR-3 will conduct a final review and must sign off on any safety-related modification.

Q. APART FROM WHAT YOU HAVE DESCRIBED, DOES FLORIDA POWER HAVE ANY OTHER ASSURANCE THAT MODIFICATIONS OF SAFETY-RELATED EQUIPMENT AND SYSTEMS ARE MADE IN AN APPROPRIATE MANNER?

A. Yes, it does. As I have described, in connection with modifications of safety-related equipment or systems, the Company completes in writing an analysis of the NRC's safety requirements prescribed in 10 C.F.R. § 50.59. This analysis is not only checked and cross-checked within the Company itself, but is made available to the NRC as well. The Company provides 50.59 reviews to the NRC within six months of each refueling outage. In addition, the Company must provide the NRC with an update of its Final Safety Analysis Report every two years, which includes changes since the last update by the Company.

Further, the Company maintains two plant review committees, as required by NRC regulations. I previously described one (specifically called the "Plant Review Committee"). The second committee is called the Nuclear General Review Committee (or "NGRC") and consists of twelve members, over half of whom are not Florida Power employees. The non-Florida Power members typically have been former executives from other nuclear power plants. The outside members have provided an additional source of information about industry experience. The remaining members come from various areas within CR-3, including the Licensing, Training, Operations, and Health Physics units. I served as Chairman of the Nuclear General Review Committee from 1992 through 1996.

The Nuclear General Review Committee periodically reviews the operations at CR-3 for safety compliance. Among other things, the committee reviews selected safety-related modifications. The committee evaluates the technical adequacy of modifications based on the past experience of other plants and the committee members' understanding of nuclear operations and systems. Committee members call upon their own diverse backgrounds in assessing whether hypothetical accident situations may arise that the Company may have failed to address, whether modifications are consistent with the plant's Final Safety Analysis Report and Technical Specifications, and whether the 50.59 review has been documented in good fashion.

In addition, over the years, the Company has implemented programs to improve and enhance the integrity of its modification documentation process. In the mid-1980's, Florida Power developed an Enhanced Design Basis Document ("EDBD"), consolidating key design basis information. The Company completed a five-year program, at an expense of $5 million, with the assistance of Babcock & Wilcox and Gilbert Commonwealth, Inc. to collect design basis information and to improve its accessibility.

In the same time frame, the Company established the Configuration Management Improvement Program, which generated computer programs to facilitate locating and collecting safety-related documents. In certain situations where information was incomplete, the Company recreated the design basis of the equipment. At the same time, the Company evaluated the equipment at the plant to determine whether it continued to meet safety requirements. In some instances, modifications were made to correct inadvertent deviations from safety standards.

IV. <u>REGULATORY DEVELOPMENTS AND HISTORICAL MODIFICATIONS TO CR-3'S ENGINEERED SAFEGUARDS SYSTEM</u>

A. <u>IMPACT OF THE THREE MILE ISLAND ACCIDENT</u>

Q. WHAT IS THE BACKGROUND TO THE MODIFICATIONS NOW BEING MADE AT CR-3?

A. The modifications that are being made during the current outage have their genesis in a series of regulatory changes that occurred in the aftermath of the accident at the Three Mile Island Unit 2 nuclear power plant in 1979. This Three Mile Island nuclear unit suffered partial damage to its fuel core during that accident. Since that time, Florida Power and other licensees responsible for operating nuclear power plants that were designed and licensed before the Three Mile Island accident have been engaged in making modifications to their plant equipment and systems to help prevent problems similar to that experienced at Three Mile Island.

Q. HOW DID THE ACCIDENT AT THREE MILE ISLAND AFFECT THE STRATEGY OF THE NRC AND THE NUCLEAR POWER INDUSTRY IN SAFETY PLANNING?

A. The accident at Three Mile Island Unit 2 started with a leak in the primary coolant system of significantly smaller size than 36 inches. This situation is often referred to as a "Small Break Loss of Coolant Accident." This accident changed the way the industry and the NRC look at nuclear safety issues. Specifically, the Three Mile Island accident highlighted the need to rely for safety purposes not only on the High Pressure Injection pump, Low Pressure Injection pump, and pressurized core flood tank that are designed to pump water directly into the reactor vessel (which contains the reactor fuel), but also on a separate set of pumps used to pump water into the two steam generators. The latter pumps are called Emergency Feedwater pumps.

This is so because, although Emergency Feedwater pumps are not needed to mitigate against a Large Break Loss of Coolant Accident, the Emergency Feedwater system is critical to coping with a Small Break Loss of Coolant Accident. Specifically, in the event of a Large Break Loss of Coolant Accident, large amounts of water would be pumped into the reactor vessel to cool the reactor fuel within licensing limits. The Emergency Feedwater system, which removes heat from the steam generators in certain emergency situations, would not come into play because the steam generators would not be functioning during a Large Break Loss of Coolant Accident. In the event of a Small Break Loss of Coolant Accident, however, there would be a loss of much less water from the primary system that cools the core, so significantly smaller amounts of cool water would be pumped into the reactor vessel. Therefore, additional cooling would be required through operation of the steam generator. Thus, the Emergency Feedwater pumps may be necessary to provide emergency feedwater to the steam generators to assist in removing heat produced by the reactor.

After the Three Mile Island accident, the NRC increased efforts to tighten already rigorous safety standards applicable to nuclear power plants. The NRC issued numerous new regulatory announcements or requirements over the ensuing years. Among other things, the NRC issued

NUREG 0737, requiring nuclear plants to design against the situation of a Small Break Loss of Coolant Accident and, in particular, to treat the Emergency Feedwater system as a vital, "safety-related" system. Though imposing this requirement, the NRC did not prescribe specifically how it was to be carried out. Nor is there any single "right" way to do it.

Because the original Emergency Feedwater components were designed as part of an auxiliary pumping system and were not required to be safety-related, they were not built and designed with the same extraordinary safety margins required of safety-related equipment installed during the construction of CR-3. CR-3 and other plants like it therefore were required to upgrade their auxiliary feedwater systems.

B. <u>POST-THREE MILE ISLAND MODIFICATIONS</u>

Q. HOW DID THE INDUSTRY IN GENERAL, AND FLORIDA POWER IN PARTICULAR, RESPOND TO THIS REQUIREMENT?

A. In the ensuing years, owners of facilities like CR-3 made a number of modifications to their Engineered Safeguards system. In this vein, in 1979, Florida Power and the owners of other Babcock & Wilcox plants formed an owners' group to address the regulatory issues arising out of the event at Three Mile Island and to share technological information and operating experience. These companies worked together with information they obtained from Babcock & Wilcox concerning the reliability of B&W plant Emergency Feedwater systems and studied engineering approaches that might enable them to implement new NRC requirements. The individual member companies evaluated this information as part of each company's own assessment of the steps it must take to meet all applicable regulatory requirements.

Further, Florida Power monitored closely reports of problems or NRC action involving other nuclear power plants in order to anticipate problems and areas of regulatory concern before they became implicated at CR-3.

Finally, in discussing and planning appropriate modifications to deal with the NRC directives issued as the result of the accident at Three Mile Island, the Company worked closely with appropriate vendors and consultants, including Coltec Industries, the manufacturer of its Emergency Diesel Generators, and Gilbert Commonwealth Inc., the Company's Architect-Engineers.

Because each nuclear facility was configured somewhat differently from the others, each company faced certain unique challenges in identifying and implementing solutions to the issues raised by the Three Mile Island accident and resulting regulatory requirements. For example, Davis Besse had two steam-driven Emergency Feedwater pumps, whereas CR-3 had one steam-driven pump and one electric-powered pump. As I will explain, the configuration of these pumps at CR-3 played an instrumental role in the modifications that Florida Power made over time to respond to regulatory directives. By way of further contrast, the Oconee nuclear facilities did not have Emergency Diesel Generators, but they used hydro-generators instead.

Following the Three Mile Island accident, Florida Power thoroughly reviewed its own safety equipment and systems to assess the steps that had to be taken to address the concerns that arose from the Three Mile

Island accident and resulting regulatory requirements. These included not only engineering modifications, but a host of system improvements, including the training of plant personnel.

In this connection, between 1979 and the present, a number of plants in the B&W owners' group, and other plants licensed before the Three Mile Island accident, made modifications to their Emergency Diesel Generators and their Emergency Feedwater systems as a means to cope with a Small Break Loss of Coolant Accident (consistent with single failure-proof planning). It is important to appreciate that these modifications were made to preserve or increase safety margins for hypothetical emergency conditions rather than to sustain the normal, safe operation of the plants. Florida Power, in particular, has made various modifications to CR-3's Engineered Safeguards system in order to address emerging regulatory requirements. To understand these modifications, however, it is helpful to appreciate the basic configuration of the Emergency Feedwater equipment that was necessitated by the treatment of the Emergency Feedwater system as a safety-related system.

In order to be prepared for a Large Break Loss of Coolant Accident, CR-3 was constructed with the set of emergency pumps and pressurized tank previously described (the High Pressure Injection pump, the Low Pressure Injection pump, and the core flood tank) that stood ready to supply water directly into the reactor vessel in the event of a rupture of the reactor coolant system. To guard against a malfunction in these pumps, CR-3 was equipped with a second set of pumps and pressurized tank just like the first. These redundant systems are referred to as the

"A" and "B" systems. Both sets of pumps are normally powered by electricity generated from sources outside the nuclear power plant (such as other Florida Power generating plants).

As discussed, in designing its safety system, Florida Power was required to hypothesize that this offsite electric power might be lost during a loss of coolant emergency. To guard against this, CR-3 was equipped with an Emergency Diesel Generator for each set of emergency pumps (the "A" Emergency Diesel Generator and the "B" Emergency Diesel Generator) and other "A" and "B" loads, too.

After 1980, Florida Power was required to treat its Emergency Feedwater pumps also as safety-related and to treat the Emergency Feedwater cooling system as a safety-related system. This meant that Florida Power was required to guard against the possibility of a loss of offsite power to the Emergency Feedwater pumps during a loss of coolant accident when the pumps would be needed. At the time of the Three Mile Island accident, plants like CR-3 were generally equipped with at least two, redundant Emergency Feedwater pumps. One of Florida Power's two Emergency Feedwater pumps (the "B" Emergency Feedwater pump) is powered by a steam turbine; so the possibility of a loss of offsite electrical power would not present a problem to the operation of this pump. The first of the two pumps, however (the "A" Emergency Feedwater pump) is powered by a motor that requires electricity The "A" and "B" Emergency Feedwater pumps are different in order to provide diversity in the design such that a latent manufacturing or design problem would not affect both pumps.

In 1980, with the benefit of and consistent with guidance developed through the B&W owners' group, Florida Power modified its Engineered Safeguards system to ensure that the "A" Emergency Diesel Generator would actuate automatically to provide the "A" Emergency Feedwater pump with a source of back-up power in the event of a loss of offsite power. Prior to that time, operators had to load the "A" Emergency Feedwater pump manually onto the "A" Emergency Diesel Generator. Of course, the same Emergency Diesel Generator (the "A" generator) was expected to actuate automatically to provide back-up electrical power to the "A" emergency equipment used to inject water directly around the reactor fuel. This meant that an increased electrical load would now be imposed upon the "A" Emergency Diesel Generator automatically in the event of certain hypothetical emergency situations, thus reducing the operating margins of the Emergency Diesel Generator.

C. <u>LOADING ON THE EMERGENCY DIESEL GENERATORS</u>

Q. DID FLORIDA POWER TAKE STEPS TO LESSEN THE LOAD ON CR-3'S EMERGENCY DIESEL GENERATORS?

A. Yes. The Technical Specifications for the Emergency Diesel Generators included maximum kW ratings for the equipment for different time intervals. These were relevant to the Company's description of the ability of its equipment and systems to cope with different contingencies. According to the Technical Specifications, at the time the plant was built, the Emergency Diesel Generators should run continuously for no longer than 30 minutes between 3001 and 3300 kW, and at lower kW levels for longer, specified periods of time. To appreciate the significance of this point, it is important to recall that CR-3 has extensive Technical Specifications (filling two volumes) that prescribe the manner in which the plant must be operated. Florida Power is required to meet all Technical Specifications for CR-3 as a condition of its federal license to operate the plant. Therefore, Florida Power had to take steps to ensure that it would not exceed the 30 minute rating for the "A" Emergency Diesel Generator in a hypothetical emergency situation where the "A" Emergency Feedwater pump would be loaded onto that generator.

In order to accomplish this, in 1980, Florida Power installed a kW monitor system that would sound an alarm if the "A" Emergency Diesel Generator entered the 30 minute rating and that would repeat the alarm 25 minutes into the 30 minute rating. The alarms were intended to afford the plant operator an opportunity to manage the load on the Emergency Diesel Generators so as to tailor the operation of electric-powered equipment within the loading limits of the Emergency Diesel Generators. No two emergency situations will be exactly alike, and thus no two emergencies will require exactly the same equipment at the same time. Thus, plant operators would be in a position to disconnect certain equipment in an actual emergency to maintain Emergency Diesel Loading margins.

As a precaution, Florida Power added a circuit that would automatically turn off (or "trip") the "A" Emergency Feedwater pump if it operated in the 30 minute rating for 30 minutes, thus necessitating operator intervention.

Florida Power made this modification after consulting with its Architect-Engineer, Gilbert Commonwealth Inc. and with the manufacturer of the Emergency Diesel Generators, Coltec Industries.

At the time that Florida Power made this modification, the Company believed that it would adequately deal with all known loading constraints on the Emergency Diesel Generators. The Company continued to modify loads, however, on an ongoing basis. It was necessary to recalculate loads every time maintenance work or modifications were made to any piece of equipment that might be powered by the Emergency Diesel Generators, as these actions could theoretically affect loading limits.

Florida Power made additional modifications over a period of years either to manage the load on the Emergency Diesel Generators or to increase the capacity of those generators. In 1987, Florida Power specifically considered replacing its Emergency Diesel Generators with more powerful equipment and ultimately rejected this alternative. This decision was based on research indicating that new Emergency Diesel Generators would have to be placed in a different physical location from its existing generators, that cables would have to be changed out to handle the greater load, and that the modification would entail a major disruption in the configuration of the plant. After careful analysis, Florida Power concluded that it could avoid such a costly modification by making less extensive electrical re-configurations of its own Engineered Safeguards system.

Accordingly, in October 1987, Florida Power undertook to have the "B" Emergency Feedwater pump, which was steam-powered, provide direct and automatic back-up support for the electric powered "A" Emergency Feedwater pump. Florida Power concluded that this would be a cost-effective modification appropriate to the needs of CR-3 that would ensure that adequate operating margins on the Emergency Diesel Generators could be maintained.

This modification was accomplished by arranging to have the valve that operated the "B" pump powered by the "A" Emergency Diesel Generator so that when the "A" system was turned on both the electric powered Emergency Feedwater pump and the steam-powered Emergency Feedwater pump would operate simultaneously. This valve is referred to as "ASV 204." See PFM Ex. 2 (consisting of MAR documentation for temporary and subsequent permanent installation for this modification). Because the steam-powered "B" pump would assist the electric-powered "A" pump in this situation, the work performed by the "A" pump was reduced, which in turn reduced load on the "A" Emergency Diesel Generator. This modification provided the added benefit of ensuring that the "A" Emergency Diesel Generator would be available to power the valve to the "B" Emergency Feedwater Pump in the event of the loss of the "B" batteries that operated that valve.

Subsequently, in 1988, in communication with the NRC, Florida Power committed to employ certain emergency diesel load management programs, and adopted certain testing (or "surveillance") requirements for its Emergency Diesel Generators.

In 1988, as part of its ongoing monitoring of its Engineered Safeguards system, Florida Power determined that it would be desirable to reduce loading further on the Emergency Diesel Generators by an electronic re-configuration in order to avoid reducing the operating margins of the diesels in certain situations. The Company engaged its Architect-Engineer, Gilbert Commonwealth Inc., to design a modification that would cause the "A" Emergency Feedwater pump to turn off automatically (or "trip off") when the water pressure in the reactor core fell to 500 pounds per square inch gauge (or "psig") in a hypothetical situation involving a loss of offsite power. This would occur during a significant loss of coolant. The Low Pressure Injection pump is actuated when pressure reaches that point. The Low Pressure Injection pump and the Emergency Feedwater pumps are not needed at the same time. Thus, an opportunity existed to shut down the Emergency Feedwater pumps in this situation and to reduce load on the Emergency Diesel Generators at that time. The Company installed this modification in 1990. See PFM Ex. 3 (consisting of MAR documentation for this modification).

Also in 1990, the Company upgraded both of its Emergency Diesel Generators to increase their engine ratings by, among other things, creating a larger air intake, modifying the turbocharger, and adding new lube oil coolers. As a result of these upgrades, the Company was able to increase the maximum 30 minute ratings of the generators from 3300 kW to 3500 kW. These modifications were designed by Coltec, the Emergency Diesel Generator manufacturer. That same year, the Company modified its Engineered Safeguards system to manage better the block loading of electric-powered equipment onto the Emergency Diesel Generators during emergency situations.

"Block loading" refers to the fact that load from various pieces of electric-powered safety-related equipment was placed on the Emergency Diesel Generators by one piece of equipment at a time, each accounting for a certain quantity -- or "block" -- of power produced by the Emergency Diesel Generators. Because equipment tends to draw more electrical load when it is first loaded than during normal operation, it is necessary to stagger load blocks to avoid overloading the Emergency Diesel Generators.

In contrast to the modifications that we were making at CR-3, Florida Power & Light installed two new Emergency Diesel Generators in 1990 at its Turkey Point nuclear facilities, taking both nuclear units out of service for approximately a year to accomplish the modifications. We hoped to avoid extensive hardware modifications of this nature, and related outages, by modifying CR-3's electronic control systems and existing hardware.

As part of its standard operating procedures and in conformance with NRC-required surveillance procedures, the Company continued to monitor Emergency Diesel Generator loading and to calculate hypothetical loading situations, both through its own engineers and with the assistance of the Company's Architect-Engineer, Gilbert Commonwealth Inc. Any change or maintenance to any of the equipment powered by the diesels -- and even wear and tear on plant equipment -- could affect diesel loading margins, and calculations had to be run after each refueling outage. A licensee is required to test and closely monitor the condition of all safety-related equipment, including Emergency Diesel Generators. In addition, as new analytical tools and technology became available that might enable the Company to enhance and optimize its Engineered Safeguards system, the Company sought to take advantage of those opportunities.

In this connection, in November 1991, Florida Power replaced its Emergency Diesel Generator block-loading relays with highly accurate quartz timing relays to optimize the accuracy of the block-loading sequence for the Emergency Diesel Generators. At the beginning of the next year, the Company upgraded batteries capable of indirectly reducing load on the Emergency Diesel Generators by installing a new battery bank to handle non-safety-related loads. Also in 1992, Florida Power added a comprehensive engine/generator monitoring system to the Emergency Diesel Generators to improve the Company's ability to ensure that operating margins were maintained on the Emergency Diesel Generators.

From 1993 through the refueling outage that commenced in February 1996, Florida Power continued to monitor Emergency Diesel Loading operating margins and to make adjustments to its Engineered Safeguards systems to maintain those margins.

D. <u>EMERGENCY FEEDWATER SYSTEM</u>

Q. DID FLORIDA POWER CORPORATION ADDRESS OTHER SIGNIFICANT ISSUES INVOLVING THE EMERGENCY FEEDWATER SYSTEM THAT HAVE A BEARING ON THE CURRENT OUTAGE?

A. Yes. As I mentioned earlier in my testimony, the Company is undertaking at this time to add cavitating venturis to the Emergency Feedwater pumps to regulate feedwater flow through the pumps. The Company made less extensive modifications to the control system over the years in order to ensure appropriate flow of feedwater through the pumps and to prevent pump failure in emergency situations.

Q. PLEASE EXPLAIN THE STEPS THAT THE COMPANY TOOK.

A. As I discussed previously, to implement new regulatory directives that the NRC issued in the 1980's, the B&W owners' group developed information that was used by each individual licensee to make determinations about how best to meet those requirements. With the benefit of this information, Florida Power, among other measures, installed a system called "Emergency Feedwater Initiation and Control" or "EFIC" to control the level of water pumped by the Emergency Feedwater system. Previously, the various pieces of equipment that make up the Emergency Feedwater system were controlled manually or by the plant non-safety control system. The Emergency Feedwater Initiation and Control system ensured that the various pumps and valves that comprise the Emergency Feedwater system could be controlled automatically and in an integrated fashion.

This modification took about four years to develop and implement. The design work began in about 1981. Florida Power consulted with its Architect-Engineer, Gilbert Commonwealth Inc. and Babcock & Wilcox, in developing the system. The Emergency Feedwater Initiation and Control system was installed in August 1985.

The Emergency Feedwater Initiation and Control system was designed, in part, to take into account possible cavitation in the Emergency Feedwater lines, and to optimize the operation of the

Emergency Feedwater system. The term "Cavitation" is used to describe a condition that occurs when water pressure at the inlet of the Emergency Feedwater drops low enough for steam bubbles to form. If the problem is extensive enough, it can cause pump failure. It may occur when water flows through the Emergency Feedwater pumps at an excessive rate or, conversely, where flow into the pump is insufficient.

The Emergency Feedwater Initiation and Control system was designed so that certain valves regulating the flow of feedwater through the pumps would remain in the open position in the event of a loss of electric power to those valves. The Company and its consultants determined that this design was acceptable because the system was specifically designed to have other valves that would stop the flow if the steam generators became too full, called "overfill protection."

In 1987, Florida Power made further modifications to its Emergency Feedwater system to help guard against pump runout or cavitation. Specifically, Florida Power engaged Gilbert Commonwealth Inc. to design and install a flow limiting circuit to the Emergency Feedwater Initiation and Control system to keep the flow from exceeding a maximum limit.

Florida Power continued thereafter to monitor the performance of its Emergency Feedwater system and to consider the adequacy of the system to respond to various hypothetical accident situations. For example, in 1992, the Company evaluated the system in the hypothetical emergency situation where the batteries that power the "B" Emergency Feedwater equipment would fail at the same time as offsite power would be lost and a loss of coolant accident occurred. The Company investigated the possibility that this might result in an excessive flow of feedwater. The Company tested this situation with the use of a training simulator and concluded that the results were acceptable.

Q. DID THE COMPANY CONSIDER INSTALLING CAVITATING VENTURIS TO PREVENT PUMP CAVITATION PRIOR TO THE CURRENT OUTAGE?

A. Yes, it did both for the High Pressure Injection system and the Emergency Feedwater system. The Company sought to avoid resorting to this modification because of the difficulties that it posed. A cavitating venturi is a device that limits the flow of, for example, feedwater through the pipes leaving the Emergency Feedwater pumps. To oversimplify the matter, it is like installing a washer with a small opening inside a pipe, forcing the water to stream through the smaller opening. The water flows through the opening, reaching sonic velocity, and the pressure created by this action slows the movement of the water through the opening.

In order to install cavitating venturis, the pipes must be cut and the devices welded into the pipes. The material must be x-rayed after every pass, and any imperfection must be ground out and the welding repaired. It is possible that it will be necessary to repeat the procedure with a new length of pipe if repeated repairs are made in the original pipe. In short, the modification is complex and requires an extended outage to complete.

In addition, Florida Power was aware in the late 1980's that Arkansas Power & Light had installed cavitating venturis in the High Pressure Injection system of its nuclear unit ("ANO-1"), which, like CR-3, had been designed in part by Babcock & Wilcox. The venturis created excessive vibration, resulting in secondary damage to the pipe supports. As a result, Arkansas Power had to modify the devices.

Similarly, in 1989, Duke Power considered installing cavitating venturis to prevent cavitation in the Emergency Feedwater pumps at its Oconee nuclear units. After designing, manufacturing, and installing these devices in a test loop, Duke Power became concerned that cavitating venturis would create excessive vibration in the Emergency Feedwater configuration at the Oconee units, which might damage other equipment. Accordingly, Duke Power abandoned its plans to install hardware modifications to deal with the cavitation concern. Instead, like Florida Power, Duke Power sought to cope with this concern by means of operator training and procedures. See PFM Ex. 4 (Duke Power correspondence).

Against this background, Florida Power sought to address concerns about cavitation by making control system and electrical modifications to its Emergency Feedwater system and by emphasizing operator training and procedures. At the time, this appeared to be a more controllable, proven, and cost-effective solution to the problem in the context of the system configuration that existed at CR-3.

E. <u>THE 1996 REFUELING OUTAGE</u>

Q. **DID FLORIDA POWER MODIFY ITS ENGINEERED SAFEGUARDS SYSTEM DURING THE 1996 REFUELING OUTAGE?**

A. Yes. At that time, the Company determined that it was necessary to disconnect the ASV-204 valve that controlled the flow of steam into the

"B" Emergency Feedwater pump turbine from the "A" Emergency Diesel Generator in order to avoid pump cavitation in certain emergency circumstances. This had the effect of "reversing" the modification that the Company had made in 1987, when it elected to power the "B" ASV-204 valve by means of the "A" Emergency Diesel Generator to reduce flow through the "A" Emergency Feedwater pump and thereby to reduce loading on the "A" Emergency Diesel Generator.

Q. WHAT LED THE COMPANY TO MAKE THIS MODIFICATION?

A. As part of the ongoing process of monitoring its systems, the Company began focusing in 1994 on the accuracy of plant instrumentation. With the benefit of more sophisticated computers and analytical tools, the Company came to question calculations that had relied on existing instrumentation. The Company modified its calculations to assume a greater margin for error than it previously assumed.

Our engineers determined in 1996 that the kilowatt meters that were used by operators in performing load management actions showed uncertainties at load levels over 200 kW. So the Company took steps to install more accurate components in the Emergency Diesel Generator kilowatt meter indication loops beginning March 1996. In connection with this modification, plant personnel were required to consider the interrelationship of this equipment with other features of the Engineered Safeguard system. In the course of doing this work, our engineers examined the then-existing engineering judgment (arising out of the modification we performed in 1987) that load would be reduced by the concurrent operation of the "A" Emergency Feedwater pump (which was powered by electricity) and the "B" Emergency Feedwater pump (powered by steam).

In re-analyzing this assumption, plant personnel considered the possibility that flow control for the "B" Emergency Feedwater pump might not be available to provide controlled back-up to the "A" Emergency Feedwater pump if a failure occurred in the batteries used to power valves on equipment on the "B" side of the Engineered Safeguards system. The valves that control the flow of feedwater through the "B" Emergency Feedwater pump are normally open and require electric power to close. If the batteries used to power those valves were to fail, the "B" pump would operate only at its maximum capacity. This would afford plant operators less control over the operations of the "B" Emergency Feedwater pump in a Small Break Loss of Coolant Accident.

In investigating this situation further, our engineers became concerned that the flow of feedwater through the Emergency Feedwater pumps in the circumstances just described might result in the loss of Net Positive Suction Head, which can lead to cavitation, the condition that I have previously described. The Company conducted detailed calculations in 1996 -- which factored in more conservative assumptions based on our knowledge then of instrument error -- which indicated that this risk might well materialize in certain hypothetical emergency situations.

To address this issue, CR-3 engineers from Operations, Licensing, Electrical and Mechanical Design Engineering, and Emergency Operating Procedures considered the option of modifying the Engineered Safeguards system so as to remove the automatic actuation of the "B" Emergency Feedwater pump whenever the "A" pump was initiated, thus reversing the 1987 modification. Although the 1987 modification had been implemented to reduce load on the Emergency Diesel Generators, the generators were upgraded in 1390 to increase their ratings, and we had obtained more accurate information about their capability from Coltec in the intervening years. Based on this information, our engineers concluded that we could safely reverse the 1987 modification.

CR-3 personnel were aware that the Emergency Diesel Generators had been assigned ratings by Coltec indicating that the diesels were made to operate continuously at a maximum of 3500 kW for 30 minutes. CR-3 personnel were further aware that calculations showed that the "A" Emergency Diesel Generator would spike up to 3700 kW for up to a few seconds if the "A" Emergency Feedwater pump were assumed to run without back-up from the "B" pump during a hypothetical Small Break Loss of Coolant Accident. CR-3 engineers were able to arrive at this determination because the Company had only recently acquired the ability to calculate loads in small temporal increments. Previous calculations had averaged loads over longer periods of time and lacked the precision to detect such transient spikes.

The Electrical Design Engineering Department contacted Coltec to confirm that the generators could in fact handle this load. Coltec assured the Company that this was the case. See PFM Ex. 5 (correspondence from Coltec). Based on this information, the Electrical Design Supervisor approved the modification reversing the 1987 modification. The modification was completed in May 1996. See PFM Ex. 6 (consisting of MAR documentation on this modification.)

The concern that led to our making the modification arose approximately two months into the February 1996 refueling outage. Had the Company not developed an adequate solution at the time, the Company would have had to keep the plant shut down until one could be developed. This is so because, based on the Company's then most recent calculations, CR-3 would fail to meet the Technical Specifications for operable Emergency Feedwater pumps and therefore Florida Power would be out of compliance with the technical requirements of its license. As a result of the solution described, the Company was able to restart the plant on May 17, 1996.

Q. **BASED ON ALL THE INFORMATION AVAILABLE TO THE COMPANY BY THE BEGINNING OF 1996, DID MANAGEMENT BELIEVE THAT IT HAD TAKEN REASONABLE AND EFFECTIVE STEPS TO MEET ALL REGULATORY REQUIREMENTS INVOLVING ITS ENGINEERED SAFEGUARDS SYSTEM?**

A. Yes. As I have described, the Company made a number of modifications to this system over the years that appeared fully effective in meeting the applicable safety requirements.

V. THE CAUSE OF THE CURRENT OUTAGE

A. THE RUPTURED LUBE OIL PIPE

Q. WHAT PRECIPITATED THE CURRENT OUTAGE?

A. On September 2, 1996, a lube oil pipe in the main turbine ruptured. Plant operators detected a loss of lube oil pressure in the main turbine on August 30, 1996, which caused an automatic start of the electric back-up bearing oil pump. (This system is used to lubricate the bearings on the main turbine.) CR-3 personnel commenced an immediate investigation of the problem and contacted the system vendor, Westinghouse, for guidance. Westinghouse indicated that a valve was partially blocked and that CR-3 could be operated while our investigation continued. We took steps to adjust the valve to correct the problem.

Plant operators monitored the lube oil pressure, standing ready to shut down the plant if oil pressure dropped to a specified level. Oil pressure continued to fall. On September 2, plant operators detected a drop in the main turbine lube oil reservoir level of about 400 to 600 gallons over the prior 24 hour period. Our investigating engineers determined that there was significant foaming of the oil within the reservoir. Based upon this problem and the continued loss of lube oil pressure, we decided to decrease reactor power and to place the turbine on turning gear. By 9:00 pm, September 2, we took CR-3 off line.

On September 3, our engineers opened several lube oil tank access ports and detected a long pipe crack and oil spewing inside the tank. Confronted with this problem, we continued to cool down the plant because the necessary repair work would require the shut down of the turbine and the breaking of the condenser vacuum.

Upon further inspection of the system, we identified a four and a half foot crack in a ten foot long pipe just inside the top of the tank. In addition, the flange bolts in the downstream end of this pipe had loosened, permitting much of the gasket to blow out. The crack and the condition of the gasket allowed lube oil to leak out of the pipe, explaining the loss of pressure.

Our engineers also determined that a pipe support was damaged, another was missing, and a third one had been previously repaired. The pipe was subject to vibration in normal operation, placing stress on these three-inch pipe supports. The deterioration of the pipe supports resulted in still greater vibration, which loosened the flange bolts. An offsite metallurgical laboratory detected a high number of inclusions (or malformaties) in the failed pipe section, which made the pipe susceptible to fracture. The vibration combined with these inclusions caused the pipe to break.

Q. BASED ON THE INFORMATION AVAILABLE TO THE COMPANY PRIOR TO THE RUPTURE OF THIS PIPE, DID THE COMPANY ANTICIPATE THAT THIS MIGHT OCCUR?

A. No. The pipe had been in use since the plant's inception, and we had not experienced any problem like this before. The pipe was naturally subject to vibration during the normal operation of the plant, for which the pipe and the pipe supports had been designed. Although our engineers had made occasional repairs to the pipe support structures during outages conducted for unrelated reasons, these repairs fell into the category of routine maintenance on account of the normal wear and tear on the plant, and there was no indication that the pipe itself had ever suffered or would suffer any structural impairment. We were unaware, of course, of the inclusions in the prefabricated pipc material. During operation of the plant, the pipe is not observable. We were unaware of the loosening of the flange bolts. Finally, to our knowledge, there have been no industry advice letters or similar failures at other nuclear units.

Q. WHAT STEPS DID THE COMPANY TAKE TO REPAIR THE PIPE?

A. After shutting the plant down, we continued our investigation of the problem and started repair work. By September 7, we had removed the tank manway and inspected supports and hangers. We had to hand wash the tank so that welding could be perf rmed. By September 9, we had received a replacement pipe section, and by September 11 it was bolted in place. New hangers and supports were fabricated as needed. By September 14, we had completed the repairs, cleaned the tank, and refilled it with oil. The lube oil system was restored to service by September 14. In the normal course, we would have been able to return CR-3 to service in approximately a week from that date, in order to allow for final inspections and verifications and the restart of the unit.

B. THE EXTENDED OUTAGE

Q. WHY DID THE COMPANY KEEP THE PLANT OUT OF SERVICE?

A. In the process of ongoing analysis of the diesel loading issues and the Emergency Feedwater system, and with oversight by the NRC, the Company determined that the modification that it had made during the February 1996 refueling outage would not be sufficient to ensure compliance with the plant's operating license.

Specifically, the May 1996 modification gave rise to several Unreviewed Safety Questions that CR-3 personnel did not recognize at the time of the modification, but that were identified after considerable analysis and discussion with the NRC. These Unreviewed Safety Questions arose from the fact that the modification led to a situation where the Emergency Diesel Generators might exceed the plant's Technical Specifications in certain hypothetical emergency situations, and where the Emergency Feedwater pumps might not be available when needed to deal with a Small Break Loss of Coolant Accident.

The Company has now determined that it must address these concerns by completing a more extensive upgrade of its Emergency Diesel Generators and that it must install cavitating venturis in its Emergency Feedwater system. Facing the necessity of these changes, the Company is taking the opportunity of this outage to complete other modifications that will ensure that the Company is able to maintain necessary safety margins in the future.

VI. <u>REASONABLENESS OF COMPANY'S ACTIONS</u>

Q. AT THIS TIME, IT APPEARS THAT EXTENSIVE MODIFICATIONS MUST BE MADE TO RESTORE THE COMPANY'S ENGINEERED SAFEGUARDS SYSTEM TO FULL COMPLIANCE WITH LICENSING REQUIREMENTS. WHY DIDN'T THE COMPANY MAKE THESE MODIFICATIONS SOONER?

A. As I have explained, management of CR-3 believed that the modifications that we made over the years were a prudent and effective means to address all applicable safety requirements. We felt that it would have been unwarranted in the context of our particular plant to make more extensive modifications sooner. It appeared that there were other reasonable and effective alternatives that would enable the Company to avoid the capital expenditures and extended outages associated with the more extensive modifications that the Company is making today.

Q. WHY DIDN'T FLORIDA POWER EMULATE OTHER NUCLEAR PLANTS THAT MADE MODIFICATIONS IN THE PAST SIMILAR TO THOSE FLORIDA POWER IS MAKING TODAY?

A. Although various nuclear plants may be similar, each one is different, and each plant must adopt strategies that provide appropriate solutions for that plant. Even among B&W plants, there are significant design differences, and the differences have been exaggerated by various modifications that each has made over the years. For example, the Oconee nuclear plant was built with no Emergency Diesel Generators. Three Mile Island has two very small electric-powered Emergency Feedwater pumps, as distinguished from one large electric-powered

Emergency Feedwater pump at CR-3 that imposes a lot of load; and Davis Besse has only steam-driven Emergency Feedwater pumps.

Florida Power has pursued strategies over the years that appeared to provide a cost-effective means of meeting all safety requirements without imposing undue costs on the ratepayers. In addition, as I have described, in the past various facilities were encountering drawbacks associated with certain major modifications, and we were reluctant to assume the risk of those modifications.

Q. WILL THE COMPANY BE ABLE TO BENEFIT FROM ITS OWN EXPERIENCE AND THE EXPERIENCE OF OTHERS IN MAKING THE CURRENT MODIFICATIONS?

A. Yes. Had the Company made these modifications sooner, it would not have had the benefit of the experience gained from all the previous efforts to address these issues that I have described. This experience has provided a much more complete understanding of the system and equipment at CR-3 than the Company had in the last decade, for example. This should enable the Company not only to make the modifications efficiently but to make better modifications. No engineer is clairvoyant or has complete knowledge of what the future will bring. But the Company has significant experience now in operating CR-3 and is well situated to make the current modifications in an effective and efficient manner. The same holds true of the experience the Company has gained from efforts by other plants to make the same or similar modifications.

VII. REASONABLE NECESSITY OF THE OUTAGE

Q. IF THE COMPANY HAD UNDERTAKEN TO MAKE THE SAME MODIFICATIONS SOONER, WOULD IT HAVE BEEN NECESSARY TO HAVE THE SAME KIND OF OUTAGE AS THE CURRENT OUTAGE TO ACCOMPLISH THESE MODIFICATIONS?

A. Yes. It is important to understand that, based on what we now know, the Company has been able to determine that the plant currently is incapable of satisfying the requirement under its license that it be able to respond to a hypothetical emergency situation that may occur once in 11.6 billion years (with resulting damage to the core occurring once in over a trillion years). Although this is an extremely remote statistical risk, it is one that has very immediate regulatory consequences, namely the Company must keep the plant out of service until the problem is fixed. See PFM Ex. 7 (Technical Specifications limiting conditions).

If we are supposing that the Company would have recognized the need to make these extensive modifications sooner, in fairness we must also assume that the Company would have known why these modifications were required. But if the Company had recognized in the past what it came to recognize in September 1996, it would have found itself in the same place. That is, it would have faced the necessity of taking or keeping the plant out of service until this deficiency in its Engineered Safeguards system could be rectified. The exigency of the modifications then, as now, would preclude advance planning.

Florida Power's history of modification experience demonstrates that there are no reasonable alternatives to making the extensive modifications now underway. The Company implemented a series of modifications over a period of seventeen years that were reasonable, technologically feasible alternatives designed to enable the Company to avoid the extensive costs associated with the current outage. Again, if we are supposing that the Company would have recognized at any time in the past what it now knows, the Company would have simply embarked in the past to make the same extensive modifications.

In addition, there is no reason to suppose that the Company should have recognized the need to make these modifications on the verge of entering any other outage, such as a regularly scheduled fuel outage. One might just as easily suppose that the recognition might have occurred while the plant was in full operation. In any case, based on my knowledge of CR-3's past outages, I am able to state that the Company could not have accommodated the kind of modifications we are undertaking at this time within the umbrella of any past outage. The Company was fully occupied dealing with other exigencies during each past outage. The most reasonable conclusion to draw from all the facts and circumstances is that had the Company arrived earlier at the same conclusions reached in September 1996, the result would have been an outage at least as long as the current one. Indeed, had we attempted to implement the current modifications before CR-3's electronic circuitry was upgraded in the 1985 (with the installation of the Emergency Feedwater Initiation and Control system that I have described), the modifications would have been harder to implement.

Q. IS THE COMPANY INCURRING ALTERNATIVE FUEL COSTS AS A RESULT OF THIS OUTAGE?

A. Yes.

Q. IS IT REASONABLY NECESSARY TO INCUR THESE COSTS?

A. Yes. The Company had to enter into an extended shutdown in order to make the necessary modifications. These modifications could not be made while the plant is in service. They include extensive upgrades to the Emergency Diesel Generators to increase their capacity by 150 kW and the dismantling of the Emergency Feedwater System in order to add cavitating venturis that will regulate the flow of feedwater from the Emergency Feedwater pumps.

As noted, these modifications are not discretionary. The Company must make these modifications in order to restore compliance with CR-3's license.

Finally, as I have described, the Company has no alternative means to satisfy the requirements of its license other than by making these extensive modifications. The Company has made every reasonable attempt over the past seventeen years to meet applicable safety requirements by means of less extensive modifications that could be made and have been made with minimal downtime. The Company has exhausted these options and has no alternative at this time other than to keep CR-3 out of service for an extended period of time in order to make the current modifications.

Q. DOES THIS COMPLETE YOUR DIRECT TESTIMONY

A. Yes, it does.

260

Florida Power CORPORATION

BEFORE THE
FLORIDA PUBLIC SERVICE COMMISSION

DOCKET NO. 970261-EI

In Re: Review of Nuclear Outage at Florida Power Corporation's Crystal River Unit No. 3

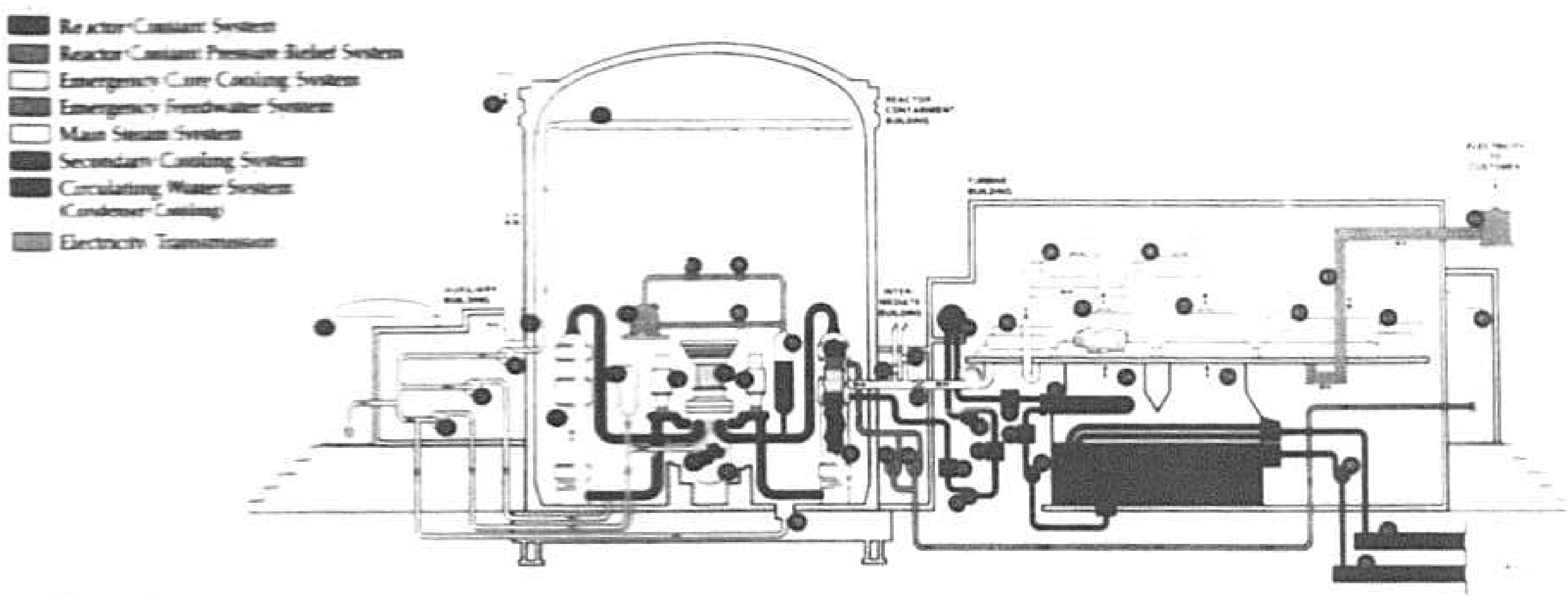
EXHBITS TO THE TESTIMONY OF

PAUL F. McKEE

For Filing April 14, 1997

EXHIBITS TO THE TESTIMONY OF
PAUL F. MCKEE

EXHIBIT NO. ___ (PFM-1)
BASIC SCHEMATIC DIAGRAM OF CR-3



Schematic

1. Borated Water Storage Tank
2. Low Pressure Injection Pumps (2)
3. High Pressure Injection Pumps (3)
4. Reactor Building Spray Pumps (2)
5. Reactor Building Purge Valve
6. Reactor Building & Auxiliary Building Vent
7. Reactor Building Spray Headers (2)
8. Pilot Operated Relief Valve
9. Pressurizer Block Valve
10. Pressurizer Safety Valve (2)
11. Reactor Coolant Drain Tank
12. Steam Generator
13. Core Flood Tanks (2)
14. Reactor Coolant Pumps (4)
15. Control Rods
16. Reactor Core
17. Reactor Vessel
18. Reactor Building Sump
19. Pressurizer
20. Atmospheric Dump Valves (2)
21. Main Steam Safety Valves (16)
22. Main Steam Isolation Valves (4)
23. Motor Driven Emergency Feed Pump
24. Turbine Driven Emergency Feed Pump
25. Main Feedwater Pumps (2)
26. Feedwater Heaters (2)
27. Feedwater Heaters (2)
28. Condensate Pumps (2)
29. Demineralizers (6)
30. Feedwater Booster Pumps (2)
31. Condensate Heaters (2)
32. Condensate Heaters (4)
33. Main Steam Lines (4)
34. Deaerator
35. High Pressure Turbine
36. Moisture Separator Reheaters (4)
37. Low Pressure Turbines (2)
38. Condensers (2)
39. Intake Canal
40. Discharge Canal
41. Circulating Water Pumps (4)
42. Electric Generator
43. Electricity Transmission Lines
44. Exciter
45. Emergency Feedwater Tank
46. Transformers (3)

EXHIBITS TO THE TESTIMONY OF
PAUL F. MCKEE

EXHIBIT NO. ___ (PFM-2)
MAR DOCUMENTATION FOR 1987 MODIFICATION

MODIFICATION APPROVAL RECORD

Florida Power

Crystal River Unit 3

Modification Number	Work Request	Safety Related	
MAR T87-10-09-01	94907	X Yes ☐ No	☐ Yes ☐ No

Project Name: ASV-5/204 POWER SEPARATION

Systematic Number	Priority Code	Cost Accounting Data
AS	X100	

Type of Modification:
☐ Permanent ☒ *Temporary Expiration Date 10-1-88 *W.R. No. for Removal 94707
☐ Mechanical ☒ Electrical ☐ I&C ☐ Structural ☐ Hanger

See Attached Sheet 1 of 1

NOTE: Supplemental Information To Be Added

See Attached Sheet 1 of 1

ORIGINAL

Design Engineer	Date		Date
E. A. Nhook	10/20/87	[signature]	10/30/87
		[signature]	10/30/87

	SIGNATURE	DATE
Quality Programs Inspection Plan Review	[signature]	11-14-87
Meeting Number 87-93 Plant Review Committee (PRC)	[signature]	11-15-87
NGRC Review Required ☐ Yes ☒ No NGRC Approval		
Installation Authorized	Nuclear Plant Manager: [signature] for (on call)	11/15/87
Cancellation	Nuclear Plant Manager	
Temporary Modification Removed	NA made Permanent by MAR 87-10-09-01 A	
Installation and Test Complete and all Documentation Enclosed	[signature]	7/13/88

CR3 085374

Florida Power Corporation	ANALYSIS / CALCULATION Crystal River Unit 3	SHEET 1 OF 1

REI/MAR No.	MAR T87-10-09-01	Date	OCTOBER 28, 1987
Project:	ASV-5/204 POWER SEPARATION		

ATTACHMENT TO MODIFICATION APPROVAL RECORD

DESCRIPTION OF PROPOSED MODIFICATION

Electrically separate Motor Operated Valve ASV-204 from ASV-5 and assign a new power source for ASV-204 from the 250/125 Vdc ES 'A' power system. Install separate remote manual control and automatic EFIC interlocks for ASV-204.

REASON FOR MODIFICATION

Presently, ASV-204 and ASV-5 are electrically connected in parallel and powered from a common 250/125 Vdc ES 'B' power source, and operate via common manual and automatic control interlocks. ASV-204 is being repowered from a 250/125 Vdc ES 'A' power source and provided with separate controls in order to enable the Turbine Driven EFW Pump (EFP-2) to be operable with a failure of the 250/125 Vdc ES 'B' power system. With this capability, EFP-2 will be available to share the EFW flow requirements with the motor driven EFW pump and, thus, decrease the load on Emergency Diesel Generator EDG-1A for scenarios requiring EFW coincident with loss-of-offsite-power and failure of the 'B' power system.

Design Engineer	Date	Verification Engineer	Date	Supervisor, Nuclear Engineering	Date
D.A. Rinda	10/28/87	[illegible]	10/28/87	[illegible] for LEV	10/28/87

CR3 085375

MODIFICATION SAFETY EVALUATION

MAR NO. T87-_10_.09_.01_

SAFETY EVALUATION: Answer the following questions and provide specific justification (use attachment if necessary).

1. Is the probability of an occurrence or the consequence of an accident or malfunction of equipment important to safety as previously evaluated in the Final Safety Analysis Report, INCREASED? YES ___ NO _X_
   Because:
   See Attached Sheet

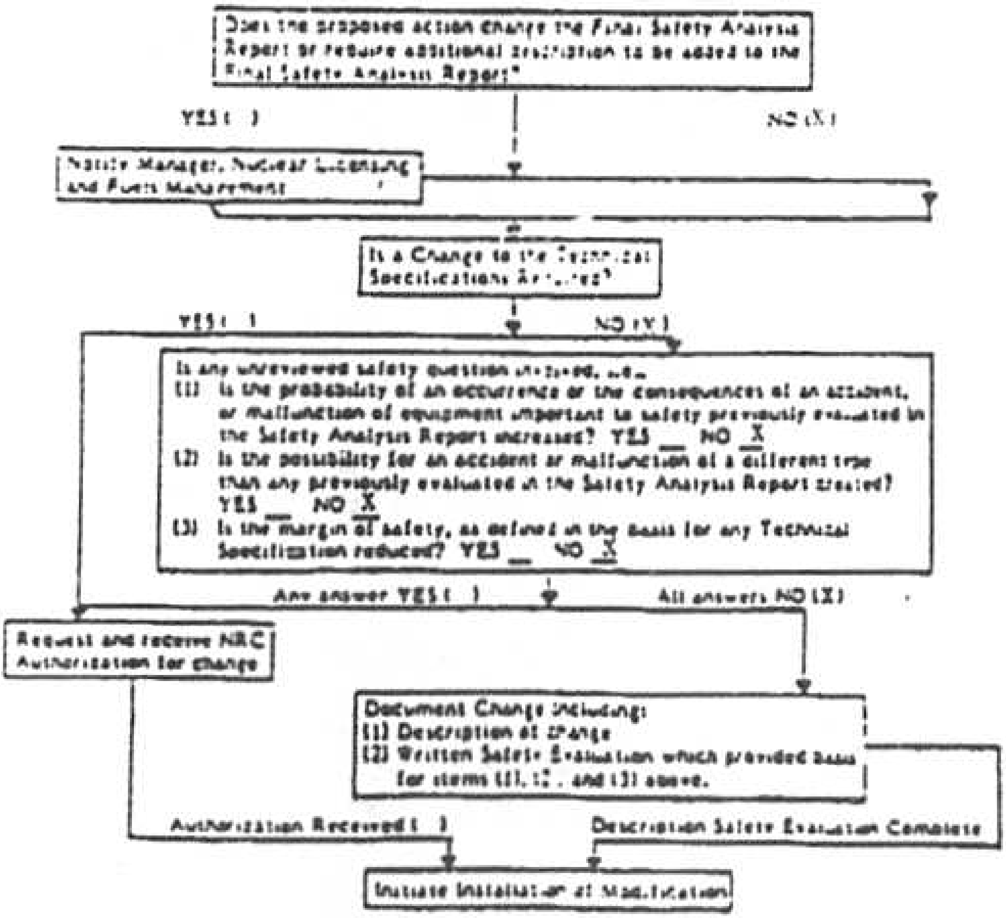
2. Is the possibility for an accident or malfunction of a different type than any previously evaluated in the Final Safety Analysis Report, CREATED? YES ___ NO _X_
   Because:
   See Attached Sheet

3. Is the margin of safety, as defined in the basis for any Technical Specification, REDUCED? YES ___ NO _X_
   Because:
   See Attached Sheet

LICENSE REVISION REQUIRED:

Final Safety Analysis Report:	YES ___	NO X
Technical Specifications:	YES ___	NO X
NRC Authorization for Change Required:	YES ___	NO X
Semi-Annual Reporting to NRC Required:	YES ___	NO X

10CFR50.59 CHECKLIST



CR3 085376

*Required changes to Technical Specifications

Prepared by: [illegible signature]

Florida Power Corporation

ANALYSIS / CALCULATION
Crystal River Unit 3

REI/MAR No.	MAR T87-10-09-01	Date	OCTOBER 19, 1987
Project:	ASV-5/204 POWER SEPARATION		

ATTACHMENT TO MODIFICATION SAFETY EVALUATION

1. ASV-5 and ASV-204 are motor operated valves having identical functions of supplying steam to the turbine driven Emergency Feedwater Pump (EFP-2). Since EFP-2 is the ES "B" channel pump, ASV-5 and 204 were electrically connected in parallel to a common 250/125 VDC ES "B" channel power and control source. This modification electrically separates ASV-204 from ASV-5 and repowers ASV-204 from 250/125 VDC ES "A" channel power. Also, separate control room controls and separate "A" channel EFIC interlocks are being provided for ASV-204. Automatic control logic of ASV-204 has not changed. Therefore, the probability of an occurance or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the FSAR is not increased since the logic of automatically opening ASV-204 whenever the EFIC System calls for emergency feedwater has not been altered. The reliability of EFP-2 has actually been increased because with this modification either "A" or "B" train power will control and operate one of the steam inlet valves to EFP-2 as opposed to both valves being "B" train powered. FSAR Sections 7.2.4, 8.2.2.6 and 10.2.1.6 have been reviewed.

2. The electrical separation of ASV-204 from ASV-5 does not impact the design function of either valve to supply steam to the EFP-2 turbine. Power and control for ASV-5 is not affected by this modification and ASV-5 retains its automatic control logic, remote manual control, local manual control and remote shutdown isolation and control. ASV-204 is being powered from the redundant power channel, and will be provided with its own remote manual control and with separate EFIC interlocks for automatic operation. The type of remote manual control and automatic operation of ASV-204 is the same as for ASV-5. Therefore, based on the above, the possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR is not created. FSAR Sections 7.2.4, 8.2.2.6, and 10.2.1.6 have been reviewed.

CR3 085377

Design Engineer	Date	Verification Engineer	Date	Supervisor, Nuclear Engineering	Date
W. G. Black	10/2/87	H. D. Schaffe	10/3/87	[illegible] for LEV	10/10/87

Florida Power Corporation

ANALYSIS / CALCULATION
Crystal River Unit 3

REI/MAR No.	MAR T87-10-09-01	Date	OCTOBER 19, 1987
Project:	ASV-5/204 POWER SEPARATION		

3. This modification enables the turbine driven Emergency Feedwater Pump (which is the "B" channel pump) to be operational even if a failure should occur on the "B" channel power system for which shutdown operation would be via the "A" channel systems. With this capability, the turbine driven EFW pump is able to operate and share the EFW requirements with the "A" channel motor driven EFW pump. This will reduce the electrical load on the "A" channel diesel generator for the condition of an ES actuation coincident with a loss-of-offsite-power and failure of the "B" channel power system. Consequently, with this modification the margin of safety, as defined in the basis for any Technical Specification, is not reduced. It is actually enhanced because of the increased availability of the turbine driven Emergency Feedwater Pump. Technical Specification Sections 3/4.7.1 and 3/4.8.1 have been reviewed.

CR3 085378

Design Engineer	Date	Verification Engineer	Date	Supervisor, Nuclear Engineering	Date
[illegible]	[illegible]	[illegible]	10/30/87	[illegible]	[illegible]

ENCLOSURE 9

SAFETY EVALUATION REVIEW

TO: Nuclear Technical Specification Coordinator

A copy of MAR T87-1[illegible]-0901 FCN ______________ is attached for your review of the MAR Safety Evaluation for Technical Specification Conformance and Unreviewed Safety Implications.

- [x] ACCEPTABLE
- [ ] UNACCEPTABLE — Return immediately to the Nuclear Modification Supervisor for resolution

COMMENTS:

11/16/87

[illegible]

Return to Nuclear Modifications Supervisor

CR3 085379

Florida Power

REGULATORY/ENVIRONMENTAL REVIEW

Crystal River Unit 3

DOCUMENT NUMBER

MAR T87-10-09-01

REF. NUCLEAR OPERATIONS DEPARTMENT PROCEDURE: NOD-11

1. **10 CFR 50.54 Reviews**

Does this modification or document revision change what is described in any of the following plans/programs? If unable to determine, contact responsible person designated below.

Quality Program Description (FSAR Section 1.7) Contact: Director, Quality Programs	☐ Yes	☑ No
Licensed Operator Requalification Program (FSAR Section 12.2.3.4 and Appendix 12 C) Contact: Manager, Nuclear Operations Training	☐ Yes	☑ No
Modified Amended Security Plan Contact: Manager, Nuclear Licensing	☐ Yes	☑ No
Safeguards Contingency Plan Contact: Nuc. Security & Special Project Superintendent	☐ Yes	☑ No
Security Guard Training and Qualification Plan Contact: Nuc. Security & Special Project Superintendent	☐ Yes	☑ No
Radiological Emergency Response Plan Contact: Manager, Site Nuclear Services	☐ Yes	☑ No

If any are "yes":

a. Contact appropriate responsible person identified above to perform the evaluation and attach. The evaluation must be approved by the Division Head responsible for the plan/program.	Evaluation ☐ Complete	☑ N/A
b. NRC approval received if needed and attach.	☐ Complete	☑ N/A

2. **Environmental Protection Plan Review**

Could this change affect the environment in a non-radiological way?	☐ Yes	☑ No
If "yes", contact the Manager, Nuclear Licensing	☐ Complete	☑ N/A

3. **Review for change to Radioactive Waste System (10 CFR 50.34a and Appendix I / T. S. 6.16.1.1)**

Will this change to a radioactive waste system (liquid, gaseous or solid) result in an increase of radioactive material released to the environment?	☐ Yes	☑ No
If "yes", submit change to Manager, Nuclear Licensing to evaluate the reporting requirements.	☐ Submitted	☑ N/A

IF ALL OF ABOVE ARE CHECKED NO, NO FURTHER REVIEWS ARE REQUIRED.

PREPARED BY	DATE	APPROVED BY	DATE
R. S. [illegible]	11/10/87	[illegible] / C B [illegible]	11/13/87 / 11/11/87

CR3 085380

QUALITY SYSTEMS DOCUMENT EVALUATION

QS LOG # ____________

DOCUMENT # T-87-10-09-01 FCN — REV — CHANGE —

SAFETY RELATED: ✓ YES ___ NO

RELATED DOCUMENTS: ____________

____

QA REVIEW:

___ (1) APPROVED AS IS

✓ (2) APPROVED WITH ADMINISTRATIVE COMMENTS

___ (3) REJECTED
(Asterisked items must be resolved for approval. This rejection must be returned for subsequent reviews of the above document.)

INSPECTION REQUIREMENTS:

___ (1) INSPECTIONS REQUIRED

___ (2) NO INSPECTION REQUIRED

___ (3) NO CHANGE TO EXISTING INSPECTION

___ (4) EXISTING INSPECTION ARE REVISED

✓ (5) INSPECTION REQUIREMENTS TO BE DETERMINED AT THE WORK PACKAGE LEVEL

QUALITY INSPECTION PLAN # ____________

COMMENTS: 1) T-MAR does not comply with 10CFR50 Appendix B criteria VI and ANSI N18.7 para 5.2.15
2) T-MAR does not comply with SRGP #6 para VB(2) removal instructions.

This document has been review by Quality Systems for Quality Assurance (QA) requirements and for the identification of inspections which are necessary to assure compliance with the engineering design and with the material, fabrication, assembly, erection, installation, and examination and test requirements.

NOTE: For definition of Hold Point, Witness Point, and Surveillance, refer to QAP No. 25, Section 4.8.

REVIEWED BY: [signature] DATE: 11-14-87

REV 7/87

CR3 085381

Florida Power Corporation

ENGINEERING INSTRUCTIONS
Crystal River Unit 3

MAR/REI No.	Date
T87-10-09-01	10/24/87

Project: ASV-5/204 Power Separation

A. SUMMARY FUNCTIONAL DESCRIPTION

This modification electrically separates the turbine driven emergency feedwater pump steam inlet isolation valves ASV-5 and ASV-204 in that ASV-204 is being powered from the 250/125 VDC ESA power system. ASV-5 will remain powered from the 250/125 VDC ESB power system. ASV-204 will become an ES channel 'A' valve and will have its own control switch and position indicator lights on the ESA/EFIC section of the main control board. The control circuit for ASV-204 will also be provided with contacts from the EFIC 'A' channel logic for automatic operation. Auxiliary relays will be added to the control circuitry in order to provide alarm logic for alarms which are common to ASV-5 and ASV-204.

This modification will enable the turbine-driven emergency feedwater pump to be operable in the event of failure of the 250/125 VDC ESB power system.

B. ATTACHMENTS

1. Project Assignment Memo
2. Design Data Sheets (4)
3. Design Input Record [illegible]
3a. Safety Classification [illegible] form with attachment [illegible] and page 2-15.
4. Verification Report
4a. GKE [illegible] (sheet 2/2)
5. Fire Protection Review – with [illegible] sheet.
6. Modification Safety Evaluation with sheets 2-3 of 3
7. Regulatory Environmental Review
8. Drawings:

Sketch 1 - Electrical Block Diagram

Sketch 2 - Electrical Conduit Layout, Intermediate Bldg. EL. 95-0

Sketch 3 - Electrical Conduit Layout, Intermediate Bldg. EL. 119-0

HWC 11/13/87

Design Engineer	Date	Verification Engineer	Date	Supervisor Nuclear Engineering	Date
D.A. R[illegible]	10/28/87	[illegible]	10/30/87	[illegible] for LEV	10/30/87

Rev 7/87

RET 25 Yr RESP. Nuc. Eng 912241

CR3 085382

MAR/SEI No.	T87-10-09-01	Date	10/24/87
Project:	ASV-5/204 Power Separation		

Sketch 4 - Electrical Conduit Layout, Complex EL. 108

Sketch 5 - Electrical Conduit Layout, Control Complex EL. 124-0

Sketch 6 - Electrical Conduit Layout, Control Complex EL. 134-0

Sketch 7 - Electrical Interconnection Wiring Diagram, ASV-5

Sketch 8 - Electrical Interconnection Wiring Diagram, Motor Starter for ASV-204

Sketch 9 - Electrical Interconnection Wiring Diagram, TB-AS-9

Sketch 10 - Electrical Interconnection Wiring Diagram, Motor Starter for ASV-5

Sketch 11 - Electrical Interconnection Wiring Diagram, ASV-204

Sketch 12 - Electrical Interconnection Wiring Diagram, TB-AS-01

Sketch 13 - Electrical Arrangement, DPDP-8A

Sketch 14 - Electrical Arrangement, ACDP-54

Sketch 15 - Electrical Arrangement, E.S. Aux. Relay Rack Assy. RR3A

Sketch 16 - Electrical Aux. Relay Rack 3A Internal Wiring

Sketch 17 - Electrical Aux. Relay Rack 3A Internal Wiring

Sketch 18 - Electrical Aux. Relay Rack 3A Terminal Boards

Sketch 19 - Electrical Aux. Relay Rack 3A Terminal Boards

Sketch 20 - Electrical Aux. Relay Rack RR5B1 Terminal Boards

Sketch 21 - Electrical Main Control Board Primary and Secondary Auxiliary Assembly

Sketch 22 - Escutcheon Plate Detail

Design Engineer	Date	Verification Engineer	Date	Supervisor Nuclear Engineering	Date
D.G. Rbsule	10/31/87	H.D. Shaeff	10/31/87	Heati for LEV	10/31/87

Florida Power Corporation

ENGINEERING INSTRUCTIONS
Crystal River Unit 3

MAR/REI No.		Date	
	T87-10-09-01		10/24/87

Project: ASV-5/204 Power Separation

Sketch 23 - Electrical Main Control Board PSA/EFIC Internal Wiring

Sketch 23A - Electrical Main Control Board PSA/EFIC Interconnecting CBO 11/13/87

Sketch 24 - Electrical Main Control Board Primary and Secondary Aux. Terminal Boards CB 11/13/87

~~Sketch 25 - Electrical Main Control Board PSA/EFIC Internal Wiring~~ CBO 11/13/87 11/13/87

Sketch 26 - Electrical Main Control Board Primary and Secondary Aux. Terminal Boards HWC 11/13/87

Sketch 27 - Electrical Elementary Wiring Diagram, ASV-204

Sketch 28 - Electrical Elementary Wiring Diagram, ASV-204

Sketch 29 - Electrical Elementary Wiring Diagram, ASV-204

Sketch 30 - Electrical Elementary Wiring Diagram, ASV-5

Sketch 31 - Electrical Elementary Wiring Diagram, ASV-5

Sketch 32 - Electrical Elementary Wiring Diagram, ASV-5

Sketch 33 - Electrical Elementary Wiring Diagram, ASV-5

Sketch 34 - Electrical Elementary Wiring Diagram, EFIC Matrix 'A' Control and EFW Actuation 'A'

Sketch 35 - Electrical Elementary Wiring Diagram, EFIC Matrix 'A' Control and EFW Actuation 'A'

Sketch 36 - Electrical Elementary Wiring Diagram, EFIC Matrix 'A' Control and EFW Actuation 'A'

E/I Sketch 'A' - Demolition Dwg., Intermediate Bldg. EL. 95-0

9. Bill of Material #1, Revision 0

10. Analysis Calculations

   a. DC-5510-126.0-EE, Rev. 0

Design Engineer	Date	Verification Engineer	Date	Supervisor Nuclear Engineering	Date
G. C. Pikul	10/24/87	[signature]	10/31/87	[signature] for ESV	10/31/87

Rev. 1/87 RET 25 Yr RESP: Nuc Eng 912141

CR3 085384

MAR/REI No.	Date
T87-10-09-01	10/24/87

Project: ASV-5/204 Power Separation

b. DC-CR3-017-EE, pages 1, 2, 24, 29, 31, 32, 33, 34 and 40, all Rev. 1

c. DC-5510-126.1-EE, Rev. 0

11. Cable Pulling, Termination and Test Data Sheets for the following circuits:

ASE 27	ASE 30	ASE 34	ASE 37	ASK 1
ASE 28	ASE 31	ASE 35	ASF 31	ASK 29
ASE 29	ASE 33	ASE 36	ASF 32	ASK 30

12. ALARA Analysis/Calculation Sheet

C. REFERENCES

1. FCS-8915 dated 10/12/87
2. Telecon D. A. Rhoads/M. U. Rahman dated 10/16/87
3. Telecon K. Shirk/I. R. Pressley dated 10/22/87
4. Emergency Diesel Generator Loading Evaluation Report dated 10/23/87

D. MATERIALS

See attached Bill of Material number 1.

E. INSTALLATION INSTRUCTIONS

1. Charge all time and materials to Work Request #94907 for installation and to Work Request #94708 for removal.
2. All cable pulling, termination, splicing, and conduit installation shall be in accordance with applicable FPC maintenance procedures.
3. Conduit Installation at ASV-204.

*a. Install the following aluminum conduits safeguards 'A' Red per Sketch No. 2 and detailed conduit routing sketches.

ASE 34 - 2"
ASE 33 - 1"
ASE 36 - 2"

Design Engineer	Date	Verification Engineer	Date	Supervisor, Nuclear Engineering	Date
D. A. Rhoads	10/30/87	[signature]	10/30/87	[signature] for [signature]	10/..87

MAR/REI No. T87-10-09-01 | Date 10/24/87

Project: ASV-5/204 Power Separation

*b. Install the following aluminum conduit non-safeguards 'X" White per Sketch Nos. 2 and 3 and detailed conduit routing sketches.

ASF 32 - 1"

4. Conduit Installation at DPDP-8A.

*a. Install the following aluminum conduit safeguards 'A' Red per Sketch No. 4 and detailed conduit routing sketches.

ASE 34-2"

5. Conduit Installation at RR3A and RR5B1.

*a. Install the following aluminum conduit non-safeguards 'X' White per Sketch No. 5 and detailed conduit routing sketches.

ASK 30-3/4"

6. Conduit installation at floor opening 5?A to Main Control Board Section PSA.

*a. Install the following aluminum conduit safeguards 'A' Red per Sketch No. 6 and detailed conduit routing sketches.

ASE 36-2"

*Note: Detailed conduit routing sketches to be issued per subsequent FCNs to this MAR.

7. Instructions for MOV ASV-5.

a. Determinate circuit ASK1 as follows and as indicated on Sketch No. 7.

Wire Color/Wire Mark	Terminal
Black (1)/100AL	13A
White(2)/100AL1	13
Shield / ---	Floating

Design Engineer	Date	Verification Engineer	Date	Supervisor Nuclear Engineering	Date
G.C. Roberts	10/24/87	[signature]	10/31/87	[signature] for EV	11/1/87

Rev 2/87

RET 25 Yr RESP Nuc Eng 912241

CR3 085386

MAR/REI No. T87-10-09-01

Date 10/24/87

Project: ASV-5/204 Power Separation

b. Retag wires in ASV-5 as follows:

CKT	Wire Color/Wire Mark	To Wire Mark
EFK42	Red (3)/100 AL1	100 AL2
	Green (4)/100 AL	100 AL1

c. Pull cable out of conduit ASK1-3/4" to cable tray #302. Pull cable back (east) in tray #302 to conduit ASE36-2" and coil in tray to be re-routed in E/I Step 8.

8. Pull the following cables as indicated on the applicable cable pulling data sheets:

ASK 1 (Pulled back in Tray 302)
ASE 34
ASE 35
ASF 32
ASE 36
ASE 37
ASE 33
ASK 29
ASK 30

9. Wiring instructions for Motor Starter for MOV ASV-5, Sketch No. 10:

a. Open breaker #20 in ACDP-12

b. Pull fuse #5 in DPDP-8B

c. Pull fuse #6 in DPDP-8B

d. Determinate the following circuits (circuits to be deleted):

Circuit No.	Wire Color/Wire Mark	Terminal Number
ASF 31	Black (1)/26	3L1
	White (2)/27	3L2
	Red (3)/Spare	---

Design Engineer	Date	Verification Engineer	Date	Supervisor Nuclear Engineering	Date
E G Pfender	10/26/87	D H Shepp	10/27/87	H Adak for CEV	10/30/87

RETURN Vr. RESP: Nuc Eng

Rev. 7/87

Florida Power Corporation

ENGINEERING INSTRUCTIONS
Crystal River Unit 3

MAR/REI No. T87-10-09-01	Date 10/24/87
Project: ASV-5/204 Power Separation	

Circuit No.	Wire Color/Wire Mark	Terminal Number
ASE 27	Black (1)/1 (P)	1L1
	White (2)/6 (N)	1L2
	Red (3)/7	C1

10. Wiring instructions for Terminal Box AS-61, Sketch No. 12.

    a. Determinate the following circuit (circuit to be deleted):

Circuit No.	Wire Color/Wire Mark	Terminal Number
ASE 30	Black (1)/Spare	---
	White (2)/Spare	---
	Red (3)/8	TBB-3
	Green (4)/7	TBB-1
	Orange (5)/15	TBB-9
	Blue (6)/18	TBB-10
	White/Black (7)/20	TBB-11
	Red/Black (8)/22	TBB-12
	Green/Black (9)/36	TBA-7

11. Wiring instructions for motor starter for MOV ASV-204, Sketch No. 8.

    a. Determinate the following circuits (circuits to be deleted):

Circuit No.	Wire Color/Wire Mark	Terminal Number
ASF 31	Black (1)/26	3L1
	White (2)/27	3L2
	Red (3)/-	---
ASE 27	Black (1)/1(P)	1L1
	White (2)/6 (N)	1L2
	Red (3)/7	2L1
ASE 28	Black (1)/38	C2
	White (2)/8	2L2
	Red (3)/50	C4

    b. Remove resistors R1, R2, and R3 and save to be reused later this MAR.

Design Engineer	Date	Verification Engineer	Date	Supervisor Nuclear Engineering	Date
D. A. Kl[illegible]	10/30/87	[illegible signature]	10/30/87	[illegible signature] for [illegible]	10/31/87

Rev. 2/87

RET. 25 Yr. RESP. Nuc Eng 91[illegible]

CR3 085388

MAR/REI No. T87-10-09-01 Date 10/24/87

Project: ASV-5/204 Power Separation

c. Terminate the following cables:

Circuit No.	Wire Color/Wire Mark	Terminal Number
ASF 32	Black (1)/21	3L1
	White (3)/22	3L2
*ASE 34	Black (1)/1 (P)	1L1
	White (3)/ 2 (N)	1L2
*ASE 35	Black (1)/8	2L1
	White (2)/9	2L2
ASE 33	Black (1)/8	2L1
	White (2)/9	2L2
	Red (3)/12	C2
	Green (4)/14	C4
	Orange (5)/Spare	---

* Tape end of cables safeguard 'A' Red.

d. Tape the end of the cable ASE 31 safeguard "A" Red and revise wire marks as follows:

From Wire Mark	To Wire Mark
24	5
25	6
23	3
33	4
28	7

12. Wiring instructions for terminal Box AS-9, Sketch No. 9.

a. Determinate the following circuits:

Circuit No.	Wire Color/Wire Mark	Terminal Number
ASE 29	Black (1)/7	TBA-2
	White (2)/Spare	—
	Red (3)/15	TBA-3

Design Engineer	Date	Verification Engineer	Date	Supervisor Nuclear Engineering	Date
D.A. R[illegible]	10/29/87	[signature]	10/30/87	[signature] for CEV	10/30/87

Florida Power Corporation

ENGINEERING INSTRUCTIONS
Crystal River Unit 3

MAR/REI No. T87-10-09-01 | Date 10/24/87

Project: ASV-5/204 Power Separation

Circuit No.	Wire Color/Wire Mark	Terminal Number
ASE 29 (cont.)	Green (4)/50	TBA-9
	Orange (5)/18	TBA-4
	Blue (6)/20	TBA-5
	White/Black (7)/22	TBA-6
	Red/Black (8)/38	TBA-8
	Green/Black (9)/36	TBA-7

(This ckt. to be reterminated later this MAR)

Circuit No.	Wire Color/Wire Mark	Terminal Number
ASE 28	Black (1)/38	TBA-8
	White (2)/8	TBA-1
	Red (3)/50	TBA-9

(This ckt. to be deleted)

Circuit No.	Wire Color/Wire Mark	Terminal Number
ASE 30	Black (1)/Spare	---
	White (2)/Spare	---
	Red (3)/8	TBA-1
	Green (4)/7	TBA-2
	Orange (5)/15	TBA-3
	Blue (6)/18	TBA-4
	White/Black (7)/20	TBA-5
	Red/black (8)/22	TBA-6
	Green/Black (9)/36	TBA-7

(This ckt to be deleted)

Circuit No.	Wire Color/Wire Mark	Terminal Number
EFK 42	Red (3)/100AL1	TBB-5
	Green (4)/100AL	TBB-4

(Wires to be re-terminated later this MAR)

Circuit No.	Wire Color/Wire Mark	Terminal Number
EFK 43	Red (3)/100AL1	TBB-5

(Wire to be reterminated later this MAR)

Design Engineer	Date	Verification Engineer	Date	Supervisor Nuclear Engineering	Date
D.A. Rhoads	10/29/87	[signature]	10/31/87	[signature] for REV	10/30/87

Rev. 1/87

DEY 25 Yr. RESP., Nuc Eng 9/2/81

CR3 085390

MAR / EEI No. T87-10-09-01 Date 10/24/87

Project : ASV-5/204 Power Separation

b. Install Zener Diodes as follows:

	From Terminal	To Terminal
ZD-1	TBA-8	TBA-12
ZD-2	TBA-8	TBA-10
ZD-3	TBA-8	TBA-11

c. Install resistors as follows (Previously removed from motor starter for ASV-204):

	From Terminal	To Terminal
R1	TBA-10	TBB-10
R2	TBA-11	TBB-11
R3	TBA-12	TBB-12

d. Install #14, AWG SIS Red jumper wire as follows:

From Terminal	To Terminal
TBA-2	TBA-8

e. Terminate circuits as follows:

Circuit No.	Wire Color/Wire Mark	Terminal Number
ASE 29	Black (1)/8	TBA-1
(Was	White (2)/Spare	---
determinated	Red (3)/13	TBA-6
earlier	Green (4)/14	TBA-7
this MAR.)	Orange (5)/15	TBB-10
	Blue (6)/17	TBB-11
	White/Black (7)/19	TBB-12
	Red/Black (8)/12	TBA-5
	Green/Black (9)/11	TBA-4

(Tape end of cable safeguard 'A' Red)

Design Engineer	Date	Verification Engineer	Date	Supervisor Nuclear Engineering	Date
R.a. Rhodes	10/29/87	[signature]	10/31/87	[signature] for CFV	10/31/87

Rev 7/82 RET 25 Yr RESP. Nuc Eng 912241

CR3 085391

Florida Power Corporation

ENGINEERING INSTRUCTIONS
Crystal River Unit 3

MAR/EEI No. T87-10-09-01 | Date 10/24/87

Project: ASV-5/204 Power Separation

Circuit No.	Wire Color/Wire Mark	Terminal Number
ASE 33	Black (1)/8	TBA-1
	White (2)/9	TBA-2
	Red (3)/12	TBA-5
	Green (4)/14	TBA-7
	Orange (5)/Spare	---
ASE 36	Black (1)/8	TBA-1
	White (2)/9	TBA-2
	Red (3)/13	TBA-6
	Green (4)/10	TBA-3
	Orange (5)/16	TBA-10
	Blue (6)/18	TBA-11
	White/Black (7)/20	TBA-12
	Red/Black (8)/Spare	---
	Green/Black (9)/Spare	---
ASE 37	Black (1)/8	TBA-1
	White (2)/11	TBA-4
	Red (3)/13	TBA-6
	Green (4)/10	TBA-3
	Orange (5)/Spare	---
EFK 43	Red (3)/100 AL2	TBB-6
EFK 42	Red (3)/100 AL2	TBB-6
	Green (4)/100 AL1	TBB-5
ASK 1	Black (1)/100 AL	TBB-4
	White (2)/100 AL1	TBB-5
	Shield/---	Float

Design Engineer	Date	Verification Engineer	Date	Supervisor Nuclear Engineering	Date
D.A. Rhode	10/24/87	H. Shauff	10/30/87	Stadie for REV	10/30/87

Rev 2/83 | RET 25 Yr RESP. Nuc Eng

MAR/REI No.	Date
T87-10-09-01	10/24/87

Project : ASV-5/204 Power Separation

13. Wiring instructions MOV ASV-204, Sketch No. 11.

    a. Retag the following wire marks as follows:

From Wire Mark	To Wire Mark
33	4 (2 places)
24	5
25	6
23	3
28	7
36	11 (2 places)
38	12 (2 places)
50	14
7	8 (2 places)
20	17
22	19
15	13
18	15
100 AL1	100 AL2

    b. Tape the ends of cables ASE 29 and ASE 13 safeguard 'A' Red.

14. Wiring instructions for DPDP-8A, Sketch No. 13.

    a. Terminate the following circuits as follows:

Circuit No.	Wire Color/Wire Mark	Fuse Terminal
ASE 35	Black (1)/8 (PN)	15-(PN)
	White (2)/9 (N)	15-(N)

(Tape ends of cable safeguard "A" red)

Circuit No.	Wire Color/Wire Mark	Fuse Terminal
ASE 34	Black (1)/1 (P)	17-(P)
	White (2)/2 (N)	17-(N)

(Tape ends of cable safeguard 'A' Red)

Design Engineer	Date	Verification Engineer	Date	Supervisor Nuclear Engineering	Date
R. A. Rhode	10/24/87	H. L. Shaff	10/31/87	[illegible] for REV	10/31/87

Rev 2/83

Florida Power Corporation

ENGINEERING INSTRUCTIONS
Crystal River Unit 3

MAR / REI No.	Date
T87-10-09-01	10/26/87

Project: ASV-5/204 Power Separation

15. Wiring instructions for ACDP-54, Sketch No. 14:

    a. Terminate the following circuit as follows:

Circuit No.	Wire Color/Wire Mark	Breaker
ASF 32	Black (1)/21	4
	White (2)/22	4

16. Wiring instructions for Relay Rack RR3A, Sketch Nos. 15, 16, 17, 18, 19.

NOTE: Reference MAR T87-10-96-01 which removes relays from the spaces where relays are being added via this MAR.

a. Install relays TMA and TMB as indicated on Sketch 15.

b. Install #14 AWG SIS Red wires as follows:

From	To
AR-1-Sketch 17	TMA-1-Sketch 16
BA-1-Sketch 17	TMB-1-Sketch 16
BA-2-Sketch 17	TMB-2-Sketch 16
AR-3-Sketch 17	TB14-21-Sketch 18
AR-4-Sketch 17	TB14-22-Sketch 18
TMA-2-Sketch 16	TMB-2-Sketch 16

c. Convert contact ~~BA~~ AR-3/4 from N.O. to N.C.

d. Install #14 AWG SIS Gray wires as follows:

From	To
TMA-3-Sketch 16	TB13-12-Sketch 19
TMA-4-Sketch 16	TB13-13-Sketch 19
TMB-3-Sketch 16	TB13-10-Sketch 19
TMB-4-Sketch 16	TB13-11-Sketch 19

Design Engineer	Date	Verification Engineer	Date	Supervisor Nuclear Engineering	Date
E. A. Pland	10/26/87	[signature]	10/30/87	[signature] for CEV	10/30/87

Rev. 1/83

RET 25 Yr. RESP.: Nuc Eng 4/22/83

CR3 085394

MAR / REI No.	Date
T87-10-09-01	10/24/87

Project: ASV-5/204 Power Separation

e. Terminate circuits as follows:

Circuit No.	Wire Color/Wire Mark	Terminal Number
ASK 29	Black (1)/1706AL3	TB13-10-Sketch 19
	White (2)/1706AL4	TB13-11-Sketch 19
	Shield/ --	Float
ASK 30	Black (1)/100K	TB13-12-Sketch 19
	White (2)/100AL	TB13-13-Sketch 19
	Shield/ --	Float
ASE 37	Black (1)/8	TB14-19-Sketch 18
	White (2)/11	TB14-22-Sketch 18
	Red (3)/13	TB14-20-Sketch 18
	Green (4)/10	TB14-21-Sketch 18
	Orange (5)/Spare	---

17. Wiring instructions for RR5B1, Sketch No. 20.

a. Terminate circuit ASK 30 as follows:

Wire Color/Wire Mark	Terminal Number
Black (1)/100K	TB12-27
White (2)/100 AL1	TB12-28
Shield/---	Float

18. Installation instructions for MCB Section PSA/EFIC, Sketch Nos. 21 and 22.
19 RCL 11/10/87 [illegible] CGD 11/10/87

NOTE: Reference MAR T87-10-04-01 which removes existing 'AD' push button.

RCL 11/10/87 a. Install Blank ~~plate~~ plate over "AD" and remove blank ~~three holes for items TM1, TM2, and TM3 on the main control board per Sketch No. 21~~ plate from spared switch and lights per sketch 21.

[illegible] 11/10/87 b. Fabricate escutcheon plate per Sketch No. 22.

CGD 11/10/87 c. Install escutcheon plate and panel items TM1, TM2, TM3, and TM4 per Sketch No. 21.

d. Engrave and install nameplate per Sketch Nos. 21 and 22.

Design Engineer	Date	Verification Engineer	Date	Supervisor Nuclear Engineering	Date
D.G. R[illegible]	10/24/87	[signature]	10/30/87	[signature] for [illegible]	10/30/87

Rev. 1/83 REF: [illegible] RESP. Nuc Eng. 9/22/83

CR3 085395

Florida Power Corporation

ENGINEERING INSTRUCTIONS
Crystal River Unit 3

MAR/REI No. T87-10-09-01

Date 10/24/87

Project: ASV-5/204 Power Separation

INSERT PER PAGE 15A CBP 11/3/87

~~Install MIMIC per existing arrangements (Sheet 3 of 15 Sht)~~ ~~Install demarcation boarders (Reference B/M Item No. 81) per Sketch No. 24.~~ RRF 11/13/87

19. Wiring instructions for MCB Section PSA/EFIC, Sketch Nos. 23, 24, ~~25~~ 23A, and 26. CBP 11/13/87

a. Install #18 AWG SIS red wiring as follows:

From	To
TM1-1-Sketch 23	TB59-1?-Sketch 24
TM1-2-Sketch 23	TB56-1-Sketch 24
TM1-2-Sketch 23	TM2-2-Sketch 23
TM2-?-Sketch 23	TM3-2-Sketch 23
TM2-1-Sketch 23	TB59-2?-Sketch 24
TM3-1-Sketch 23	TB59-2?-Sketch 24

b. Install #14 AWG SIS red wiring as follows:

From	To
TM4-3-Sketch 23	TB56-2-Sketch 24
TM4-3-Sketch 23	TM4-7-Sketch 23
TM4-4-Sketch 23	TB59-15-Sketch 24
TM4-8-Sketch 23	TB59-3-Sketch 24

c. Remove #14 AWG SIS green wire as follows:

RRC 11/13/87 CBP 11/13/87 11/13/87

From	To
AE4-11-Sketch ~~25~~ 23A	TB40-3-Sketch 26
AE4-9-Sketch ~~25~~ 23A	TB40-2-Sketch 26
AE4-10-Sketch ~~25~~ 23A	TB40-1-Sketch 26

d. Install #14 AWG SIS gray wire as follows:

From	To
TB40-5-Sketch 26	TM4-1-Sketch 23
TM4-2-Sketch 23	AE4-11-Sketch ~~25~~ 23A

CBP 11/13/87 11/13/87

Design Engineer	Date	Verification Engineer	Date	Supervisor Nuclear Engineering	Date
D.A. Rhodes	10/24/87	H. D. Graff	11/3/87	Hladki for REV	10/24/87

Rev. 1/87

RET: 25 Yr. RESP: Nuc Eng. 912141

CR3 085396

Florida Power

ENGINEERING INSTRUCTIONS
Crystal River Unit 3

Sheet 15A of 17

Request No.	Date
T87-10-09-01	11/13/87

Project: ASV-5/204 POWER SEPARATION

18. e. DETERMINATE, THE SWITCHBOARD WIRING FROM DEVICES AE1, AE2, AE3/AEN, AQ, AP, AN AND AM. REMOVE THESE DEVICES ~~[illegible]~~ FROM THE MAIN CONTROL BOARD. RETAIN THEM FOR REINSTALLATION IN STEP 18K.

f. FABRICATE A COVER PLATE FOR THE AFFECTED AREA OUT OF #10 GAGE STEEL PLATE. THE PLATE SHALL BE 3.5 ± 0.1 IN HIGH AND 3.5 ± 0.1 IN WIDE.

g. LOCATE THE PLATE ON THE MAIN CONTROL BOARD PER AEA, SKETCH 21. LOCATE AND PUNCH HOLES FOR DEVICES AE1, AE2, AE3, AQ, AP, AN AND AM PER SKETCH 21. THE SWITCH FOR ASV-5 SHALL BE VERTICALLY ALIGNED WITH THE SWITCH FOR ASV-204. THE EFIC CH B TO EFP-2 START LOGIC LIGHTS SHALL BE VERTICALLY ALIGNED TO THE SIMILAR LIGHTS ABOVE FOR CHANNEL A. HORIZONTAL LOCATIONS SHOULD BE APPROXIMATELY AS SHOWN.

h. USING THE PREPARED COVER PLATE AS A TEMPLATE, CUT OR DRILL HOLES IN THE MAIN CONTROL BOARD TO MATCH.

i. PAINT THE COVER PLATE TO MATCH THE MAIN CONTROL BOARD. SEE THE PAINT SPECIFICATION IN MAR 80-08-10-11

j. INSTALL THE COVER PLATE ON THE PANEL USING INSTRUCTIONS FOUND IN THE NOTES OF DRAWING E-201-334, REV 1.

K. INSTALL DEVICES AE1, AE2, AE3, AEN, AQ, AP, AN AND AM AND RETERMINATE THE WIRING PER SKETCH 23A

Design Engineer	Date	Implementation Engineer	Date	Supervisor, Nuclear Engineering	Date
C. B. Doyle	11/13/87	[signature]	11/13/87	[signature]	11/13/87

CR3 085397

MAR / REI No. -	T87-10-09-01	Date	10/24/87
Project:	ASV-5/204 Power Separation		

e. Terminate circuit ASE 36 as follows:

Circuit No.	Wire Color/Wire Mark	Terminal Number
ASE 36	Black (1)/8	TB56-2-Sketch 24
	White (2)/9	TB56-1-Sketch 24
	Red (3)/13	TB59-8-Sketch 24
	Green (4)/10	TB59-15-Sketch 24
	Orange (5)/16	TB59-16-Sketch 24
	Blue (6)/18	TB59-26-Sketch 24
	White/Black (7)/20	TB59-27-Sketch 24
	Red/Black (8)/ --	Spare
	Green/Black (9)/ --	Spare

20. Demolition of cable and conduit for ASV-5 and ASV-204

a. Remove the following cables per E/I Sketch A and their respective cable pulling data sheets.

ASE 30
ASE 28
ASF 31
ASE 27

b. Remove the following conduits per E/I Sketch A

ASK 1-1/2"
ASE 28-3/4"
ASE 27-1"
ASF 31-1"
ASE 30-1 1/2"

c. Plug all holes in motor starter ASV-204, Terminal Box AS-1, Terminal Box AS-9, motor starter ASV-5 and MOV ASV-5.

21. REMOVAL INSTRUCTIONS FOR THIS TEMPORARY MAR TO BE ISSUED WHEN THE MAR IS ISSUED FOR THE PERMANENT SOLUTION TO THE DIESEL PROBLEM. GGD 11/13/87

11/13/87

Design Engineer	Date	Verification Engineer	Date	Supervisor Nuclear Engineering	Date
D.A. Rhonda	10/29/87	[signature]	10/30/87	[signature] for RV	10/30/87

Florida Power Corporation

ENGINEERING INSTRUCTIONS
Crystal River Unit 3

MAR/REI No.	Date
T87-10-09-01	10/24/87

Project: ASV-5/204 Power Separation

F. EXAMINATION

Visually inspect component installations and wiring terminations to assure proper installation and termination in accordance with the MAR sketches and maintenance procedures.

G. TESTING

Upon completion of this modification, motor operated valves ASV-204 and ASV-5 shall be functionally tested to assure proper valve operation, and all limit switch functions (i.e., indicator lights, alarms, interlocks) shall be monitored to assure that the valve control logic has not been altered by this modification. After installation, all wiring shall be tested for continuity and insulation integrity, as applicable, in accordance with FPC standard procedures.

H. IN-SERVICE INSPECTION

Not applicable.

Design Engineer	Date	Verification Engineer	Date	Supervisor Nuclear Engineering	Date
[signature]	10/29/87	[signature]	10/29/87	[signature]	10/30/87

RET 25 Yr RESP Nuc Eng 912241

Florida Power

PROJECT ASSIGNMENT
Nuclear Engineering

REVMAR NUMBER
MAR T87-10-09-01

PROJECT TITLE
ASV-5/204 POWER SEPARATION

The below listed personnel are assigned the responsibility for the subject project:

DESIGN ENGINEER(S)

C/CI (Lead)

R. E. Olson

VERIFICATION ENGINEER(S)

/CI.

C. Iwachow

COMMENTS

SUPERVISOR, NUCLEAR ENGINEERING
C. B. Doyle

DATE
11/11/87

Original: Lead Design Engineer
cc: Other Design/Verification Engineers
File: REVMAR Number
Supervisor (w/REVMAR form only)

Florida Power

DESIGN DATA SHEET
Crystal River Unit 3

SHEET 1 OF 2

NO / MAR NUMBER	DATE
T87-10-09-01	OCTOBER 14, 1987
PROJECT	**SYSTEM**
ASV-5/204 POWER SEPARATION	AS

SAFETY CRITERIA: [X] Safety Listing Rev. 24 Dated 7-10-87 Page I-15
[X] Safety Classification Review Form (attach copy)

SAFETY RELATED: [X] Yes [ ] No

	YES	NO	APPLICABLE DESIGN INPUT REQUIREMENTS
1.	[X]	[ ]	Basic Functions of each structure, system and component.
2.	[X]	[ ]	Performance requirements such as capacity, rating, system output.
3.	[X]	[ ]	Codes, standards, and regulatory requirements including the applicable issue and/or addenda.
4.	[X]	[ ]	Design conditions such as pressure, temperature, fluid chemistry and voltage.
5.	[X]	[ ]	Loads such as seismic, wind, thermal and dynamic.
6.	[X]	[ ]	Environmental conditions anticipated during storage, construction and operation such as pressure, temperature, humidity, corrosiveness, site elevation, wind direction, nuclear radiation, electromagnetic radiation and duration of exposure. 10CFR50.49 applicability — For electrical equipment only, reference above Safety Listing page or attach copy of Environmental Qualification Requirements form.
7.	[X]	[ ]	Interface requirements including definition of the functional and physical interfaces involving structures, systems and components.
8.	[X]	[ ]	Material requirements including such items as compatibility, electrical insulation properties, protective coating and corrosion resistance.
9.	[ ]	[X]	Mechanical requirements such as vibration, stress, shock and reaction forces.
10.	[ ]	[X]	Structural requirements covering such items as equipment foundations and pipe supports.
11.	[ ]	[X]	Hydraulic requirements such as pump net positive suction heads (NPSH), allowable pressure drops, and allowable fluid velocities.
12.	[ ]	[X]	Chemistry requirements such as provisions for sampling and limitations on water chemistry.
13.	[X]	[ ]	Electrical requirements such as source of power, voltage, raceway requirements, electrical insulation and motor requirements.
14.	[X]	[ ]	Layout and arrangement requirements, to include potential adverse affects of non-seismically qualified masonry walls.
15.	[X]	[ ]	Operational requirements under various conditions such as plant startup, normal plant operation, plant shutdown, plant emergency operation, special or infrequent operation, and system abnormal or emergency operation.
16.	[X]	[ ]	Instrumentation and control requirements including indicating instruments, controls and alarms required for operation, testing, and maintenance. Other requirements such as the type of instrument, installed spares, range of measurement, and location of indication should also be included.
17.	[ ]	[X]	Access and administrative control requirements for plant security.
18.	[X]	[ ]	Redundancy, diversity and separation requirements of structures, systems and components.
19.	[X]	[ ]	Failure effects requirements of structures, systems and components, including a definition of those events and accidents which they must be designed to withstand.
20.	[X]	[ ]	Test requirements including in-plant tests and the conditions under which they will be performed.
21.	[ ]	[X]	Accessibility, maintenance, repair and inservice inspection requirements for the plant including the conditions under which these will be performed.

DESIGN ENGINEER	DATE	VERIFICATION ENGINEER	DATE	SUPERVISOR, NUCLEAR ENGINEERING	DATE
[illegible]	[illegible]	[illegible]	10/21/87	[illegible] for [illegible]	[illegible]

Florida Power

DESIGN DATA SHEET

Crystal River Unit 3

SHEET 2 OF 2

MAR/MAN NUMBER	T87-10-09-01	DATE OCTOBER 14, 1987
PROJECT	ASV-3/204 POWER SEPARATION	SYSTEM AS

YES NO **APPLICABLE DESIGN INPUT REQUIREMENTS: (Continued)**

22. ☒ ☐ Personnel requirements and limitations including the qualification and number of personnel available for plant operation, maintenance, testing and inspection and permissible personnel radiation exposures for specified areas and conditions. (ALARA)
23. ☐ ☒ Transportability requirements such as size and shipping weight, limitations, I.C.C. regulations.
24. ☐ ☒ Fire protection or resistance requirements: (Check applicable letter)
    - ☐ a. Changes or additions/deletion of fire detection suppression systems or equipment.
    - ☐ b. Changes or additions to the plant configuration that change the effectiveness of existing fire detection/suppression systems.
25. ☐ ☒ Handling, storage and shipping requirements
26. ☐ ☒ Other requirements to prevent undue risk to the health and safety of the public.
27. ☒ ☐ Materials, processes, parts and equipment suitable for application.
28. ☒ ☐ Safety requirements for preventing personnel injury including such items as radiation hazards, restricting the use of dangerous materials, escape provisions from enclosures, and grounding of electrical systems.
29. ☐ ☒ Addition or relocation of safe shutdown equipment, systems, components, or circuits that require compliance with the separation criteria stated in 10CFR50, Appendix R.
30. ☐ ☒ The modification will modify an item that: (Check applicable letter(s))
    - ☐ a. was originally procured and installed in an unmodified state and for which spare parts are stocked; or,
    - ☐ b. is being procured as a part of this modification and will be modified after receipt; or,
    - ☐ c. is currently in inventory (FIMIS) in an unmodified state and will be modified prior to installation and/or restocking
    - ☐ d. will be completely or partially replaced by an item of different design or materials.

    If "Yes" to any of the criteria, forward a completed copy of the Design Data Sheet, Design Input Record, Engineering Instructions, PR/BOM/material specifications (as applicable) and other applicable supporting documentation (Instruction Manuals, Vendor Drawings, FPC Drawings, Sketches, Figures, etc.) to Site Nuclear Procurement Engineering
31. ☒ ☐ The modification adds or relocates control room equipment entailing operator interface that requires Human Factors design review per the criteria stated in NUREG-0700 (Ref: FPC Specification SP-5145)

CR3 085402

DESIGN ENGINEER	DATE	VERIFICATION ENGINEER	DATE	SUPERVISOR NUCLEAR ENGINEERING	DATE
[illegible signature]	[illegible]	[signature]	10/21/87	[signature] for [illegible]	10/30/87

Florida Power Corporation

DESIGN INPUT RECORD
Crystal River Unit 3

Sheet 1 of 5

REI/MAR No. MAR T87-10-09-01 | Date OCTOBER 14, 1987

Project: ASV-5/204 POWER SEPARATION

1. The function of this modification is to assure that the turbine driven emergency feedwater pump EFP-2 will be operable in the event of a failure of the ESB 250/125V DC system coincident with loss-of-offsite-power and an ES actuation. Under this scenario EFP-2 will be relied upon to share the emergency feedwater load with the motor driven emergency feedwater pump in order to decrease the electrical load on diesel generator EDG-1A. In order to assure that EFP-2 is operable for the above condition, this modification removes the 250/125VDC ES "B" power from ASV-204 and repowers it from 250/125VDC ES "A" power. This involves separating the ASV-5 and ASV-204 power and control circuits which are presently wired in parallel, and providing a separate control switch, indicator lights, and EFIC interlocks for ASV-204.

2. The capacity and rating of ASV-204 will not be changed by this modification. However, it will not be required to comply to the separation requirements of 10CFR50, Appendix R, because ASV-5 will remain in compliance. Therefore, remote shutdown isolation will not be required for ASV-204 circuitry.

3. IEEE-323-1974
   IEEE-344-1975
   IEEE-383-1974
   10CFR50, Appendix R

Design Engineer	Date	Verification Engineer	Date	Supervisor, Nuclear Engineering	Date
D G Rhode	10/29/87	[signature]	10/30/87	[signature] for KEV	10/30/87

Rev. 1/87

RET Life Of Plant RESP: Nuclear Engineering

CR3 085403

Florida Power Corporation

DESIGN INPUT RECORD
Crystal River Unit 3

REI/MAR No.	MAR T87-10-09-01	Date	OCTOBER 14, 1987
Project:	ASV-5/204 POWER SEPARATION		

4. Power and control voltage to ASV-204 shall remain at 250/125 VDC.

5. New circuits which will be classified as safety related shall be routed in seismically qualified raceway. Relay racks and the main control board shall be reviewed for seismic integrity to account for the addition of any new components.

6. This modification involves wiring changes at ASV-204 and ASV-5 (Environmental Zone 14), adds a control switch and indicator lights in the main Control Room and adds two relays in the Relay Room (Environmental Zones 13 and 58, respectively). The zone environmental data sheets defining the environmental parameter requirements for the above zones are attached.

7. The control circuit for ASV-204 shall interface with the EFIC System as follows:

   a. A normally open EFIC actuation contact shall be wired in parallel with the control switch contact in the valve "opening" circuit to provide automatic opening of ASV-204 when EFIC is actuated.

   b. A normally closed EFIC actuation contact shall be wired in series with the control switch contact in the valve "closing" circuit to interrupt the closing of ASV-204 when EFIC is actuated.

Design Engineer	Date	Verification Engineer	Date	Supervisor, Nuclear Engineering	Date
E C Pikade	10/20/87	H. Shaeff	10/30/87	[illegible]	10/30/87

Florida Power Corporation

DESIGN INPUT RECORD
Crystal River Unit 3

REI/MAR No.	MAR T87-10-09-01	Date	OCTOBER 16, 1987
Project:	ASV-5/204 POWER SEPARATION		

12. Safety related electrical cables shall meet the requirements of IEEE-383-1974. New components required for this modification such as relays, and control switches shall be qualified for the application in which they are to be used.

13. Power for ASV-204 shall be 250 VDC ES Channel "A" power for the motor operator and 125 VDC ES Channel "A" power for the valve control circuit. The circuits shall be routed in accordance with CR3 ES Channel Separation Criteria. The motor starter space heater for ASV-204 is presently powered in parallel with the starter space heater for ASV-5 from a common non-1E 120 VAC power source. Although space heaters are non-1E, in order to avoid any circuitry common to the starters for ASV-5 and ASV-204, the ASV-204 starter space heater shall be powered from a separate non-1E 120 VAC source.

14. The control switch and indicator lights for ASV-204 shall be located on the PSA/EFIC Section of the Main Control Board in ~~the space where pushbutton switch "AD2" is mounted.~~ a existing spared switch cutout. Switch "AD2" is being removed via MAR T87-10-19-01, and a [illegible] blanked in [illegible]

15. With this modification, ASV-204 is required to be operational for a loss-of-power condition coincident with an ES actuation and a single failure of the 125 VDC ES Channel "B" power system.

Design Engineer	Date	Verification Engineer	Date	Supervisor, Nuclear Engineering	Date
[illegible]	10/29/87	[illegible]	10/30/87	[illegible]	10/30/87

Florida Power Corporation

DESIGN INPUT RECORD
Crystal River Unit 3

REI/MAR No.	MAR T87-10-09-01	Date	OCTOBER 14, 1987
Project:	ASV-5/204 POWER SEPARATION		

16. Remote manual control and indication shall be provided from the main control room via a control switch and red, green, and amber position indicator lights. The manual control shall be such that the control switch must be held in the open or close position to permit valve travel. Automatic control of ASV-204 shall be as described in Item 7 above. Existing alarms utilizing ASV-204 and ASV-5 control logic shall be maintained. Auxiliary relays shall be used, as required, to provide the alarm logic from ASV-204.

18. Reference DIR Items 2 and 13 above.

19. The failures and events that this modification is to be designed to withstand are as described in DIR Items 5 and 15 above.

20. Manual and automatic operation of ASV-204 and ASV-5 including alarms and indicator lights (and remote shutdown operation of ASV-5) shall be tested upon completion of this modification.

22. Cable routing design and component location shall consider maintaining doses ALARA.

27. Cables, relays, controls, indicator lights, and any other components to be installed by this modification shall be qualified to the seismic and/or environmental criteria applicable to the equipment or areas in which they are to be installed.

Design Engineer	Date	Verification Engineer	Date	Supervisor, Nuclear Engineering	Date
D.G. P. Bowie	10/29/87	[signature]	10/30/87	[signature] for [illegible]	10/30/87

Rev. 7/87

RET: Life Of Plant RESP: Nuclear Engineering

CR3 085406

Florida Power Corporation

DESIGN INPUT RECORD
Crystal River Unit 3

REI/MAR No.	MAR T87-10-09-01	Date	OCTOBER 14, 1987
Project:	ASV-5/204 POWER SEPARATION		

30. Existing plant electrical grounding procedures shall apply.

31. With the implementation of this modification, there will be separate control switches and indicator lights for valves ASV-5 and ASV-204 in lieu of just one common switch and lights that presently exists for both valves. The new switch and indicator lights for ASV-204 shall be located per Specification SP-5145, Human Factors Design Conventions.

Design Engineer	Date	Verification Engineer	Date	Supervisor, Nuclear Engineering	Date
A. A. F[illegible]	10/29/87	[illegible signature]	10/30/87	[illegible] for RSV	[illegible]

Florida Power

SAFETY CLASSIFICATION REVIEW
Crystal River Unit 3

ITEM	COMPONENT
Valve Motor Operator ASV-204	Turbine Driven Emergency Feedwater Pump Inlet Isolation Valve

Generic ☐ Yes ☒ No Application

SYSTEM	REQ. NO.	P.O. NO.	[illegible] NO.
AS	N/A	N/A	N/A

Answer each question:

1. Does the item/service assure the integrity of the reactor coolant system boundary (i.e., "Pressure-retaining" as defined in ASME Boiler and Pressure Vessel Code)? Yes ☐ No ☐

Justification: (Answer the following questions - If both are Yes, check "Yes" above, otherwise check "No".

	Yes	No
• Is the item/service a part of reactor coolant system?	☐	☐
• Is the item/service pressure-retaining per Code?	☐	☐

Comments: N/A

2. Does the item/service assure the capability to shut down the reactor and to maintain it in a safe shutdown condition? Yes ☐ No ☐

Justification: N/A

3. Does the item/service assure the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those referred to in 10CFR100.11? Yes ☐ No ☐

Justification: N/A

Is the answer to any question above "Yes"?

☐ Yes → Item/Service is Safety-related

☐ No → Item/Service is non Safety-related

NOTE:
If item is considered to be electrical equipment, the Environmental Qualification Requirements Review form must be completed to determine 10CFR50.49 applicability.

DESIGN ENGINEER	DATE	VERIFICATION ENGINEER	DATE	SUPERVISOR NUCLEAR ENGINEERING	DATE
D Q Rhode	10/29/87	[signature]	10/30/87	[signature]	10/[illegible]

CR3 085408

Florida Power Corporation

ANALYSIS / CALCULATION
Crystal River Unit 3

SHEET 1 OF 1

REI/MAR No.	MAR T87-10-09-01	Date	OCTOBER 14, 1987
Project:	ASV-5/204 POWER SEPARATION		

ATTACHMENT TO "SAFETY CLASSIFICATION REVIEW" FOR VALVE MOTOR OPERATOR ASV-204

Safety Listing Revision 24, Page 1-15 lists valve motor operator ASV-204 as Safety Channel B. The modification of MAR T87-10-09-01 will change the motor and control power to ASV-204 from ES "B" 250/125 VDC to ES "A" 250/125 VDC. Consequently, the channel designation for ASV-204 on Page 1-15 of the Safety Listing must be changed from "B" to "A".

Design Engineer	Date	Verification Engineer	Date	Supervisor, Nuclear Engineering	Date
D O Rhode	10/20/87	[signature]	10/23/87	[signature] for [illegible]	10/2[illegible]

CR3 085409

Florida Power

VERIFICATION REPORT
Crystal River Unit 3

T87-10-09-01 | Date: 10/29/87

PROJECT: ASV-5/204 Power Separation

VERIFICATION METHOD: ☑ Design Review (see checklist below) ☐ Alternate Calculations (attach analysis calculation sheet) ☐ Qualification Testing (see attachments as necessary)

	YES	NO	N/A	
1.	☑	☐		Were the inputs correctly selected and incorporated into design?
2	☑	☐	☐	Are assumptions necessary to perform the design activity adequately described and reasonable? Where necessary, are the assumptions identified for subsequent reverifications when the detailed design activities are completed?
3	☑	☐		Are the appropriate quality and quality assurance requirements specified?
4	☑	☐		Are the applicable codes, standards and regulatory requirements including issue and addenda properly identified and are their requirements for design met?
5	☑	☐	☐	Have applicable construction and operating experience been considered?
6	☑	☐	☐	Have the design interface requirements been satisfied?
7	☑	☐		Was an appropriate design method used?
8.	☑	☐		Is the output reasonable compared to inputs?
9	☑	☐	☐	Are the specified parts, equipment, and processes suitable for the required application?
10	☑	☐	☐	Are the specified materials compatible with each other and the design environmental conditions to which the material will be exposed?
11	☑	☐	☐	Have adequate maintenance features and requirements been specified?
12	☑	☐	☐	Are accessibility and other design provisions adequate for performance of needed maintenance and repair?
13	☐	☐	☑	Has adequate accessibility been provided to perform the in-service inspection expected to be required during the plant life?
14	☑	☐	☐	Has the design properly considered radiation exposure to the public and plant personnel?
15	☑	☐	☐	Are the acceptance criteria incorporated in the design documents sufficient to allow verification that design requirements have been satisfactorily accomplished?
16	☑	☐	☐	Have adequate pre-operational and subsequent periodic test requirements been appropriately specified?
17	☑	☐	☐	Are adequate handling, storage, cleaning and shipping requirements specified?
18	☑	☐	☐	Are adequate identification requirements specified?
19	☑	☐	☐	Are requirements for record preparation review, approval, retention, etc. adequately specified?
20	☑	☐	☐	Has seismic adequacy been considered and evaluated and are the results acceptable?
21	☑	☐	☐	Have fire protection considerations been adequately addressed?

I have performed a verification on the subject MAR design package and find the results
☑ acceptable ☐ unacceptable

COMMENTS

Verifier: [signature] Date: 10/30/87 | Supervisor Nuclear Engineering: [signature] for CCV Date: 10/30/87

Gilbert Commonwealth Companies

POWER AND INDUSTRIAL SYSTEMS DIVISION – READING
DESIGN VERIFICATION RECORD

PAGE 1 OF

A PROJECT: CRYSTAL RIVER UNIT 3 MAR T87-10-09-01

SUBJECT: ASV-5/204 Power Separation 2D • DV-5510-121-EE

SECTION NAME AND NUMBER: Elect. Engr. / 0421

W.O.: 04-5510-121

D. A. Rhoads
ORIGINATOR

R. P. Cronk
PROJECT ENGINEER

THIS DOCUMENT CONTAINS PRELIMINARY DATA/ASSUMPTIONS:

NO ✓ YES ______ PAGE(S) ______

A COMPUTER PROGRAM WAS:

✓ NOT USED ______ USED (CERTIFIED PER CAM) ______ USED (NOT CERTIFIED–TO BE VERIFIED WITH CALCULATION)

PROGRAM SYSTEM NAME	REV.		REV.
(1)		(4)	
(2)		(5)	
(3)		(6)	

VERIFICATION PACKAGE (IDENTIFY EACH ITEM)

DOCUMENTS TO BE VERIFIED	REV.		REV.
(1) CWP for MAR T87-10-09-01		(4)	
(2)		(5)	
(3)		(6)	

SUPPORTING DOCUMENTS	REV.		REV.
(1) As listed in Section 5		(7)	
(2) of Engineering Instructions		(8)	
(3)		(9)	
(4)		(10)	
(5)		(11)	
(6)		(12)	

D. A. Rhoads
ORIGINATOR'S SIGNATURE

10/24/87
DATE

B NO VERIFICATION REQUIRED PER DCP 2.05

REASON:

VERIFICATION REQUIRED (CHECK METHOD(S))

DESIGN REVIEW ✓ ALTERNATE CALCULATION ______ QUALIFICATION TESTING ______

IDENTIFICATION OF VERIFIER/~~VERIFICATION TEAM~~: H. I. Grace, EE

PROJECT ENGINEER'S SIGNATURE

10/30/87
DATE

PAGE 2 OF

C. CONCURRENCE WITH SELECTION OF VERIFIER(S):

SECTION MANAGER'S SIGNATURE

10/30/87
DATE

D. EXTENT OF VERIFICATION:

Documents were verified for conformance to FPC SREP's and G/C, Inc. DCP's and that the modification meets the design intent. Documents were reviewed for accuracy of inputs and correct design methodology.

RESULTS OF VERIFICATION:

Inputs were correctly selected and applied and design methodology is reasonable and applicable and appropriate. Applicable operating conditions have been considered in the design. Outputs are reasonable compared to the inputs. The final product meets the intent of the design. Verification comments have been resolved with the originator. Results are acceptable.

BACK SIDE OF
PREVIOUS DOCUMENT

ATTESTATION:

THIS DESIGN VERIFICATION WAS PERFORMED IN ACCORDANCE WITH DCP 3.3.

H. E. Chaoff
VERIFIER'S SIGNATURE

10/30/87
DATE

E. COMPLETION OF VERIFICATION:

PROJECT ENGINEER'S SIGNATURE

11/3/87
DATE

FIRE PROTECTION REVIEW
Crystal River Unit 3

Florida Power

MAR NUMBER: MAR T87-10-09-01	DATE: OCTOBER 28, 1987
PROJECT: ASV-5/204 POWER SEPARATION	SYSTEM: AS

YES NO

1. Does the design/design change involve the modification, addition, removal, or relocation of any of the following? If yes, explain in space provided.

X ☐ a. Combustibles (oil, hydraulic fluid, grease, wood-based materials, cloth, charcoal, PVC, cable insulation, carpet, etc.)

See Attached Sheet

☐ X b. Available Fire Protection (detectors, fire extinguishers, hose stations, sprinklers, halon system, CO_2 system, etc.)

☐ X c. Equipment, components, or cables that would interfere with the operation of existing fire detection, emergency lighting, or other fire protection features.

X ☐ d. Penetrations, penetration seals, or conduit seals. (If yes, update Penetration Seals List if applicable)

See Attached Sheet

☐ X e. Space Separators (walls, ceilings, floors, doors, curbs, dampers, etc.)

X ☐ f. Fireproofing, exposure fire protection, cable tray covers/wrapping, conduit wrapping, etc.

See Attached Sheet

2 ☐ X Will this design/design change require a revision to the Fire Hazards Analysis? Reference EG-4, Sect. V.C.

3. ☐ X Does this design/design change deviate from any applicable NFPA Fire Code requirement in safety-related or safe shutdown areas/zone? If yes, explain below and obtain Fire Protection Engineer review and concurrence.

QCA 10/23/87

ENGINEER	DATE	FIRE PROTECTION ENGINEER (if concurrence required by item 3)	DATE
D. A. Rhodes	10/28/87		

Florida Power Corporation	ANALYSIS / CALCULATION Crystal River Unit 3	SHEET 1 OF 1
REI/MAR No. MAR T87-10-09-01		Date OCTOBER 28, 1987
Project: ASV-5/204 POWER SEPARATION		

ATTACHMENT TO FIRE PROTECTION REVIEW

Item 1.a.

This modification involves the installation of an approximate total of 3,200 feet of various power, control and instrumentation cable. Most of this cable is routed either in conduit or fire protected cable tray and, thus, does not contribute to the combustible load in the respective fire areas. The only fire areas where some of the above cable is routed in open cable trays are the Cable Spreading Room and the Relay Room. In the Cable Spreading Room, most of the cable tray route is in tray less than 50% full. Since the Fire Hazard Analysis is based on trays 50% filled, the additional cables do not affect the overall combustible load in this area. The cables in the Relay Room are very short runs where as they penetrate the ceiling and drop into the relay racks. The cables consist of a 5/C #14 control cable and two 2/C #16 instrument cables. The amount of combustible load contributed by these cables is considered insignificant.

Item 1.d.

This modification requires a conduit penetration through the Intermediate Building EL-119 floor, which requires resealing to the equivalent rating of the floor. Cables will penetrate through existing sealed penetrations in the following fire barriers:

Intermediate Building Floor EL-119
Control Complex West Wall On EL-134
Control Complex Floors EL-134 And 145

Upon completion of cable installation, these penetrations shall be resealed to the equivalent fire rating of the respective fire barriers.

Item 1.f.

Fire protected cable trays that need to be opened to accommodate cable installation shall be resealed to the required fire rating of the respective trays.

Design Engineer	Date	Verification Engineer	Date	Supervisor, Nuclear Engineering	Date
[illegible signature]	[illegible]	[illegible signature]	11/3/87	[illegible signature]	[illegible]

MODIFICATION SAFETY EVALUATION

MAR NO. T87 - 10 - 09 - 01

SAFETY EVALUATION: Answer the following questions and provide specific justification (use attachment if necessary).

1. Is the probability of an occurrence or the consequence of an accident or malfunction of equipment important to safety as previously evaluated in the Final Safety Analysis Report, INCREASED? YES ___ NO X
Because:
See Attached Sheet

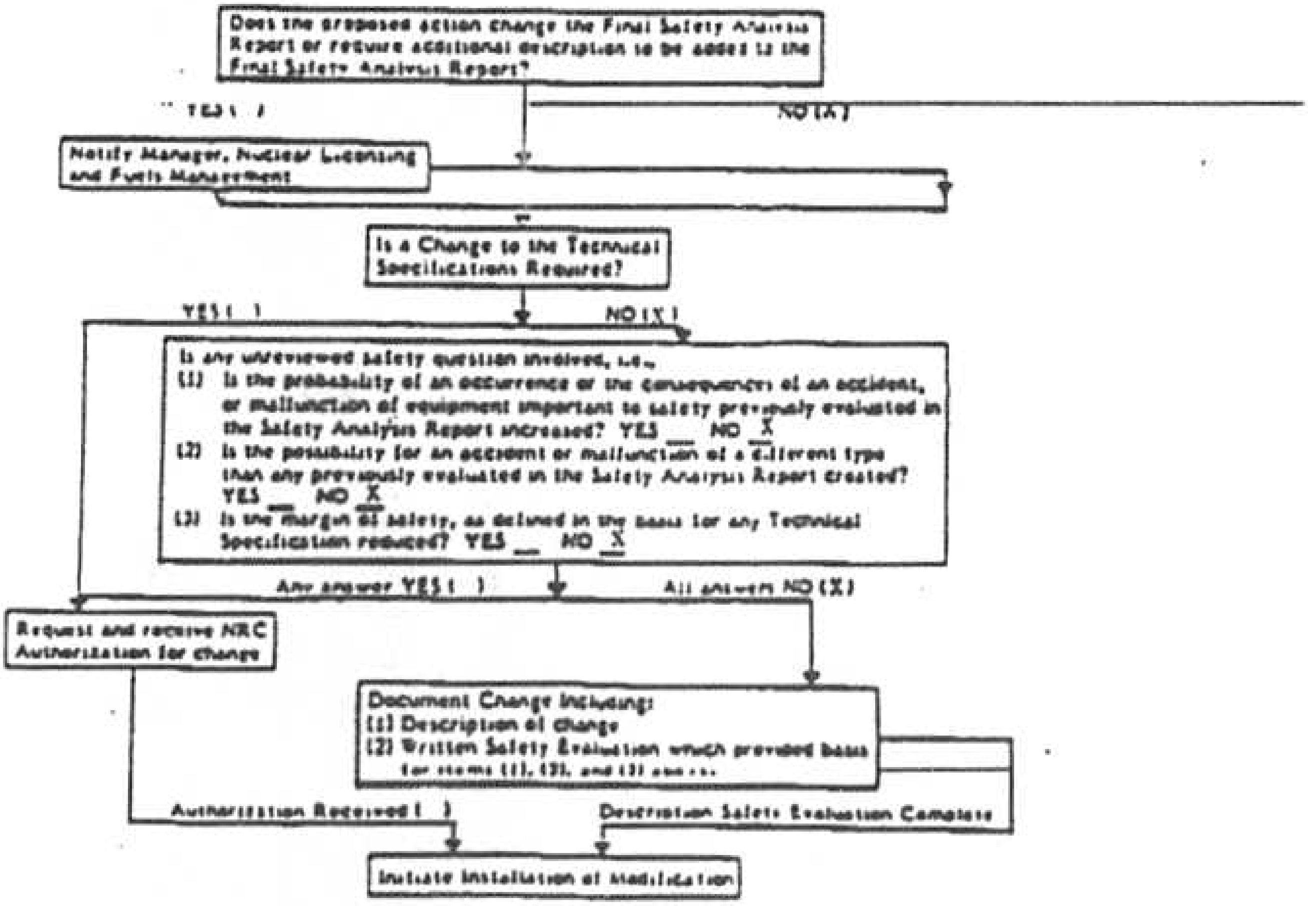
2. Is the possibility for an accident or malfunction of a different type than any previously evaluated in the Final Safety Analysis Report, CREATED? YES ___ NO X
Because:
See Attached Sheet

3. Is the margin of safety, as defined in the basis for any Technical Specification, REDUCED? YES ___ NO X
Because:
See Attached Sheet

LICENSE REVISION REQUIRED:

Final Safety Analysis Report:	YES ___	NO X
Technical Specifications:	YES ___	NO X
NRC Authorization for Change Required:	YES ___	NO X
Semi-Annual Reporting to NRC Required:	YES ___	NO X

10CFR50.59 CHECKLIST



Florida Power Corporation

ANALYSIS / CALCULATION
Crystal River Unit 3

Ref / MAR No.	MAR T87-10-09-01	Date	OCTOBER [illegible], 1987

Project: ASV-5/204 POWER SEPARATION

ATTACHMENT TO MODIFICATION SAFETY EVALUATION

1. ASV-5 and ASV-204 are motor operated valves having identical functions of supplying steam to the turbine driven Emergency Feedwater Pump (EFP-2). Since EFP-2 is the ES "B" channel pump, ASV-5 and 204 were electrically connected in parallel to a common 250/125 VDC ES "B" channel power and control source. This modification electrically separates ASV-204 from ASV-5 and repowers ASV-204 from 250/125 VDC ES "A" channel power. Also, separate control room controls and separate "A" channel EFIC interlocks are being provided for ASV-204. Automatic control logic of ASV-204 has not changed. Therefore, the probability of an occurance or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the FSAR is not increased since the logic of automatically opening ASV-204 whenever the EFIC System calls for emergency feedwater has not been altered. The reliability of EFP-2 has actually been increased because with this modification either "A" or "B" train power will control and operate one of the steam inlet valves to EFP-2 as opposed to both valves being "B" train powered. FSAR Sections 7.2.4, 8.2.2.6 and 10.2.1.6 have been reviewed.

2. The electrical separation of ASV-204 from ASV-5 does not impact the design function of either valve to supply steam to the EFP-2 turbine. Power and control for ASV-5 is not affected by this modification and ASV-5 retains its automatic control logic, remote manual control, local manual control and remote shutdown isolation and control. ASV-204 is being powered from the redundant power channel, and will be provided with its own remote manual control and with separate EFIC interlocks for automatic operation. The type of remote manual control and automatic operation of ASV-204 is the same as for ASV-5. Therefore, based on the above, the possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR is not created. FSAR Sections 7.2.4, 8.2.2.6, and 10.2.1.6 have been reviewed.

Design Engineer	Date	Verification Engineer	Date	Supervisor, Nuclear Engineering	Date
[illegible]	[illegible]	[illegible]	10/2/87	[illegible]	[illegible]

Florida Power Corporation

ANALYSIS / CALCULATION
Crystal River Unit 3

MAR No.	MAR T87-10-09-01	Date	OCTOBER 19, 1987
Project	ASV-3/204 POWER SEPARATION		

3. This modification enables the turbine driven Emergency Feedwater Pump (which is the "B" channel pump) to be operational even if a failure should occur on the "B" channel power system for which shutdown operation would be via the "A" channel systems. With this capability, the turbine driven EFW pump is able to operate and share the EFW requirements with the "A" channel motor driven EFW pump. This will reduce the electrical load on the "A" channel diesel generator for the condition of an ES actuation coincident with a loss-of-offsite-power and failure of the "B" channel power system. Consequently, with this modification the margin of safety, as defined in the basis for any Technical Specification, is not reduced. It is actually enhanced because of the increased availability of the turbine driven Emergency Feedwater Pump. Technical Specification Sections 3/4.7.1 and 3/4.8.1 have been reviewed.

Design Engineer	Date	Verification Engineer	Date	Supervisor, Nuclear Engineering	Date
N G Plunk	10/24/87	[signature]	10/30/87	[signature]	11/3/87

CR3 085417

REGULATORY/ENVIRONMENTAL REVIEW
Crystal River Unit 3

Florida Power

DOCUMENT NUMBER

MAR T87-10-09-01

REF. NUCLEAR OPERATIONS DEPARTMENT PROCEDURE NOD-11

1. **10 CFR 50.54 Reviews**

   Does this modification or document revision change what is described in any of the following plans/programs? If unable to determine, contact responsible person designated below.

Quality Program Description (FSAR Section 1.7) Contact: Director, Quality Programs	☐ Yes	☑ No
Licensed Operator Requalification Program (FSAR Section 12.2.3.4 and Appendix 12 C) Contact: Manager, Nuclear Operations Training	☐ Yes	☑ No
Modified Amended Security Plan Contact: Manager, Nuclear Licensing	☐ Yes	☑ No
Safeguards Contingency Plan Contact: Nuc. Security & Special Project Superintendent	☐ Yes	☑ No
Security Guard Training and Qualification Plan Contact: Nuc. Security & Special Project Superintendent	☐ Yes	☑ No
Radiological Emergency Response Plan Contact: Manager, Site Nuclear Services	☐ Yes	☑ No

   If any are "yes":

   a. Contact appropriate responsible person identified above to perform the evaluation and attach. The evaluation must be approved by the Division Head responsible for the plan/program. — Evaluation ☐ Complete ☑ N/A

   b. NRC approval received if needed and attach. — ☐ Complete ☑ N/A

2. **Environmental Protection Plan Review**

   Could this change affect the environment in a non-radiological way? — ☐ Yes ☑ No

   If "yes", contact the Manager, Nuclear Licensing — ☐ Complete ☑ N/A

3. **Review for change to Radioactive Waste System (10 CFR 50.34a and Appendix I / T. S. 6.16.1.1)**

   Will this change to a radioactive waste system (liquid, gaseous or solid) result in an increase of radioactive material released to the environment? — ☐ Yes ☑ No

   If "yes", submit change to Manager, Nuclear Licensing to evaluate the reporting requirements. — ☐ Submitted ☑ N/A

IF ALL OF ABOVE ARE CHECKED NO, NO FURTHER REVIEWS ARE REQUIRED.

PREPARED BY	DATE	APPROVED BY	DATE
R. S. [signature]	11/10/87	[signature] / C. G. [signature]	11/13/87 / 11/11/87

Rev. 1/87

(REF. Gen. BESP Nuclear Engineering)

Gilbert Commonwealth Companies

POWER AND INDUSTRIAL SYSTEMS DIVISION – READING
DESIGN VERIFICATION RECORD

A PROJECT: CRYSTAL RIVER UNIT 3 DC-5510-126EE

SUBJECT: ADDITION OF ASV-204 TO DPDP-8A

SECTION NAME AND NUMBER: Elect. Engr. /0421

W.A. 04-5410-126

D. A. Rhoads
ORIGINATOR

R. P. Crink
PROJECT ENGINEER

THIS DOCUMENT CONTAINS PRELIMINARY DATA/ASSUMPTIONS:

NO ✓ YES ______ PAGE(S) ______

A COMPUTER PROGRAM WAS:

✓ NOT USED ______ USED (CERTIFIED PER CAM) ______ USED (NOT CERTIFIED—TO BE VERIFIED WITH CALCULATION)

PROGRAM SYSTEM NAME	REV.		REV.
(1)		(4)	
(2)		(5)	
(3)		(6)	

VERIFICATION PACKAGE (IDENTIFY EACH ITEM)

DOCUMENTS TO BE VERIFIED	REV.		REV.
(1) DC-5510-126EE	0	(4)	
(2)		(5)	
(3)		(6)	

SUPPORTING DOCUMENTS	REV.		REV.
(1) As Listed in 'Design Inputs'		(7)	
(2) of DC-5510-126EE		(8)	
(3)		(9)	
(4)		(10)	
(5)		(11)	
(6)		(12)	

D. A. Rhoads
ORIGINATOR'S SIGNATURE

10/24/87
DATE

B NO VERIFICATION REQUIRED PER DCP 2.05

REASON: ______

VERIFICATION REQUIRED (CHECK METHOD(S))

DESIGN REVIEW ✓ ALTERNATE CALCULATION ______ QUALIFICATION TESTING ______

IDENTIFICATION OF VERIFIER/~~VERIFICATION TEAM~~ H. T. Graeff

[signature]
PROJECT ENGINEER'S SIGNATURE

10/24/87
DATE

CR3 085410

Gilbert/Commonwealth CALCULATION	SUBJECT: CRYSTAL RIVER UNIT 3 ADDITION OF ASV-204 TO DPDP-8A	IDENTIFIER: DC-5370-[illegible]26.0-EE	PAGE: [illegible]3
	REV. 0		OF 4
	ORIGINATOR: R. A. Rhoads		
	DATE: 10/20/87		

1. PURPOSE

Review the 250/125VDC ES 'A' power distribution system to varify that sufficient copacity is available to power motor operated valve ASV-204 from distribution panel DPDP-8A.

2. DESIGN INPUTS

A. Scope Document FC3-8915 dated 10/13/87
B. Calculation DC-CR3-017-EE, Rev. 0
C. Calculation DC-5079-275.0-EE, Rev. 0
D. Motor Data Sheet for ASV-5

3. COMPUTER PROGRAMS USED

None

4. ASSUMPTIONS

None

5. CALCULATION

Per D.I. 2.C above, the electrical requirements of ASV-204 are identical to ASV-5. The motor data sheet for ASV-5 indicates a full load current of 2.9 amps and a locked rotor current of 16.2 amps.

ASV-204 is being powered from distribution panel DPDP-8A.

Per page 29 of 40 of Calculation DC-CR3-017-EE, Rev. 0 (D.I. 2.B) the load on DPDP-8A (without ASV-204) is 94.5 amps for the first minute of the battery duty cycle and 4.5 amps after the first minute.

Allowing one amp for control circuit power and conservatively assuming that the motor current for the first minute is equal to the LRA value, the addition of ASV-204 (which will operate during the first minute of the battery duty cycle) increases the load on DPDP-8A to 111.7 amps for the first minute and 5.5 amps after the first minute. This is well within the 400 amp continuous rating of DPDP-8A.

Per page 34 of 40 of Calculation DC-CR3-017-EE, Rev. 0 (D.I.2.B) the total Ampere-Hour load on the 250/125VDC ESA system is 1111A-H without the addition of ASV-204. With the addition of ASV-204 to DPDP-8A the total Ampere-Hours becomes 1113 A-H (Ref. DC-CR3-017-EE, Rev 1). The rating of the 250/125VDC ESA battery is 1590A-H.

6. CONCLUSION

Based on the above analysis the new total load on the 250/125VDC ESA system with the addition of ASV-204 to DPDP-8A is well within the system capacity.

Gilbert Commonwealth Companies

POWER AND INDUSTRIAL SYSTEMS DIVISION – READING
DESIGN VERIFICATION RECORD

PAGE 1 OF 4

A PROJECT: CRYSTAL RIVER UNIT 3 DC-5510-1261-EE

SUBJECT: ASV-5/204 Power Separation Qualification Criteria

SECTION NAME AND NUMBER: Elect. Engr. / 0421

W.O. 08-5510-126

D. A. Rhoads
ORIGINATOR

R. P. Crank
PROJECT ENGINEER

THIS DOCUMENT CONTAINS PRELIMINARY DATA/ASSUMPTIONS:

NO ✓ YES ______ PAGE(S) ______

A COMPUTER PROGRAM WAS:

✓ NOT USED ______ USED (CERTIFIED PER CAM) ______ USED (NOT CERTIFIED—TO BE VERIFIED WITH CALCULATION)

PROGRAM SYSTEM NAME	REV.		REV.
(1)		(4)	
(2)		(5)	
(3)		(6)	

VERIFICATION PACKAGE (IDENTIFY EACH ITEM)

DOCUMENTS TO BE VERIFIED	REV.		REV.
(1) DC-5510-1261-EE	0	(4)	
(2)		(5)	
(3)		(6)	

SUPPORTING DOCUMENTS	REV.		REV.
(1) As listed in "DESIGN INPUTS"		(7)	
(2) on page 3 of 4		(8)	
(3)		(9)	
(4)		(10)	
(5)		(11)	
(6)		(12)	

D. A. Rhoads
ORIGINATOR'S SIGNATURE

10/29/87
DATE

B NO VERIFICATION REQUIRED PER DCP 2.05:

REASON: ______

VERIFICATION REQUIRED (CHECK METHOD(S)):

DESIGN REVIEW ✓ ALTERNATE CALCULATION ______ QUALIFICATION TESTING ______

IDENTIFICATION OF VERIFIER/~~VERIFICATION TEAM~~: T. D. Biss

[signature]
PROJECT ENGINEER'S SIGNATURE

10/29/87
DATE

GAI-466 3-81

CR3 085422

Gilbert Commonwealth Companies	POWER AND INDUSTRIAL SYSTEMS DIVISION — READING DESIGN VERIFICATION RECORD	PAGE 1 OF 40
	A PROJECT: CRYSTAL RIVER UNIT 3 — ID = DC-CR3-017-EE	
	SUBJECT: BATTERY 3A SIZE VERIFICATION	
	SECTION NAME AND NUMBER: Elect. Eng. / 0421	W.O. 04-5510-126

D. A. Rhoads
ORIGINATOR

R. P. Cornit
PROJECT ENGINEER

THIS DOCUMENT CONTAINS PRELIMINARY DATA/ASSUMPTIONS:

NO ✓ YES ____ PAGE(S) ____

A COMPUTER PROGRAM WAS:

✓ NOT USED ____ USED (CERTIFIED PER CAM) ____ USED (NOT CERTIFIED–TO BE VERIFIED WITH CALCULATION)

PROGRAM SYSTEM NAME	REV.		REV.
(1)		(4)	
(2)		(5)	
(3)		(6)	

VERIFICATION PACKAGE (IDENTIFY EACH ITEM)

DOCUMENTS TO BE VERIFIED	REV.		REV.
(1) DC-CR3-017-EE, Pages	1	(4)	
(2) 24, 29, 31-34, 40		(5)	
(3)		(6)	

SUPPORTING DOCUMENTS	REV.		REV.
(1) DC-5510-126-EE	0	(7)	
(2)		(8)	
(3)		(9)	
(4)		(10)	
(5)		(11)	
(6)		(12)	

D. A. Rhoads
ORIGINATOR'S SIGNATURE

10/24/07
DATE

B NO VERIFICATION REQUIRED PER DCP 2.05:

REASON: ____

VERIFICATION REQUIRED (CHECK METHOD(S)):

DESIGN REVIEW ✓ ALTERNATE CALCULATION ____ QUALIFICATION TESTING ____

IDENTIFICATION OF VERIFIER ~~/QUALIFICATION TEAM~~: H. I. Graeff

PROJECT ENGINEER'S SIGNATURE

10/24/07
DATE

CR3 085423

QUALITY PROGRAM INSPECTION PLAN REVIEW

Florida Power

FIELD CHANGE NOTICE

Crystal River Unit 3

PLANT REVIEW COMMITTEE APPROVAL — Sheet 1 of 6

MAR NUMBER	FCN NUMBER	SAFETY RELATED	QI REVIEW REQUIRED
T87-10-07-01	1A	☒ Yes ☐ No	☐ Yes ☒ No

PROJECT NAME: KSV-5 & KSV-204 Power Separation

SAFETY EVALUATION REQUIRED	REGULATORY ENVIRONMENTAL REVIEW REQUIRED	PRC REVIEW REQUIRED
☐ Yes ☒ No	☐ Yes ☒ No	☐ Yes ☒ No

DRAWINGS AFFECTED (List All):

SEE SHEET #2

DESCRIPTION OF FIELD CHANGE (Add Additional Sheets if Necessary)

THIS FCN PROVIDES THE CONDUIT ROUTING AND SEISMIC CONDUIT SUPPORT WEIGHT CALC SHEETS FOR KSV-5 & KSV-204 POWER SEPARATION AS REQUIRED FOR THE ADDITION OF CONDUIT IN THE SUBJECT MAR.

ORIGINAL

FOR ATTACHMENTS ADDED, SEE SHEET #3.

REASON FOR FIELD CHANGE (Add Additional Sheets if Necessary)

ADDITIONAL CONDUIT ROUTING IS REQUIRED PER INSTALLATION INSTRUCTIONS 3A, 3B, 4A, 5A, AND 6A ON SHEETS 4 AND 5 OF 17 OF THE ENGINEERING INSTRUCTIONS OF THE SUBJECT MAR.

TECHNICAL JUSTIFICATION: EACH CONDUIT SUPPORT HAS AN ACTUAL LOAD WITHIN ITS CALCULATED ALLOWABLE AS SHOWN IN INDIVIDUAL SUPPORT CALCULATION PACKAGES.

NUCLEAR ENGINEERING APPROVALS

DESIGN ENGINEER	DATE	VERIFICATION ENGINEER	DATE	OTHER DISC. APPROVAL
Jürgen F. Emhardt	11/16/87	E. O. [illegible]	11-16-87	☐ Yes ☐ No ☒ N/A
		ENGINEERING SUPERVISOR: G. E. Coller	11-16-87	

QUALITY PROGRAMS INSPECTION PLAN REVIEW

REVIEWER	DATE	INSPECTION PLAN FOR APPROVED FCN
[illegible signature]	11-19-87	☐ No Change ☐ Revised & Issued W/Working Copy ☐ Revised & Attached ☐ To Be Added at W.P. Level

PLANT REVIEW COMMITTEE APPROVAL

PRC APPROVAL BY (PRC Chairman)	DATE	MEETING NUMBER
N/A		N/A

INSTALLATION AUTHORIZED BY (Nuclear Plant Manager)	DATE
N/A	N/A

NUCLEAR PLANNER VERIFICATION

THIS FCN HAS BEEN INCORPORATED INTO THE MAR PACKAGE BY (Print Name)	POSITION
MICHAEL K. WILSON	NPS

COMPANY NAME	DATE
FPC	7/17/92

Rev 11/86

CR3 085424

Florida Power

FIELD CHANGE NOTICE
Crystal River Unit 3

☒ To Be Added at W.P. Level ☐ Revised & Issued w/Working Copy ☐ Revised 8

MAR NUMBER	FCN NUMBER	SAFETY RELATED		QC REVIEW REQUIRED	
T87-10-09-01	2A	☒ Yes	☐ No	☐ Yes	☒ No

PROJECT NAME
ASV-5/204 Power Separation

SAFETY EVALUATION REQUIRED		REGULATORY ENVIRONMENTAL REVIEW REQUIRED		ISI REVIEW REQUIRED	
☐ Yes	☒ No	☐ Yes	☒ No	☐ Yes	☒ No

DRAWINGS AFFECTED
Ref EC-209-008 AS-6

DESCRIPTION OF FIELD CHANGE (Use Additional Sheets if Necessary)

1. Revised Safety Listing To ~~include~~ revise Channel on AS-4 Term. Box And ASV-204 Starter
2. Added Note Identifying Channel Change of ASV-204 Motor Starter from Green To Red
3. Revised Wiring To assist field in installation.
4. Corrected Typographical error.

FCN Attachment - BIM - Attachment 1

REASON FOR FIELD CHANGE (Use Additional Sheets if Necessary)

This FCN is written to assist the field in wiring Changes To ASV-204, and Corrects omissions to the Safety Listing.

ORIGINAL

NUCLEAR ENGINEERING APPROVAL

DESIGN ENGINEER	DATE	VERIFICATION ENGINEER	DATE	SINGLE SIGN-OFF APPROVED FOR
[signature]	11/23/87	C. S. Kramer	11/23/87	☒ Yes ☐ No ☐ N/A
		ENGINEERING SUPERVISOR	DATE	
		[signature]	11-23-87	

QUALITY PROGRAMS INSPECTION PLAN REVIEW

REVIEWER	DATE	INSPECTION PLAN FOR APPROVED FCN
[signature]	11-27-87	☐ No Change ☐ Revised & Issued w/Working Copy ☐ Revised & Attached ☒ To Be Added at W.P. Level

PLANT REVIEW COMMITTEE APPROVAL (Required if checked YES)

PRC APPROVAL BY PRC CHAIRMAN	DATE	MEETING NUMBER
N/A		N/A
INSTALLATION AUTHORIZED BY Nuclear Plant Manager	DATE	
N/A		

NUCLEAR PLANNER VERIFICATION

THIS FCN HAS BEEN INCORPORATED INTO THE WORK PACKAGE	POSITION
MICHAEL K. WILSON	NPS
COMPANY NAME	DATE
FPC	7/11/92

NOTE: Form with NEDM Nuclear Ops. Engineering

CR3 085425

Florida Power

FIELD CHANGE NOTICE

Crystal River Unit 3

PLANT REVIEW COMMITTEE APPROVAL ... Sheet 1 of 1

MAR NUMBER	FCN NUMBER	SAFETY RELATED	QI REVIEW REQUIRED
T87-10-09-01/A	3A	☒ Yes ☐ No	☐ Yes ☒ No

PROJECT NAME: ASV-5/204 Power Separation

SAFETY EVALUATION REQUIRED	REGULATORY ENVIRONMENTAL REVIEW REQUIRED	PRC REVIEW REQUIRED
☐ Yes ☒ No	☐ Yes ☒ No	☐ Yes ☒ No

DRAWINGS AFFECTED AND REV. NO.

None

DESCRIPTION OF FIELD CHANGE (Use Additional Sheets if Necessary)

Add note to Clarify use of Brand Rex Wire B/M Item 23 in Equipment Located in a Harsh Environment.

ORIGINAL

REASON FOR FIELD CHANGE (Use Additional Sheets if Necessary)

Brand Rex 219 wire is the only qualified Jumper wire for use in a harsh environment

NUCLEAR ENGINEERING APPROVAL

ORIGINATING ENGINEER	DATE	REVIEWED BY	DATE	
[signature]	11/29/87	Larry R. McDougal		☒ Yes ☐ No ☐ N/A
		ENGINEERING SUPERVISOR: Steve Hildebrant	11-24-87	

QUALITY PROGRAMS INSPECTION PLAN REVIEW

REVIEWER: [signature] DATE: 11-27-87

INSPECTION PLAN FOR ADVANCED FCN: ☒ To Be Added at W.P. Level ☐ No Change ☐ Revised & Issued W/Working Copy ☐ Revised & Attached

PLANT REVIEW COMMITTEE APPROVAL (Required if checked YES ...)

PRC APPROVAL BY PRC Chairman: N/A DATE: | MEETING NUMBER: N/A

INSTALLATION AUTHORIZED BY Nuclear Plant Manager: N/A DATE:

NUCLEAR PLANNER VERIFICATION

THIS FCN HAS BEEN INCORPORATED INTO THE MAR PACKAGE BY (Print Name): MICHAEL K. WILSON | POSITION: NPS

COMPANY NAME: FPC | DATE: 7/13/92

Rev. 11/85

CR3 085426

FIELD CHANGE NOTICE

Florida Power — Crystal River Unit 3

Sheet 1 of 10

FCN Number	FCN Number	Safety Related	ISI Review Required
T87-10-09-01	4A01	☒ Yes ☐ No	☐ Yes ☒ No

Project Name: #ASV-57/-204 Power Separation

Safety Evaluation Required	Regulatory Environmental Review Required	PRC Review Required
☐ Yes 4HAE ☒ No	☐ Yes ☒ No	☐ Yes ☒ No

Drawings Affected:

SKETCH 1 (REF. DWG. 55-211-008, AS-06)
SKETCH 9 (REF. DWG. EC-209-008, AS-06)
SKETCH 18 (REF. DWG. EC-210-624)
SKETCH 24 (REF. DWG. EC-210-124)
SKETCH 27 (REF. DWG. B-208-005)
SKETCH 57 (REF. DWG. EC-209-018, AS-06)
SKETCH 36 (REF. DWG. E-201-092)

Description of Field Change:

THIS FCN ADDS LOCAL CONTROLS & INDICATION FOR VALVE ASV-204 TO TERM. BOX AS-9.

SEE SHEET 2 FOR DETAILED CHANGES.

FCN attachments, None.

Insert added per sheet 8A.

Reason for Field Change:

Local Control station provided for Personnel and Equipment safty and protection.

ORIGINAL

RR 12/1/87

Design Engineer	Date		Date
L. D. Kirby	11/25/87	R. R. ... 12-1-87	☐ Yes ☐ No ☒ N/A
R. E. Plemer	11/25/87	Engineering Supervisor: C. B. Paul	12/1/87

Reviewer	Date	Inspection Plan
Gene Brandon	12-7-87	☐ No Change ☐ Revised & Issued W/Working Copy ☐ Revised & Attached ☑ To Be Added at W.P. Level

PRC Approval by PRC Chairman: N/A — Date: — Meeting Number:

Installation Authorized by Nuclear Plant Manager: N/A — Date:

MICHAEL K. WILSON — Position: NPS

Company Name: FPC — Date: 7/13/92

FIELD CHANGE NOTICE

Crystal River Unit 3

NA
T87-10-04-01

5A

Yes NA

No

ASH-5/ 209 POWER SEPARATION

MC-Yes-01 No

Yes No

Yes No

SKETCH 17 (REF. DWG EC-210-622)
SKETCH 18 (REF. DWG EC-210-624)
SKETCH 27 (REF DWG GLR-BLDY-ASH 206)
SKETCH 36 (REF B208-026, EF-15)

TB14 TERMINALS 19 THRU 22 ARE IN USE. THIS FCN MOVES ASE37 TO TB15 TERMINALS 17 THRU 20.

CHANGE E.I. SECTION E.16.b, PER ATTACHED & E.16.c, PER ATTACHED.

INTERFERENCE WITH MAR 86-05-25-01

ORIGINAL

12-6-87

12/4/87

12/7/87

12-08-87

To Be Added at W.P. Level

No Change Revised & Issued ... Copy Revised & Attached

NA

NA

NA

MICHAEL K. WILSON

NPS

FPC

7/13/92

CR3 085428

1 . 11

Florida Power

QUALITY PROGRAMS INSPECTION PLAN REVIEW

FIELD CHANGE NOTICE

Crystal River Unit 3

PLANT REVIEW COMMITTEE APPROVAL

☒ To Be Added at W.P. 4

☐ Revised & Issued W/Working Copy ☐ Revised &

FCN NUMBER	REV NUMBER	SAFETY RELATED		QC REVIEW REQUIRED	
T87-10-09-01	6A	☒ Yes	☐ No	☐ Yes	☒ No

PROJECT NAME, AUTHORIZED

ASV-5/204 POWER SEPARATION

SAFETY EVALUATION REQUIRED ☐ Yes ☒ No | REGULATORY ENVIRONMENTAL REVIEW REQUIRED ☐ Yes ☒ No | PRC REVIEW REQUIRED ☐ Yes ☒ No

DRAWINGS AFFECTED (LIST OR)

SKETCH 9, SKETCH 27, SKETCH 37 (All Rev A) [ARH]

DESCRIPTION OF FIELD CHANGE AND ADDITIONAL SKETCHES & REFERENCES

1) REVERSE ZENER DIODES ZD1, ZD2 & ZD3 ON SKETCH 9.
2) DELETE RESISTORS R4, R5 & R6 FROM SKETCH 27 & 37.

ORIGINAL

REASON FOR FIELD CHANGE AND ADDITIONAL SKETCHES & REFERENCES

1) INCORRECTLY SHOWN, THIS FCN MAKES SKETCH 9 AGREE WITH SKETCH 27, WHICH IS CORRECT.
2) LIGHTS ARE 125V, RESISTORS ARE NOT REQUIRED.

NUCLEAR ENGINEERING APPROVAL

DESIGN ENGINEER	DATE		DATE	
John M Kurtz	12-11-87	Paul Stoner	12/11/87	☒ Yes ☐ No ☐ N/A
		Dave Hildebrandt	12-11-87	

QUALITY PROGRAMS INSPECTION PLAN REVIEW

REVIEWER	DATE	INSPECTION PLAN FOR	
J.D. Martin	12/16/87	☒ To Be Added at W.P. Level	☐ No Change ☐ Revised & Issued W/Working Copy ☐ Revised & Attached

PLANT REVIEW COMMITTEE APPROVAL

PRC APPROVAL BY PRC Chairman	DATE	MEETING NUMBER
NA		NA
INSTALLATION AUTHORIZED BY Nuclear Plant Manager	DATE	
NA		

	POSITION
MICHAEL KWIKSON	NPS
COMPANY NAME	DATE
FPC	7/13/92

CR3 085429

FIELD CHANGE NOTICE

Crystal River Unit 3

PLANT REVIEW COMMITTEE APPROVAL REQUIRED — YES

Sheet 1 of 6

Safety Related: X Yes — No

Review Required: — Yes X No

TB7-10-89-01 NA

7A

ASV 5/204 Power Separation

Safety Evaluation Required: — Yes X No

— Yes X No

— Yes X No

SKETCH 32, 26, 19, 20 & NEW SKETCHES 20A & 39

1) ADD SKETCHES 20A & 39 TO SECTION B - ATTACHMENTS

2) REVISE SKETCHES 32, 26, 19 & 20 AS SHOWN ON PAGES 2, 3, 4 & 5 OF THIS FCN

3) ADD THE FOLLOWING INSTALLATION INSTRUCTIONS TO SECTION E.

NOTE: THESE STEPS STAND ALONE.

1. AT THE PSA SECTION OF MCB (REF. SK 26), DETERM AND SPARE BLK (1) CONDUCTOR FROM ASK29 CABLE, NOW LANDED ON TB40-3

2. INSTALL JUMPER BETWEEN TB40-3 & TB40-5 (REF. SK 26).

3. AT RR3A (REF SK 19) DETERM AND SPARE BLACK CONDUCTOR (1) 12-27-87 FROM CABLE ASK29, NOW LANDED ON TB13-10.

(CONTINUED ON PAGE 2)

AS PRESENTLY CONNECTED, THE EFP #2 'FAILURE TO START' ALARM WILL FUNCTION ONLY WHEN BOTH 'A' & 'B' EFIC SIGNALS ARE PRESENT TO START THE PUMP.

THIS CHANGE ALLOWS THIS ALARM TO FUNCTION IF THE 'PUMP START' SIGNAL COMES FROM EITHER OF THE EFIC SIGNALS 12-27-87

ORIGINAL

Design Engineer: [signature] 12-27-87

[signature] 12/28/87

Engineering Supervisor: [signature] 12/28/87

Inspection plan for advanced FCN: To be added at W.P. level; No Change; Revised & Issued W/Working Copy; Revised & Attached

Reviewer: Steve Broadd[?] 12/29/87

PRC Approval by PRC Chairman: NA

Date / Meeting Number: NA

Date

Installation Authorized by Station Plant Manager: NA

MICHAEL K WILSON

Position: RDS

Date: 7/13/92

Company Name: FPC

CR3 085430

Chris S. Sterner -89

PRC #156-
7-18-89

FIELD CHANGE NOTICE

Florida Power

PLANT REVIEW COMMITTEE — Crystal River Unit 3 — Sheet 1 of 3

MAR NUMBER	FCN NUMBER	SAFETY RELATED	QI REVIEW REQUIRED
T87-10-09-01	8	☒ Yes ☐ No	☐ Yes ☒ No

PROJECT NAME: ASV-5/204 POWER SEPARATION

SAFETY EVALUATION REQUIRED	REGULATORY ENVIRONMENTAL REVIEW REQUIRED	PRC REVIEW REQUIRED
☐ Yes ☒ No	☐ Yes ☒ No	☒ Yes ☐ No

DRAWINGS AFFECTED:

SEE ATTACHED SHEET #2

DESCRIPTION OF FIELD CHANGE (Use Additional Sheets if Necessary):

INCORPORATE INTERIMS OF CRITICAL CONTROL ROOM DRAWINGS INTO THE MAR BY COMPOSITING THE MAR SKETCHES [ ] RELEVANT FCN SKETCHES.

"DOCUMENTATION ONLY" FCN

ORIGINAL

SEE SHEET #3 FOR ATTACHMENTS INCLUDED

REASON FOR FIELD CHANGE (Use Additional Sheets if Necessary):

CRITICAL CONTROL ROOM DRAWINGS MUST BE AVAILABLE TO CONTROL ROOM PERSONNEL AND INTERIM DRAWINGS ARE THE PRESCRIBED METHOD EFFECTING THIS FOR PROPOSED CHANGES.
SEVERAL CONSTRAINTS AFFECTING THE ISSUANCE OF THE PERMANENT MAR AND PERFORMING A HELB REVIEW NECESSITATE MAKING THESE CRITICAL DRAWING AVAILABLE UNTIL THE MAR CAN BE MADE PERMANENT AND AS-BUILT.

DESIGN ENGINEER	DATE	VERIFICATION ENGINEER	DATE	CONCUR WITH ADVANCED FCN
Chris S. Sterner	6/26/89	[signature]	6/27/89	Yes No ☒ N/A
		ENGINEERING SUPERVISOR: [signature]		

REVIEWER	DATE	INSPECTION PLAN FOR ADVANCED FCN — To Be Added at W.P. Level
J.D. Martin	7/12/89	☐ No Change ☐ Revised & Issued W/Working Copy ☐ Revised & Attached

PRC APPROVAL BY	DATE	MEETING NUMBER
[signature]	7/18/89	89-29

INSTALLATION AUTHORIZED BY Nuclear Plant Manager	DATE
[signature]	7/19/89

THIS FCN HAS BEEN INCORPORATED INTO THE MAR PACKAGE BY	POSITION
MICHAEL R. WILSON	NPS
COMPANY NAME: FPC	DATE: 7/13/92

CR3 085431

7-41-07

Florida Power — FIELD CHANGE NOTICE — Crystal River Unit 3

PLANT REVIEW COMMITTEE (Required if checked YES by Nuclear Engineering) — Sheet 1 of 2

3 9/11/89

MAR NUMBER	FCN NUMBER	SAFETY RELATED	QA REVIEW REQUIRED
T87-10-09-01	9	☒ Yes ☐ No	☐ Yes ☒ No

PROJECT NAME: ASV-5/204 POWER SEPARATION

SAFETY EVALUATION REQUIRED	REGULATORY ENVIRONMENTAL REVIEW REQUIRED	PRC REVIEW REQUIRED
☐ Yes ☒ No	☐ Yes ☒ No	☒ Yes ☐ No 9/14/89

DRAWINGS AFFECTED (LIST ALL):

201-071 DP-15A (INTERIM REV. A)
208-008 AS-07 (INTERIM REV. A)

DESCRIPTION OF FIELD CHANGE (USE ADDITIONAL SHEETS IF NECESSARY):

CORRECT LOAD COMPONENT IDENTIFICATIONS FOR ASV-204 FEEDS TO READ:

SW. NO.	LOAD COMPONENT IDENTIFICATION
15	125V DC CONTROL TO (ASV-204)
17	250V DC POWER TO (ASV-204)

FCN ATTACHMENTS - PARTIAL DWG 201-071 DP-15A & PARTIAL DWG 208-008 AS-07

REASON FOR FIELD CHANGE (USE ADDITIONAL SHEETS IF NECESSARY):

POWER FEED IDENTIFICATIONS FOR THE POWER AND CONTROL FOR ASV-204 WERE INCORRECTLY IDENTIFIED BY FCN #8 TO MAR T87-10-09-01.

NUCLEAR ENGINEERING APPROVAL

DESIGN ENGINEER	DATE	VERIFICATION ENGINEER	DATE	CONCUR WITH ADVANCED FCN
Chris S. Sterner	9/11/89	[signature]	9/11/89	☐ Yes ☐ No ☒ N/A
NMES FIELD ENGINEER	DATE	PSWP REQUIRED ☐ Yes ☐ No	[signature] RWBeith	DATE 9/11/89

QUALITY SYSTEMS REVIEW FOR QUALITY REQUIREMENTS

REVIEWER: J.D. Martin — DATE: 9/14/89

PLANT REVIEW COMMITTEE APPROVAL (Required if checked YES by Nuclear Engineering)

PRC APPROVAL BY PRC CHAIRMAN: DR — DATE — MEETING NUMBER

INSTALLATION AUTHORIZED BY — DATE

NUCLEAR PROJECTS SPECIALIST

THIS FCN HAS BEEN INCORPORATED INTO THE MAR PACKAGE BY (PRINT NAME): MICHAEL K. WILSON — POSITION: NPS

COMPANY NAME: FPC — DATE: 2/13/92

Florida Power

FIELD CHANGE NOTICE

Crystal River Unit 3

PLANT REVIEW COMMITTEE APPROVAL (Required if checked YES by Nuclear Engineering) Sheet 1 of 3

MAR NUMBER	FCN NUMBER	SAFETY RELATED	ISI REVIEW REQUIRED
T87-10-09-01	10	☒ Yes ☐ No	☒ Yes ☐ No

PROJECT NAME: ASV-5/204 POWER SEPARATION

EQ REVIEW REQUIRED: ☒ Yes ☐ No

SAFETY EVALUATION REQUIRED	REGULATORY ENVIRONMENTAL REVIEW REQUIRED	PRC REVIEW REQUIRED
☐ Yes ☒ No	☐ Yes ☒ No	☐ Yes ☒ No

DRAWINGS AFFECTED (List all)

NONE

DESCRIPTION OF FIELD CHANGE (Use Additional Sheets if Necessary)

1) CORRECT ISI REVIEW REQUIREMENT ON THE MAR FORM & ADD GUIDELINES TO SECTION "H" OF ENGINEERING INSTRUCTIONS.

2) INDICATE THAT "E.Q. REVIEW" IS REQUIRED.

ORIGINAL

REASON FOR FIELD CHANGE (Use Additional Sheets if Necessary)

1) THIS CORRECTION IS IN RESPONSE TO RECA 87-015-01.

2) ORIGINAL ISSUANCE OF MAR WAS PRIOR TO THE MAC FORM HAVING A CHECK-BLOCK FOR E.Q. REVIEW.

NUCLEAR ENGINEERING APPROVAL

DESIGN ENGINEER	DATE	[illegible]	DATE
L. M. James	3/7/90	[signature]	3/7/90
SITE FIELD ENGINEER	DATE	[illegible]	DATE
		[signature]	3/7/90

SITE SUPERVISOR EQ GROUP (if EQ REVIEW IS REQ'D)

QUALITY SYSTEMS REVIEW FOR QUALITY REQUIREMENTS

REVIEWER: [signature] DATE: 3/22/90

PLANT REVIEW COMMITTEE APPROVAL (Required if checked YES by Nuclear Engineering)

PRC APPROVAL BY PRC CHAIRMAN: [signature] DATE: MEETING NUMBER:

INSTALLATION AUTHORIZED BY Nuclear Plant Manager DATE:

NUCLEAR PROJECTS SPECIALIST

THIS FCN HAS BEEN INCORPORATED INTO THE MAR PACKAGE BY (Print Name): MICHAEL R WILSON

POSITION: NPS

COMPANY NAME: FPC

DATE: 7/13/92

CR3 085433

QUALITY SYSTEMS REVIEW FOR QUALITY REQUIREMENTS

FIELD CHANGE NOTICE

Crystal River Unit 3

PLANT REVIEW COMMITTEE APPROVAL (Required if checked YES by Nuclear Engineering) Sheet 1 of 8

8-3-90

T87-10-09-01 N/A	11	SAFETY RELATED: ☒ Yes ☐ No	QA REVIEW REQUIRED: ☐ Yes ☒ No
ASY-5/204 Power separation	N/A		EQ REVIEW REQUIRED: ☒ Yes ~~☒ No~~ JPC 7/30/90
SAFETY EVALUATION REQUIRED: ☐ Yes ☒ No	REGULATORY ENVIRONMENTAL REVIEW REQUIRED: ☐ Yes ☒ No	PRC REVIEW REQUIRED: ☐ Yes ☒ No	

DRAWINGS AFFECTED:

208-008 sh. AS-07, Interim Rev.A

DESCRIPTION OF FIELD CHANGE:

See sheet 2.

THIS FCN INCLUDES ATTACHMENTS:

#1 (CHECKLIST FOR EQ DOCUMENT REVIEW, 2 SHEETS)

JPC 8/3/90 MR

REASON FOR FIELD CHANGE:

This FCN is required for documentation only to assist in making this T-MAR permanent.

ORIGINAL

NUCLEAR ENGINEERING APPROVAL

DESIGN ENGINEER: Robert L. Caine	DATE: 7/30/90	VERIFICATION ENGINEER: S. K. Bollard	DATE: 8/1/90
FIELD ENGINEER:	FOMP REQUIRED: ☐ Yes ☐ No	ENGINEERING SUPERVISOR: D. K. Koch	DATE: 8/3/90
		SUPERVISOR – EQ GROUP: Hugh Stilston	8-3-90

QUALITY SYSTEMS REVIEW FOR QUALITY REQUIREMENTS

REVIEWER: Steve Barnard | DATE: 8/7/90

PLANT REVIEW COMMITTEE APPROVAL (Required if checked YES by Nuclear Engineering)

PRC APPROVAL BY PRC CHAIRMAN: N/A | DATE: | MEETING NUMBER: N/A

INSTALLATION AUTHORIZED BY: N/A | DATE:

NUCLEAR PROJECTS SPECIALIST

THIS FCN HAS BEEN INCORPORATED: MICHAEL K WILSON | POSITION: NPS

COMPANY NAME: FPC | DATE: 7/13/92

T-MAR # T87-10-09-01
FCN # 11
SHEET 2 OF 8
DWG # N/A

FCN# 11 TO T-MAR T87-10-09-01

DESCRIPTION OF FIELD CHANGE:

This FCN revises drawings and sketches to incorporate changes made by FCN 39 & 42 of MAR 87-03-13-02. FCN 39 was initiated to de-terminate the motor starter heater circuit of ASV-204. FCN 42 was initiated to assist the field in the implementation of FCN 39 and to correct the discrepancies between FCN 39 and T87-10-09-01. There is no field work involved, this FCN is for documentation only.

Interim drawings 201-061 sheets AC-19 & AC-19A are not being revised by this FCN, FCN 42 of 87-03-13-02 correctly updates these drawings.

SUMMARY:

Additional information is being added to the original MAR package in sections C. REFERENCE.

ADD TO: C. REFERENCES:

1. EQ# 90-1903
2. MAR 87-03-13-02 FCN's 39 & 42

RCT COPY

MODIFICATION APPROVAL RECORD

Florida Power

Crystal River Unit 3

MODIFICATION NUMBER	WORK REQUEST	SAFETY RELATED	ISI REVIEW REQUIRED
87-10-09-01A	105572	X Yes ☐ No	X Yes ☐ No

PROJECT NAME: ASV-5/ASV-204 Power Separation

EO REVIEW REQUIRED: ☒ Yes ☐ No

SYSTEM/TAG NUMBER: AS

SOURCE DOCUMENTS: ~~IEC~~ N/A [initials] 3/5/92

TYPE OF MODIFICATION

☒ Permanent ☐ *Temporary Expiration Date ______ *W.R. No. for Removal ______

☐ Mechanical ☐ Electrical ☒ I&C ☐ Structural ☐ Support

PLANT STATUS

For Implementation ______ On-Line (Full load) ______ On-Line (Reduced load) __X__ Outage Required

DESCRIPTION OF PROPOSED MODIFICATION (Use Additional Sheets if Necessary)

This modification converts temporary modification MAR T87-10-09-01 into a permanent installation. Temporary MAR T87-10-09-01 was installed in refuel 6. Minor plant changes are needed to be performed for this modification before the T-MAR can be documented as a permanent installation. Three electrical devices installed per the T-MAR need to be relabelled to correct plant item tag designation. In addition, conduit routing near control station AS-9 need to be removed from the plant.

REASON FOR MODIFICATION (Use Additional Sheets if Necessary)

The temporary modification is being converted into a permanent installation in conformance with engineering design procedures. The temporary installation has been determined to be an acceptable method for electrical load demand reduction for the "A" emergency diesel generator. Electrical devices were only given a T-MAR numbering system instead of a proper equipment tag assignment designation. T-MAR deleted and requested that conduit routings be eliminated and actually work steps to remove them has not been accomplished per the T-MAR Engineering Instructions.

NUCLEAR ENGINEERING APPROVAL

DESIGN ENGINEER	DATE	ENGINEERING SUPERVISOR	DATE
CC Strumpk [initials] 3/5/92	3/2/92	C. S. Kamer	3/2/92
VERIFICATION ENGINEER	DATE	SUPERVISOR, SHIFT... GROUP IF ISI RELATED OR NA	DATE
J D Patel	3/2/92	[signature] for AH Fulcher	3/4/92

MODIFICATION APPROVALS

	SIGNATURE	DATE
FEWP REQUIRED ☐ Yes ☐ No	SHES FIELD ENGINEER	
Meeting Number ______ Plant Review Committee (PRC)	CHAIRMAN	
NGRC Review Required ☐ Yes ☐ No NGRC Approval	CHAIRMAN	
Installation Authorized	NUCLEAR PLANT MANAGER	
Cancellation	NUCLEAR PLANT MANAGER	
Temporary Modification Removed	NUCLEAR PROJECTS SPECIALIST	
Modification Testing Complete	MAR TEST SUPERVISOR	
Modification Installation Complete and All Documentation Enclosed	NUCLEAR PROJECTS SPECIALIST	

Florida Power

SAFETY EVALUATION

Check to indicate applicable review and enter applicable document number:

☒ MAR NO. 87-10-01-01A ☒ Procedure No.: NEP 0211 Rev. 08 TC ____

☐ FCN No. ____ ☐ Other ____

SAFETY EVALUATION: Answer the following questions and provide specific justification (use attachment if necessary).

1. Is the probability of an occurrence or the consequence of an accident or malfunction of equipment important to safety, as previously evaluated in the Final Safety Analysis Report, INCREASED? ☐ Yes ☒ No
Because:

See Attached Sheet

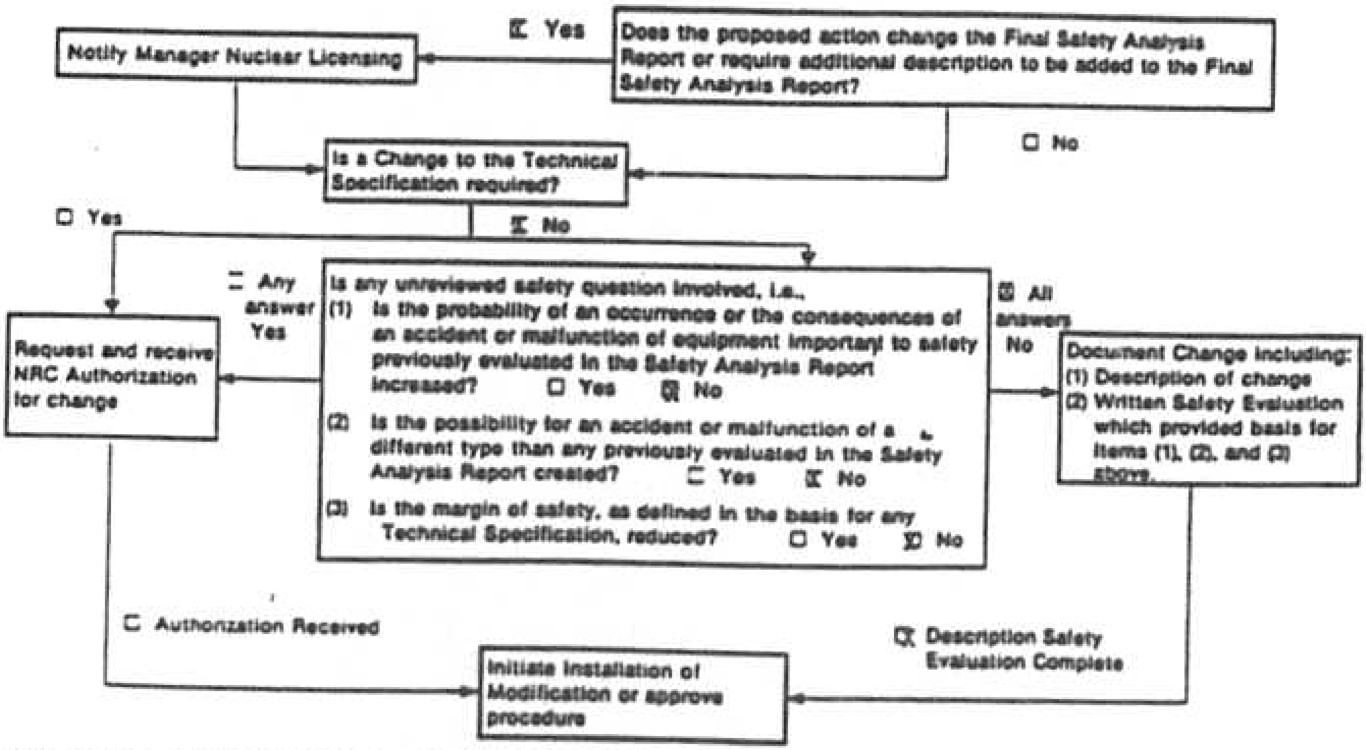
2. Is the possibility for an accident or malfunction of a different type than any previously evaluated in the Final Safety Analysis Report CREATED? ☐ Yes ☒ No
Because:

See Attached Sheet

3. Is the margin of safety, as defined in the basis for any Technical Specification, REDUCED? ☐ Yes ☒ No
Because:

See Attached Sheet

10CFR50.59 CHECKLIST



Required changes to Technical Specifications should be processed in parallel to this checklist.

ENGINEER	DATE	VERIFICATION ENGINEER	DATE	SUPERVISOR	DATE
C. Stumpf	3/2/92	TD (Patel)	3/2/92	C. D. Kremin	3/2/92

12/91

REF. Life of Plant RESP: Nuclear Engineering 600 246

CR3 005863

Florida Power CORPORATION

ANALYSIS / CALCULATION

Crystal River Unit 3

REI / MAR NUMBER MAR 87-10-09-01A

PROJECT ASV-5/204 POWER SEPARATION

ATTACHMENT TO MODIFICATION SAFETY EVALUATION

1. ASV-5 and ASV-204 are motor operated valves having identical functions of supplying steam to the turbine driven Emergency Feedwater Pump (EFP-2). Since EFP-2 is the ES "B" channel pump, ASV-5 and 204 were electrically connected in parallel to a common 250/125 VDC ES "B" channel power and control source. This modification electrically separates ASV-204 from ASV-5 and repowers ASV-204 from 250/125 VDC ES "A" channel power. Also, separate control room controls and separate "A" channel EFIC interlocks are being provided for ASV-204. Automatic control logic of ASV-204 has not changed. Therefore, the probability of an occurrence or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the FSAR is not increased since the logic of automatically opening ASV-204 whenever the EFIC System calls for emergency feedwater has not been altered. The reliability of EFP-2 has actually been increased because with this modification either "A" or "B" train power will control and operate one of the steam inlet valves to EFP-2 as opposed to both valves being 'B" train powered. FSAR Sections 7.2.4, 8.2.2.6 and 10.5.3 have been reviewed.

2. The electrical separation of ASV-204 from ASV-5 does not impact the design function of either valve to supply steam to the EFP-2 turbine. Power and control for ASV-5 is not affected by this modification and ASV-5 retains its automatic control logic, remote manual control, local manual control and remote shutdown isolation and control. ASV-204 is being powered from the redundant power channel, and will be provided with its own remote and local manual control and with separate EFIC interlocks for automatic operation. The type of manual control and automatic operation of ASV-204 is the same as for ASV-5. Therefore, based on the above, the possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR is not created. FSAR Sections 7.2.4, 8.2.2.6, and 10.5.3 have been reviewed.

3. This modification enables the turbine driven Emergency Feedwater Pump (which is the 'B" channel pump) to be operational even if a failure should occur on the "B" channel power system for which shutdown operation would be via the 'A" channel systems. With this capability, the turbine driven EFW pump is able to operate and share the EFW requirements with the 'A" channel motor driven EFW pump. This will reduce the electrical load on the "A" channel diesel generator for the condition of an ES actuation coincident with a loss-of-offsite-power and failure of the 'B" channel power system. Consequently, with this modification the margin of safety, as defined in the basis for any Technical Specification, is not reduced. It is actually enhanced because of the increased availability of the turbine driven Emergency Feedwater Pump. Technical Specification Sections 3/4.7.1 and 3/4.8.1 have been reviewed.

DESIGN ENGINEER	DATE	VERIFICATION ENGINEER	DATE	SUPERVISOR, NUCLEAR ENGINEERING	DATE
C. Stuempke	3/2/92	[signature]	3/2/92	C. D. Kasmer	3/2/92

A. SUMMARY FUNCTIONAL DESCRIPTION:

This modification MAR 87-10-09-01A converts temporary modification MAR T87-10-09-01 into a permanent installation.

Temporary MAR T87-10-09-01 was installed in Refuel 6. The design function that is to be made permanent by this MAR is to assure that the turbine driven emergency feedwater pump (EFP-2) will be operable in the event of a failure of the Engineered Safeguard "B" 250/125V DC system coincident with loss-of-offsite power and an engineered safeguard actuation. Under this scenario, EFP-2 will be relied upon to share the emergency feedwater load with the motor driven emergency feedwater pump (EFP-1) in order to decrease the electrical load demand on emergency diesel generator EDG-3A. To assure this capability, the temporary modification removed the 250/125V DC distribution source Engineered Safeguard "B" power from ASV-204 and repowered the valve from 250/125V DC Engineered Safeguard "A" distribution source. This involved separating ASV-5 and ASV-204 power and control circuitry (which were wired in parallel) and providing a separate main control board control switch, local pushbuttons, indicator lights, and EFIC actuation signal to ASV-204.

This modification requires that some minor plant changes be implemented before the T-MAR can be documented as a permanent installation. The changes involve the re-labelling of electrical equipment devices in the rear of the PSA/EFIC section of the main control board and auxiliary relay rack cabinet RR3A. The device label designation was not per engineering tagging convention and were assigned a label prefix "TM" which corresponded to mean "Temporary Modification Item number". The originators of the T-MAR did not acquire the appropriate equipment item tag designation. Therefore, the following electrical devices need to be relabelled as follows:

a) The valve position status indicating lights for ASV-204 labelled as "TM1", "TM2" and "TM3" need to be relabelled as item numbers "BM1", "BM2" and "BM3".

b) The three position GE CR2940US203E selector switch furnished on the PSA/EFIC bench board of the MCB along with the above indicating lights was labelled as board device item "TM4" and needs to be relabelled as "BM4".

c) In auxiliary relay rack RR3A relay devices labelled as "TMA" and "TMB" needs to be relabelled as device items "DB" and "DA" respectively.

d) This modification separates the alarm circuits for the valves. An alarm "steam supply not ready" will alarm if the valve is closed or valve control power is lost.

DESIGN ENGINEER	DATE	VERIFICATION ENGINEER	DATE	SUPERVISOR, NUCLEAR ENGINEERING	DATE
C C Strumpfl	3/2/92	T D Patel	3/2/92	C. J. Kraemer	3/2/92

ENGINEERING INSTRUCTIONS CONTINUATION SHEET

Crystal River Unit 3

RE: MAR NUMBER

MAR 87-10-09-01A

A. SUMMARY FUNCTIONAL DESCRIPTION: (Cont'd)

In the engineering instruction section for T-MAR 87-10-09-01 under installation instructions, it was instructed that a number of conduit routings be deleted and removed from in-plant service. This may not have been accomplished and possibly still remain in-place. To convert the T-MAR into a permanent installation, the below listed conduit routings may need to be removed (if not ready done so) from the plant. The conduit routings are:

ASK1 - 1-1/2"
ASE27 - 1"
ASE28 - 3/4"
ASE30 - 1-1/2"
ASF31 - 1"

B. ATTACHMENTS:

1. Design Data Sheets (2 Sheets with 3 Page Attachment)
2. Design Input Record (11 Sheets with EDG Loading Justification Sht.)
3. Verification Report (1 Sheet)
4. Fire Protection Review (with 2 Page Attachment)
5. HELB Review (1 Sheet with 1 Page Attachment)
6. Modification Safety Evaluation (2 Sheets)
7. Regulatory/Environmental Review (1 Sheet)
8. MCB Bill of Material Input Record (8 Sheets)
9. ALARA Analysis/Calc. Form (' Sheet)
10. Environmental Qualification Applicability Review (1 Sheet)
11. Memo, J. A. Lese to R. Iwachow, July 12, 1991 (1 Sheet)
12. G/C, Inc. Interim Drawings (All Revision ~~A~~, EXCEPT AS NOTED) 3/5/92

CSI-95-4, Sheet 1 (BOTH REV. A ~~& B~~)	211-008 AS-01
CSI-95-4, Sheet 2 (BOTH REV. A ~~& B~~)	211-008 AS-06
CSI-95-4B 3/16/92 3/5/92	211-026 EF-01
CSI-95-287	211-026 EF-12
201-172	212-008 ASE1
208-008 AS-01 (2 PARTIAL SHEETS) 3/16/92	212-008 ASE3
208-008 AS-07	212-008 ASE4
208-026 EF-01	212-008 ASF4
208-026 EF-12	212-008 ASK1
208-026 EF-15	212-008 ASK2
209 ~~208~~-008 AS-01 (2 PARTIAL SHEETS) 3/16/92	212-008 ASK3
209-008 AS-06 (2 PARTIAL SHEETS) 3/16/92	212-008 ASK4
209-054-SS-01 (2 PARTIAL SHEETS) 3/16/92	212-026 EFC2
209-106 SH. 1	212-026 EFC5
209-136 SH. 3 (2 PARTIAL SHEETS) 3/16/92	212-026 EFK2
209-166 (2 PARTIAL SHEETS)	212-026 EFK4
210-120	212-026 EFK5
210-123	212-026 EFM1
210-124	215-032 SH. 1
210-543 (2 PARTIAL SHEETS) 3/16/92	215-032 SH. 2 (2 PARTIAL SHEETS) 3/16/92

3/5/92

CR3 005866

Florida Power
CORPORATION

DESIGN INPUT RECORD

Crystal River Unit 3

MAR NUMBER				
. 87-10-09-01A	[ ] Electrical	[x] I&C	[ ] Mechanical	[ ] Structural

PROJECT:
ASV-5/ASV-204 Power Separation

1. The function of this modification is to convert temporary modification T-Mar 87-10-09-01 to a permanent installation which assures that the turbine driven emergency feedwater pump (EFW-2) will remain operable in the event of a failure of Emergency Safeguard "B" 250/125 Volt DC system coincident with loss-of-offsite-power and an Engineered Safeguard actuation. Under this scenario, emergency feedwater pump EFP-2 will be relied upon to share the emergency feedwater flow requirements with the motor driven emergency pump (EFP-1) in order to decrease the electrical load demand on emergency diesel generator EDG-3A. In order to assure that EFP-2 is operable for the above condition, this modification documents the removal of the 250/125V DC Engineered Safeguard "B" power from ASV-204 and repowers the valve onto the 250/125V DC Engineered Safeguard "A" distribution system. This involves the separation of ASV-5 and ASV-204 power and control circuits which were wired in parallel; thereby, providing redundant steam admission to EFP-2 for increase reliability. The physical wiring separation and circuit rerouting were accomplished under temporary MAR T87-10-09-01 which also provided for the addition of a control switch on the main control, local control station, pushbuttons, indicating lights and EFIC 'A' actuation signals to ASV-204.

   This modification requires that some minor plant changes need to be implemented before the T-MAR can be documented as a permanent installation. These additional changes are as follows:

   a. Electrical devices added by the T-MAR were labelled with only a numbering system instead of a proper equipment tag assignment designation. The item tag designation for the three position selector switch and its position status lights needs to be re-labelled from "TM1 thru TM4" to read as "BM1 thru BM4". Also, two Clark type relays installed in auxiliary relay rack cabinet RR3A have been tagged with the temporary designation "TMA" and "TMB" and should be relabelled as "DB" and "DA" respectively.

   b. In addition, the temporary modification separated power and control circuits but did not provide for separation of alarm circuitry between safeguard channels. This modification incorporates design changes to reroute and rewire alarm circuits for ASV-5 and ASV-204. Alarm (100AL) "Steam Supply Not Ready" is being retained for ASV-5 and a new alarm (1521AL) "Steam Supply Not Ready" is being added for ASV-204. The logic for these alarms is also being change to "loss of power" or "valve not open". To accomplish this, T-MAR relay "TMA" is being SPARED and an existing 125VDC (non-time delay) relay in RR3A is being wired into ASV-204 control circuitry. Since the ASV-5 alarm relay is 120 VAC, 5 second delay, it will be adjusted to the minimum

DESIGN ENGINEER	DATE	VERIFICATION ENGINEER	DATE	SUPERVISOR, NUCLEAR ENGINEERING	DATE
C Strumpfk	3/2/92	I. D. Patel	3/2/92	C. S. Kr...	3/2/92

delay (approximately 0.2 seconds). The differences in the relays is only a utilization of existing relays and not a diversification design requirement.

c. This permanent modification also spares alarm circuits that were affected by modification MAR 89-04-15-02 and MAR 89-04-15-03. The alarm circuits being spared provided an interlock interface for starting the old EFW chemical addition system which has now been abandoned.

Design changes for this modification will occur at the following plant components:

a) at both motor operated valves ASV-5 and ASV-204

b) at both terminal boxes AS-1 and AS-9

c) behind the PSA/EFIC section of the main control board

d) in auxiliary relay rack cabinets: RR-3A, RR-5B1, RR-PSA

e) in events recorder cabinet 5

f) at the secondary cycle sampling analyzer panel SSCP-1

g) at the 480V water treatment motor control center 3B, unit 6B

h) at the 4160V switchgear engineered safeguard bus 3A, Unit 3A3.

2. The performance requirements will remain unchanged for ASV-204. The same holds true for the capacity and rating requirements for the valve. The valve is required to open on an EFIC "A" actuation signal. This permits the start of steam turbine driven emergency feedwater pump (EFP-2) to account for the plant configuration change accomplished by the temporary modification to lessen the electrical demand on the "A" emergency diesel generator EDG-3A.

3. Code, Standards, Regulatory Requirements and other documents:

a) IEEE 323-1974, Qualifying Class 1E Equipment for Nuclear Power Generating Stations.

b) IEEE 344-1975, Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations.

c) IEEE 383-1974, Type Test of Class 1E Electric Cables, Field Splices and Connections for Nuclear Generating Stations.

d) Code of Federal Regulations 10CFR50, Appendix R.

e) FPC Calculational Document No. E-87-0001, Revision 0, titled "Addition of ASV-204 to DPDP-8A".

f) FPC Calculational Document No. E-87-0002, Revision 0, titled "ASV-5 and ASV-204 Power Separation Qualification Review".

g) FPC Calculational Document No. E-87-0004, Revision 1, titled "Battery 3A size Verification".

h) FPC Calculation Document E-88-0019, Revision 1, titled "Conversion of Terminal Box AS-9 to a Control Station".

i) Vendor Qualification Package No. CNTL-G080-02, Revision 0, titled "FPC (General Electric) - Local Control Station (Pushbutton/Indicator Lights), Volumes 1 and 2".

j) Vendor Qualification Package No. VLV-L200-06, Revision 0, titled "Limitorque Model SMB Class H".

k) Vendor Qualification Package No. CABL-L080-05, Revision 0, titled "Kerite FR/FR Control Cable".

l) Vendor Qualification Package No. CABL-K080-06, Revision 0. titled "Kerite HTK/FR Power Cable".

m) Vendor Qualification Package No. CNTL-N431-03, Revision 0, titled "Nutherm 125VDC Motor Starter" Volumes 1 thru 3.

n) Vendor Qualification Package No. CNTL-N431-01, Revision 0, titled "Nutherm 125VDC Control Station", Volumes 1 thru 3.

4. Valve ASV-204 is required to be powered from a Engineered Safeguard "A" power distribution source. The 250V DC power for the valve is provided from distribution panel DPDP-8A, Fuse 17. The 125V DC control power for manipulating the valve is provided from the same distribution panel and fed from Fuse 15.

5. This modification does not alter the design modifications accomplished under the temporary modification T-MAR 87-10-09-01. The design change regarding the seismic integrity of the circuity changes were evaluated under the T-MAR.

6. Valve ASV-204 is identified per CMIS as an essential component that is required to support safe shutdown. This modification changes the power and control distribution source requirements for the valve's motor operator (ASV-204-MO). This modification does not change the Code Key Classifications. The component is still required to meet 10CFR50.49

criteria. CIDP NO. 91012303 dated 01/23/91 has been issued to correct the channel designation form "B" to "A" for the motor starter for the motor operator. This CIDP also updates the document cross reference listing with additional design drawing information. Refer to CR-3 Walkdown Package No. 0042 dated 10/06/89 pages 1 thru 52.

7. The design modification implemented under the T-MAR and this permanent modification occur in the control complex on building elevations 108', 124', 134', and 145'. Specifically in the areas of the main control room, CRDM rooms, EFIC Room B, the cable separating room and the 4160V Switchgear Room A. These floor elevations (or areas) are classified as mild environmental areas and the environmental conditions expected are listed on the environmental zone sheets as follows:

   a) Zone 13, Rev. 4; dated 8/90 titled "Elev. 145 - Control Complex, Control Room".

   b) Zone 43, Rev. 3; dated 8/90 titled "Elev. 95' and 108'- Control Complex".

   c) Zone 58. Rev. 3; dated 8/90 titled "Elev. 124' and 134'- Control Complex".

   Modification changes have occurred in the areas of elevation 95' and 119' of the Intermediate Bldg. The design changes occurred in the vicinity of the steam turbine driven emergency feedwater pump EFP-2. Power and control cables had to be routed from this area into the control complex by utilization of the plant's existing cable tray system. Both floor elevations of the "Intermediate Bldg. are listed as "Harsh" Environmental areas and their expected environmental conditions are listed on environmental zone sheets as follows:

   a) Zone 14, Rev. 5; dated 8/90 titled "Elev. 95' - Intermediate Bldg. Emergency Feedwater Pump Room".

   b) Zone 19, 4; dated 8/90 titled "Elev. 95' - Intermediate Bldg. HVAC Area".

   c) Zone 20, Rev. 4; dated 8/90 titled "Elev. 95' - Intermediate Bldg. Miscellaneous Equipment Area'.

   d) Zone 57, Rev. 5; dated 8/90 titled "Elev. 119' - Intermediate Bldg. Persónal Hatch Area".

   Reference Environmental and Seismic Qualification Program Manual, Rev. 5 dated August, 1990 for determination of zone information.

Florida Power CORPORATION

DESIGN INPUT RECORD CONTINUATION SHEET

Crystal River Unit 3

FED/MAR NUMBER
. MAR 87-10-09-01A

The following Vendor Qualification Package Reports have been updated to incorporate the design changes initiated by the temporary modification:

a) VQP CNTL-N431-01, Revision 0, titled "Nutherm 125V DC Control Station", Volumes 1 thru 3.

b) VQP CNTL-N431-03, Revision 0, titled "Nutherm 125V DC Motor Station", Volumes 1 thru 3.

c) VQP VLV-L200-06, Revision 0, titled "Limitorque Model SMB Class H".

d) VQP CABL-K080-05, Revision 0, titled "Kerite FR/FR Control Cable".

e) VQP CABL-K080-06, Revision 0, titled "Kerite HTK/FR Power Cable".

f) VQP CNTL-G080-02, Revision 0, FPC (General Electric) - Local Control Station (Pushbutton/Indicator Lights)", Volumes 1 and 2.

8. The design modification interfaced with various plant components and cabinets. Interface occurred at the following locations:

a) The PSA/EFIC bench board section of the main control board was changed to add a GE CR2940 selector switch to permit the plant operator to take remote manual control of ASV-204. The bench board had to be reworked to accommodate the switch on the upper half of the EFW "A" channel flowpath.

b) Two time delay relays were ad' d in relay cabinet 3A to pick-up the EFIC actuation signal for alarming the opening of valve ASV-204.

c) Terminal Box AS-9 was modified to become a local control station for ASV-204.

d) The control and power sources to valve ASV-204 which were commonly shared with ASV-5 were separated and assigned to an electrical safeguard channel "A" DC power distribution source. The DC power is fed from Distribution Panel DPDP-8A.

e) The 120VAC power supply to the motor starter space heater was disconnected and the actual work was accomplished under Modification MAR 87-03-13-02, FCN's 39 and 42.

f) The interlock alarms which interfaces with the EFW chemical addition pump had been spared and the chemical addition pump EFP-3 has been realigned to act morpholine transfer pump. This realignment of pump duty was done under modification MAR 89-04-15-02 and MAR 89-04-15-03.

9. The power and control cables were furnished from safety related cable inventory maintained at CR-3 site stores. The General Electric control switches selected for installation within local control station (AS-9) are qualified for the harsh environment and documented in VQP Report CNTL-G080-02. A General Electric three position selector switch type CR2940US203E was used for the installation on the control board PSA section. Indicating lamps used for valve ASV-204 position status were installed as Drake Gemlite model 5160 series type. Joslyn-Clark 120 VAC time delay relays model type 7313-PMT, were used and located in auxiliary relay cabinet RR3A by the T-MAR. One of these relays is being SPARED and an existing SPARE Clark 125 VDC relay, model 4U8-4, is being utilized in the alarm circuit.

14. The electrical source of power for ASV-204 motor operator and control circuitry is supplied from the "A" Engineered Safeguard distribution bus. The motor operator power is 250VDC and fed from distribution panel DPDP-8A, Fuse #17. The control power is 125VDC and also fed from distribution panel DPDP-8A, Fuse #15. The motor starter space heater for ASV-204 was supplied from a 120VAC distribution panel ACDP-54, Breaker #4; however, modification MAR 87-03-13-02, FCN's 39 and 42 disconnected the power feeder to the space heater.

    FPC Document No. E-87-0001, Rev. 0, titled "Addition of ASV-204 to DPDP-8A", evaluated the design capacity of the distribution panel to accept addition voltage loading and determine if this added load is within the design margin of the distribution panel. Also, another calculation was established to determine that the additional voltage load does not exceed the design capacity of DC Battery 3A. This determination is recorded in FPC Document No. E-87-0004, Revision 0, dated 10/20/87 titled "Battery 3A Size Verification". Both engineering calculations have resulted in showing that the added voltage loads are within the design margins of the DC distribution system.

15. The physical valve arrangements for ASV-5 and ASV-204 remain unchanged and both valves assemblies are parallel to each other and permit steam admission to the steam turbine driven emergency feedwater pump (EFP-2). Both valves ASV-5 and ASV-204 shared common power sources and control devices to actuate and cause motion of the valves. The commonality between the two was separated and terminal box AS-9 was converted to a local control station. In addition, the control devices located on the lower half of the PSA/EFIC bench section mimic for the steam turbine driven EFP-2 was altered to allow addition of a selector switch for ASV-204. The PSA/EFIC control board mimic arrangement for both emergency feedwater flowpaths were changed to allow addition of the three position selector switch.

16. Valve ASV-204 is operational during all plant modes except in plant start-up and shutdown modes where the EFIC actuation signals are bypassed to prevent spurious start of the steam turbine driven EF pump. ASV-204 is

required to be functional for a loss-of-power event coincident with an ES actuation and a single failure of the 125VDC ES channel "B" power system.

17. In the main control room a three position selector switch type GE CR2940US203E with position status lights of red, green and amber has been furnished for remote manual manipulation of valve ASV-204. The control function of the selector switch is such that the control switch must be held in either the open or close position to permit valve travel. Local control station AS-9 has been provided with two pushbutton switches of the model type GE CR2490YA202C and CR2940YA202E and three position status indicating lights of the model type GE CR2940UC212B2, CR2940UC212C2 and CR2940UC212D2. The control circuitry for ASV-204 interfaces with the EFIC actuation signals at the Relay Cabinet RR3A to provide the following system functions:

    a) A normally open EFIC actuation contact is wired in parallel with the selector switch and the pushbutton switch contacts in the valve's "open" circuit. This contact provides the automatic opening of valve ASV-204 upon and EFIC Channel "A" actuation signal.

    b) A normally closed EFIC actuation contact is wired in series with the selector switch and pushbutton switch contacts in the valves "close" circuit. This contact provides an interruption to the valves closure signal when an EFIC Channel "A" actuation signal is initiated to open the valve.

Auxiliary relays have been added and identified as Items "TMA" and "TMB" to provide alarm status for ASV-204. These item designations need to be changed since they were not assigned a permanent plant item identifier. Items "TMA" and "TMB" need to be retagged and relabelled as "DB" and "DA". The 120 VAC time delay relay "DB" will be made SPARE. An existing SPARE 125 VDC relay tagged "H" will be utilized for this alarm separation logic.

19. ASV-204 was added to the auxiliary steam supply system by MAR 80-11-48. The valve was installed to provide the Turbine Driven Emergency Feedwater Pump (EFP-2) with a parallel path (around ASV-5) for providing motive power. ASV-204 was added by MAR 80-11-48 to improve the reliability of the steam supply by providing a redundant path. However, since the pump was considered to be "B train", ASV-204 was powered with "B train" electric power. The power to ASV-204 was changed to "A train" by the temporary MAR T87-10-09-01 which is being made permanent by this MAR. This change provides a potential reduction of dependency on EFP-1 thus reducing loading concerns. This change also increases the reliability of achieving EFP-2 pump start by providing electrically and physically redundant steam supply valves. The "A train" circuits to ASV-204 are separated per the plant separation criteria from the "B train" circuits of ASV-5. The "A train" and "B train" signals to open ASV-204 and ASV-5 originate from redundant sensors and logic channels. Electric power is also fed from redundant, independent sources. A failure evaluation (FPC document E-90-0111 Rev. 0)

was performed to demonstrate sufficient independence existed between the "B train and "A train" to prevent simultaneous failure of the redundant trains of emergency feedwater.

20. Temporary Modification TMAR 87-10-09-01A was implemented during Refuel 6 to reduce electrical load demand on emergency diesel generator EDG-3A. To achieve this load reduction on EDG-3A, the two parallel steam emission valves (ASV-5 and ASV-204) to the steam driven emergency feedwater pumps were reconfigured to permit the valves to be electrically powered from two separate distribution sources versus both being powered from the same power source. The Temporary modification accomplished this realignment of power sources by removing auxiliary steam emission valve ASV-204 from the 250/125 volt DC, Engineered Safeguard "B" source to the 250/125 volt DC, Engineered Safeguard "A" source. This effort involved the physical separation of the power and control circuits which were commonly shared between ASV-5 and ASV-204. The temporary modification did not touch the actual valve assembly and it still remains in parallel to ASV-5. FPC Failure Modes and Effects Analysis Document No. E-90-0111, Revision 0, titled "ASV-204 Failure Evaluation" demonstrates that no propagated failures can result due to valve assembly ASV-204 and it's electrical circuitry which are still within the vicinity of the "B" side steam turbine driven emergency feedwater pump EFP-2. The analysis shows that no fault or failures could cause impairment of the entire "A" engineered safeguard train due to initiating events on the "B" side.

28. Material selection has been based on compatibility with as-installed system and existing plant design. Control devices for ASV-204 mounted in control station AS-9 are similar to design configuration of qualified control stations furnished by Nutherm. This similarity in design is documented in VQP #CNTL-G080-02. Relay "DB", "DA" and "H" located in RR3A are Clark-Joslyn type which are similar in design to other existing relays in RR3A.

30. This modification provides for the documentation of cable routings for electrical circuits designated as ASE-34, ASE-35, ASE-36, ASE-37 and ASK-1. The circuits identified are part of the emergency feedwater system which is designated as a safe shutdown system. The circuit paths are routed through the Intermediate Bldg. and the Control Complex and located in the following fire areas/zones:

Fire Area IB-95-200C
Fire Area IB-119-201
Fire Area AB-95-003B
Fire Area CC-134-118A
Fire Area CC-108-108
Fire Area CC-124-117
Fire Area CC-145-118B

The fire areas/zones have been determined as listed in the Crystal River Unit 3 Fire Hazards Report No. 03-0920-1103, Revision 3, and visual inspection of Appendix R Fire Area Layout drawings 213-021, Revision 3, 213-023, Revision 3 and 213-028, Revision 01.

a) The modification has occurred in fire areas containing safe shutdown equipment.

b) The modification does not add, delete, or relocate systems, structures, or components into a zone or area where an unprotected opposite train exists.

c) The modification does not add, delete or relocate non-safety related circuits that share power supplies, signal sources, enclosures and raceways with safety related circuits.

d) Emergency lighting is not being obstructed or relocated by this modification.

This modification documents the addition of power and control circuits into the exiting plant cable tray system in the above fire areas. The increase in combustible loading does not exceed the maximum permissible loading in each zone.

32. FPC Specification SP-5145, Revision 1 titled "Human Factors Design Conventions for the Control Room" was consulted for addition of the control switch for ASV-204 on the bench board of the PSA/EFIC section. This arrangement of the EFW board included the reconfiguration of the EFW pump display of both emergency feedwater flowpath mimics and the pump matrix switches in the area of the EFW pump controls.

34. This modification has an impact on the loading of the Emergency Diesel Generator and the loading evaluation has been addressed and documented in FPC, IOC WPN 966-0434 dated April 15, 1988. This modification has intentions on reducing the continuous duty loading of the motor driven emergency feedwater pump EFP-1 on the electrical load demand requirements for the "A" emergency diesel generator. The modification relies on having the steam turbine driven emergency feedwater pump EFP-2 share the emergency feedwater flow requirements with the motor driven emergency feedwater pump EFP-1. This was accomplished by having the power source for ASV-204 removed (which commonly shared the same power source with ASV-5 from the 250/125VDC, Engineered Safeguard "B" source) and assigned to a 250/125VDC, Engineered Safeguard "A" source. This change permits the start of the steam turbine driven EFW pump (EFP-2) both on a EFIC Channel "A" and Channel "B" actuation signal. Valve ASV-204 is auto-loaded on ES load Block 1.

Valve ASV-204 motor operator previously received 250 VDC power from Distribution Panel DPDP-8B, Fuse #5. The 125 VDC control power for ASV-204 is previously received from Distribution Panel DPDP-8B, Fuse #6. The load reduction to this panel is 4520.25 watts. (4462.75 watts of motor operator locked rotor load from Fuse #5 and 57.5 watts of control power load from Fuse #6.)

The motor operator locked rotor DC load of 4462.75 watts is being added to Distribution Panel DPDP-8A, Fuse #17. This load is intermittent duty and auto-loaded onto the "A" EDG. The estimated time of ASV-204 energization is one minute. Reference FPC Document No. E-90-0105, Revision 0 dated 12/11/90 titled "Electrical DC System Revalidation Program DC Master Data Base (G/CI Report #2851)", Volume 2, Section 1.0, pages DPDP8A17-1 thru DPDP8A17-4.

The control power DC load of 57.5 watts (in-rush) and 25.0 watts (steady-state) is being added to Distribution Panel DPDP-8A, Fuse #15. The DC load for the in-rush condition is intermittent duty and auto-loaded onto the "A" EDG for a period of one minute. The steady-state load is continuous and also auto-loaded onto the "A" EDG. Determination of the control power electrical loads is based on input data found in FPC Document No. E-90-0105, Revision 0 dated 12/11/90 titled "Electrical DC System Revalidation Program DC Motor Data Base (G/CI Report #2851)", Volume 2, Section 1.0 pages DPDP8A15-1 thru DPDP8A15-4.

38. MOVATs testing is not required since the modification deals with electrical wiring changes and has no affect on the pressure retaining characteristics of the valve.

39. The modifications were performed in the following Dominant Area of Concern:

   a) Room No. CB303 - Relay/CRD Switchgear Room

   b) Room No. CBEFB - EFIC "B" Room

   c) Room No. CB208 - 4160V Switchgear Room

   d) Room No. IB095 - EFWP Room

   e) Room No. CB504 - Control Room

   An SBO review form has been completed and evaluation by the SBO Reviewer. Valve ASV-204 is identified as a major component required to meet Station Black-out requirements.

40. FPC Documents No. E-90-0110, Revision 0, titled "HELB Evaluation for Valve ASV-204 and FPC IOC NEA90-0991 dated June 9, 1990 have been prepared to assess the impact in the operability of ASV-204 during and after exposure to a HELB event in the vicinity of the steam driven emergency feedwater pump EFP-2.

REI/MAR NUMBER
MAR 87-10-09-01A

42. The modification will cost less than $50,000 based on:

    1) the majority of the modification was installed by T-MAR 87-10-19-01.

    2) the additional changes to be completed can be installed by less than $10,000 ... conservatively assuming: (2 men for 1 week) x 2 for paper work times 1.5 for cost increase, and including material.

EXHIBITS TO THE TESTIMONY OF
PAUL F. MCKEE

EXHIBIT NO. ___ (PFM-3)

MAR DOCUMENTATION FOR 1990 MODIFICATION

RCT COPY

Florida Power

FIELD CHANGE NOTICE
Crystal River Unit 3

PRC #1392

Sheet 1 of 4

MAR NUMBER	FCN NUMBER	SAFETY RELATED	NI REVIEW REQUIRED
88-05-24-01	19	☒ Yes ☐ No	☐ Yes ☒ No

PROJECT NAME	EQ REVIEW REQUIRED
DHP/EFP ES Start Modification	☐ Yes ☒ No

SAFETY EVALUATION REQUIRED	REGULATORY/ENVIRONMENTAL REVIEW REQUIRED	PRC REVIEW REQUIRED
☒ Yes ☐ No	☐ Yes ☒ No	☒ Yes ☐ No

DRAWINGS AFFECTED

NONE

DESCRIPTION OF FIELD CHANGE (Use Additional Sheets if Necessary)

1) Revise Safety Evaluation Sh 1 of 11 per FCN # 19 Sheet 3 of 4.

2) Revise Engineering Instructions Sheet 1 of 1 per FCN # 19 Sheet 2 of 4.

ORIGINAL

REASON FOR FIELD CHANGE (Use Additional Sheets if Necessary)

1) Safety Evaluation is revised to include the receipt of NRC approval for Technical Specification Change.

2) Add MAR 88-05-24-01 reference (18). It provides approval of Technical Specification change (Cover letter provided on FCN #19 sheet 4 of 4).

NUCLEAR ENGINEERING APPROVAL

DESIGN ENGINEER	DATE	VERIFICATION ENGINEER	DATE	CONCUR WITH ADVANCED FCN
[signature]	6/7/90	[signature]	6/7/90	☐ Yes ☐ No ☒ N/A

SNES FIELD ENGINEER	DATE	FEWP REQUIRED	ENGINEERING SUPERVISOR	DATE
		☐ Yes ☐ No	DH Beach	6/7/90

SNES SUPERVISOR - EQ GROUP (if EQ RELATED OR N/A): N/A

QUALITY SYSTEMS REVIEW FOR QUALITY REQUIREMENTS

REVIEWER	DATE
J.D. Martin	6/7/90

PLANT REVIEW COMMITTEE APPROVAL (Required if checked YES by Nuclear Engineering)

PRC APPROVAL	DATE	MEETING NUMBER
[signature]	6/8/90	90-23

INSTALLATION AUTHORIZED BY	DATE
P.F. McKee	6/8/90

NUCLEAR PROJECTS SPECIALIST

THIS FCN HAS BEEN INCORPORATED INTO THE MAR PACKAGE BY (PRINT NAME)	POSITION
DON Karger	NPS

COMPANY NAME	DATE
FPC	10·01·90

MAR # 88-05-24-01

FCN # 19

SHEET 2 OF 4

DWG # NA

C. References

(1) MAR 87-10-19-01A, MAR 87-07-08-01, MA
(2) G/CI Letter FCS #10419 "System Flows for
(3) G/CI telecon memo, C.C. Strempke to Don points".
(4) G/CI Letter FCS-10128, dated 1-25-89, sec
(5) G/CI Letter FCS-9894, dated 11-4-88, conc
(6) G/CI telecon memo, C.C. Strempke tc "Annunciator windows."
(7) G/CI speed memo, ES panels & cabinet seis
(8) G/CI Letter FCS-10434, dated 5-16-89, Pu
(9) Calculation DC-3515-004-1.02 CS, and DC-
(10) G/CI Letter FCS-10420, dated 4-7-89, FSAR Transmittal.
(11) G/CI Letter FCS-10784, dated 8-24-89, revised scope letter.
(12) G/CI Letter FCS-10814, dated 9-12-89, "EDG loads".
(13) IOC, M.W. Averett to R.E. Clauson, dated 7-11-88, "Reliability Analysis for the Proposed EFP-1/LPI Logic Modification".
(14) B&W Report 51-117603-00 - "Acceptability of a 500 psig start of LPI for all Loss of Coolant Accidents".
(15) G/C, Inc. Drawings: 201-129, Main Control Board Equipment Cutout Details
201-130, Main Control Board Equipment Cutout Details
(16) Design Basis Document Temporary Change No. 67,68,& Emergency Feedwater (Sect 6/3), Decay Heat Removal Syst (Sect 6/3), & ES Safeguards Actuation Syst. (Sect 6/3)

(17) NRC Letter to FPC [illegible] [illegible] Approval of Technical Specification Change.

FCN #19

D. Material

See attached Bill of Material

E. Installation Instructions

1. Charge all time and material to NU88052401.

2. All cable handling, pulling, termination, conduit installation, equipment removal and installation shall be in accordance with applicable FPC maintenance procedures. All relays removed shall be returned to Nuclear Materials Coordinator.

3. De-energize ES system Loop A.

1a. This MAR shall not be made operational until NRC approval for the change has been received. A new Safety Evaluation will be added to the MAR via FCN after NRC approval

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

June 4, 1990

Docket No. 50-302

MAR #	98-05-24-01
FCN #	19
SHEET	4 OF 4
DWG #	NA

Mr. Percy M. Beard, Jr.
Senior Vice President,
Nuclear Operations
Florida Power Corporation
ATTN: Manager, Nuclear Operations
Licensing
P. O. Box 219-NA-2I
Crystal River, Florida 32629

Dear Mr. Beard:

SUBJECT: CRYSTAL RIVER UNIT 3 - ISSUANCE OF AMENDMENT RE: EMERGENCY FEEDWATER PUMP AND LOW PRESSURE INJECTION SYSTEM BLOCK/TRIP AND ESF RESPONSE TIMES (TAC NO. 76135)

The Commission has issued the enclosed Amendment No. 130 to Facility Operating License No. DPR-72 for the Crystal River Unit No. 3 Nuclear Generating Plant (CR-3). This amendment consists of changes to the Technical Specifications (TS) in response to your application dated February 13, 1990.

This amendment changes the TS to allow elimination of the automatic simultaneous operation of the motor-driven emergency feedwater pump and the low pressure injection system when offsite power is not available, and changes response times for the low pressure and high pressure injection systems.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Harley Silver, Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 130 to DPR-72
2. Safety Evaluation

cc w/enclosures:
See next page

RCT COPY
1-11-90

Florida Power

MODIFICATION APPROVAL RECORD
Crystal River Unit 3

MODIFICATION NUMBER	WORK REQUEST	SAFETY RELATED	QA REVIEW REQUIRED
88-05-24-01	172483	☒ Yes ☐ No	☐ Yes ☒ No

PROJECT NAME: DHP/EFP ES START MODIFICATION

SYSTEM/TAG NUMBER: ES   PRIORITY CODE: 100

TYPE OF MODIFICATION
☒ Permanent ☐ *Temporary Expiration Date N/A *W.R. No. for Removal N/A
☐ Mechanical ☐ Electrical ☒ I&C ☐ Structural ☐ Support

DESCRIPTION OF PROPOSED MODIFICATION AND ADD'L SHEETS IF NECESSARY

1. To provide a DH/EF pump interlock to automatically prevent concurrent operation of the DH and EFP-1 pump during LOOP events by starting LPI at 500 psig RC pressure and tripping EFP-1.
2. To evenly load the EDG by changing the sequence logic block loading as follows:
   a. DHP-1A(1B) from Block 1 to Block 4 set at 15 seconds.
   b. RWP-3A(3B) from Block 4 to Block 5 set at 20 seconds.
   c. DCP-1A(1B) from Block 4 to Block 6 set at 25 seconds.
   d. BSP-1A(1B) from Block 3 to Block 6 set at 25 seconds.
   e. EFP-1 from Block 5 to Block EFW set at 5 seconds after EDG breaker closing "OR" 5 seconds after Block 3 loading.

REASON FOR MODIFICATION AND ADDITIONAL SHEETS IF NECESSARY

To more evenly load the EDG during pump starts and to prevent concurrent operation of the DH and EFW pump EFP-1 during LOOP events. These changes are part of the solution to keep the EDG within its 2,000 hour, 3,000 KW rating.

ORIGINAL

E.G. REVIEW: [signature] 11/89 PER TELECON WITH A.H. GELSTON SIGNATURE CONCURRENCE WITH R. PINNEY, NCA

DESIGN ENGINEER	DATE		DATE
[signature]	10.4.89	[signature]	10.9.89
		[signature]	10/12/89

	SIGNATURE	DATE
Quality Programs Inspection Plan Review	REVIEWER Barry Chastain	11/29/89
Meeting Number ___ Plant Review Committee (PRC)	CHAIRMAN [signature]	
NGRC Review Required ☐ Yes ☐ No NGRC Approval	CHAIRMAN	
Installation Authorized	NUCLEAR PLANT MANAGER [signature] for [signature]	1/11/90
PEWP REQUIRED ☐ Yes ☐ No	SITES FIELD ENGINEER	
Cancellation	NUCLEAR PLANT MANAGER	
Temporary Modification Removed	NUCLEAR PROJECTS SPECIALIST	
Modification Testing Complete	NUCLEAR PROJECTS SPECIALIST [signature]	07/28/90
Modification Installation Complete and All Documentation Enclosed	LEAD TEST SUPERVISOR [signature]	7/29/90

Florida Power

SAFETY EVALUATION

Check to indicate applicable review and enter applicable document number:

☒ SMAR NO. 8 8 – 0 5 – 2 4 – 0 1

☐ FCN No. ______

☐ Procedure No.: NEP 2 1 1 Rev. 0 2 TC __

SAFETY EVALUATION: Answer the following questions and provide specific justification (use attachment if necessary).

1. Is the probability of an occurrence or the consequence of an accident or malfunction of equipment important to safety as previously evaluated in the Final Safety Analysis Report, INCREASED? ☐ Yes ☒ No
Because:

SEE ATTACHED

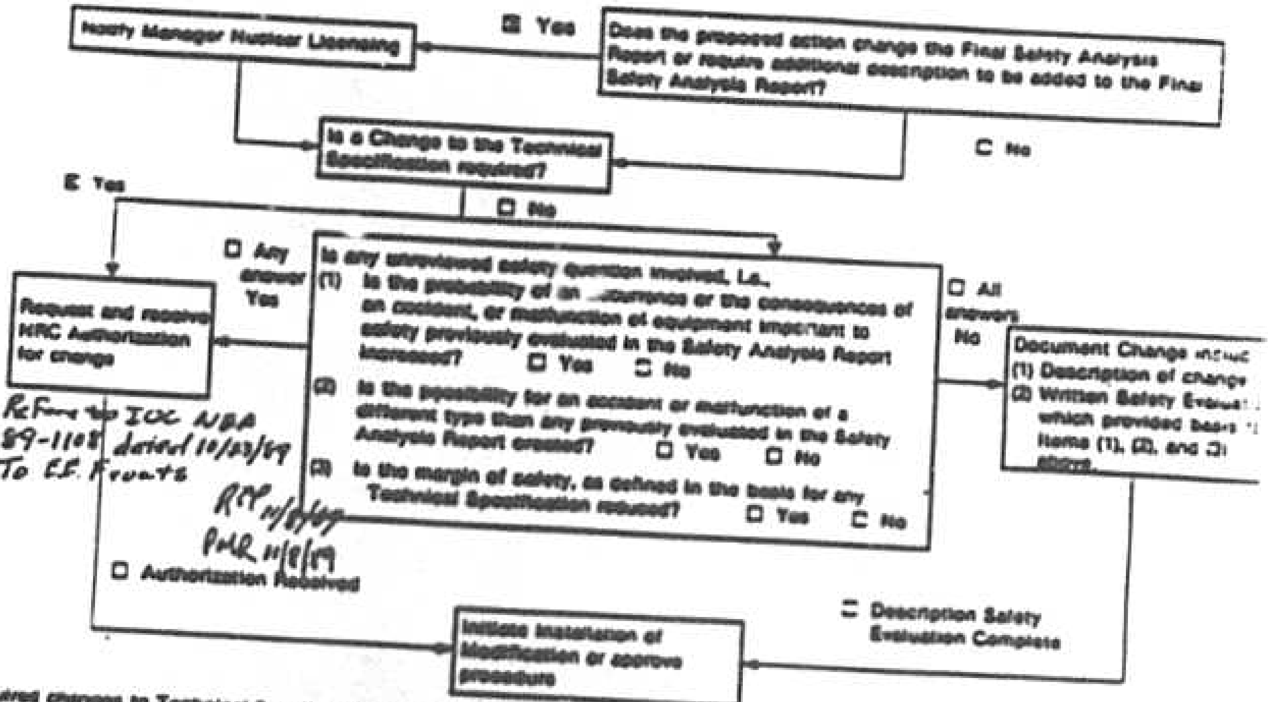
2. Is the possibility for an accident or malfunction of a different type than any previously evaluated in the Final Safety Analysis Report, CREATED? ☐ Yes ☒ No
Because:

SEE ATTACHED

3. Is the margin of safety, as defined in the basis for any Technical Specification, REDUCED? ☐ Yes ☒ No
Because:

SEE ATTACHED

10CFR50.59 CHECKLIST



*Required changes to Technical Specifications should be processed in parallel to this checklist.

ORIGINATOR	DATE	INDEPENDENT REVIEWER	DATE	SUPERVISOR	DATE
C C Straight	10/12/89	[signature]	10.13.89	[signature]	10/13

Rev. 10/88

NES Life of Plant Nuclear Engineering Assurance

Florida Power Corporation

ANALYSIS / CALCULATION

Crystal River Unit 3

REV 1 ...

88-05-24-01

PROJECT DHP/EFP ES START MODIFICATION

This logic limits operation of the LPI and EF pumps to their designed accident scenario. The EF pump functions during a small break LOCA to remove RC system heat while the HPI system provides the necessary makeup water to the RC system to maintain core coverage. During the large break LOCA, the coolant is lost through the break at a rate greater than the HPI pumps capacity. This accident relies on the LPI pumps to maintain core coverage and cooling.

This modification does not change the piping configuration of equipment operating conditions during startup, shutdown, normal power generating modes. Therefore, the probability of an occurrence as previously analyzed in the FSAR is not increased.

Equipment operation for accident mitigation is changed by this modification; however, equipment function remains unchanged. During a small break LOCA, the LPI pumps currently start and run in a recirculation mode. This is because the RC pressure remains above 500 psig. The FSAR accident analysis; therefore, cannot take credit for any cooling from LPI pumps. During a large break LOCA, circulation through the OTSGs cannot be maintained; therefore, the steam generators quickly lose heat transfer capability and the EF system cannot remove RC heat. The EFIC system currently will cause the EF pumps to go to a recirculation mode of operation since steaming in the OTSGs does not occur. The FSAR analysis; therefore, does not take credit for EF system cooling.

An intermediate size LOCA will rely on both the LPI and EF systems to provide core cooling. However, as identified above, cooling from both systems is not required concurrent. Therefore, the accident evaluation in the FSAR will still provide the bounding case after this modification is completed.

FSAR Sections reviewed are 6.0, 7.0, 8.0, 9.0, 14.0.

2. Accidents or system failures other than those evaluated in the FSAR are not introduced, since it is the pump logic after the accident has occurred which is changed. Malfunctions in the pump logic can occur, but these do not result in an unanalyzed scenario since it is designed to the same standards as the original logic. Since this is part of a protection system the following criteria of IEEE 279 have been applied to the ESAS which initiates the new logic:

   1. Single Failure Criterion - Redundant independent actuation channels are provided such that a single failure will not prevent actuation at the system level.

DESIGN ENGINEER	DATE	VERIFICATION ENGINEER	DATE	SUPERVISOR/NUCLEAR ENGINEERING	DATE
C.C. [signature]	10/18/89	[signature]	10.18.89	[signature]	10/19/89

2. Quality of Components - Components are specified and procured through a 10CFR50, Appendix B program.
3. Equipment Qualification - Components are qualified in accordance with applicable requirements of IEEE344-1971.
4. Channel Integrity - Protection system channels are designed and procured to function under anticipated extremes of environment, energy supply, malfunctions, and accidents.
5. Channel Independence Redundant channels are independent and physically separated from each other based on plant separation criteria.
6. Control & Protection System Interaction - Control and protection systems are maintained independent to prevent interaction which could cause failure of the protection system.
7. Derivation of System Inputs - Protection system inputs are provided by direct measurement of the desired variable.
8. Capability For Sensor Checks - Sensor checks can be performed by cross checking between channels.
9. Capability For Test And Calibration - Capability for testing and calibrating of channels during plant operation is provided.
10. Channel Bypasses - Two out of three logic is used for protection channel actuation. This enables the capability to remove a channel from operation for testing and calibration.
11. Operating Bypasses - Operating bypasses are designed such that the bypass will be automatically removed whenever the permissive conditions are not met. Manual bypasses are indicated to the operator. Automatic bypasses are not used.
12. Indication of Bypasses - Bypasses are indicated by status lights and annunciator.
13. Access to Means for Bypassing - Access to bypasses is provided on the Main Control Board.
14. Multiple Setpoints - Different setpoints for particular modes of operation are not used.
15. Completion of Protective Action - The system uses "seal-ins" to assure that once initiated actuation at the system level will go to completion.
16. Manual Initiation - Manual actuation at the system level is provided via switches on the Main Control Board.
17. Access To Setpoint Adjustments - Setpoint adjustments require access to locked system cabinets.
18. Identification Of Protective Actions - Protective actions are indicated by ES status lights.
19. Information Read-Out - The system is designed with sufficient visual displays to permit timely evaluation of system status by the operator.

Design Engineer	Date	Verification Engineer	Date	Supervisor, Nuclear Engineering	Date
CC Stumpf	10/18/89	K Kotyw...	10.18.89	R J Heppel	10/18/89

Florida Power Corporation

ANALYSIS / CALCULATION

Crystal River Unit 3

MO/MAR NUMBER 88-05-24-01

PROJECT DHP/EFP ES START MODIFICATION

20. System Repair - The system is designed to facilitate recognition, location, replacement, repair, or adjustment of malfunctioning components.

    FSAR Sections 6.0, 7.0, 8.0, 9.0, 14 reviewed.

3. The operating time of equipment is maintained in the analyzed range. However, the sequence loading of the LPI pumps has been changed from 25 seconds to 35 seconds including Emergency Diesel Generator loading. This is still within B&W analyzed time per B&W Calc. 51-1172948100.

   Technical Specification sections reviewed are 3.5.2, 3.5.3, 4.5.2, 4.5.3, and 3/4.3.2.

DESIGN ENGINEER	DATE	VERIFICATION ENGINEER	DATE	SUPERVISOR, NUCLEAR ENGINEERING	DATE
C.C. Stramphe	10/18/89	[signature]	10.18.89	[signature]	10/19/89

DOCKET COPY

Florida Power
CORPORATION

Walter S. Wilgus
Vice President
Nuclear Operations

[illegible] DOCKET FILE
[illegible] (SF ONLY)
[illegible] (SF ONLY)
[illegible]

July 22, 1988
3F0788-18

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: S.A. Varga

Subject: Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72
Key Projects Status

Dear Sir:

In recent discussions I became aware that you have not been formally updated on several key subjects. Subsequent conversations among our respective staff identified four key issues which warranted an update prior to our August 16 meeting(s). Attached are detailed discussions of three of these issues (Additional Feedwater Pump, Emergency Diesel Generator and Ultimate Heat Sink). The fourth, ATWS, is the subject of a B&W Owners Group meeting next week. Following development of a consensus among the Owners a meeting with the Staff will be held to assure timely closure of the issues not accepted in the Staff SER. FPC has initiated design efforts on the balance of the system in a good faith effort to support Refuel VII installation if the issues can be resolved.

Note: Additional Feedwater Pump and Ultimate Heat Sink Attachments are not part of MAR 88-05-24-01, therefor, are not included in Safety Evaluation.

July 22, 1988
3F0788-18
Page 2

I am confident the attached highlights FPC's efforts to move ahead expeditiously on these key issues. We look forward to discussing these with you and your staff in the coming weeks.

Sincerely,

W.S. Wilgus
Vice President
Nuclear Operations

WSW/KRW/dhd
Attachment

xc: Regional Administrator, Region II

Senior Resident Inspector

88-05-24-0

Pg 8 of 21
RCA 8/1/88

EMERGENCY DIESEL GENERATOR LOADING CONCERNS

On March 30, 1988, FPC met with the NRC staff to discuss the emergency diesel generator loading concerns at CR-3. FPC presented a proposed EFW pump block/trip design concept that would resolve the EDG load issue. FPC would like to re-confirm that it will install this block/trip modification during Refuel VII which is currently scheduled for the Fall of 1989. The conceptual design of this modification will be completed by November, 1988 and submitted to the NRC at that time.

In a May 12, 1988 letter summarizing the March 30 meeting with FPC, the NRC requested answers to four issues (a - d) regarding emergency diesel generator (EDG) loading. A similar request for information on five issues (1 - 5) was contained in a May 26 letter to FPC. This section responds to those concerns. Responses to similar comments from the two letters are grouped together to avoid repetition.

1. During certain events, both the EFW and DH pumps may be necessary for plant control in the long-term recirculation mode. In general, the question of long-term need for loads not needed early in an accident should be addressed.

RESPONSE: A study of the long-term EDG loads which could be required under post-accident conditions has been conducted since the March 30 meeting. The study extended the previously analyzed short term (30 minutes) accident scenarios out to seven days, including consideration of long term support equipment requirements. A preliminary report predicts EDG loading during the 7 day post accident period to be less than the 30 minute rating. This report is currently under review by FPC and B&W. The report considered maintenance of high pressure injection during the recirculation phase of the smaller LOCA accidents by utilizing the low pressure injection pumps to supply high pressure injection pump suction. The EDG loading during this HPI/LPI series operation is below its 30 minute rating for the "B" EDG failure case because the turbine driven emergency feedwater pump (EFP-2) can be utilized and motor driven pump (EFP-1) can be secured. For the EFP-2 failure case with continued operation of the motor driven emergency feedwater pump, the loads from the Nuclear Services Closed Cycle Cooling and Seawater pumps are reduced due to shared flow with the "B" train pumps. This also results in a load below the 30 minute rating. The current schedule for finalization of the report is <u>September 1</u>, 1988, and will be submitted by September 15.

2. The proposed long-term solution control system may pose additional risk or problems in more likely scenarios. The reliability of the Engineered Safeguard Features (ESF) system should be addressed to assure that failures which could prevent operation of EFW or DH removal are not made more likely, and that defeating a safety

CR3 005808

98.05.24

Pg 18 of 21
SC-1/88/89

function (i.e., EFW) will not have a significant effect on ESF reliability. The adequacy of two-out of three logic should also be examined.

RESPONSE: The proposed changes to the Engineered Safeguards (ES) low pressure injection (LPI) actuation logic provides for starting LPI on either less than 500 psig Reactor Coolant System (RCS) pressure or greater than 4 psig Reactor Building (RB) pressure when off-site power is available, but modifies the logic to start on only 500 psig RCS pressure when the EDG is supplying power. In addition, when the EDG is supplying power, the proposed change will prevent the motor driven emergency feedwater pump from starting automatically and/or trip it if it is operating upon an ES actuation caused by less than 500 psig RCS pressure. For the situation where off-site power is available the emergency feedwater pump will not be affected by the 500 psig RCS pressure ES actuation.

The existing ES Actuation System logic is based on a two out of three redundancy for reliability and testability. The system is powered from vital power supplies and is designed as de-energize to actuate on the two out of three level, while within the redundant train logic it is energize to actuate to preclude power failures from causing actuation.

The above changes provide the loading intelligence to preclude the automatic operation of both the motor driven emergency feedwater pump and the LPI pump, thus the EDG loading is maintained below its 30 minute rating. In addition, the proposed modification provides the following:

a. Like the existing CR-3 ES Actuation System, the proposed LPI/EFW actuation changes will meet all requirements of FSAR Section 7.1 Protection Systems, including single failure, on-line testability, status indication and operational bypasses.

b. The emergency feedwater pump and the LPI pump are started within the safety analysis timing specifications when either off-site power or on-site power (EDG) is available.

c. Signal diversity is maintained in its present configuration for accidents where off-site power is available. The diversity is reduced only in the starting of the LPI pump on on-site power. Since loss of off-site power concurrent with an intermediate or large break LOCA is an extremely low probability event, diversity of actuation for this combination of events provides little added protection.

48-05-2

Pg 12 of 21
Rev 11/8/89

d. Human factor principles are maintained by not introducing new controls and using the same actuation logic design for both actuation trains.

Analysis of the preliminary design was done using probabilistic risk assessment (PRA) techniques. This analysis shows that the additional circuitry will have a negligible effect on the overall reliability of the motor driven emergency feedwater pump train. A major reason this impact is negligible is that the circuitry is enabled only when the "A" EDG output breaker is closed. Thus, the possibility of inadvertent actuation is eliminated any time the EDG is not running. In addition, the changes to the LPI actuation parameters have a negligible effect on the overall reliability of the LPI trains. The reason for this, like that above, is that diverse actuation is removed only when the EDG output breaker is closed.

3. Currently, the Low Pressure Injection System is initiated by diverse signals, namely 4 psig containment pressure or low RCS pressure. In the proposed design, only the latter is used. The use of an appropriate diverse signal should be addressed.

RESPONSE: The present design utilizes a 4 psig RB pressure as a diverse actuation signal for LPI. This logic provides an anticipatory actuation in the event of a large or intermediate LOCA. However, it is an unnecessary initiation which results in operation of both LPI and EFW pumps for other accidents such as small break LOCA's or steam and feedwater line breaks. The following options have been considered for maintaining diversity without the anticipatory nature of the 4 psig signal:

a. Maintain the 4 psig signal on "B" train only.

This approach was discarded on the basis of poor human factors.

b. Actuate LPI on RB pressure of 30 psig.

This approach was not considered viable because of uncertainty in the calculation of passive heat sinks and RB cooler heat removal capabilities. This reduces confidence of the correlation to an RCS pressure of 500 psig (i.e., greater than expected heat removal capability could preclude reaching the 30 psi setpoint during the larger LOCA's).

c. Disable the 4 psig signal when off-site power is not available.

Diversity provides protection against common mode

CR3 005810

95-05-24-01

failures. This protection is most beneficial for high probability events. Intermediate and large break LOCA, combined with loss of off-site power, are low probability events.

Since the diesel loading concern occurs only during a loss of off-site power concurrent with an intermediate or large break LOCA, diversity of actuation for this combination of events provides little added protection and could be removed.

The proposed modification maintains the 4 psig RB pressure diversity at all times except when the EDG output breaker is closed.

4. The battery charger and other loads as appropriate should be added back on the EDG.

RESPONSE: FPC agrees and intends to add the battery charger loads back on the EDG. The design of the modification will be completed by October 1988. When the EFW/LPI block/trip modification is installed during Refuel VII, the automatic tripping of the "A" battery charger will be eliminated. The tripping of the heat tracing will be retained. This load can be reapplied by the operator if needed.

5. Although the proposed long-term solution is conceptually acceptable, the EDGs would not have a desirable capacity margin, particularly with the return of loads previously removed. You have indicated that one group of EDG upgrades will be implemented by the next refueling, and that others are being considered. The capacity benefit of these upgrades and of those you propose to implement should be addressed.

RESPONSE: FPC response to this comment is in the section of this submittal entitled Emergency Diesel Generator Capacity Upgrades.

Florida Power
CORPORATION

MAR TRANSMITTAL MEMO
INTEROFFICE CORRESPONDENCE

Nuclear Operations Engineering (Office) C21 (Mail Code)

Dist. 11/21/89

SUBJECT: Crystal River Unit No. 3
Project Title DHP/EFP ES Start Modification
File: REI/MAR No. 88-05-24-01

TO: Nuclear Projects Superintendent
Crystal River No. 3

Attached is the completed and approved design package for the subject MAR.

Remarks

NRC approval for this change is required before it is made operational.

A new Safety Evaluation will be added via FCN at a later date.

"THIS MAR AFFECTS EDG LOADING"

Responsible A/E: G/C I

Design Engineer	Date	Supervisor Nuclear Engineering	Date
R. E. Clauson	11/9/89	P. M. Rubio	11/9/89

cc: Readers
Nucl. Ops. Sched. Analyst
Mgr., Site Nucl. Engrg. Serv. w/attach and dwgs
Mgr., Nucl. Engrg. Assurance w/attach, w/o dwgs
File (REI/MAR) w/attach; w/o dwgs.
Micromedia w/Engrg. Instructions & orig. dwgs
Drafting and CADD Section w/Section B of Engrg. Instructions
Nucl. Proj. Super. w/orig. & 7 copies; w/5 sets dwgs.
CR-3 Document Control w/attach; 4 sets
Nucl. Train. Control Supv. (NU47) w/attach; 1 set
Gilbert/Commonwealth w/1 a/c and Sec. B. of Eng. Inst.
Babcock & Wilcox w/1 a/c and Sec. B. of Eng. Inst.
Impell w/1 a/c and Sec. B. of Eng. Inst. (except B&W drawings)
A/E N/A w/attach and dwgs.
Mgr., Nucl. Config. Mgmt. w/attach, w/o dwgs.

NOE Clerk w/dwgs only
Mgr., Nucl. Fuel Mgmt. and Safety Analysis
Mgr., Nuc. Licensed Op. Tr. w/attach and dwgs.
Mgr., Nucl Plant Sys. Eng. w/attach.
Supervisor, SNES - Environmental Qualification
Mgr., Nuc. Procurement Eng. Services w/attach. and dwgs ☒ Yes ☐ No
Supervisor, SNES - Field Engineering
P.M. Rubio
R.E. Clauson, w/Eng. Inst. & Dwgs.
A. Petrowski, w/Eng. Inst. & Dwgs.
G. B. Poore w/attach, w/o Dwgs.
Nuclear Operations Engineering Supervisor Electrical, w/attach & Dwgs.
R. Merchese, w/attach & Dwgs.

RECEIPT OF DOCUMENTATION

MAR No.	Acknowledged By	Date
88-05-24-01	Joyce Greene	11-24-89

Return to: Power Plant Engineering Services Document Control, C2E, St. Petersburg, Florida

CR3 005812

REGULATORY/ENVIRONMENTAL REVIEW
Crystal River Unit 3

DOCUMENT NUMBER

MAR 88-05-24-01 ES ACTUATION EFW/DH START

REF. NUCLEAR OPERATIONS DEPARTMENT PROCEDURE NOD-11

1. **10 CFR 50.54 Reviews**

   Does this modification or document revision change what is described in any of the following plans/programs? If unable to determine, contact responsible person designated below.

Quality Program Description (FSAR Section 1.7) Contact: Director, Quality Programs	☐ Yes	☑ No
Licensed Operator Requalification Program (FSAR Section 12.2.3.4 and Appendix 12 C) Contact: Manager, Nuclear Operations Training	☐ Yes	☑ No
Modified Amended Security Plan Contact: Manager, Nuclear Licensing	☐ Yes	☑ No
Safeguards Contingency Plan Contact: Nuc. Security & Special Project Superintendent	☐ Yes	☑ No
Security Guard Training and Qualification Plan Contact: Nuc. Security & Special Project Superintendent	☐ Yes	☑ No
Radiological Emergency Response Plan Contact: Manager, Site Nuclear Services	☐ Yes	☑ No

   If any are "yes":

   a. Contact appropriate responsible person identified above to perform the evaluation and attach. The evaluation must be approved by the Division Head responsible for the plan/program. — Evaluation ☐ Complete ☑ N/A

   b. NRC approval received if needed and attach. — ☐ Complete ☑ N/A

2. **Environmental Protection Plan Review**

   a. Could this change affect the environment in a non-radiological way? — ☐ Yes ☑ No

      If "yes," contact the Manager, Nuclear Licensing. — ☐ Complete ☑ N/A

   b. Does the modification affect the National Pollutant Discharge Elimination System permit or associated state permit? — ☐ Yes ☑ No

      If "yes," contact the Supervisor, Water Programs Section to evaluate reporting requirements. Notification of such action must also be made to the Manager, Nuclear Licensing. — ☐ Submitted ☑ N/A

      EPA and/or FDER approval received, if needed, and attached. — ☐ Yes ☑ No

3. **Review for change to Radioactive Waste System (10 CFR 50.34a and Appendix I / T. S. 6.15.1.1)**

   Will this change to a radioactive waste system (liquid, gaseous or solid) result in an increase of radioactive material released to the environment? — ☐ Yes ☑ No

   If "yes", submit change to Manager, Nuclear Licensing to evaluate the reporting requirements. — ☐ Submitted ☑ N/A

IF ALL OF ABOVE ARE CHECKED NO, NO FURTHER REVIEWS ARE REQUIRED.

PREPARED BY	DATE	APPROVED BY	DATE
[signature]	7/27/89	[signature]	8/3/89

CR3 005813

ENCLOSURE 6

SAFETY EVALUATION REVIEW

TO: Nuclear Safety Supervisor

A copy of MAR 89-05-24-01 FCN — is attached for your review of the MAR Safety Evaluation for Technical Specification Conformance and Unreviewed Safety Implications.

* ☒ ACCEPTABLE

☐ UNACCEPTABLE Return immediately to the MAR Office for resolution

COMMENTS: * The stated safety evaluation appears to be acceptable; however, it should be noted that the NRC Authorization for FSAR and Tech Spec changes has not been received as of this date. (Letter NGA89-[illegible])

12/1/89
Date

[signature]
Nuclear Safety Supervisor
(Return to the MAR Office)

**SETPOINT CHANGE/PROCEDURE REVISION VERIFICATION
(PART A)**

MAR 88 - 05 - 24 - 01 - ___ FCN ___

INSTRUCTIONS: The attached MAR/FCN is routed to you for your review to determine if procedures existing or proposed within your area of responsibility will need to be revised upon completion of this modification.

RETURN THIS FORM TO THE PROJECTS DEPARTMENT (NAIB) UPON OF YOUR REVIEW. DO NOT ROUTE TO THE NEXT PERSON ON THE LIST.

NR3B NUC. OPS.

___ NO ✓ YES PROCEDURE CHANGES ARE REQUIRED (LIST #)

AP-380 EP-390 ~~VP-580~~ [illegible]

AR-301 AR-304 per Kov [illegible]

___ NO ✓ YES ___ N/A THIS MAR/FCN REQUIRES TURNOVER TO OPERATIONS IN ACCORDANCE WITH NPP-107. (N/A FOR FCN REVIEW UNLESS THE MAR HAS ALREADY BEEN TURNED OVER TO OPERATIONS OR CLOSED.)

___ NO ✓ YES ___ N/A PROCEDURE CHANGES ARE REQUIRED PRIOR TO MAR TURNOVER TO OPERATIONS. (LIST #)

SEE ABOVE !

COMMENTS REFER TO OPERATIONS IOC TO ENGINEERING...

PRINT NAME K.O. Vogel EXT. 4196 DATE 12/4/89

SETPOINT CHANGE/PROCEDURE REVISION VERIFICATION
(PART A)

NA1E NUCLEAR PLANT SYSTEMS ENGINEERING

___ NO X YES PROCEDURE CHANGES ARE REQUIRED (LIST #) SP-130 SP-136A,B,C SP-13 SP-357 SP-358A,B,C SP-41 SP416 SP-456 SP-457 SP-902

X NO ___ YES ___ N/A PROCEDURE CHANGES ARE REQUIRED PRIOR TO MAR TURNOVER TO OPERATIONS.

______ ______ ______

COMMENTS ____________________

PRINT NAME J.S. ENOSLEY EXT. 4446 DATE 12-6-89

NA2G NUCLEAR ISI SPECIALIST

___ NO ___ YES PROCEDURE CHANGES ARE REQUIRED (LIST #)

______ ______ ______

___ NO ___ YES ___ N/A PROCEDURE CHANGES ARE REQUIRED PRIOR TO MAR TURNOVER TO OPERATIONS.

______ ______ ______

COMMENTS ____________________

PRINT NAME ________ EXT. ________ DATE ________

SETPOINT CHANGE/PROCEDURE REVISION VERIFICATION
(PART A)

NR3A NUCLEAR CHEM/RAD PROT. SUPT.

___ NO ___ YES PROCEDURE CHANGES ARE REQUIRED (LIST #)

_______ _______ _______

___ NO ___ YES ___ N/A PROCEDURE CHANGES ARE REQUIRED PRIOR TO MAR TURNOVER TO OPERATIONS.

_______ _______ _______

COMMENTS ____

____

____

PRINT NAME ____________________ EXT. ________ DATE ____________

NA2E NUCLEAR SECURITY

___ NO ___ YES PROCEDURE CHANGES ARE REQUIRED (LIST #)

_______ _______ _______

___ NO ___ YES ___ N/A PROCEDURE CHANGES ARE REQUIRED PRIOR TO MAR TURNOVER TO OPERATIONS.

_______ _______ _______

COMMENTS ____

____

____

PRINT NAME ____________________ EXT. ________ DATE ____________

SETPOINT CHANGE/PROCEDURE REVISION VERIFICATION
(PART A)

NA1B	NUCLEAR PROJECTS SPECIALIST	___ NO _✓_ YES	PROCEDURE CHANGES ARE REQUIRED AND ARE IDENTIFIED.

PRINT NAME _Mike Jensen_ EXT. _[illegible]_ DATE _12-11-89_

ENCLOSURE 7
(Page 5 of 7)

Rev
11/28/89

SETPOINT CHANGE/PROCEDURE REVISION VERIFICATION
AND PREVENTIVE MAINTENANCE UPDATE
(PART B)

MAR 88 - 05 - 24 - 01 - ___ FCN ___ - ___

INSTRUCTIONS: The attached MAR/FCN is routed to you for your review to determine if procedures existing or proposed within your area of responsibility will need to be revised upon completion of this modification.

RETURN THIS FORM TO THE PROJECTS DEPARTMENT (NA1B) UPON COMPLETION OF YOUR REVIEW. DO NOT ROUTE TO THE NEXT PERSON ON THE LIST.

NA1E SITE NUCLEAR ENGINEERING SERVICES (ENVIRONMENTAL QUALIFICATION) X NO ___ YES PROCEDURE CHANGES ARE REQUIRED (LIST #)

___ ___ ___

Please indicate any EQ (10CFR50.49) requirements which must be incorporated into the MPs/PMs upon completion of this modification.

PRINT NAME Serrano EXT. 4302 DATE 12/18/89

NR3C NUCLEAR SUPT., PROJECTS/ NUCLEAR PROJECTS SPECIALIST X NO ___ YES PROCEDURE CHANGES ARE REQUIRED (LIST #)

___ ___ ___

X NO ___ YES PREVENTATIVE MAINTENANCE CHANGES ARE REQUIRED.

PRINT NAME RHMilliken EXT. 4704 DATE 1/8/90

CR3 005819

**SETPOINT CHANGE/PROCEDURE REVISION VERIFICATION
AND PREVENTIVE MAINTENANCE UPDATE
(PART B)**

NA2H NUCLEAR FIRE PROTECTION SPECIALIST ___ NO ___ YES PROCEDURE CHANGES ARE REQUIRED (LIST #)

_______ _______ _______

PRINT NAME ____________________ EXT. _______ DATE __________

NA2F NUCLEAR SAFETY AND RELIABILITY X NO ___ YES PROCEDURE CHANGES ARE REQUIRED (LIST #)

_______ _______ _______

PRINT NAME Raleigh Smith EXT. 4477 DATE 1-12-90

OTHERS:

___ ____________ ___ NO ___ YES PROCEDURE CHANGES ARE REQUIRED (LIST #)
____
____

_______ _______ _______

PRINT NAME ____________________ EXT. _______ DATE __________

___ ____________ ___ NO ___ YES PROCEDURE CHANGES ARE REQUIRED (LIST #)
____
____

_______ _______ _______

PRINT NAME ____________________ EXT. _______ DATE __________

SETPOINT CHANGE/PROCEDURE REVISION VERIFICATION AND PREVENTIVE MAINTENANCE UPDATE (PART B)

NAIB	NUCLEAR PROJECTS SPECIALIST	✓ NO ___ YES PROCEDURE CHANGES ARE REQUIRED AND ARE IDENTIFIED.

PRINT NAME Mike Jones EXT. 4404 DATE 5-15-91

DESIGN DATA SHEET
Crystal River Unit 3

SHEET 1 OF 2

RID / MAR NUMBER

MAR 88-05-24-01

PROJECT

DHP/EFP ES START MODIFICATION

SYSTEM

ES

DISCIPLINE: ☐ Electrical ☒ I & C ☐ Mechanical ☐ Structural

SAFETY CRITERIA: ☒ Safety List System

☐ Safety Classification Review Form (attach copy)

SAFETY RELATED: ☒ Yes ☐ No

YES	NO	APPLICABLE DESIGN INPUT REQUIREMENTS
1. ☒	☐	Basic Functions of each structure, system and component.
2. ☒	☐	Performance requirements such as capacity, rating, system output.
3. ☒	☐	Codes, standards, and regulatory requirements including the applicable issue and/or addenda.
4. ☒	☐	Design conditions such as pressure, temperature, fluid chemistry and voltage.
5. ☒	☐	Loads such as seismic, wind, thermal and dynamic.
6. ☒	☐	Environmental conditions anticipated during storage, construction and operation such as pressure, temperature, humidity, corrosiveness, site elevation, wind direction, nuclear radiation, electromagnetic radiation and duration of exposure. 10CFR50.49 applicability — For electrical equipment only, attach copy of Safety List page or attach copy of Environmental Qualification Requirements form if the system item is to be added or changed in the Safety List System.
7. ☒	☐	Interface requirements including definition of the functional and physical interfaces involving structures, systems and components.
8. ☒	☐	Material requirements including such items as compatibility, electrical insulation properties, protective coating and corrosion resistance.
9. ☐	☒	Mechanical requirements such as vibration, stress, shock and reaction forces.
10. ☐	☒	Structural requirements covering such items as equipment foundations and pipe supports.
11. ☐	☒	Hydraulic requirements such as pump net positive suction heads (NPSH), allowable pressure drops, and allowable fluid velocities.
12. ☐	☒	Chemistry requirements such as provisions for sampling and limitations on water chemistry.
13. ☒	☐	Electrical requirements such as source of power, voltage, raceway requirements, electrical insulation and motor requirements.
14. ☐	☒	Layout and arrangement requirements, to include potential adverse affects of non-seismically qualified masonry walls.
15. ☒	☐	Operational requirements under various conditions such as plant startup, normal plant operation, plant shutdown, plant emergency operation, special or infrequent operation, and system abnormal or emergency operation.
16. ☒	☐	Instrumentation and control requirements including indicating instruments, controls and alarms required for operation, testing, and maintenance. Other requirements such as the type of instrument, installed spares, range of measurement, and location of indication should also be included.
17. ☐	☒	Access and administrative control requirements for plant security.
18. ☒	☐	Redundancy, diversity and separation requirements of structures, systems and components.
19. ☒	☐	Failure effects requirements of structures, systems and components, including a definition of those events and accidents which they must be designed to withstand.
20. ☒	☐	Test requirements including in-plant tests and the conditions under which they will be performed.
21. ☒	☐	Accessibility, maintenance, repair and inservice inspection requirements for the plant including the conditions under which these will be performed.

DESIGN ENGINEER	DATE	VERIFICATION ENGINEER	DATE	SUPERVISOR NUCLEAR ENGINEERING	DATE
C.C. Stauffer	7/13/89	Roman Ostrowski	7.14.89	R E L	7/21/89
C.C. Stauffer	10/3/89	Roman Ostrowski	10.4.89	R E L	10/12/89

Rev. 6/88

CR3 005822

DESIGN DATA SHEET

Crystal River Unit 3

MAR 88-05-24-01

	YES	NO	APPLICABLE DESIGN INPUT REQUIREMENTS: (Continued)
22.	☐	☒	Personnel requirements and limitations including the qualification and number of personnel available for installation, plant operation, maintenance, testing and inspection as they pertain to permissible personnel radiation exposures for specified areas and conditions. (ALARA)
23.	☐	☒	Transportability requirements such as size and shipping weight, limitations, I.C.C. regulations.
24.	☐	☒	Fire protection or resistance requirements: (Check applicable letter(s)) ☐ a. Changes or additions/deletion of fire detection/suppression systems or equipment. ☐ b. Changes or additions to the plant configuration that change the effectiveness of existing fire detection/suppression systems.
25.	☐	☒	Handling, storage and shipping requirements. Include equipment protection requirements for in-place storage or system lay-up as applicable.
26.	☐	☒	Other requirements to prevent undue risk to the health and safety of the public.
27.	☒	☐	Materials, processes, parts and equipment suitable for application.
28.	☐	☒	Safety requirements for preventing personnel injury including such items as radiation hazards, restricting the use of dangerous materials, escape provisions from enclosures, and grounding of electrical systems.
29.	☒	☐	Addition or relocation of safe shutdown equipment, systems, components, or circuits that require compliance with the separation criteria stated in 10CFR50, Appendix R.
30.	☐	☒	The modification will modify an item that: (Check applicable letter(s)) ☐ a. was originally procured and installed in an unmodified state and for which spare parts are stocked; or, ☐ b. is being procured as a part of this modification and will be modified after receipt; or, ☐ c. is currently in inventory (FIMIS) in an unmodified state and will be modified prior to installation and/or restocking ☐ d. will be completely or partially replaced by an item of different design or materials. If "Yes" to any of the criteria, forward a completed copy of the MAR to the Manager, Nuclear Procurement Engineering Services.
31.	☒	☐	The modification adds or relocates control room equipment entailing direct operator interface that requires Human Factors design review per the criteria stated in NUREG-0700 and/or adds, deletes or modifies any other equipment which could affect the characteristics of plant operation as displayed to the plant operator. (Ref: FPC Specification SP-5145)
32.	☐	☒	The modification represents a replacement which is governed by ASME Section XI, IWA-7000.
33.	☒	☐	The modification involves additions and/or deletions to EDG loading.
34.	☐	☒	High Energy Line Break (HELB) design requirements.
35.	☐	☒	The modification will delete or modify inputs to the Safety Parameter Display System (SPDS).
36.	☒	☐	The modification involves a location classified as a dominant area of concern (DAC) for Station Blackout Equipment (SBO). If "Yes," forward a completed copy of the MAR to the Nuclear Engineering Supervisor, Electrical.
37.	☐	☒	The modification will affect exterior physical structures in the protected area and/or within 20 feet outside the protected area.

Florida Power CORPORATION

DESIGN INPUT RECORD

Crystal River Unit 3

MAR NUMBER	88-05-24-01	☐ Electrical	☒ I&C	☐ Mechanical	☐ Structural

Project: (Required for page one only) DHP/EFP ES START MODIFICATION

1. Emergency Diesel Generator (EDG) load capacity limitations have made it necessary to modify the Engineered Safeguards (ES) loading sequence to maintain the EDG loads below the 3000 kW, 2000 hour rating. During a March 30, 1988 meeting with the NRC, FPC presented a proposed EFW pump block/trip design concept which would resolve the loading concerns. This meeting was followed by Letter 3F0788-18 on July 22, 1988 and a subsequent meeting on August 16, 1988. In a September 30, 1988 letter to the NRC (3F0988-22), FPC advised that the ES actuation logic would be changed to prevent simultaneous actuation of the emergency feedwater and low pressure injection pumps. This modification implements the ES logic change based on the Conceptual Design forwarded by FCS-9894 (November 4, 1988), FPC's request for scope letter, G/CI #2821 (December 30, 1988), and the revised scope per FPC's request per FCS #10784 (August 24, 1988).

Specifically, the modification precludes simultaneous actuation of both the EFP-1 and DHP-1A, based on a 500 psi Reactor Coolant System pressure and create the following ES load blocks:

<u>Load Block 1</u>

High Pressure Injection Pump MUP-1A (MUP-1B)
Miscellaneous MCC Loads And Motor Operated Valves
Inverters
Control Complex Lighting
Miscellaneous AC Distribution Panels
Transformer
Battery Chargers
Emergency Lighting

<u>Load Block 2</u>

Reactor Building Fan AHF-1A (AHF-1B)
Emergency Nuclear Services Seawater Pump RWP-2A (RWP-2B)
Flush Water Pump DOP-2A (DOP-2B)

<u>Load Block 3</u>

Emergency Nuclear Services Closed Cycle Cooling Water Pump SWP-1A (SWP-1B)
Decay Heat Closed Cycle Cooling Fan AHF-15A (AHF-15B)

Design Engineer	Date	Verification Engineer	Date	Supervisor, Nuclear Engineering	Date
C.C. Stampfle	10/3/89	R. Kalynowski	10.9.89	R E Vega	10/12/89

EF Block

Motor Driven Emergency Feedwater Pump EFP-1A.

Load Block 4

Decay Heat Removal Pump DHP-1A (DHP-1B) permissive.

Load Block 5

Decay Heat Seawater Pump RWP-3A (RWP-3B).

Load Block 6

Decay Heat Closed Cycle Cooling Water Pump DCP-1A (DCP-1B)
Reactor Building Spray Pump BSP-1A (BSP-1B) permissive.

The basic function of the Engineered Safeguards (ES), the Decay Heat System, and the Emergency Feedwater (EFW) system is not changed due to this MAR. However, the systems is modified as follows:

a. The ES actuation matrix for starting the LPI pumps (DHP-1A and DHP-1B) is changed from a 1500 psig Reactor Coolant (RC) pressure to a 500 psig RC pressure energize to trip.

b. The ES 4 psig Reactor Building (RB) pressure diverse actuation signal for LPI pump start remains when OFFSITE-POWER exists. However, the 4 psig RB signal is blocked from starting the LPI pumps when offsite power is unavailable.

c. The ES actuation matrix for sequence starting the Decay Heat Closed Cycle Cooling pumps (DCP-1A and DCP-1B) is moved from Block 4 to Block 6. Block 6 is set to start at 25 seconds.

d. The ES actuation permissive matrix for sequence starting the Reactor Building Spray pumps (BSP-1A and 1B) is moved from Block 5 to Block 6. Block 6 is set to start at 25 seconds.

e. The ES actuation matrix for sequence starting the Raw Water pumps (RW-03A and 03B) is moved from Block 4 to Block 5. Block 5 is set to start at 20 seconds.

f. The ES actuation permissive matrix for sequence starting the LPI pumps (DHP-1A and 1B) is moved from Block 1 to Block 4. Block 4 is set to start at 15 seconds.

g. The Emergency Feedwater motor driven pump (EFP-1) has a dedicated ES sequence start matrix to start 5 seconds after EDG loading on Loss of offsite power (Loop) without ES actuation and 5 seconds after Block 3 loading with ES actuation. If an ES condition occurs after LOOP the EFP-1 is tripped and re-sequenced. If the ES condition is a LOCA that allows RC pressure to decrease below 500 psig with a LOOP event, EFP-1 is tripped due to LPI pumps starting. There is no EFP-1 trip function during non-LOOP events.

h. The ES status lights for actuation and testing of the above items (a) through (g) is re-arranged as necessary on the ESF section of the main control board, and ES relay cabinets, actuation cabinets, and test light panel.

i. The manual ES actuation of LPI will not cause automatic start of DHP-1A and DHP-1B pumps until HPI is either automatically or manually ES actuated to properly sequence load the pump by the ES sequence timing permissive matrix.

j. The manual actuation of LPI with a LOOP will trip EFP-1.

2. The performance requirements of the ES, DC, DH, BS, RW, and EF systems have not changed from the original plant installation due to this MAR.

3. a. IEEE 279-1968, Criteria for Protective Systems for Nuclear Power Generating.

   b. 10CFR50 Appendix R, Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979.

   c. EP-5145, Human Factors Design Conventions for the Control Room.

   d. Control Board and Relay Rack ES Separation Criteria Report for Crystal River Unit 3, May 18, 1972.

4. This MAR does not affect the design conditions of the fluid process for the BS, DC, DH, RC, RW, or EF systems. Voltage for ES status lights on the MCB shall be 28 V. Status lights in relay panels shall be powered from their control circuits. Power supplies for control circuits shall be 125V DC or 120V AC.

Florida Power Corporation

DESIGN INPUT RECORD CONTINUATION SHEET

Crystal River Unit 3

88-05-24-01

5. The additional status lights and cut-out holes in the ESF section of the Main Control Board have been reviewed and by engineering judgement have negligible effect on the seismic static loads. The additional test lights, relays, and relay contacts to be located in the ES actuation cabinets have been reviewed and by engineering judgement have negligible effect on the seismic static loads.

   Relays added for this MAR shall be the same as currently used in the racks to assure seismic operability.

6. All equipment modification and wiring occurs within the mild environment of the Control Building EQ Zone 13. 10CFR50.49 does not apply.

7. This modification interfaces the BS, DC, DH, ES, EF, and RW systems. The ES actuation matrices are being modified for pump sequence loading and for LPI actuation and EFW pump EFP-1 trip.

   EFP-1 is loaded 5 seconds after EDG Breaker Closure upon Loss of Offsite Power without ES actuation. The EFW turbine driven pump (EFP-2) is unaffected by this modification. No modification is being made to the EFIC system for actuation logic and control.

   MAR 87-07-08-01 "ES Matrices Bypass Status Lamps", MAR 87-10-06-01 "Application of ES Clark Relays", and MAR 87-07-06-01 "BS Pump Testing Circuit" interfaces with this modification.

8. Additional relays and relay contacts shall be similar to and compatible with existing relays. All wiring shall be from existing, approved site stock and will be installed in the control building.

13. All control wiring shall have 600 volt insulation. Status light wiring may have 300 volt insulation. No change in power source is required for this MAR. Relay contact requirements are similar to existing relays.

15. Operational requirements for plant startup, normal operation, shutdown and emergency operation remain the same. However, the operational procedures will need revision to (1) reflect the starting of LPI pumps on 500 psig instead of 1500 psig, (2) the tripping of EFP-1 upon 500 psig RC pressure, and (3) to assure that the HPI is manually tripped whenever LPI is manually tripped. Tripping of the motor driven EF pump has no effect on the turbine driven EF pump. Operation of the steam driven turbine EF pump is covered by existing procedures.

    Operation of the turbine EF pump may continue as long as steam is available to drive the pump. If steam is not available and EFW is required, then the ES must be manually reset and EFP-1 re-started manually until Decay Heat can be made available to the RC System.

[illegible] EFP-2 [illegible] EFW REQUIRED?

16. Sequence loading of the EDG occurs whenever there is an ES actuation or a LOOP event. To evenly load the EDG and prevent excessive voltage or frequency dips the ES sequence block loading is modified as follows:

   a. The DHP-1A & 1B loading is moved <u>from</u> Block 1 ES matrix <u>to</u> Block 4 ES matrix.

   b. The DCP-1A & 1B loading is moved <u>from</u> Block 4 ES matrix <u>to</u> Block 6 ES matrix.

   c. The RWP-3A & 3B loading is moved <u>from</u> Block 4 ES matrix <u>to</u> Block 5 ES matrix.

   d. The BSP-1A & 1B loading is moved <u>from</u> Block 5 ES matrix <u>to</u> Block 6 ES matrix.

   e. The EFP-1 loading is moved from Block 5 to a new dedicated EFW Block to load 5 seconds after the EDG is loaded if a LOOP event occurs without ES conditions. For non-LOOP events the EFP-1 pump starts immediately. For Large LOCA conditions with a LOOP event the EFP-1 pump will be tripped by the LPI coincident logic. For a LOOP event followed by an ES condition after EFP-1 has been sequence loaded, EFP-1 will be ES tripped and re-sequenced to load 5 seconds after Block 3.

The sequence logic shown on a typical elementary 208-028 ES-A20 is described as follows:

Block	Relay	Time Delay
1	AG, AH, AL, AM	NONE
2	AS	5 Seconds
3	AV	10 Seconds
4	AQ	15 Seconds
5	CD	20 Seconds
6	CE	25 Seconds
EFW	CH	5 Seconds after Block 3 conditional

These sequencing relays are normally energized Agastat (time delay to de-energize). Since these Agastat relays have limited contacts auxiliary relays (AW, AX, AY, AZ, CF, CG, & CJ) are used to provide contacts for status lights and ES actuation matrices.

ES Actuation (Offsite Power Available)

Block 1 relays are automatically de-energized by an ES 'OR' logic of 1500 psig RC pressure, 500 psig RC pressure or a 4 psig RB pressure. Blocks 2, 3, 4, 5 and 6 are sequenced at 5 second intervals and are also de-energized by logic similar to Block 1. The EFW block relay CH will de-energize 5 seconds after Block 3 and provides a two-out-of-three matrix logic to start EFP-1; however, with offsite power available, the ES HPI signal which initiates sequencing will also initiate EFIC causing both Block 1 loads and EFP-1 to start at the same time. The Block 4 load for LPI provides a two-out-of-three matrix permissive start signal to the DH pumps. The actual start signal for LPI is controlled by the 4 psig RB pressure and 500 psig RC pressure 'OR' logic or the 'OR' logic of the normally energized 'S' and 'W' relays as depicted by a typical elementary ES-A35.

LOOP Conditions (ES not actuated)

For LOOP conditions, the undervoltage relays initiate (close) the EDG buss loaded contact 27XA as depicted on typical elementary 208-028 ES-A20. Since a LOOP condition exists and ES has not been actuated, EFP-1 as depicted on elementary 208-026 EF-01 will be started by the ES sequence matrix ES-A29 closing 5 seconds after the EDG is loaded. Since an HPI condition does not exist, no other sequencing occurs. The LOOP condition will energize the BS relay which will prevent the 4 psig RB pressure from actuating LPI as depicted on typical elementary 208-028 ES-A35.

9/22/88

Florida Power Corporation

ENGINEERING INSTRUCTIONS CONTINUATION SHEET

Crystal River Unit 3

88-05-24-01

Sheet 10 of 100

C. **References**

EDG accumulator timer installation

(1) MAR 87-10-19-01A, MAR 87-07-08-01, MAR 87-10-08-01.
(2) G/CI Letter FCS #10419 "System Flows for EDG Loading Evaluation".
(3) G/CI telecon memo, C.C. Strempke to Don Cuzzo dated 4-3-89 "SER points".
(4) G/CI Letter FCS-10125, dated 1-25-89, scope letter.
(5) G/CI Letter FCS-9894, dated 11-4-88, conceptual design.
(6) G/CI telecon memo, C.C. Strempke to Ron Clauson dated 5-16-89. "Annunciator windows."
(7) G/CI speed memo, ES panels & cabinet seismic evaluation. 5/22/89.
(8) G/CI Letter FCS-10424, dated 6-16-89, Purchase Requisitions.
(9) Calculation DC-5515-004-1.02 CS, and DC-5515-004-2 ...
(10) G/CI Letter FCS-10420, dated 4-7-89, and FCS-10805, dated 9-11-89. FSAR Transmittal.
(11) G/CI Letter FCS-10784, dated 8-24-89, revised scope letter.
(12) G/CI Letter FCS-10814, dated 9-12-89, "EDG loads".
(13) IOC, M.W. Averett to R.E. Clauson, dated 7-11-88, "Reliability Analysis for the Proposed EFP-1/LPI Logic Modification".
(14) B&W Report 51-117603-00 - "Acceptability of a 500 psig start of LPI for all Loss of Coolant Accidents".
(15) G/C, Inc. Drawings: 201-129, Main Control Board Equipment Cutout Details
201-130, Main Control Board Equipment Cutout Details
(16) Design Basis Document Temporary Change ... Emergency Feedwater (Sect 6/.3), Decay Heat Removal System (Sect 6/3), Eng Safeguards Actuat Systs. (Sect 6/3) ...

D. **Material**

See attached Bill of Materia.

E. **Installation Instructions**

1. Charge all time and material to NU88052401.

2. All cable handling, pulling, termination, conduit installation, equipment removal and installation shall be in accordance with applicable FPC maintenance procedures. All relays removed shall be returned to Nuclear Materials Coordinator.

3. De-energize ES system Loop A.

1a. This MAR shall not be made operational until NRC approval for the change has been received. A new Safety Evaluation will be added to the MAR via FCN after NRC approval

11/8/89

18. The ES system provides redundant trains of actuation which is not changed by this modification. (There is no motor driven EFW pump on ES "B" train.) The starting of the LPI pumps has been modified on both trains of ES from 1500 psig RC pressure to 500 psig. The status lights for actuation and testing have been modified on both trains of the ES. No sharing of circuits of either ES train shall occur by this modification.

Diversity of initiating LPI by RB pressure and low RC pressure is retained when offsite power is available. Whenever offsite power is lost, the RB pressure initiate signal is bypassed. The probability of having a large break LOCA condition that requires LPI occurring coincident with a LOOP event is 4.8 X 10^{-8} events/year per the FPC PRA.. The ES actuation of LPI is a reliable 2 out of 3 logic and since the coincidence of LOOP and LOCA is a low probability, the loss of diversity during a LOOP is acceptable. See page 8a for continuation

19. The failure effects of the LPI and EF systems are not changed by this modification. The ES system failure modes also remain the same. Only the LPI actuation matrix has changed from a 1500 psig RC pressure signal to a 500 psig RC pressure signal. The actuation of LPI still occurs on diverse signals of low RC pressure (500 psig) and high RB pressure signal (4 psig) for all accidents except when coincident with loss-of-offsite-power. For loss-of-offsite-power, the 4 psig RB pressure signal is bypassed by an auxiliary relay contact requiring an energize to bypass from the 4160V switchgear breaker contacts. A single failure of a contact or relay will not cause this bypass since the bypass is upstream of the two-out-of-three matrix.

Sequence block loading has changed by this modification without introducing additional failure modes. The DCP-1A and DCP-1B pumps have been moved from Block 4 to Block 6. DHP-1A & DHP-1B pumps have been moved from Block 1 to Block 4. The EFP-1 pump is assigned to a new Block set at 5 seconds after Block 3 loading. The BSP-1A & BSP-1B pumps have been assigned to a new Block 6 set at 25 seconds. The RWP-3A & 3B pumps have been moved from Block 4 to Block 5.

The ES trip matrix for EFP-1 is initiated by a coincidence logic of low RC pressure (500 psig) <u>AND</u> loss-of-offsite-power. This coincident logic is upstream of the two-out-of-three matrix such that a single failure of a relay or contact does not trip the motor driven pump. The coincident logic for tripping and re-sequencing the EF pump is also upstream of the two-out-of-three matrix such that a single failure of a relay or contact does not trip the motor driven EFW pump.

This modification does not affect any HELB areas.

RPF 10/3/88

Next Page 8a of 10

Florida Power Corporation

DESIGN INPUT RECORD CONTINUATION SHEET

Crystal River Unit 3

MAR NUMBER 88-05-24-01

18. Continued

This MAR changes color coding of wires within the ES Actuation Cabinets 4A, 4B, 4C, 5A, 5B, and 5C. This is required since this MAR interchanges the output function of matrices from relays AJ and AN. The change will be identified by taping the ends of the wire with the apropriate color tape. The matrices from relays AN will be taped brown in the "A" cabinets and orange in the "B" cabinets since these are alarm ciruits. The matrices from relays AJ will be taped red and green in the "A" and "B" cabinets, respectively, since these matrices are for DH valve actuation.

This MAR also changes color coding of wires within the ES Logic Cabinet 2B. This change was initiated to avoid confusion over color code in the cabinet and to make the color code consistent with its function. The change in color code will be identified by wrapping the ends of the wire with green tape. The change will be applied for the LPI logic from the fuse within the cabinet to the relays. Since the wires to be taped are orange (and orange and green are run together) rerouting and/or rebundling of the wires is not required.

10/31/88

11/8/89

Next Page 9 of 10

20. The test requirements for the ES system remains the same as currently designed. The test procedures will need modification to reflect load sequencing, new setpoints and bypass functions. Additional test switches for testing the LOOP logic have been added to the ES actuation cabinets, 4D and 5D.

A test switch contact and loss-of-offsite-power relay contact have been added to each DHP-1A and DHP-1B pump trip circuit.

The EFP-1 START and TRIP circuitry has been simplified by this change. The pump will be load sequenced and Block loaded only upon Loss of Offsite Power. The pump will be tripped upon a logic condition of LPI and Loss of Offsite Power. If an ES condition occurs after the pump has been sequence loaded it will be tripped and re-sequenced.

The Building Spray pumps (BSP-1A & 1B) logic has been simplified with a Block 6 load sequence permissive signal from an ES matrix set at 25 seconds T.D.

The Decay Heat Pumps (DHP-1A & 1B) logic has been changed from Block 1 to Block 4 at 15 seconds T.D. and the 1500 psig RC pressure initiate signal is removed. Upon LOOP conditions the 4 psig RB pressure signal is bypassed.

The RWP-3A & 3B pumps are load sequenced at 20 seconds T.D.

The DCP-1A & 1B pumps are load sequenced at 25 seconds T.D.

The BS pump test lights for ES matrix is a special configuration per MAR 87-07-06-01. Existing test procedures provide the testing method. This MAR 88-05-24-01 changes the single relay contact permissive to an ES matrix permissive as depicted on dwgs. ES-A30, B30, A66 and B66.

The DH pump test lights for ES matrices are a special configuration per this MAR 88-05-24-01 as depicted on 208-021 DH01 and DH-02. A single set of lights are utilized to test both the HPI permissive matrix using the HPI test switch and the LPI actuation matrix using the LPI test switch.

21. All equipment of this modification is accessible for maintenance and testing. There are no additional in-service inspection requirements.

27. All material shall be from site stock or be similar to existing wire, switches, and relay contacts suitable for safety systems.

29. The additional DHP-1A and DHP-1B pump trip circuit being installed by this modification for loss-of-offsite-power shall be routed and/or protected to prevent tripping of both pumps due to a fire.

88-05-24-01

31. The ES actuation status lights on the ESF section of the main control board (MCB) shall be re-configured to:

    a. Reflect the 500 psig trip signal for LPI

    b. Reflect the block loading of DHP-1A & 1B, DCP-1A & 1B, EFP-1, BSP-1A & 1B and RWP-3A & 3B.

    The ES test matrix status lights on the actuation cabinets shall be reconfigured reflect items (a) and (b).

Layout and labeling shall follow the guidance of SP-5145.

33. The purpose of this modification is to reduce EDG loading. Both the LPI pumps and the EFW pumps need not operate simultaneously during LOCA events coincident with loss-of-offsite-power. Hence, either the LPI or the EFW pumps will be running during this scenario.

To accomplish the above the ES system relay logic and status light's requires changes resulting in the following maximum load changes.

Haberstroh 11-8-89

EDG

A	VB3A	VBDP-3	BKR #1	,	96.42 watt increase
A	VB3C	VBDP-5	BKR #1	,	96.42 watt increase
B	VB3B	VBDP-4	BKR #1	,	96.42 watt increase
A	VB3A	VBDP-3	BKR #7	,	17.92 watt increase
B	VB3B	VBDP-4	BKR #2	,	65.28 watt increase
A	VB3C	VBDP-5	BKR #2	,	65.28 watt increase
B	VB3D	VBDP-6	BKR #2	,	65.28 watt increase
B	VB3B	VBDP-4	BKR #7	,	15.68 watt increase
A	VB3A	VBDP-3	BKR #6	,	8.43 watt decrease
A	BUSS3A	DPDP-5A	Fuse #3	,	7[illegible].28 watt increase
B	BUSS3B	DPDP-5B	Fuse #3	,	81.84 watt increase
A	BUSS3A	DPDP-5A	Fuse #10	,	1.28 watt decrease
B	BUSS3B	DPDP-5B	Fuse #10	,	.05 watt increase

36. The modification occurs in the Control Building at elevations 108', 124', 134' and 145'. DAC's are CB303 CB304, CB305, CB504, CB208 and CB209.

CR3 005835

Florida Power

EMERGENCY DIESEL GENERATOR LOADING JUSTIFICATION

DIR Sheet 10A

MAR / PCN NUMBER	DATE
MAR 88-05-24-01	July 3, 1989
PROJECT TITLE	
EF/DH START MODIFICATION	

One of the below listed items must be checked.

☐ The load is needed to bring the plant to a safe condition after a design basis accident.

☐ The load is required to be EDG backed per NRC regulations (e.g., Reg. Guide, NUREG, IE Bulletin, etc.). The NRC regulation is identified as follows:

☑ The load belongs to a system previously analysed and documented in the FSAR as requiring to be EDG backed. The applicable FSAR section is identified as follows:

Section 6, 7, 8.0 and 10

☐ Other (Explain)

Last page

CR3 005836

VERIFICATION REPORT
Crystal River Unit 3

Florida Power

REI / MAR No.
88-03-24-01

PROJECT: DHP/EFP ES START MODIFICATION

VERIFICATION METHOD: [X] Design Review ☐ Alternate Calculations
☐ Qualification Testing

	YES	NO	N/A	
1.	X	☐		Were the inputs correctly selected and incorporated into design?
2.	☐	☐	X	Are assumptions necessary to perform the design activity adequately described and reasonable? Where necessary, are the assumptions identified for subsequent reverifications when the detailed design activities are completed?
3.	X	☐		Are the appropriate quality and quality assurance requirements specified?
4.	X	☐		Are the applicable codes, standards and regulatory requirements including issue and addenda properly identified and are their requirements for design met?
5.	X	☐	☐	Have applicable construction and operating experience been considered?
6.	X	☐	☐	Have the design interface requirements been satisfied?
7.	X	☐		Was an appropriate design method used?
8.	X	☐		Is the output reasonable compared to inputs?
9.	X	☐	☐	Are the specified parts, equipment, and processes suitable for the required application?
10.	X	☐	☐	Are the specified materials compatible with each other and the design environmental conditions to which the material will be exposed?
11.	X	☐	☐	Have adequate maintenance features and requirements been specified?
12.	X	☐	☐	Are accessibility and other design provisions adequate for performance of needed maintenance and repair?
13.	X	☐	☐	Has adequate accessibility been provided to perform the in-service inspection expected to be required during the plant life?
14.	X	☐	☐	Has the design properly considered radiation exposure to the public and plant personnel?
15.	X	☐	☐	Are the acceptance criteria incorporated in the design documents sufficient to allow verification that design requirements have been satisfactorily accomplished?
16.	X	☐	☐	Have adequate pre-operational and subsequent periodic test requirements been appropriately specified?
17.	X	☐	☐	Are adequate handling, storage, cleaning and shipping requirements specified?
18.	X	☐	☐	Are adequate identification requirements specified?
19.	☐	☐	X	Are requirements for record preparation review, approval, retention, etc. adequately specified?
20.	X	☐	☐	Has seismic adequacy been considered and evaluated and are the results acceptable?
21.	X	☐	☐	Have fire protection considerations been adequately addressed?
22.	☐	☐	X	Have High Energy Line Break (HELB) considerations been adequately addressed?

I have performed a verification on the subject MAR design package and find the results
[X] acceptable ☐ unacceptable

COMMENTS
Subject documents were reviewed against one another. All questions resolved.

Verification Engineer: [illegible] Date: 7.14.89 | [illegible] 7/27/89
[illegible] 10.9.89 | [illegible] 10/12/89

CCS

CR3 005837

Florida Power Corporation

VERIFICATION REPORT
Crystal River Unit 3

MDI / MAR No.
MAR 88-05-24-01

PROJECT: EF/DH START MODIFICATION

VERIFICATION METHOD: ☒ Design Review (see checklist below) ☐ Alternate Calculations (attach analysis calculation sheet) ☐ Qualification Testing (see attachments as necessary)

	YES	NO	N/A	
1.	☒	☐		Were the inputs correctly selected and incorporated into design?
2.	☐	☐	☒	Are assumptions necessary to perform the design activity adequately described and reasonable? Where necessary, are the assumptions identified for subsequent reverifications when the detailed design activities are completed?
3.	☒	☐		Are the appropriate quality and quality assurance requirements specified?
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7.	☒	☐		Was an appropriate design method used?
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21.	☐	☐	☒	Have fire protection considerations been adequately addressed?
22.	☐	☐	☒	Have High Energy Line Break (HELB) considerations been adequately addressed?
23.	☐	☐	☒	Have Environmental Qualifications requirements been identified and adequately addressed?

I have performed a verification on the subject MAR design package and find the results
☒ acceptable ☐ unacceptable

COMMENTS
Design review of changes attached to FCS-10981 against Design Inputs.

Verification Engineer: R. A. Kamme for R. Pstynowski Date: 11/6/89 Paul [illegible] Date: 11/8/89

Florida Power Corporation

VERIFICATION REPORT
Crystal River Unit 3

MAR 88-05-24-01

PROJECT: EF/DH START MODIFICATION

VERIFICATION METHOD: ☒ Design Review (see checklist below) ☐ Alternate Calculations (attach analysis calculation sheet) ☐ Qualification Testing (see attachments as necessary)

#	YES	NO	N/A	
1.	☒	☐		Were the inputs correctly selected and incorporated into design?
2.	☐	☐	☒	Are assumptions necessary to perform the design activity adequately described and reasonable? Where necessary, are the assumptions identified for subsequent reverifications when the detailed design activities are completed?
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6.	☒	☐	☐	Have the design interface requirements been satisfied?
7.	☒	☐		Was an appropriate design method used?
8.	☒	☐		Is the output reasonable compared to inputs?
9.	☒	☐	☐	Are the specified parts, equipment, and processes suitable for the required application?
10.	☐	☐	☒	Are the specified materials compatible with each other and the design environmental conditions to which the material will be exposed?
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12.	☒	☐	☐	Are accessibility and other design provisions adequate for performance of needed maintenance and repair?
13.	☐	☐	☒	Has adequate accessibility been provided to perform the in-service inspection expected to be required during the plant life?
14.	☐	☐	☒	Has the design properly considered radiation exposure to the public and plant personnel?
15.	☒	☐	☐	Are the acceptance criteria incorporated in the design documents sufficient to allow verification that design requirements have been satisfactorily accomplished?
16.	☐	☐	☒	Have adequate pre-operational and subsequent periodic test requirements been appropriately specified?
17.	☐	☐	☒	Are adequate handling, storage, cleaning and shipping requirements specified?
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19.	☐	☐	☒	Are requirements for record preparation review, approval, retention, etc. adequately specified?
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21.	☐	☐	☒	Have fire protection considerations been adequately addressed?
22.	☐	☐	☒	Have High Energy Line Break (HELB) considerations been adequately addressed?
23.	☐	☐	☒	Have Environmental Qualifications requirements been identified and adequately addressed?

I have performed a verification on the subject MAR design package and find the results
☒ acceptable ☐ unacceptable

COMMENTS

CUSTOMER REQUIREMENTS WERE INCORPORATED AS REQUESTED IN THE MAR PACKAGE

[signature] Date 11.1.89 [signature] 11/1/89

Florida Power CORPORATION

VERIFICATION REPORT
Crystal River Unit 3

[illegible] MAR 88-05-24-01

PROJECT: EF/DH START MODIFICATION

VERIFICATION METHOD: [x] Design Review (use checklist below) [ ] Alternate Calculations (attach analysis calculation sheet) [ ] Qualification Testing (use attachments as necessary)

	YES	NO	NA	
1.	[x]	[ ]		Were the inputs correctly selected and incorporated into design?
2.	[ ]	[ ]	[x]	Are assumptions necessary to perform the design activity adequately described and reasonable? Where necessary, are the assumptions identified for subsequent reverifications when the detailed design activities are completed?
3.	[x]	[ ]		Are the appropriate quality and quality assurance requirements specified?
4.	[x]	[ ]		Are the applicable codes, standards and regulatory requirements including issue and addenda properly identified and are their requirements for design met?
5.	[x]	[ ]	[ ]	Have applicable construction and operating experience been considered?
6.	[x]	[ ]	[ ]	Have the design interface requirements been satisfied?
7.	[x]	[ ]		Was an appropriate design method used?
8.	[x]	[ ]		Is the output reasonable compared to inputs?
9.	[x]	[ ]	[ ]	Are the specified parts, equipment, and processes suitable for the required application?
10.	[ ]	[ ]	[x]	Are the specified materials compatible with each other and the design environmental conditions to which the material will be exposed?
11.	[ ]	[ ]	[x]	Have adequate maintenance features and requirements been specified?
12.	[x]	[ ]	[ ]	Are accessibility and other design provisions adequate for performance of needed maintenance and repair?
13.	[ ]	[ ]	[x]	Has adequate accessibility been provided to perform the in-service inspection expected to be required during the plant life?
14.	[ ]	[ ]	[x]	Has the design properly considered radiation exposure to the public and plant personnel?
15.	[x]	[ ]	[ ]	Are the acceptance criteria incorporated in the design documents sufficient to allow verification that design requirements have been satisfactorily accomplished?
16.	[ ]	[ ]	[x]	Have adequate pre-operational and subsequent periodic test requirements been appropriately specified?
17.	[ ]	[ ]	[x]	Are adequate handling, storage, cleaning and shipping requirements specified?
18.	[x]	[ ]	[ ]	Are adequate identification requirements specified?
19.	[ ]	[ ]	[x]	Are requirements for record preparation review, approval, retention, etc. adequately specified?
20.	[ ]	[ ]	[x]	Has seismic adequacy been considered and evaluated and are the results acceptable?
21.	[ ]	[ ]	[x]	Have fire protection considerations been adequately addressed?
22.	[ ]	[ ]	[x]	Have High Energy Line Break (HELB) considerations been adequately addressed?
23.	[ ]	[ ]	[x]	Have Environmental Qualifications requirements been identified and adequately addressed?

I have performed a verification on the subject MAR design package and find the results
[x] acceptable [ ] unacceptable

COMMENTS
Design review of changes attached to FCS-10937 against Design Inputs

[illegible] [illegible] [illegible] 11/6/89

Florida Power

VERIFICATION REPORT
Crystal River Unit 3

MAR No.: MAR 88-05-24-01

PROJECT: EF/DH START MODIFICATION

VERIFICATION METHOD: ☒ Design Review (use checklist below) ☐ Alternate Calculations (attach analysis calculation sheet)
☐ Qualification Testing (use attachments as necessary)

	YES	NO	N/A	
1.	☒	☐		Were the inputs correctly selected and incorporated into design?
2.	☐	☐	☒	Are assumptions necessary to perform the design activity adequately described and reasonable? Where necessary, are the assumptions identified for subsequent reverifications when the detailed design activities are completed?
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4.	☒	☐		Are the applicable codes, standards and regulatory requirements including issue and addenda properly identified and are their requirements for design met?
5.	☒	☐	☐	Have applicable construction and operating experience been considered?
6.	☒	☐	☐	Have the design interface requirements been satisfied?
7.	☒	☐		Was an appropriate design method used?
8.	☒	☐		Is the output reasonable compared to inputs?
9.	☒	☐	☐	Are the specified parts, equipment, and processes suitable for the required application?
10.	☐	☐	☒	Are the specified materials compatible with each other and the design environmental conditions to which the material will be exposed?
11.	☐	☐	☒	Have adequate maintenance features and requirements been specified?
12.	☒	☐	☐	Are accessibility and other design provisions adequate for performance of needed maintenance and repair?
13.	☐	☐	☒	Has adequate accessibility been provided to perform the in-service inspection expected to be required during the plant life?
14.	☐	☐	☒	Has the design properly considered radiation exposure to the public and plant personnel?
15.	☒	☐	☐	Are the acceptance criteria incorporated in the design documents sufficient to allow verification that design requirements have been satisfactorily accomplished?
16.	☐	☐	☒	Have adequate pre-operational and subsequent periodic test requirements been appropriately specified?
17.	☐	☐	☒	Are adequate handling, storage, cleaning and shipping requirements specified?
18.	☒	☐	☐	Are adequate identification requirements specified?
19.	☐	☐	☒	Are requirements for record preparation review, approval, retention, etc. adequately specified?
20.	☐	☐	☒	Has seismic adequacy been considered and evaluated and are the results acceptable?
21.	☐	☐	☒	Have fire protection considerations been adequately addressed?
22.	☐	☐	☒	Have High Energy Line Break (HELB) considerations been adequately addressed?

I have performed a verification on the subject MAR design package and find the results
☒ acceptable ☐ unacceptable

COMMENTS

NO FURTHER QUESTIONS RAISED.

Verification Engineer	Date	Supervisor, Nuclear Engineering	Date
[signature]	10.18.89	[signature]	10/19/89

Rev 1/88 Mgr. of Plant Engineering Nuclear Engineering Assurance

CR3 005841

Operation of this system does not depend on the operation of any other ES. However, the system can be operated in series with the Decay Heat Removal System, if needed, in the recirculation mode.,

6.1.3.2 Low Pressure Injection System (LPI)

One pump can deliver approximately 3,000 gpm to the reactor vessel. Assuming the reactor had been operating at full power prior to the accident, the decay heat being generated in the core at 30 minutes after the accident is approximately 1.9% of full power.

Following a LOCA assuming a simultaneous loss of normal power sources, the emergency power source and the LPI System will be in full operation within the required ~~35~~ 25 seconds.

The LPI System is connected with other ES Systems in four respects, i.e., (1) the HPI, the LPI, and the Reactor Building Spray System take their suction from the BWST; (2) the LPI pumps and the reactor building spray pumps share common suction lines from the reactor building sump during the coolant recirculation mode; (3) the LPI System and the Core Flooding System utilize common injection nozzles on the reactor vessel; and (4) the HPI System may be operated through the LPI system if recirculation through the HPI pumps is required.

6.1.3.3 Core Flooding System

Injection response of the Core Flooding System (see Figure 6-2) is dependent upon the rate of reduction of RCS pressure. The capability of the Core Flooding System to reflood the core is described in Chapter 14.

The core flooding nozzles and lines are designed to assure that they will accommodate the differential temperature which occurs between the injection mode and the recirculation mode.

6.1.4 TESTS AND INSPECTIONS

All active components of the ECCS, listed in Table 6-5, are tested periodically to demonstrate system readiness. Performance of active systems is tested by establishing flow and observing pressures and flows during scheduled shutdowns. The HPI System is inspected periodically during normal operation for leaks from pump seals, valve packing, and flanged joints. During operational testing of the LPI pumps, the portion of the system subjected to pump pressure is inspected for leaks. Pump seals, valve packing, flange gaskets, heat exchangers, and safety valves are inspected for leaks to the atmosphere.

will not inhibit the overall ES functions. Where a motor-operated or a solenoid-operated valve is driven by either of two matrices, one is from actuation channel A and one from actuation channel B. Redundant ES pumps are controlled from separate and independent actuation channels.

The bistable units and output relays in the instrumentation channels are de-energized to trip, and similarly, the logic matrices in the actuation channels are de-energized to trip and actuate the ES equipment except for the "LPI pump start" and "EFW pump trip upon LPI start" matrices which are energized to trip for reliable operation of the EFW pump. Separate essential service and DC power supplies are used for each actuation channel.

Actions initiated by the ESAS are summarized in Table 7-3, Figure 7-4, sheets 1 and 2 and Figure 7-5.

7.1.3.2.3 Description of Protection Channels

a. HPI and Loading Sequence

Referring to one of three transmitters shown in Figure 7-4 Sheet 1, a signal proportional to the reactor coolant pressure is applied to a safeguards bistable and to a bypass bistable. The design of safeguard bistables is such that when the reactor coolant pressure is above the setpoint and control power is available, bistable interposing relay R3 is energized.

The bypass enabling contact of the bypass bistable, however, closes when the pressure is below its setpoint (1700 psig) and control power is available. This permits manual bypass for normal shutdown.

HPI is initiated by de-energizing the multiple contact output relays constituting loading sequence block 1, in 2-out-of-3 channels. The multiple contact output relays can be de-energized by the manual actuation relay, by their related test contact and by an 'OR' function made up of contacts which open when the reactor coolant pressure is below 1,500 psig (R3), the reactor building pressure exceeds 4 psig (R12) or the reactor coolant pressure is below 500 psig, respectively.

The ES load sequencer timers are normally energized and will de-energize to time out in 5 second intervals. As shown on Figure 7-4 Sheet 1, the initiation of the load sequencer is a similar 'OR' logic of contacts which open when RC pressure is below 1,500 psig or RB pressure exceeds 4 psig or the RC pressure is below 500 psig or a manual operator initiation occurs. This 'OR' initiation logic is in a 'AND' logic with the ES 4160 volt bus 2-out-of-3 undervoltage scheme to keep the timer relays of Blocks 2, 3, 4, 5, 6 and EFW energized until the EDG is started and loaded on the ES bus.

The 'OR' initiation logic is in a 'AND' logic with an automatic reset scheme for timers 4 and 5 in the event the loading sequence would be re-initiated by an ES subsequent signal. A manual reset of timers 4 and 5 is provided such that the HPI portion of ES can be taken to manual control if required without affecting automatic actuation of the DH and BS systems.

Start

The EFW Block timer is a conditional logic specifically for the EFW motor driven pump EFP-1 of the ES 'A' train to provide for anticipatory of the EFP-1 pump in 5 seconds in the event of a LOOP event without ES actuation and to provide sequence loading 5 seconds after Block 3 should a ES condition be present. This is

Gm

which

accomplished by the auxiliary relay contact of timer 2 is normally closed. The Block 1 relay contact is normally open and the auxiliary relay contact for LOOP is a normally de-energized closed contact.

Hence, for a LOOP event without HPI initiation, the EFW Block timer times out in 5 seconds. For an ES HPI initiation without a LOOP event, the EFW Block timer times out 5 seconds after Block 3 (timer 2) or 15 seconds after the EDG is loaded. For a LOOP event after a ES HPI condition the EFW timer and associated timer 2 are reset such that EFW will again time out 5 seconds after Block 3 (timer 2). For an ES HPI initiation after a LOOP event, the EFW timer will again time out in 5 seconds after Block 3 (timer). A logic circuit of relay contacts from Block 1, Block 3, LOOP auxiliary relay and 500 psig RC pressure trips the EFW pump EFP-1 when a LOOP event is followed by an ES HPI initiation or a 500 psig RC pressure as shown of Figure 7-5.

The interlocks with the undervoltage scheme is such that, in case of a blackout, block 2, 3, 4, 5, 6 and EFW are kept in their reset status as long as there is no voltage on the related engineered safeguards bus. As soon as the emergency diesel generator is connected on the bus, auxiliary contacts from the diesel generator circuit-breaker are used to defeat the action of the undervoltage relay scheme irrespective of the voltage on the bus thus preventing erratic loading or simultaneous energization of load blocks 2, 3, 4, 5, 6 and EFW. The emergency diesel generator is connected automatically on the bus only when correct voltage and frequency are reached.

De-energizing output relays of block 1, 2, 3, 4, 5, 6 and EFW in two of the three channels will start, in sequence, the following equipment.

Block 1	HPI pumps Injection and nuclear service valves
Block 2	Reactor building cooling fans to low speed Emergency nuclear service seawater pumps
Block 3	Emergency nuclear services closed cycle cooling water pumps Decay heat pump air handling units
Block 4 OR	LPI pump (for which minimum flow recirculation is provided) Start Permit
Block 5	Decay heat service seawater pumps
Block 6	Reactor building spray pump start-permit Decay heat close cycle cooling water pumps
EFW Block	Motor driven emergency feedwater pump

b. Low Pressure Injection

The channels of LPI are equipped with bistables similar to those used for HPI but which are adjusted to actuate at a lower setpoint. The output relays can be de-energized by the manual actuation relay, by their related test contact, and by an 'OR' function made up of contacts which are open when

If a LOOP event [illegible]
automatically BYPASSED by an auxiliary relay contact which is energized closed by 'AND' logic comprised of 4160 V feeder breaker contacts and the EDG breaker closed auxiliary relay contacts

the reactor coolant pressure is below 500 psig, or when the reactor building pressure exceeds 4 psig. A typical channel is shown on Figure 7-4, Sheet 1. The output of the bistables will de-energize the ~~same~~ output relay ~~to the HPI bistables~~ at 500 psig for actuation of the LPI valves and energize the output relay at 500 psig for LPI pump actuation and

The bypass enabling contact of the bistable closed when the reactor coolant pressure is below its setpoint (900 psig) and control power is available. This action permits manual bypass of the channel ~~for normal shutdown of the system.~~

if a LOOP event is present trip EFP-1 pump

ILLEGIBLE ORIGINAL

c. Reactor Building Isolation and Cooling

The channels of reactor building isolation and cooling are similar in design to the channels of HPI and loading sequence except for the bistable and the bypass circuit, as shown in Figure 7-4, Sheet 2. When the reactor building pressure is below 4 psig and control power is available, pressure switch interposing relay R10 is energized to the reset state by means of the bypass reset pushbutton. A subsequent loss of power or rise in building pressure above setpoint will drop out R10.

The continuous bypass of a channel is possible only after a 2-out-of-3 actuation. De-energizing the output relays of 2-out-of-3 channels initiates reactor building isolation, starts reactor building emergency cooling, and opens all valves required for reactor building spray. The reactor building pressure is sensed by two sets of three pressure switches, and the bypass can only be energized after a 2-out-of-3 actuation.

Also, reactor building isolation of systems not required for containment or RCS heat removal is initiated by HPI actuation. Bypass capabilities are the same as those for HPI, however, they do not affect the 4 psig building isolation logic.

d. Reactor Building Spray

Sequence start permit

Reactor building spray is initiated by starting the pumps at 30 psig in the reactor building coincident with HPI actuation. This is achieved, as shown in Figure 7-4, Sheet 2, by sensing the reactor building pressure with two sets of three pressure switches. Each set of three pressure switches, which are wired in a 2-out-of-3 matrix, controls the closing coils of the circuit breaker of one spray pump.

ES "A" trouble and ES "B" trouble alarms

ES "A" not reset and ES "B" not reset

ES "A" not bypassed and ES "B" not bypassed alarms

ES "A" reactor building isolation on and ES "B" reactor building isolation on

ES "A" diesel failed to start and ES "B" diesel failed to start

ES "A" reactor building spray on and ES "B" reactor building spray on

Channel RC-1 1,500 psi trip bistable tripped

Channel RC-2 1,500 psi trip bistable tripped

Channel RC-3 1,500 psi trip bistable tripped

Channel RC-4 500 psi trip bistable tripped

Channel RC-5 500 psi trip bistable tripped

Channel RC-6 500 psi trip bistable tripped

ES aux., test, and channel cabinets fan failure

Transmitter power supply cabinet "A" standby power

Transmitter power supply cabinet "B" standby power

Automatic closure interlock decay heat valve system trouble

Actuation "A" and Actuation "B" for loading sequence Block 1

Actuation "A" and Actuation "B" for loading sequence Block 2

Actuation "A" and Actuation "B" for loading sequence Block 3

Actuation "A" and Actuation "B" for loading sequence Block 4

Equipment and cabling associated with the ESAS are distinguished by color coding, marking, and equipment location. Each ESAS train is completely separate from the other as stated above.

Actuation "A" [redacted] for loading sequence Block EFW

7.1.3.3.2 Electrical Isolation

The use of buffer or isolation amplifiers prevents any faults (shorts, grounds, or cross connection of signals) on any analog signal leaving the system from being reflected into or propagating through the system. The direct connection of any analog signal to a source of electrical power can, at most, negate information from the measured variable involved. A de-energize to actuate design does not require an isolated power supply. To prevent false actuation due to a single power supply failure each transmitter and its associated bistables and channels are connected to separate vital buses.

7.1.3.3.3 Physical Isolation

Physical isolation is provided between channel transmitter- bistable combinations and pressure switches monitoring the same variable. Each of the twelve channels comprising actuation system "A" or "B" is isolated from the others up to the point where the 2-out-of-3 matrices are formed.

Complete separation also exists between the 2-out-of-3 matrices of the actuation systems "A" and "B" used to drive the redundant ES equipment.

7.1.3.3.4 Periodic Testing and Reliability

The safeguards actuation system is designed to be tested any time during plant operation or shutdown and fully complies with the requirements of IEEE 279-1968.

The pressure sensors are periodically checked by comparing their output signals with the output signal of similar sensors monitoring the same parameter.

The bistables may be tested one at a time by substitution of signals at the isolation amplifier, allowing accurate adjustment of the set-point. The two channels originating from one bistable will be tripped each time their initiating bistable is tested, assuring that protective action cannot be defeated.

A test switch associated with each engineered safeguards bistable which will permit demonstrating the bistable's capability to trip its associated channels.

The ESAS includes all the devices from sensor through the 2 out of 3 logic matrix. The ESAS contains the Automatic Actuation Logic which is the logic performed by the 2 out of 3 matrix. Additionally, the Reactor Building Spray Automatic Actuation Logic includes all the logic provided by the 30 psi RB pressure switches coincident with an HPI ~~signal.~~ The end device includes the final component that is being actuated, i.e., valves, circuit breakers, pumps, etc.

Sequence Start permit

Table 7-3

Engineered Safeguards Actuation Conditions

Function	Plant Variable	Trip Setpoint, psig	Bypass
High pressure injection and loading sequence and EFP-1 start	Low reactor coolant pressure -or	≥1,500	Manual bypass permitted below 1,700 psig. Bypass removed above 1,700 psig.
	Low-low reactor coolant pressure - or	≥500	Manual bypass permitted below 900 psig. Bypass removed above 900 psig.
	High reactor building pressure	≤4	Manual bypass permitted after actuation of any 2 out of 3 channels and during post-accident to permit reconstituting system.
Low pressure injection	Low-low reactor coolant pressure - or	≥500	Manual bypass permitted below 900 psig. Bypass removed above 900 psig.
	High reactor building pressure ##	≤4	Manual bypass permitted after actuation of any 2 out of 3 channels and during post-accident to permit reconstituting system.
Reactor building isolation and reactor building cooling	High reactor building pressure -or HPI actuation*	≤4 -	Manual bypass permitted after actuation of any 2 out of 3 channels and during post-accident. Same as HPI above.
Reactor building spray	High-high reactor building pressure and high pressure injection	<30	None.
EFP-1: Trip -	Low-low reactor coolant pressure ###	≥500	repeat

*Isolation only of those systems not required for Reactor Building or Reactor Coolant System heat removal.

4 psig RB Pressure bypassed on Loss of Offsite Power (LOOP) conditions

LOOP event only

CR3 005848

back to the Dedicated Emergency Feedwater (EFT-2) Tank. The motor-driven emergency feedwater pump (EFP-1) receives power from the 4160 volt Engineered Safeguards (ES) Bus 3A. The motor is tripped on an AC power failure at the bus. In the event of this condition, ~~coincident with loss of both main feedwater pumps or low level in either OTSG,~~ the motor-driven emergency feedwater pump will automatically start five seconds after ES ~~Block 4~~ bus is loaded onto the "A" train diesel generator. The turbine-driven emergency feedwater pump is independent of AC power and starts by opening the 125V DC motor operated steam admission valves (ASV-5 and ASV-204) when activated by the EFIC System.

10.5.2.1 Motor-Driven Emergency Feedwater Pump

The Motor-Driven Emergency Feedwater Pump (EFP-1) is powered from emergency diesel generator 3A when normal AC power is not available. It has a rated capacity of 740 gpm at 1300 psig with a design recirculation flowrate of 200 gpm with the pump discharge closed. Motor, gear cooling, and lube oil cooling are supplied from the Nuclear Services Closed Cycle Cooling System (SW). Manual controls (both locally and on the main control board) provide the operator with the capability of shutting down the pump if the extra feedwater capacity is not required.

10.5.2.2 Turbine-Driven Emergency Feedwater Pump

The Turbine-Driven Emergency Feedwater Pump (EFP-2) provides redundant 100% capacity to the EF System and automatically starts on the same initiating signals as the motor-driven emergency feedwater pump (EFP-1). It has a rated capacity of 740 gpm at 1300 psig with a design recirculation flowrate of 200 gpm with the pump discharge closed. Manual controls (both locally and on the main control board) provide the operator with the capability of shutting down the pump is the extra feedwater capacity is not required. Lube oil cooling is supplied by the pump discharge water.

10.5.2.3 Emergency Feedwater Pump Turbine Drive

The Turbine Drive (EFTB-1) provides rapid start-up for EFP-2. The steam supply for EFTB-1 is obtained from both OTSGs through six-inch lines containing normally-open DC motor operated stop-check valves (MSV-55 and MSV-56) and check valves (MSV-186 and MSV-187). The check valves and motor operated valves provide redundant isolation capability to preclude blowing down the operable steam generator in the event of main steam line or main feedwater line break. Downstream of check valves (MSV-186 and MSV-187) the lines join to form a common supply to the pump turbine. Upstream of the turbine are redundant, normally closed DC motor operated valves (ASV-5 and ASV-204) which are open upon actuation from the EFIC System. A description of the controls for these valves is contained in Section 7.2.4.

Turbine exhaust is vented to the atmosphere.

SENDER AND KEEP COPY. 000-213

INTER-OFFICE COMMUNICATION

TO: JACK TLASTIL DATE 12-16-88

SUBJECT EDG LOADING CALCULATIONS

EDG LOADING WILL CHANGE AS A RESULT OF THE FOLLOWING MARS THAT ARE TO BE INSTALLED DURING REFUEL VII. A RUNNING LOG OF EDG LOADING IS CURRENTLY BEING KEPT. FORMAL REVISION TO THE CALCULATIONS WILL BE MADE AFTER REFUEL VII AND FSAR UPDATES REFLECTING THESE CHANGES WILL BE PROVIDED AT THAT TIME.

MAR 88-05-04-01
88-08-01-01
88-05-24-01

c:c RE CLAYSON
G.W. CASTLEBERRY

SIGNED [signature]

RETURN TO: DATE

SIGNED

CR3 005850

C. **References**

(1) MAR 87-10-19-01A, MAR 87-07-08-01, MAR 87-10-06-01.
(2) G/CI Letter FCS #10419 "System Flows for EDG Loading Evaluation".
(3) G/CI telecon memo, C.C. Strempke to Don Guzzo dated 4-3-89 "SER points".
(4) G/CI Letter FCS-10125, dated 1-25-89, scope letter.
(5) G/CI Letter FCS-9894, dated 11-4-88, conceptual design.
(6) G/CI telecon memo. C.C. Strempke to Ron Clauson dated 5-16-89. "Annunciator windows."
(7) G/CI speed memo, ES panels & cabinet seismic evaluation. 5/22/89.
(8) G/CI Letter FCS-10424, dated 6-16-89, Purchase Requisitions.
(9) Calculation DC-5515-004-1.02 CS, and DC-5515-004-2.__ __
(10) G/CI Letter FCS-10420, dated 4-7-89, and FCS-10805, dated 9-11-89. FSAR Transmittal.
(11) G/CI Letter FCS-10784, dated 8-24-89, revised scope letter.
(12) G/CI Letter FCS-10814, dated 9-12-89, "EDG loads".
(13) IOC, M.W. Averett to R.E. Clauson, dated 7-11-88, "Reliability Analysis for the Proposed EFP-1/LPI Logic Modification".
(14) B&W Report 51-117603-00 - "Acceptability of a 500 psig start of LPI for all Loss of Coolant Accidents".
(15) G/C, Inc. Drawings: 201-129, Main Control Board Equipment Cutout Details
201-130, Main Control Board Equipment Cutout Details
(16) Design Basis Document Temporary Change __. 67, 68, 69; Emergency Feedwater (Sect 6/.3), Decay Heat Removal System (Sect 6/2), Emer Safeguards Actuat Systs. (Sect 6/8). __/89

D. **Material**

See attached Bill of Materia.

E. **Installation Instructions**

1. Charge all time and material to NU88052401.

2. All cable handling, pulling, termination, conduit installation, equipment removal and installation shall be in accordance with applicable FPC maintenance procedures. All relays removed shall be returned to Nuclear Materials Coordinator.

3. De-energize ES system Loop A.

1a. This MAR shall not be made operational until NRC approval for the change has been received. A new Safety Evaluation will be added to the MAR via FCN after NRC approval

__ 11/8/89

EXHIBITS TO THE TESTIMONY OF
PAUL F. MCKEE

EXHIBIT NO. ___ (PFM-4)

DUKE POWER CORRESPONDENCE

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12 1989

DUKE POWER COMPANY
P.O. BOX 33189
CHARLOTTE, N.C. 28242

HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

December 7, 1989

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Subject: Oconee Nuclear Station
Doc. Nos. 50-269, -270, -287
Installation of Cavitating Venturi

On October 17, 1989, Duke Power personnel met with the NRC Staff to discuss pump runout protection for the Oconee Emergency Feedwater (EFW) pumps. This meeting was requested by me in a letter dated June 20, 1989. The purpose of the meeting was to discuss, in depth, the reasons for revising our previous commitment to implement a hardware fix to reduce operator burden during postulated accidents with low steam generator pressure. As discussed during the October 17, 1989, meeting we have been evaluating various options to address this concern, and to date have not been able to identify an optimum solution. Accordingly, we do not intend to implement any hardware fix to address this concern. As such, I am withdrawing my commitment to install cavitating venturies or any other modification in order to relieve operator burden in ensuring that the EFW pumps are protected from runout in postulated accidents with low steam generator pressure. The basis for this change was discussed in depth during the October 17, 1989 meeting. We will continue to assure operator awareness of this potential problem, through training and procedures.

In addition during the October 17, 1989 meeting, the vibration problems with installing cavitating venturies into Oconee EFW system was discussed. The NRC requested that Duke provide additional information regarding this problem. To this end, please find attached the requested information.

Very truly yours,

J W Hampton
for

H. B. Tucker

PFG/78/td

Attachment

OCONEE UNITS 1, 2, and 3
Cavitating Venturi
Vibration Problems

To solve the potential problem with EFW pump runout and enhance the operation of the EFW system, cavitating venturis were selected as a possible passive solution that might effectively reduce operator burden without complicating the system.

Following the design and manufacture of the venturis, the venturis were installed in a temporary test loop and operated to verify functional performance and measure vibration created in downstream piping. The piping response under postulated worst case cavitating conditions resulted in vibration velocities in excess of 10 ips-pk (pegging the meter) and accelerations in excess of 60 g's.

While it is generally accepted that the piping vibration is highly dependent upon the piping geometry and support configurations, it is also known from failure experiences that cantilevered vents and drains, as well as valve actuators, can have vibration responses several times higher than the run piping for cavitation type input. The greater response of these components to the high level of "white noise" is attributed to the poor overall damping available in combination with excitation of natural frequencies from the broad frequency range forcing function.

Inasmuch as (1) the Duke piping in question near the proposed location of the cavitating venturi does contain cantilevered attachments and (2) we expect that, based on our experience, this piping would be highly susceptible to short term vibration fatigue, we cannot recommend installation of the cavitating venturi as a solution to our problem with EFW pump runout.

This opposition is further reinforced by our concerns that if we were to install the venturi, a lengthy data acquisition testing effort would be required to adequately evaluate the systems functional integrity under the conditions generated by the cavitating venturi. Following data collection, the unit would be considered inoperable for days or perhaps weeks while analytical efforts were made to qualify the vibration and calculate the allowable remaining fatigue stress cycles not consumed by the testing. Efforts to limit stress cycles and allow a greater alternating stress intensity would quickly reduce the allowable run time. This could force tracking of the stress cycles during actual unit operation resulting in extensive permanent instrumentation, monitoring equipment, and evaluation expense.

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Also, the demands of the Duke system configuration would have required placement of a venturi immediately upstream of a secured cabinet containing several valves and branch connections. Because of the high vibration susceptibility of these components, extensive data collection and additional compensatory security measures in these areas would have been required during the attempted qualification testing.

Furthermore, discussions with another utility having cavitating venturis installed in a simple piping configuration reinforced the belief that qualification would likely be much more difficult, if at all possible, for Duke's complicated piping geometry.

In summary, many factors pointed to the decision not to attempt installation and qualification of the cavitating venturis. Among them were previous experiences with high vibration failures and comments from other utilities concerning qualification difficulties for much simpler piping systems, as well as, vibration results from the test loop.

EXHIBITS TO THE TESTIMONY OF
PAUL F. MCKEE

EXHIBIT NO. ___ (PFM-5)

COLTEC CORRESPONDENCE

Coltec Industries

Fairbanks Morse
Engine Division
701 White Avenue
Beloit, WI 53511-5492
608/364-4411
FAX: 608/364-0382

April 17, 1996

Florida Power - CR3
15760 W. Powerline Street
Crystal River, FL 34428

Attention: M.U. Rahman

Subject: Crystal River Unit 3
Emergency Diesel Generators
Load Excursion

Reference (a) Florida Power NED96-0227, dated 4/10/96

Gentlemen,

In response to reference (a), please be advised that a one time excursion of 3500/3700 KW for up to 2 seconds during motor starting is not expected to have an adverse affect on the genset

It should be noted however, that multiple excursions will have a cumulative effect that can increase potential for component distress

Known affects of engine overload include.

- piston ring breakage
- piston insert cracking
- liner/jacket cracking
- piston bushing extrusion

Coltec recommends that should an event(s) of this nature occur, engine operating parameters should closely monitored to determine performance changes and inspections should be scheduled to determine any component distress.

ATTACHMENT

If you have any questions or require additional information, please contact the writer at (608) 364-8424.

Very truly yours,

T.M. Stevenson
Manager Parts Sales, Nuclear and Export

TMS:jaw

cc J. Eves
T. Miller
T. Stull
G. Kasel
G. Gutoski

ATTACHMENT

Coltec Industries

Fairbanks Morse
Engine Division
701 White Avenue
Beloit, WI 53511-5492
608/364-4411
FAX 608/364-0382

April 25, 1996

Florida Power - CR3
15760 W. Powerline Street
Crystal River, FL 34428

Attention M.U Rahman

Subject: Crystal River Unit 3
Emergency Diesel Generators
Load Excursion

Reference (a) Florida Power NED96-0264, dated 4/24/96
(b) Coltec Letter dated 4/17/96

Gentlemen:

In response to Reference (a), please be advised that a one-time event identified per Reference (a) is not expected to have an adverse effect on the genset

Please refer to Reference (b) for multiple event effects on the genset

If you have any questions or request additional information, please contact me at 608-364-8424

Very truly yours,

T. M. Stevenson
Manager, Nuclear Parts Sales

☐ ILLEGIBLE
☐ POOR QUALITY, BUT READABLE
☒ POOR XEROX COPY

lm

cc: R. Blessin
T. Miller

ATTACHMENT

EXHIBITS TO THE TESTIMONY OF
PAUL F. MCKEE

EXHIBIT NO. ___ (PFM-6)

MAR DOCUMENTATION FOR 1996 MODIFICATION

Florida Power CORPORATION

MAR REVIEW CHECKLIST

MAR 96-04-12-01

Number of FCNs ∅

✓ MAR Page No.(s) checked and MAR number correct on all documents

✓ Safety Related / ISI Review / EQ Review / ALARA Review blocks checked yes or no

✓ Required Signatures on MAR / FCN Approval Sheet: If req'd make sure signature / date written

✓ PRC Committee Chairman signature (if checked yes)

N/A NGRC signature (if marked yes)

✓ Installation Authorized by Nuclear Plant Manager signature

N/A Temporary Modification Removed signature (N/A for permanent MAR)

✓ MAR Test Supervisor signature

✓ Installation Complete signature Nuclear Project Specialist

✓ Safety Evaluation (10CFR50.59) (may be N/A if a minor MAR)

N/A NRC authorization received (if req. on the safety evaluation)

✓ All listed attachments (Section B) included and completed (except drawings)

✓ EQ Review Form (if required) (may be included in main MAR)

✓ MAR Transmittal Memo

___ As-Built Completion Notice

✓ ISI Review Form (if required)

✓ Post Mod. Functional Test Form

N/A Safety Evaluation Review Form (if required)

✓ Fire Protection Review Form

✓ Modification Procedure Review Forms (yellow/green)

N/A T-MAR Extension Forms (N/A for permanent MARs)

✓ Operations Acceptance Form

✓ Walkdown and System Acceptance

✓ Completion of Modification Testing Notice

___ Procedure Review Records

✓ Completion of Training Notice

✓ Fast Turnaround Drawing Update Notice (if required)

✓ MAR Drawings / As-Built Notice

✓ Data Transmittal Sheets

✓ Completion of EQ Qualification Notice (if required)

✓ Work packages reviewed for completeness (all required procedures / signoffs complete)

✓ All test procedures included

___ All documents signed and dated

E. J. Brancik
MAR Reviewer

6/18/96
Date

___
Nuclear Project Specialist

___
Date

CR3 005889

ASTER

ENCLOSURE 2

FLORIDA POWER CORPORATION
QUALITY ASSURANCE RECORD
TRANSMITTAL

ATTENTION: RECORDS MANAGEMENT SECTION
CRYSTAL RIVER UNIT NO. 3

DOCUMENTS TRANSMITTED: (1) MAR 96-04-12-01 (2) T/O package (3) IOC NED 96-0272 (4) TP's 2 @ original & working copy (5) WR# 335012

Need:

(1) As Blt Comp Notice
(2) PRR's for SP-349A ~~SP-[illegible]A~~

The Quali[illegible] r
inclusion[illegible]

These rec[illegible] [illegible] Florida
Power Corporation's Quality Program.

E. Bosworth
Responsible Supervisor/Designee DATE ________

RECEIPT ACKNOWLEDGEMENT BY:

________________________ DATE ________
Manager, Nuclear Information Resources/Designee

FUTURE RETENTION OF THESE RECORDS IS THE RESPONSIBILITY OF RECORDS MANAGEMENT.

CR3 005890

Florida Power

PLANT DESIGN BASIS DOCUMENT TEMPORARY CHANGE
Nuclear Engineering

TEMPORARY CHANGE NUMBER: 511

PREPARED BY	DATE	REVIEWED BY	DATE
C. B. Doyel	5/30/96		
AFFECTED SYSTEM(S)/ACCIDENT(S): EF		AFFECTED SECTION(S): TAB NO.(S), PAGE NO.(S): 6/13, pp36 and 63	

I. To be completed by Design Engineer

Temporary Change to: (Check one)

EDBD TC No. 511 Sheet 1 of 4

- ☐ Design Basis Document (DBD)
- ☑ Enhanced Design Basis Document (EDBD)
- ☐ Analysis Basis Document (ABD)
  NOTE: Concurrence by Nuclear Fuel Management and Safety Analysis required if ABD is changed.

CONCURRENCE, MGR., NUCLEAR FUEL MGT. & SAFETY ANALYSIS	DATE

- ☐ Topical Design Basis Document (TDBD)

Effective date: (Check one)

- ☑ This Temporary Change is effective upon approval.
- ☐ This Temporary Change is to be included in the appropriate document for information only until implementation of MAR/PEERE ______.

PRC Review required ☑ Yes ☐ No (If No, NA PRC Review)
If Yes: PRC Review/Date ______.

II. To be completed by CMI, except as noted in instructions.

YES	N/A		
☑	☐	1.	Analysis / Calculation Form with Technical Justification included.
☑	☐	2.	Marked Up Pages of the appropriate document included.
☐	☑	3.	10CFR50.59 Safety Evaluation performed (Required if Safety Evaluation covering the change is not located in another document. If located in another document, list source document MAR 96-04-12-01)
☐	☑	4.	Nuclear Engineering Safety Impact Assessment performed. (Required if assessment covering the change is not located in another document. If located in another document, list source document ______.)
☐	☑	5.	Revised Table of Contents included.
☐	☐	6.	Other ______.

III. CMI Review Comments / Resolution:

- ☐ No comment
- ☐ Comment / Resolution

DXREF: Design
Upon completion, distribute as follows:

Records Management (Original)
Supervisor, NPTS - Maintenance / Component Engineering ☐ Yes ☑ No
MAR Office ☑ Yes ☐ No (If "Yes," send copy of page 1 only)

CR3 005891

Florida Power Corporation

ANALYSIS/CALCULATION

Crystal River Unit 3

DOCUMENT NUMBER
EDBD TC 511

PROJECT
MAR 96-04-12-01

Statement of Changes:

1.) Remove the first part of the sentence, "EC train A EF initiate is required to open ASV-204..." under COMPONENT/PARAMETER on p36 and add to the text a second sentence as follows, "ASV-204 is a remote manual operated valve providing a redundant steam admission valve to ASV-5."

2.) Reword the Reason/Source to reflect that only ASV-5 receives an open demand signal from EFIC.

Technical Justification:

This MAR is being driven by the performance requirements of 2 major components in the plant, the Steam Driven Emergency Feedwater Pump, EFP-2, and the "A" Emergency Diesel Generator, EGDG-1A. In 1987, while struggling with the question of how to meet the loading requirements for EGDG-1A after loading EFP-1 on it for a LOCA, concurrent with a Loss of Offsite Power (LOOP), and a "B" Battery failure, the solution was proposed, accepted, and installed to power ASV-204 from the "A" electrical train and initiate an automatic opening of the valve in the event of an EFIC actuation. By doing this, the load expected to be carried by EFP-1 would be minimized as EFP-2 would be running without any controls, i.e. at full speed. This was the only way that the projected load on EGDG-1A could remain within the rated capacity of the machine at that time. Since then, the upgrade of the diesel generators has been completed, and new electrical calculations show that EGDG-1A can handle the load of EFP-1 in this accident scenario. In addition, a concern about the NPSH requirements for EFP-2 not being met when the pump is running in this all out condition dictated that this automatic opening of ASV-204 be removed.

EDBD TC No. 511 Sheet 2 of 4

CR3 005892

DESIGN ENGINEER	DATE	VERIFICATION ENGINEER	DATE	SUPERVISOR, NUCLEAR ENGINEERING	DATE
B. Doyel	5/30/96				

3.0 COMPONENT PARAMETERS

EDBD TC No. 511 Sheet 3 of 4

COMPONENT/PARAMETER	REASONS/SOURCE	INTERFACE
FSAR	FSAR Sections 14.2.2.1, 7.2.4, 10.2.1.2	
~~EC Train A EF initiate is required to open ASV-204 and~~ EC Train B is required to open ASV-5, MSV-55, and MSV-56. ASV-204 IS A ~~MANUALLY~~ REMOTE OPERATED VALVE PROVIDING A REDUNDANT STEAM ADMISSION VALVE TO ASV-5. CSD 5/20/96	Opening ASV-5, ~~204~~ provides ~~redundant~~ steam admission for the start of EFP-2. MSV-55, 56 are normally open motorized stop check valves; however, they receive a signal to open to ensure steam flow in case of inadvertent closure for any reason. Specific component/ parameters are addressed in the AS and MS EDBD's. References 23, 32, 117	AS System MS System
FSAR	FSAR Sections 10.5.2.3, 10.2.1.4	
EC is required to control the ADV's MSV-25, and MSV-26 to a predetermined setpoint and provide manual capability thru a hand-auto station.	Allows steam generator pressure control independent of the ICS without unnecessarily challenging the MSSV's. Manual capability allows for pressure control during cooldown using EF. Reference 23	MS System
Flow orifices EF-023-FE, EF-024-FE, EF-025-FE and EF-026-FE are required to be designed with a beta ratio of 0.6197 for a design flow of 1000 gpm.	NUREG-0737 requires EF flow measurement. The orifices were originally installed with a design flow of 820 gpm. Reference 47 increases the scale to 1,000 gpm because the flow instrumentation had pegged high on several previous EF actuations. A B=0.6197 gives an adequate differential pressure for flow measurement. Reference 47, 77	

EDBD TC No. 511 Sheet 4 of 4

4.0 REFERENCES

105. Interoffice Correspondence, WPN84-0813, "Emergency Feedwater Valves 1, 2, 3, 4, 7 and 8 and Interface with EFIC RS Appendix R," September 28, 1984.

106. MAR 88-03-05-01 "System Tuning of EFW Control Valve Signal," dated July 26, 1989.

107. FPC Interoffice Correspondence NDB90-072, W. S. O'Brien to K. B. Baker, "Open Item Report 90-CM-EF-07," dated July 3, 1990.

108. FPC Calculation I85-0009, Rev. 3 "Dedicated EFW Tank - Alarm Setpoints," dated September 9, 1992.

109. MAR 93-06-16-01, "NI-14 and NI-15 Modification for NI-1 and NI-2 Backup and AMSAC Setpoint Change."

110. MAR 88-07-05-04, "FWP-7 Mechanical Tie-Ins."

111. MAR 88-07-05-01, "Add Aux Feedpump, FWP-7."

117. MAR 96-04-12-01, "ASV-204 EFIC AUTO OPEN REMOVAL."

Florida Power

PLANT DESIGN BASIS DOCUMENT TEMPORARY CHANGE
Nuclear Engineering

TEMPORARY CHANGE NUMBER: 510

PREPARED BY	DATE	REVIEWED BY	DATE
C. B. Doyel *C. B. Doyel*	5/30/96		
AFFECTED SYSTEMS/ACCIDENT(S): MS		AFFECTED SECTION(S): TAB NO.(S), PAGE NO.(S): 6/10, pp33 and 38	

I. To be completed by Design Engineer

Temporary Change to: (Check one)

EDBD TC No. 510 Sheet 1 of 4

- ☐ Design Basis Document (DBD)
- ☑ Enhanced Design Basis Document (EDBD)
- ☐ Analysis Basis Document (ABD)
  NOTE: Concurrence by Nuclear Fuel Management and Safety Analysis required if ABD is changed.

CONCURRENCE, MGR., NUCLEAR FUEL MGT. & SAFETY ANALYSIS	DATE

- ☐ Topical Design Basis Document (TDBD)

Effective date: (Check one)

- ☑ This Temporary Change is effective upon approval.
- ☐ This Temporary Change is to be included in the appropriate document for information only until implementation of MAR/PEERE ________.

PRC Review required ☑ Yes ☐ No (If No, NA PRC Review)
If Yes: PRC Review/Date ________.

II. To be completed by CMI, except as noted in instructions.

YES	N/A		
☑	☐	1.	Analysis / Calculation Form with Technical Justification included.
☑	☐	2.	Marked Up Pages of the appropriate document included.
☐	☑	3.	10CFR50.59 Safety Evaluation performed (Required if Safety Evaluation covering the change is not located in another document. If located in another document, list source document MAR96-04-12-01)
☐	☑	4.	Nuclear Engineering Safety Impact Assessment performed. (Required if assessment covering the change is not located in another document. If located in another document, list source document ________.)
☐	☑	5.	Revised Table of Contents included.
☐	☐	6.	Other ________.

III. CMI Review Comments / Resolution:

- ☐ No comment
- ☐ Comment / Resolution

DXREF: Design
Upon completion, distribute as follows:

Records Management (Original)
Supervisor, NPTS - Maintenance / Component Engineering ☐ Yes ☑ No
MAR Office ☑ Yes ☐ No (If "Yes," send copy of page 1 only)

CR3 005895

Florida Power CORPORATION

ANALYSIS/CALCULATION

Crystal River Unit 3

DOCUMENT NUMBER
EDBD TC 510

PROJECT
MAR 96-04-12-01

Statement of Changes:

1.) Remove the second sentence in the NOTE on p33 and replace it with the following, "ASV-204 is a remote manual operated valve providing a redundant steam admission valve to ASV-5."

2.) Reword the Reason/Source to reflect that only ASV-5 receives an open demand signal from EFIC.

Technical Justification:

This MAR is being driven by the performance requirements of 2 major components in the plant, the Steam Driven Emergency Feedwater Pump, EFP-2, and the "A" Emergency Diesel Generator, EGDG-1A. In 1987, while struggling with the question of how to meet the loading requirements for EGDG-1A after loading EFP-1 on it for a LOCA, concurrent with a Loss of Offsite Power (LOOP), and a "B" Battery failure, the solution was proposed, accepted, and installed to power ASV-204 from the "A" electrical train and initiate an automatic opening of the valve in the event of an EFIC actuation. By doing this, the load expected to be carried by EFP-1 would be minimized as EFP-2 would be running without any controls, i.e. at full speed. This was the only way that the projected load on EGDG-1A could remain within the rated capacity of the machine at that time. Since then, the upgrade of the diesel generators has been completed, and new electrical calculations show that EGDG-1A can handle the load of EFP-1 in this accident scenario. In addition, a concern about the NPSH requirements for EFP-2 not being met when the pump is running in this all out condition dictated that this automatic opening of ASV-204 be removed.

EDBD TC No. 510 Sheet 2 of 4

DESIGN ENGINEER	DATE	VERIFICATION ENGINEER	DATE	SUPERVISOR, NUCLEAR ENGINEERING	DATE
[signature]	5/30/96				

3.0 COMPONENT PARAMETERS

EDBD TC No. 510 Sheet 3 of 4

COMPONENT/PARAMETERS	REASON/SOURCE	INTERFACE
EF Pump Turbine Steam Inlet Valves Tag Numbers: ASV-5, ASV-204		
The valve opening stroke time is required to support the EF system achieving minimum flow within 50 seconds. NOTE: ASV-204 was installed to improve EFW reliability and is not required for operability. ~~During an ES actuation coincident with a LOOP and failure of the "B" train EDG, it is relied upon to provide margin in the "A" train EDG loading.~~ ASV-204 IS A REMOTE MANUAL OPERATED VALVE PROVIDING A REDUNDANT STEAM ADMISSION VALVE TO ASV-5	EFIC sends an ASV-5 open demand signal to the valve~~s~~ for Emergency Feedwater initiation. It ~~They~~ must open fast enough to ensure that the EF system can successfully achieve its safety function of delivering flow to the OTSGs within 50 seconds of actuation. Reference: 30, 35, 36, 52	The EFIC (EC) system sends an "OPEN" control signal for Emergency Feedwater initiation. Electrical Power: 250 VDC motive power from: ASV-5 DPDP-8B ASV-204 DPDP-8A
FSAR	FSAR Section 10.2.1.4., 14.1.2.8, 14.2.2.9	The HVAC (AH) system maintains compartment temperature, pressure, and humidity within EQ limits. Key Parameters: Heat Removal (Btu/hr)

EDBD TC No. 510 Sheet 4 of 4

4.0 REFERENCES

46. FPC IOC, Nuclear Licensing C2M, Closure Memo, Crystal River Unit 3, SPIP Recommendations, 11/2/89 for SPIP Recommendation No. TR-174-MSS, "Improve the Response of the Modulating Turbine Bypass Valves (Atmospheric and Condenser Pump Valves)."

47. B&W Letter, dated March 2, 1992, FPC 92-096, Task 847, "ADV Stroke Time Requirements."

48. MAR 91-08-04-01, "MSIV Downstream Pressure Indication."

49. FPC IOC NEA91-1080, dated 8/5/91.

50. BWNS Report 86-1219188-00, "MSSV Setpoint Tolerance Evaluation."

52. MAR 96-04-12-01, ASV-204 EFIC AUTO OPEN REMOVAL

:110F04
:110M04

FLORIDA POWER CORPORATION
CMIS - CLASSIFICATION DATA

DATE: 04/26/96
TIME: 08:26:02

#: ASV-204 TURB DRIVEN EMERG FWP INLET ISO
PONENT: OPERATOR, MOTOR SYSTEM: AS DESIGN REF: IN-PLANT
LOC: INTERMEDIATE BLDG ELEV: 95 COL/ROW: 308/H DISC: E
LOC: NPRDS REPORTABLE
PANEL BOX: ASV-204

ETY: S CHANNEL: A SEISMIC:
CODE: CODE KEY: L BB EX MR QQ NOTES:

=HELP F2=SEL CRIT F3=MULTI LIST F7/8=TAG PAGE F9=FIMIS
=PREV SCRN F11=MAIN MENU F12=LOGOFF F14=EQ F16=COMP F17=XREF
=MARS F19/20=ASSC TAG MNR/MJR F21=COMMENTS F22=MACS WR LIST
IGURATION ITEM DISPLAYED / THERE ARE COMMENTS TO DISPLAY

MAR 96-04-12-01
MAR DESCRIPTION
ATTACHMENT
p 1 of 2

E110F04 FLORIDA POWER CORPORATION DATE: 04/26/96
E110M04 CMIS - CLASSIFICATION DATA TIME: 11:30:02

; #: AS-9 FOR EFW PUMP ISO VALVE ASV-204
IPONENT: CONTROL STATION SYSTEM: AS DESIGN REF: IN-PLANT
LOC: INTERMEDIATE BLDG ELEV: 95 COL/ROW: DISC: E
LOC: ,
PANEL BOX:

'ETY: S CHANNEL: A SEISMIC:
: CODE: CODE KEY: L BB EX MR QQ NOTES:

1=HELP F2=SEL CRIT F3=MULTI LIST F7/8=TAG PAGE F9=FIMIS
0=PREV SCRN F11=MAIN MENU F12=LOGOFF F14=EQ F16=COMP F17=XREF
8=MARS F19/20=ASSC TAG MNR/MJR F21=COMMENTS F22=MACS WR LIST
FIGURATION ITEM DISPLAYED / ** NO COMMENTS TO DISPLAY **

MAR 96-04-12-01
MAR DESCRIPTION
ATTACHMENT
p 2 of 2

Florida Power
CORPORATION

FIRE PROTECTION REVIEW

5-8-96

To: Senior Nuclear Fire Protection Specialist

A copy of MAR / CGWR 96-04-12-01 FCN ______

is attached for your review. Review this document to verify compliance with National Fire Protection Association (NFPA) codes and standards.

☐ Does comply with NFPA codes and standards

☐ Does not comply with NFPA codes and standards
(Return immediately to the Manager, Nuclear Projects for resolution)

☒ NFPA codes and standards do not apply

Comments: ______

S. L. Russell
Senior Nuclear Fire Protection Specialist
(Return to the MAR office)

6-10-96
Date

FIRE PROTECTION REVIEW

MAR / CGWR and FCN Sign-off Sheet

MAR / CGWR 96-04-12-01 FCN ______

	Yes	No
1. Ensure MAR/CGWR/FCN has date-in stamp and is logged in SNS MAR/CGWR/FCN Logbook.	X	
2. Are major *in-situ* combustibles being installed? This may cause a change to the FHA or require new fire protection features.		X
3. Are large quantities of transient combustibles required by the MAR/CGWR/FCN? This may require compensatory measures to be put in place or special attention by the FNP Staff during the installation of the MAR/CGWR/FCN.		X
4. Will the MAR/CGWR/FCN cause breaching of fire barriers, fire doors, fire dampers, penetration seals, or TSI material? This will require a Fire Barrier Penetration Breach Permit and compensatory measures per CP-137 for Tech. Spec. items, and will cause compensatory measures to be invoked per the Fire Protection Plan for non-Tech. Spec. areas or items.		X
5. Will the MAR/CGWR/FCN need to incorporate "hot work" processes (welding, grinding, heat shrinking, etc.) to accomplish the MAR/CGWR/FCN? This may require Hot Work Permits in accordance with CP-118 to assure adequate fire protection features are in place.		X
6. Will the MAR/CGWR/FCN cause a fire protection system (sprinkler, fixed water spray, Halon, CO_2 standpipe and hose station, or fire detection system) to be disarmed or taken out of service? This will necessitate compensatory measures in accordance with CR-3 Tech. Spec. systems, and will necessitate compensatory measures per the Fire Protection Plan for non-Tech. Spec. systems.		X
7. Will the MAR/CGWR/FCN cause any impact on the Fire Protection Plan or fire protection procedures? Ensure that the MAR/CGWR/FCN will not inpact fire protection procedures or cause a change to existing procedures or a need for a change to the Preventative Maintenance Program.		X
8. Does the MAR/CGWR/FCN comply with the pertinent NFPA code? Ensure that the MAR/CGWR/FCN complies with the proper code, if required, and make sure the proper code is referenced and used in developing the MAR/CGWR/FCN. Fill out the sign-off sheet.	N/A	

CR3 005902

FIRE PROTECTION REVIEW

MAR / CGWR and FCN Sign-off Sheet
(Continued)

		Yes	No
9.	Do the Engineering Review Sheets, including the Fire Protection Review Sheets, indicate an impact on Appendix "R" fire protection issues?		X
	Ensure that the proper personnel are involved and following up on the Appendix "R" modifications, and that this information is going into the Appendix "R" fire study.		
10.	Does the MAR/CGWR/FCN present special fire protection concerns that would warrant a review by the NFP Group upon MAR/CGWR/FCN completion?		X
11.	Pertinent comments have been recorded in the Review Sheet's "Comments" section and the responsible Design Engineer has been notified.	N/A	
12.	MAR/CGWR/FCN returned to Nuclear Projects for processing.	X	
13.	Mark logbook to reflect MAR/CGWR/FCNs that the NFP Group should review upon completion of the MAR/CGWR/FCN.		X

Reviewed by J. L. Russell Date 6-10-96

Florida Power
CORPORATION

INSERVICE INSPECTION REQUIREMENTS

MAR 96-04-12-01 FCN —

Preservice Requirements

☐ Acceptable — As specified in Paragraph ______________________

☐ Unacceptable — Needs to be revised per below or attached comments.
See FCN No. ______________.

Inservice Requirements

☒ Acceptable — As specified in Paragraph B.7.C

☐ Unacceptable — Needs to be revised per below or attached comments.
See FCN No. ______________.

Comments:

____
Nuclear ISI Specialist

4/26/96
Date

CR3 005904

Florida Power CORPORATION

POST MODIFICATION FUNCTIONAL TEST REQUIREMENTS

MAR/CGWR 96-04-12-01 FCN ____________

☐ **No Functional Test Procedure Required**

- ☐ No Post Modification Test required
- ☐ POQAM Procedure to satisfy test requirements

  Procedure No. ____________ Sections ____________

  Section Desc. ____________

☒ **MAR Functional Testing Required (Select Option Below)**

- ☐ Work Request Test Instructions

  Enter Work Request Number ____________

- ☒ MAR Functional Test Procedure

  Briefly describe Test will simulate an output for EFIC Actuation A, to verify that AS-204 does not receive an input, and, all other functions associated with the actuation signal continue to operate satisfactorily.

Comments: ____________

[illegible] Supervisor, Nuclear Engineering

Functional Testing Discipline

4/27/96 Date

CR3 005905

Florida Power CORPORATION

MODIFICATION/PEERE/PROCEDURE REVIEW
Part A

AR / CGWR 96-04-12.01 FCN —
EERE

structions: The attached MAR/CGWR/FCN/PEERE is routed to you for your review to determine if procedures existing within your area of responsibility will need to be revised upon completion of this activity.

RETURN THIS FORM TO THE PROJECTS DEPARTMENT (NA1B) UPON COMPLETION OF YOUR REVIEW. DO NOT ROUTE TO THE NEXT PERSON ON THE LIST.

. ☑ Nuc. Ops.

Does the MAR/CGWR/FCN/PEERE affect equipment operability as defined in Tech. Specs. (i.e. does having the equipment out of service require entering an action statement) or is the equipment needed for a mode escalation?

Yes ☑ ☐ No

MAR/CGWR/PEERE should be returned to service during normal working hours. Administrative control of the equipment, ie., tag out or short term instruction, etc., may be needed until formal return to service is completed.

List operations requirements below i.e., list equipment needs or mode escalation restraint. (May require re-evaluation prior to working.) Administrative control of the equipment i.e., tag out of short term instruction etc. may be needed until return to service is completed.

MAR/CGWR/PEERE return to service requirements should be hand delivered to Operations to ensure timely return to service. The Project Manager, in cooperation with the Operation's Engineer, will inform the Control Room that MAR/CGWR/PEERE return to service requirements are to be accomplished prior to declaring equipment operable or mode changes.

__ No ✓ Yes Procedure changes are required

List # SP-349A SP-640A

✓ No __ Yes __ N/A Procedure changes are required prior to MAR/CGWR/PEERE return to service.

List #

Mode Escalation Restraint: ☐ 1 ☑ 2 ☐ 3 ☑ 4 Vogel 4/30/96 ☐ 5 ☐ 6 ☐ N/A

Operations Requirements / Comments
Use attachment if necessary

Vogel 3196 4/27/96
Print Name Ext. Date

☒ Nuclear Plant Systems Engineering ✓ No ___ Yes Procedure changes are required

List # ______ ______ ______ ______

___ No ___ Yes ✓ N/A Procedure changes are required prior to MAR/CGWR/PEERE Return to Service to Operations

List # ______ ______ ______ ______

Comments ____

M. FITZGERALD — Print Name
3444 — Ext.
4/27/96 — Date

☒ Nuclear ISI Specialist ✓ No ___ Yes Procedure changes are required

List # ______ ______ ______ ______

___ No ___ Yes ✓ N/A Procedure changes are required prior to MAR/CGWR/PEERE Return to Service to Operations

List # ______ ______ ______ ______

Comments ____

Phyllis A. Dixon — Print Name
3512 — Ext.
4/27/96 — Date

☒ Nuclear Maintenance ✓ No ___ Yes Procedure changes are required

List # ______ ______ ______ ______

✓ No ___ Yes ___ N/A Procedure changes are required prior to MAR/CGWR/PEERE Return to Service to Operations

List # ______ ______ ______ ______

Comments ____

d. C Hunsinger — Print Name
5663 — Ext.
4.29.96 — Date

CR3 005907

5. ☐ **Nuclear Chem / Rad Prot.**

___ No ___ Yes — Procedure changes are required

List # ______ ______ ______ ______

___ No ___ Yes ___ N/A — Procedure changes are required prior to MAR/CGWR/PEERE Return to Service to Operations

List # ______ ______ ______ ______

Comments ____

____

____

Print Name	Ext.	Date

6. ☐ **Nuclear Security**

___ No ___ Yes — Procedure changes are required

List # ______ ______ ______ ______

___ No ___ Yes ___ N/A — Procedure changes are required prior to MAR/CGWR/PEERE Return to Service to Operations

List # ______ ______ ______ ______

Comments ____

____

____

Print Name	Ext.	Date

7. ☒ **Nuclear Projects Specialist**

___ No ✓ Yes — Procedure changes are required and are identified

Print Name	Ext.	Date
E. Bosworth	3668	4-27-96

CR3 005908

Florida Power

MODIFICATION/PEERE/PROCEDURE REVIEW
Part B

AR / CGWR 96-0412-01 FCN ______

ERE ______

tructions: The attached MAR/CGWR/FCN/PEERE is routed to you for your review to determine if procedures existing within your area of responsibility will need to be revised upon completion of this activity.

RETURN THIS FORM TO THE PROJECTS DEPARTMENT (NA1B) UPON COMPLETION OF YOUR REVIEW. DO NOT ROUTE TO THE NEXT PERSON ON THE LIST.

☒ **Nuclear Engrg Support Services (Environmental Qualification)** ✓ No ~~X~~ Yes 4/27/96 Procedure changes are required

List # ______ ______ ______ ______

ease indicate any EQ (10CFR50.49) requirements which must be incorporated into the MPs / PMs upon completion this modification.

No new requirements

✓ No ___ Yes EQ Walkdown required prior to MAR/PEERE return to service

yes, list all components/tag numbers that require walkdown:

QC verification of terminal block torque per work package instructions satisfies EQ Rqmt.

M. H. Kimel 3285 4/27/96

nt Name Ext. Date

☒ **Reliability Centered Maintenance** X No ___ Yes Preventative Maintenance changes are required

Canstine 4/29/96

nt Name Ext. Date

CR3 005909

3. ☒ **Nuclear Fire Protection Specialist** ✓ No ___ Yes Procedure changes are required

List # ______ ______ ______ ______

J. L. Russell	3433	4/29/96
Print Name	*Ext.*	*Date*

4. ☐ **Others** ___ No ___ Yes Procedure changes are required

List # ______ ______ ______ ______

____

____

____

Print Name	*Ext.*	*Date*

5. ☐ **Others** ___ No ___ Yes Procedure changes are required

List # ______ ______ ______ ______

____

____

____

Print Name	*Ext.*	*Date*

6. ☒ **Project Manager** ___ No ___ Yes Procedure changes are required

R Bosworth	3668	4-29-96
Print Name	*Ext.*	*Date*

CR3 005910

MAR EVALUATION SUMMARY

MAR NUMBER 96-04-12-01.

MAR TITLE: ASV-204 EFIC Auto Open Removal

	ENGINEERING	LABOR	MATERIALS	TOTAL
RPA ESTIMATE	none			
ACTUAL COST	461		6	467

FCN'S: (i.e. reason, cause) NONE

REA'S: (i.e. reason, cause) NONE

AREAS FOR IMPROVEMENT: (i.e, Scheduling, Manpower, Design, Testing)

① There was some confusion over PMT completion / Verification & MAR package Prints were poor

MAR STRENGTHS: Job went well

Florida Power CORPORATION

RETURN TO SERVICE OF EQUIPMENT FOLLOWING MODIFICATIONS

MAR/CGWR/PEERE 96-04-12-01 NUMBER OF FCNs Ø TEMPORARY MAR INSTALLED N/A REMOVED N/A

RETURN TO SERVICE: X FULL ______ PARTIAL

PROJECT NAME ASV-204 RFIC Auto Open Removal

DESCRIPTION OF BOUNDARIES N/A

REQUIREMENTS FOR RETURN TO SERVICE	YES	N/A
Exhibit 15 reviewed to assure engineering design requirement satisfied.	☒	☐
Walkdown and System Acceptance (WSA)	☒	☐
Functional Testing/Post Maintenance Testing Complete	☒	☐
Procedures Required for Return to Service Revised	☐	☒
Training is Documented	☒	☐
Revised Control Room Drawings Available in the Control Room (201 sheets 61-71 / 206 / 208 / 220 sheets 221-226 / 302)	☒	☐
REA/Memo received verifying partial return to service evaluated by Engineering.	☐	☒

ABOVE REQUIREMENTS ARE SATISFIED: E. Bowrith 5/8/96

Nuclear Project Manager Date

RETURN TO SERVICE ACCEPTED/ANY ADMIN. CONTROLS REQUIRED UNTIL CLOSURE WILL BE MAINTAINED:

[signature] 5-8-96

Operations Designee Date

ADMINISTRATIVE REQUIREMENTS IN PLACE/LIST: ______________________________

CR3 005912

ENCLOSURE 4

WALKDOWN AND SYSTEM ACCEPTANCE (WSA)

PART I (INSTALLATION SUPERVISOR)

THIS FORM IS TO BE USED BY THE WORK GROUP SUPERVISOR TO RELEASE THE MAR/CGWR AFFECTED EQUIPMENT BACK TO NUCLEAR PROJECTS AFTER MODIFICATION INSTALLATION. LIST THE AFFECTED WORK PACKAGE NUMBERS AND THE WORK STATUS (I.E. COMPLETE, NEED ISLT, ETC.) IN PART 1 AND DELIVER THIS FORM TO THE PROJECT MANAGER OR RETURN IT TO NUCLEAR PROJECTS.

MAR/CGWR TITLE EFIC AUTO OPEN REMOVAL

MAR/CGWR NUMBER 96-04-12-01

WORK PACKAGE NUMBERS AND STATUS 335012 PACKAGE COMPLETE

JCZ [signature] 4/29/96

MAR/CGWR INSTALLER SUPERVISOR OR PROJECT MANAGER DATE

PART 2 (PROJECT MANAGER)

- START -

IS A WALKDOWN OF THE INSTALLATION REQUIRED BY THE PROJECT MANAGER

NO (X) YES ( )

ARE THERE ANY PROBLEMS OR EXCEPTIONS NOTED BY THE INSTALLATION SUPERVISOR OR WALKDOWN

NO (X) YES ( )

WALKDOWN IS COMPLETED AND ANY PROBLEMS HAVE BEEN DOCUMENTED

DOCUMENT NUMBER ______

PERFORMED BY ______ DATE

ALL PROBLEMS/EXCEPTIONS HAVE BEEN REVIEWED TO DETERMINE THE FOLLOWING:

1.( ) MUST BE CORRECTED PRIOR TO STARTING TESTING
( ) ABOVE PROBLEMS/EXCEPTIONS ARE NOW CORRECTED, TESTING CAN PROCEED.
2.( ) MAY BE CORRECTED AFTER TESTING HAS STARTED

INSTALLER SUPERVISOR ______ DATE

PROJECT MANAGER ______ DATE

WILL INSTALLATION BE RELEASED TO TEST ENGINEERING

E. Bosworth 4/29/96

PROJECT MANAGER DATE

NO ( ) YES (X) OK

OK

RELEASED TO TEST ENGINEERING FOR TESTING

~~E. Bosworth 4/29/96~~

PROJECT MANAGER DATE

ALL TESTING NEEDED FOR RETURN TO SERVICE TO OPERATIONS IS COMPLETED AND IS ACCEPTABLE

E. Bosworth 5/8/96

PROJECT MANAGER DATE

ALL PROBLEMS HAVE BEEN RESOLVED AND THESE WORK PACKAGES ARE READY FOR RETURN TO SERVICE TO OPERATIONS

E. Bosworth 5/8/96

PROJECT MANAGER DATE

Florida Power
CORPORATION

COMPLETION OF MODIFICATION TESTING NOTICE

MAR CGWR 96-04-12-01

☒ Complete ☐ Partial *

* Description of Partial Modification Turnover: ______

MAR Testing Requirements: ☒ Yes ☐ No
(As specified by Test Engineering)

☒ MAR Functional Test Procedure(s):
List: TP ☒ Completed

☐ Work Request Test Instructions:
WR#: ______ ☐ Completed

☐ POQAM Procedure(s):
List: ______ ☐ Completed

MAR ISI Requirements: ☒ Yes ☐ No
List: SP-349 B ☒ Completed
5/7/96
Swingsmith

MAR Examination Requirements: ☐ Yes ☒ No
List: ______ ☐ Completed

Post Maintenance Testing Listed on MAR / CGWR related work request(s): ☐ Yes ☒ No
List: ______ ☐ Completed

Comments: ______

All Modification / Post Maintenance Testing requirements for turnover are satisfied.

G. Bosworth
Project Manager

5/8/96
Date

CR3 005914

3/90

RET: Life of Plant RESP: Nuclear Engineering

Florida Power Corporation

PROCEDURE HOLD NOTIFICATION

TO: DOCUMENT CONTROL

SUBJECT: NOTIFICATION FOR PROCEDURE HOLD

THE FOLLOWING MODIFICATION OR EQUIVALENCY REPLACEMENT HAS BEEN INSTALLED AND AFFECTS PLANT PROCEDURES.

PLEASE PLACE THE LISTED PROCEDURES ON HOLD UNTIL THEY ARE REVISED TO REFLECT THIS MODIFICATION/EQUIVALENCY REPLACEMENT OR, IF AUTHORIZED, USE "AS IS" BY THE INTERPRETATION CONTACT.

MAR/CGWR No: 96 - 04 - 12 - 01 - ____

PEERE No: ____________

AFFECTED PROCEDURES:

SP-349A

SP-640A

E Bosworth

PROJECT'S REPRESENTATIVE

CC: INTERPRETATION CONTACT

These procedures ok to use "as is". Frank Dola per telecom 5/1/96 EB

RECEIPT ACKNOWLEDGEMENT BY:

S. Bornemeier

NUCLEAR DOCUMENT CONTROL SUPERVISOR/DESIGNEE

PROCEDURE REVIEW RECORD (page 1 of 3)

PRR# **Proc #** **Current Rev #** **Title:**

DESCRIPTION OF NEW PROCEDURE OR CHANGE (Detailed)

SP-349A
SP-340A
Prior to
closure

REASON AND REFERENCES FOR PROCEDURE CHANGE

- ☐ Result of scheduled procedure review notification from
  - -PRNR-A previously submitted to Document Control
  - -Enclosure 23 of AI-402B attached
- ☐ Biennial Review conducted prior to normal review frequency
  - -PRNR-B (Enclosure 10) attached
  - -Enclosure 23 of AI-402B attached (as required)
- ☐ Plant Modification: MAR Number ________
- ☐ Non-Conformance: Finding number and date ____/____
- ☐ Technical Specification Amendment: Specification Number ________

Originated By ________ Title ________ Date ________

Final Procedure Approvals

Interpretation Contact ________ Date ________

PRC Meeting No. ________

Plant Review Chairman ________ Date ________

Dir. Nuc. Plant Oper. ________ Date ________

☐ Not Required

NOTIFICATION OF NEW/REVISED PROCEDURE

Procedure # AP-770		PRR #11482		
			Reviewed	
	Check		Initial	Date
To:		Manager, Nuclear Chem/Rad Protection		
	✓	Manager, Nuclear Plant Operations		
		Supervisor, Nuclear Engineering Support Services		
		Manager, Nuclear Plant Technical Support		
		Manager, Nuclear Security		
		Supervisor, Nuclear Document Control		
		Supervisor, Nuclear Records Management		
		Manager, Nuclear Plant Maintenance		
		Manager, Nuclear Quality Assessments		
		Manager, Nuclear Engineering Design		
		Manager, Site Nuclear Services		
		Manager, Radiological Emergency Planning		
		Manager, Nuclear Licensing		
	✓	Procedure Interpretation Contact D. deMontfort		
	✓	Manager, Nuclear Licensing Operator Training		
		NOTE: For others below, state their name and address Other:		
		Other:		

NOTE: This procedure has been approved and issued for use. It is your responsibility to advise personnel in your organization affected by this new/revised procedure of its contents. Attached is a copy of the Procedure Description, Reasons, and References (Enclosure 2).

CR3 005917

DESCRIPTION OF NEW PROCEDURE OR CHANGE (Detailed)

3.29 Changed step obtain guidance from TSC if diesel fuel reserves will not be adequate to run 2 EDGs at 12 hours following an accident. Changed guidance from 10 hours at the 2 hour point in the accident to 12 hours from the start of the accident

REASON AND REFERENCES FOR PROCEDURE CHANGE

Guidance for diesel fuel management belongs in TSC as this becomes a concern 12 hours after a loss of offsite power.

Check As Appropriate:

	Reference#
[✓] Modification [ ] FCN [ ] Other Design Related Item	MAR96-03-12-01 MAR96-04-12-01
[ ] Problem Report [ ] Precursor Card [ ] NRC Violation	
[ ] ITS Amendment [ ] ITS Bases Change [ ] FSAR Change	

orig

EEB

CR3 005918

PROCEDURE DESCRIPTION, REASONS, AND REFERENCES

Procedure #: AP-770 PRR #: 11482

DESCRIPTION OF NEW PROCEDURE OR CHANGE (Detailed)

3.1 Changed SOTA to STA
3.5/3.6 split original step into two steps

REASON AND REFERENCES FOR PROCEDURE CHANGE

Editorial changes to comply with AI 402A

DESCRIPTION OF NEW PROCEDURE OR CHANGE (Detailed)

3.4 added new step to increase EDG voltage if voltage is too low to make up the READY matrix

REASON AND REFERENCES FOR PROCEDURE CHANGE

provide additional guidance to reenergize a dead bus

DESCRIPTION OF NEW PROCEDURE OR CHANGE (Detailed)

Table 1 Changed EDG Ratings to agree with Calculations for MAR 96-03-12-01 for the new EDG KW indicators

REASON AND REFERENCES FOR PROCEDURE CHANGE

These values had not previously been error corrected and are required to be so for load management.

DESCRIPTION OF NEW PROCEDURE OR CHANGE (Detailed)

Tables 2 and 3 changed to Enclosure 1 and added starting KW loads values

REASON AND REFERENCES FOR PROCEDURE CHANGE

Required information would no longer fit on facing page.

DESCRIPTION OF NEW PROCEDURE OR CHANGE (Detailed)

3.7 added new step to provide guidance when EDG load rating limits are challenged. References a new Enclosure 2 to Cross-connect EFP-2 to EFV-57 and 58 if EDG-1A limits are challenged and EFP-2 flow indication is not available

REASON AND REFERENCES FOR PROCEDURE CHANGE

Insufficient available load exists on EDG-1A during some accident scenarios. This new enclosure provides the operator with guidance that will allow shutting down EFP-1 to provide sufficient operating margin to handle Starting KW of manually applied loads.

CR3 005919

DESCRIPTION OF NEW PROCEDURE OR CHANGE (Detailed) 3.9 Changed Letdown recovery to reference EOP-14 for consistency **REASON AND REFERENCES FOR PROCEDURE CHANGE** Consistency
DESCRIPTION OF NEW PROCEDURE OR CHANGE (Detailed) 3.10 Added step to provide guidance to reenergize a faulted bus when repairs are complete **REASON AND REFERENCES FOR PROCEDURE CHANGE** Provides guidance that did not previously exist for recovering a faulted bus
DESCRIPTION OF NEW PROCEDURE OR CHANGE (Detailed) 3.11 Moved steps to restore a failed EDG to Enclosure 3 **REASON AND REFERENCES FOR PROCEDURE CHANGE** The required guidance exceeded one page. Moved to Enclosure to comply with AI-402A.
DESCRIPTION OF NEW PROCEDURE OR CHANGE (Detailed) 3.12 Reworded step to be consistent with EOPs **REASON AND REFERENCES FOR PROCEDURE CHANGE** Consistency
DESCRIPTION OF NEW PROCEDURE OR CHANGE (Detailed) Various Steps. Additional information concerning starting KW load has been added. **REASON AND REFERENCES FOR PROCEDURE CHANGE** implementing requirement to prevent exceeding 3500 KW during equipment start.
DESCRIPTION OF NEW PROCEDURE OR CHANGE (Detailed) 3.19 Reworded step to be concistent with EOPs **REASON AND REFERENCES FOR PROCEDURE CHANGE** Consistency
DESCRIPTION OF NEW PROCEDURE OR CHANGE (Detailed) 3.25 Changed details to direct starting CC emergency recirc per EOP-3 if adequate subcooling margin does not exist. **REASON AND REFERENCES FOR PROCEDURE CHANGE** Guidance for starting CREVS is required in EOP-3. This will prevent any interlocking of AP-770 and EOP-3

CR3 005920

orida
ower

DATA TRANSMITTAL SHEET

PREPARED BY FEB	DATE 05/07/96	
FILE CODE(S) MAR 96-04-12-01		
PROJECT ASV-204 EFIC AUTO OPEN REMOVAL		
PLANT CRYSTAL RIVER UNIT #3		

TYPE OF DATA	STATUS OF DATA	ACTION TO BE TAKEN	RETURN COMMENTS TO:
CWP Eng. Study or Rec. Instruction Manual Prints/Drawings Specifications Other	☐ Preliminary ☐ Interim ☒ As-built ☐ Revised (DCN or Plant Rev.) ☐ Approved/Accepted ☐ Other	☐ Receipt/Acknowledge ☐ Approve & Return ☐ See "Special Instructions" ☒ New Data for File ☐ Review & Submit Comments ☐ Certify ☐ Other	BY DATE:

No. Cop's	Route To	Pur-pose
OR	CSS	S/F

No. Cop's	Distribution	Pur-pose
1S	CSS	MF
5AC	RHC	
1AC	PP	**
1AC	FT	**
DTS	KFL	

AL INSTRUCTIONS

BUILT PER MAR 96-04-12-01

DOCUMENT NUMBER	REV	TITLE OR DESCRIPTION
		SEE ATTACHED FOR LIST OF DRAWINGS

ADDITIONAL COMMENTS OR INSTRUCTIONS

9'S - 3P RHC
0'S - 3P RHC
1'S - 1P RHC

PP (Parsons Power) - 201, 205, 206, 207, 208, 209, 210, 302, & 308 ONLY
FT (Framatome Technologies) - 205, 206, 302, & 308 ONLY

RING ASSISTANT / ENGINEER'S APPROVAL	DATE	VERIFICATION	DATE	SUPERVISOR'S APPROVAL	DATE
F.E. Blackman F. E. BLACKMAN	5/7/96	Larry Maakestad L. E. MAAKESTAD	5/7/96	Chris A. Stenner C. E. STONER	5/7/96

RET: 2 YRS. RESP: ENERGY SUPPLY

CR3 005921

DRAWING NO.	REV # ↓	PROJ →	96-04-12-01								
209-008, AS-06	8		X								
210-124	31		X								
210-619	19		X								
210-624	46		X								
211-008, AS-06	5		X								
600-061, SH. 4	2		X								
600-066, SH. 5	3		X								

lorida
ower

14-DAY DWGS DATA TRANSMITTAL SHEET

PREPARED BY	DATE	
FEB	04/30/96	
FILE CODE(S) MAR 96-04-12-01		
PROJECT ASV-204 EFIC AUTO OPEN REMOVAL		
PLANT CRYSTAL RIVER UNIT #3		

TYPE OF DATA	STATUS OF DATA	ACTION TO BE TAKEN	RETURN COMMENTS TO:
CWP Eng. Study or Rec. Instruction Manual Prints/Drawings Specifications Other	☐ Preliminary ☐ Interim ☒ As-built ☐ Revised (DCN or Plant Rev.) ☐ Approved/Accepted ☐ Other	☐ Receipt/Acknowledge ☐ Approve & Return ☐ See "Special Instructions" ☒ New Data for File ☐ Review & Submit Comments ☐ Certify ☐ Other	BY DATE:

No. Cop's	Route To	Pur-pose
OR	CSS	S/F
No. Cop's	**Distribution**	**Pur-pose**
1S	CSS	MF
5AC	RHC	
1AC	PP	**
1AC	FT	**
DTS	KFL	
*	SEE BELOW	

CIAL INSTRUCTIONS

-BUILT PER MAR 96-04-12-01

DOCUMENT NUMBER	REV	TITLE OR DESCRIPTION
08-008, AS-07	4	TURB DRIVEN EMERG FEED PUMP INLET ISOL VLV ASV-204
08-026, EF-15	9	EFIC MATRIX "A" CONTROL AND EFW ACTUATION "A"

ADDITIONAL COMMENTS O' INSTRUCTIONS

'S - 6P RHC

P (Parsons Power) - 201, 205, 206, 207, 208, 209, 210, 302, & 308 DRAWINGS ONLY
T (Framatome Technologies) - 205, 206, 302, & 308 DRAWINGS ONLY

RING ASSISTANT / ENGINEER'S APPROVAL	DATE	VERIFICATION	DATE	SUPERVISOR'S APPROVAL	DATE
e F. Blackman	4/30/96	Larry Maakestad	4/30/96	Chris A. Stevens	4/30/96
FAYE E. BLACKMAN		L. E. MAAKESTAD		C. E. STEWER	

RET: 2 YRS. RESP: ENERGY SUPPLY 807 703

CR3 005923

Florida Power
CORPORATION

MAR/CGWR/PEERE DRAWING / AS-BUILT UPDATE

To: Supervisor, Nuclear Design Support
Please initiate MAR/CGWR/PEERE drawings and "As-Built" requirements as per the below conditions

MAR/CGWR/PEERE 96-04-12-01

☐ Temporary modification removed.

☐ Permanent MAR/CGWR/PEERE is expected to be completed within 14 days, please update the fast turn around drawings.

☒ Permanent MAR/CGWR/PEERE is complete and return to service is in process or accepted by operations, please initiate the "As-Built" requirements.

☐ MAR/CGWR is partially installed and is currently ☐ in progress or ☐ on hold, please update the necessary plant drawings to reflect actual conditions.

Work Package No.(s) 335012

Partial installation consists of: ____________________

Fast Turnaround MAR/CGWR/PEERE Drawings

302/201 (061-071) / 206 / 208 / 220's (221-226)

drawings are affected: X Yes ___ No

If yes, list:

208-008 AS-07

208-026 EF-15

Fast Turnaround FCN Drawings

302/201 (061-071) / 206 / 208 / 220's (221-226)

drawings are affected: ___ Yes X No

If yes, list:

E. Bosworth
Nuclear Projects Specialist

Date 4/29/96

RECEIPT OF DOCUMENTATION	
Acknowledged by Nuclear Engineering: Chris D. Stevens	Date 4/29/96

Retain original: Return a copy to the MAR office (NA1B).

CR3 005924

3/96

RET: Life of Plant RESP: Nuclear Engineering

Florida Power

COMPLETION OF ENVIRONMENTAL QUALIFICATION (EQ) NOTICE

MAR 96-04-12-01

☒ Complete ☐ Partial

EQ requirements must be satisfied prior to turnover: ☐ Yes ☒ No

Description of Partial Modification Turnover:

MAR Title: ASV-204 EFIC Auto Open Removal

All Environmental Qualification (EQ) requirements per 10CFR50.49 for the subject MAR have been satisfactorily completed.

[signature]
Nuclear Engineering - (EQ)

4/27/96
Date

Upon completion, return to the Nuclear Projects Department (NA1B)

CR3 005925

Florida Power
CORPORATION

"FAST RETURN TO SERVICE" MODIFICATION DRAWING UPDATE NOTICE

MAR/CGWR 96-04-12-01

Control Room drawing updating is required.

Applicable Drawings: 208-008, AS-07
208-026, EF-15

The applicable "Fast Return To Service" MAR/CGWR drawing(s) have been transmitted to Document Control for Control Room updating.

E. Froelich 5/8/96
Nuclear Projects Manager — Date

Dick Craft 5-8-96
Document Control — Date

CR3 005926

Florida Power Corporation

COMPLETION OF MODIFICATION TRAINING NOTICE

MAR/CGWR 96-04-12.01

☒ Complete ☐ Partial *

* Description of Modification Boundaries:

MAR/CGWR Title: ASV-204 EFIC AUTO OPEN Removal

Training required: ☒ Yes ☐ No

Training required prior to return to service : ☒ Yes ☐ No

COMMENTS: Training will be performed prior to startup.

Training complete ZB

Johnnie Smith per telecon ZB 4/29/96

Nuclear Operations Training Supervisor/Designee Date

Upon completion, return to the Projects Department (NA1B)

CR3 005927

Florida Power CORPORATION

MAR TRANSMITTAL

MAR NUMBER	TRANSMITTAL DATE (To be entered by Project Manager)
96-04-12-01	4-27-96
TITLE	
ASV-204 EFIC Auto Open Removal	

TEMPORARY MAR EXPIRATION DATE: N/A

SOURCE DOCUMENT: PC 96-2197

SI REVIEW REQUIRED: [X] YES [ ] NO

OPEN ITEMS INCLUDED: [X] YES [ ] NO

PROJECT TEAM ENDORSEMENTS (PM may NA with justification for projects >$50,000)	
MAINTENANCE:	OTHER:
OTHER:	OTHER:
PROJECT MANAGER: [signature] / BGB	OTHER:
JUSTIFICATION FOR NA:	

PRC REVIEW REQUIRED: [X] YES [ ] NO

PRC CHAIRMAN: [signature] as alternate PRC Chairman

PRC MEETING NUMBER: 96-17

	CHAIRMAN	DATE
NRC Review Required [ ] YES [ ] NO NRC Approval		
Installation Authorized	NUCLEAR PLANT MANAGER [signature]	DATE 4/27/96
Modification Authorized	NUCLEAR PLANT MANAGER	DATE

DISTRIBUTE AS FOLLOWS:

Mgr., Nucl Projects w/orig. & 5 copies; w/3 sets dwgs.
-3 Document Control w/attach; w/3 a/c
...sons w/1 a/c and Sec. B of T.O.C.
...matone w/1 a/c and Sec. B of T.O.C.
Supervisor, Nuclear Operations Training Admin, w/attach; 1 a/c
Mgr., Nucl Licensed Op. Tr. w/attach and dwgs.
Supervisor, Nuclear Design Support w/attach and dwgs.
Supervisor, Nucl Eng (Functional Testing) w/attach and dwgs.
Supervisor, Nucl Eng (EQ) w/attach and dwgs. if EQ reviewed.
[✓] YES [ ] NO
Mgr., Nucl Procurement Eng. Services w/attach and dwgs.
NEP 228 Review [ ] YES [✓] NO
Senior Nuclear Reactor Engineer w/attach [ ] YES [✓] NO

Supervisor, Nucl Eng (Elec.) w/attach and dwgs.
EDG Review [✓] YES [ ] NO
Elec. Calc Review [ ] YES [✓] NO
SBO Review [ ] YES [✓] NO
Transmittal Only [ ] YES [✓] NO
Supervisor, Nucl Eng (I&C) [✓] YES [ ] NO (Transmittal Only)
Supv, Nucl Eng (Mech) w/attach and dwgs.
Hydraulic Calc Review [ ] YES [✓] NO
Transmittal Only [✓] YES [ ] NO
Supervisor, Nucl Eng (Civil/Struc) [ ] YES [✓] NO (Transmittal Only)
Supervisor, Maintenance/Component Engineering w/attach and dwgs.
SMOV Review [ ] YES [✓] NO
Senior Radiation Protection Engineer w/attach and dwgs. [ ] YES [✓] NO
Supervisor, NPTS (I&C) for ERDS Action w/attach [ ] YES [✓] NO
NGRC Chairman w/attach and dwgs. [ ] YES [✓] NO

CR3 005028

Florida Power CORPORATION

MAR Number 96-04-12-01

MAR APPROVALS

Crystal River Unit 3

A. WALKDOWNS: Conceptual Design/Design Development (Mandatory) ☐ Yes ☑ No

If no, provide justification: CONCEPTUAL DESIGN NOT REQUIRED DESIGNED REQUIRED FOR CRITICAL PATH TO START UP FROM REFUEL 10.

Confirmation Walkdown (Optional) ☐ Yes ☑ No

B. REQUIRED SIGNATURES

Design Engineer	☒ Yes		C.B. Doyel — Design Engineer	4/26/96 / Date
EQ Reviewer	☒ Yes	~~☒~~ No CBD 4/26/96	[signature] — EQ Reviewer	4/26/96 / Date
Fire Protection Engineer	☐ Yes	☒ No	Fire Protection Engineer	/ Date
Nuc Plant Technical Support	☒ Yes		Robert X Mazzi — NPTS	4/26/96 / Date
Nuclear Plant Operations	☒ Yes		Carl W Bergstrom — NPO	4-26-96 / Date
Site Welding Engineer (Optional)	☐ Yes	☒ No	Site Welding Engineer	/ Date
Verification Engineer	☒ Yes	☐ No	[signature] — Verification Engineer	4/26/96 / Date

Verification Method Used:

☒ Design Review
☐ Alternative Calculations
☐ Qualification Testing

Design Engineer's Supervisor	☒ Yes	BB Shook — Supervisor	4/26/96 / Date

CR3 005929

Florida Power
CORPORATION

DESIGN INPUT RECORD

Crystal River Unit 3

Sheet 1 of 1

MAR NUMBER
96-04-12-01

☐ Electrical ☑ I&C ☐ Mechanical ☐ Structural

1.) This MAR is being driven by the performance requirements of 2 major components in the plant, the Steam Driven Emergency Feedwater Pump, EFP-2, and the "A" Emergency Diesel Generator, EGDG-1A. In 1987, while struggling with the question of how to meet the loading requirements for EGDG-1A after loading EFP-1 on it for a LOCA, concurrent with a Loss of Offsite Power (LOOP), and a "B" Battery failure, the solution was proposed, accepted, and installed to power ASV-204 from the "A" electrical train and initiate an automatic opening of the valve in the event of an EFIC actuation. By doing this, the load expected to be carried by EFP-1 would be minimized as EFP-2 would be running without any controls, i.e. at full speed. This was the only way that the projected load on EGDG-1A could remain within the rated capacity of the machine at that time. Since then, the upgrade of the diesel generators has been completed, and new electrical calculations show that EGDG-1A can handle the load of EFP-1 in this accident scenario. In addition, a concern about the NPSH requirements for EFP-2 not being met when the pump is running in this all out condition dictates that this automatic opening of ASV-204 be removed.

5.) ASV-204 is classified as Essential Equipment and as QQ. This modification will delete the EFIC Actuation automatic opening of the valve. The valve will remain QQ as it will be used as a remotely operated valve. This valve provides redundancy within one train of Emergency Feedwater, but is not required for redundancy of the other train. This MAR will remove the automatic opening and the active designation from SFD 600-061 sht 4 and 600-066 sht 5, showing it as a passive component. The 10CFR50.49 qualification will not be affected.

13.) REA 96-0467 provides the new calculation demonstrating that EGDG-1A will handle the electrical load resulting from eliminating the automatic opening of ASV-204 and the running of EFP-2 in the above referenced scenario. It should also be noted that this scenario is bounding for the other scenarios where the EDG's are required, i.e. Main Steam Line Break, Feedwater Line Break, etc.

15.) The current ongoing EOP revision program has identified that the Operator action on seeing this scenario and the initiation of EFP-2, without flow control and indication, would secure EFP-2. This MAR relieves the operator from facing the scenario of having the pump operational without any flow control of indication.

18.) ASV-204 provides a redundant steam admission valve to the EFP-2 turbine, redundant to ASV-5. Removal of the automatic opening of ASV-204 doesn't eliminate the redundancy as the operator will still have the capability to open the valve manually. This capability will also continue to be from the "A" electrical train, whereas ASV-5 is powered from the "B" electrical train. It is noted that this redundancy is in one train only and does not affect the redundancy of having the other train of Emergency Feedwater.

22.) AS-9, The local control station is located in ALARA Zone III.

Florida Power CORPORATION

MAR PROGRAM IMPACT ASSESSMENT

Crystal River Unit 3

MAR/PEERE NUMBER

96-04-12-01

	YES	NO	
1.	☐	☒	The MAR affects an NPRDS reportable item or adds a new component or item.
2.	☐	☒	The MAR affects an MOV component or affects the differential pressure calculations for an MOV.
3.	☐	☒	The MAR affects the ES System Hydraulic Calculations.
4.	☐	☒	The MAR affects the Electrical System Calculations.
5.	☒	☐	The MAR affects EDG loading.
6.	☐	☒	The MAR affects the Regulatory/Environmental Review form.
7.	☐	☒	The MAR adds or relocates Control Room equipment entailing direct operator interface that requires Human Factors design review per criteria stated in NUREG-0700 and/or adds, deletes or modifies any other equipment which could affect the characteristics of plant operation as displayed to the plant operator.
8.	☐	☒	The MAR affects the Control Room Habitability Envelope.
9.	☐	☒	The MAR represents a repair or replacement which is governed by ASME Section XI, IWA-4000 or IWA-7000, respectively.
10.	☐	☒	The MAR will delete or modify inputs to the Safety Parameter Display System (SPDS).
11.	☐	☒	The MAR affects exterior physical structures in the protected area and/or within 20 feet outside the protected area.
12.	☐	☒	The MAR contains equipment that will be part of or affect an instrument loop located in a harsh environment.
13.	☐	☒	The MAR affects SBO requirements.
14.	☐	☒	The MAR affects HELB requirements.
15.	☐	☒	The MAR affects NI or calorimetric instrumentation and/or results in configuration changes which affect NI/calorimetric instrument readings.
16.	☐	☒	The MAR affects hardware or software associated with the transmitting of Emergency Response Data System data points.
17.	☐	☒	The MAR affects Fire Protection Review requirements.
18.	☐	☒	The MAR modifies an item that: ☐ a. was originally procured and installed in an unmodified state. ☐ b. is being procured as part of this modification and will be modified after receipt; or ☐ c. is currently in inventory (FIMIS) in an unmodified state and will be modified prior to installation and/or restocking; or ☐ d. will be completely or partially replaced by an item of different design or materials.
19.	☐	☒	The MAR requires duplicate ordering of parts for the Simulator.
20.	☒	☐	The MAR requires a Plant Design Basis Document change.
21.	☐	☒	The MAR affects an MCB Panel, Rack or Cabinet Bill of Material.
22.	☐	☒	The MAR requires the ordering of spare parts.
23.	☐	☒	The MAR requires a change to the Configuration Management Information System data.
24.	☐	☒	The MAR will add digital equipment of any kind to the plant.
25.	☐	☒	The MAR changes safety related fuse curve(s) contained in the Electric Design Fuse Time Current Curves or adds a new safety related fuse to CR3.

CR3 005931

Florida Power CORPORATION

EMERGENCY DIESEL GENERATOR LOADING JUSTIFICATION

MAR/FCN NUMBER 96-04-12-01	
PROJECT TITLE EFIC ASV-204 Automatic Opening Removal	CBD 4/26/96

. One of the below listed items must be checked.

☐ The load is needed to bring the plant to a safe condition after a design basis accident.

☐ The load is required to be EDG backed per NRC regulations (e.g., Reg. Guide, NUREG, IE Bulletin, etc.). The NRC regulation is identified as follows:

____

☐ The load belongs to a system previously analyzed and documented in the FSAR as requiring to be EDG backed. The applicable FSAR section is identified as follows:

______________ ______________

☒ Other (Explain)

EFP-1 loading must be recalculated for the most bounding case as the previous bounding case is no longer applicable.

Complete the following:

a. Kw load (check as applicable and enter load): ☒ added Full EFW flow load ☐ deleted __

b. EDG affected (check as applicable): ☒ A ☐ B

c. Type of load (check as applicable): ☐ continuous duty ☒ intermittent duty

If intermittent, estimated % of time the equipment will be energized: 100% until LPI pump is required at which time it will go to 0% as EFP-2 will be cross tied and used to provide EFW flow.

d. Method of loading (check as applicable): ☒ auto-loaded ☐ manual loaded

If manual, time after initiation of accident when item should be loaded: ______________

e. Identify panel / MCC / switchgear / breaker by tag number or device number to which the load is being added. If not applicable, enter NA.

panel Same as previously

switchgear Same as previously

MCC Same as previously

breaker Same as previously

REA 96-0467 HAS RECALCULATED THE MOST BOUNDING CASE AND THE LOADING IS AS FOLLOWS:
EGDG-1A 6th BLOCK LOADING = 3656.7 KW
EGDG-1A 1 MINUTE LOADING = 3159.1 KW
EDG LOADING LOG UPDATED.
Mattew W. Coleman 4-26-96

NSM ENGINEER	DATE	SUPERVISOR, NUCLEAR ENGINEERING ELECTRICAL	DATE
C.B. Doyel	4/26/96	D. Shook for S.F. Ullen	4/26/96

Florida Power CORPORATION

MAR OPEN ITEMS

Crystal River Unit 3

MAR OPEN ITEMS
(If NA, do not include in package)

EDBD Change to Chapters 6-10, Rev. 5 and 6-13, Rev. 3

Field Work Package

CR3 005933

Florida Power
CORPORATION

MAR
TABLE of CONTENTS
Crystal River Unit 3

MAR NUMBER

96-04-12-01

Section A - MAR Inclusions

1. MAR Open Items (1 sheet)
2. MAR Description / 10CFR50.59 Screening (5 sheets)
3. Nuclear Engineering Safety Impact Assessment (1 sheet)
4. Installation Instructions / Considerations (2 sheets)
5. Design Data Sheets (2 sheets)
6. Design Input Record (1 sheet)
7. MAR Program Impact Assessment (1 sheet)
8. Checklist for EQ Document Review (2 sheets)
9. Emergency Diesel Generator Loading Justification (1 sheet)
10. IOC NED96-0272, FSAR Change per MAR 96-04-12-01 (5 sheets)
11. Regulatory/Environmental Review (3 sheets)

Section B - Drawings

1. Installation Drawings (All Interim Rev. A)

   208-008, AS-07
   208-026, EF-15
   209-008, AS-06
   210-124
   210-619
   210-624
   600-061, sht 4
   600-066, sht 5

2. Sketches

   None

3. Drawings to be As-Built

   ss-211-008, Sheet AS-6, Rev.4. As Build circuit ASE36, change wire mark of conductpor #4 to 11. Circuit ASE37, spare conductors 1,2,3,4.

4. Reference Drawings (not attached)

   None

Section C - References

1. REA 96-0467
2. FSAR Sections 7.2.4 and 10.5.2, Rev. 22
3. EDBD Sections 6-10, Rev. 5, and 6-13, Rev. 3
4. ITS 3.3.11, 3.3.12, 3.3.13, 3.3.13, 3.7.5, Amend 151

MAR DESCRIPTION/10CFR50.59 SCREENING

MAR NUMBER

96-04-12-01, ASV-204 EFIC Auto Open Removal

A. 1. Safety Related ☒ Yes ☐ No

2. Safety Classification Basis

☒ Configuration Management Information System (CMIS)

☒ Screen Print

☐ Drawing(s) Used:

☐ Safety Classification Review form (List CIDP Number if not attached) (CIDP #)

B. Description of MAR

This MAR removes the EFIC Auto Open signal from the EFIC "A" Channel on ASV-204. In addition it will remove the interlock that prevents the closure of ASV-204 when there is an EFIC actuation.

C. 10CFR50.59 Screening for Applicability (Review NEP 210, Exhibit 4, prior to response).

1. Is this a change to the facility as described in the FSAR?

☒ Yes ☐ No

List FSAR Section(s) Reviewed: Should have [illegible] [illegible] ? [illegible]

7.2.4, 7.1.3.2.3, 10.5, 14.2.2.5b and [illegible] [illegible] updates

2. Does this change affect the ITS Bases?

☐ Yes ☒ No

List ITS Bases Reviewed:

B 3.3.11, B 3.3.12, B 3.3.13, B 3.3.14, B 3.7.5

3. Does this change involve changes to the Technical Specifications?

☐ Yes ☒ No

List Technical Specification Section(s) Reviewed:

3.3.11, 3.3.12, 3.3.13, 3.3.14, 3.7.5

If "Yes" to items C.1, C.2 or C.3 perform a 10CFR50.59 Safety Evaluation (Exhibit 4, Part II) and provide notification to the Manager, Nuclear Licensing per NEP 210, Instructions for Part I Exhibit 4.

Florida Power

10CFR50.59 SAFETY EVALUATION

Document No. MAR 96-04-12-01

PART II - 10CFR50.59 Evaluation

Review NEP 210, Exhibit 4, instructions before completing Part II. All "Yes" and "No" answers must be explained.

1. Is the probability of occurrence of an accident previously evaluated in the FSAR increased?

☐ Yes ☒ No

Eliminating the automatic opening upon an EFIC actuation of ASV-204 will not impact the probability of occurrence of any of the accidents analyzed in Chapter 14 of the FSAR. Those accidents involve the failure of large plant components and none of them are started by failure of the control circuitry for ASV-204. FSAR Chapter 14 reviewed.

2. Are the consequences of an accident previously evaluated in the FSAR increased?

☐ Yes ☒ No

The accidents that the opening of ASV-204 is involved in are any that involve an EFIC initiation. Since they all take a concurrent LOOP, this involves every one of them. The only single failure that may require ASV-204 to automatically open is a failure of ASV-5 to automatically open. Any single failure that causes ASV-5 to not open precludes a single failure that would effect the "A" Emergency Feedwater Train. Since the "A" EFW train would be available to mitigate the consequences of any postulated accident, the consequences of that accident would not be increased. FSAR Chapter 14 reviewed.

3. Is the possibility of an accident of a different type than any previously evaluated in the FSAR created?

☐ Yes ☒ No

Since this MAR takes out the capability of ASV-204 to open automatically, it actually decreases the possibility of an accident of a different type than any previously evaluated in the FSAR, as ASV-204 cannot spuriously open as a result of a failure in that circuit. FSAR Section 10.5.2 reviewed.

4. Is the probability of occurrence or malfunction of equipment previously evaluated in the FSAR increased?

☐ Yes ☒ No

ASV-204 becomes a redundant manually operated steam admission valve to EFP-2. This does not impact the redundancy of the Emergency Feedwater system. FSAR Section 10.5.2 reviewed.

CR3 005936

Florida Power

10CFR50.59 SAFETY EVALUATION

ocument No. MAR 96-04-12-01

5. Are the consequences of malfunction of equipment previously evaluated in the FSAR increased?

☐ Yes ☒ No

ASV-204 becomes a redundant manually operated steam admission valve to EFP-2. The Emergency Feedwater system remains fully redundant.

FSAR Section 10.5.2 reviewed.

6. Is the possibility for malfunction of equipment of a different type than any previously evaluated in the FSAR created?

☐ Yes ☒ No

Removal of the automatic opening on EFIC actuation on ASV-204 actually decreases the possibilities of malfunctions of this equipment of a different type than previously evaluated in the FSAR since it removes two circuits that could fail from the plant. FSAR Section 10.5.3 reviewed.

7. Is the margin of safety, as defined in the basis for any Technical Specification, reduced?

☐ Yes ☒ No

The margin of safety, as defined in the basis for Technical Specification 3.7.5, Emergency Feedwater System, is satisfied by one of the two trains remaining operational in the event of the defined accidents even with a single failure. Removal of the automatic opening of ASV-204 does not change the capability of the EFW System to meet that requirement. Technical Specification 3.7.5 and its Bases reviewed.

If the answer is "Yes" to any of the questions in Part II an unreviewed safety question (USQ) is involved. Contact the Manager, Nuclear Licensing per NEP 210, Exhibit 4.

PREPARED BY C. B. Doyel DATE 4/26/96	APPROVED BY D. A. Shook DATE 4/26/96
Printed Name of Preparer C. B. DOYEL	Printed Name of Approver D. A. SHOOK

CR3 005937

Florida Power CORPORATION

NUCLEAR ENGINEERING SAFETY IMPACT ASSESSMENT

DOCUMENT/NUMBER/REVISION
MAR 96-04-12-01

PART I - Screening for Impact on Plant Operations.

A. Could this change potentially reduce the level of safety of Crystal River Unit 3?

☐ Yes ☒ No

Justification: (Do not simply restate the question.)

This MAR improves the level of safety at CR3. Per the 10CFR50.59 Evaluation, it creates no increased probability or possibility of accidents, or no increased consequences of those accidents. It improves the Emergency Feedwater System response to a LOCA, LOOP and "B" Battery failure, by eliminating reliance on EFP-2 when no flow controls or indication is present.

B. Could this change possibly lead to an event that impacts the safe operation of Crystal River Unit 3.?

☐ Yes ☒ No

Justification: (Do not simply restate the question.)

Examination of the events that the EFW system is required to mitigate, and the single failure analysis of the system with the new configuration, shows that this MAR will not impact the availability of the EFW system. Eliminating the capability of ASV-204 to automatically open on EFIC actuation also eliminates possible spurious failures.

If either answer is "Yes," PRC review is required.

PREPARED BY C. B. Doyel DATE 4/26/96	APPROVED BY D. A. Shook 4/26/96 DATE
Printed Name of Preparer C. B. DOYEL	Printed Name of Approver D. A. SHOOK

CR3 005938

MAR 96-04-12-01

Attachment 10 Page 3 of 5

10.5.2 System Description

The Emergency Feedwater System is required to assure an adequate emergency supply to the OTSGs to remove reactor decay heat until suitable conditions are attained to start the Decay Heat Removal System. The EF System consists primarily of redundant feedwater pumps powered from diverse power sources and the Emergency Feedwater Initiation and Control System (EFIC) which are required to function during periods when main feedwater supply and/or the electrical supply to the vital main feedwater auxiliaries has been lost. The EFIC System, a four channel safety grade control system, activates EF System components upon (1) both main feedwater pumps are tripped and reactor power is greater than 20%, (2) low level in either steam generator, (3) low pressure in either steam generator, (4) all four reactor coolant pumps are tripped, (5) or HPI actuation on both A and B ESAS channels. The EFIC System also controls the atmospheric steam dump valves and isolates the affected OTSG in event of a high energy line break. Related EF flow and tank level instrument signals are processed through EFW Auxiliary Cabinets for operator information in the main control room and remote shutdown panel. The EF System pumps (EFP-1 and EFP-2) are supplied from either the Dedicated Emergency Feedwater Tank (preferred), Condensate Storage Tank or the condenser hotwell (both secondary sources), with the water boiled off by the steam generators during cooldown being vented to the atmosphere. To provide indication of proper system operation, alarms are provided indicating pump auto start, pump failure to auto start, and motor-driven pump discharge pressure low when its control switch is in the normal after start position. A detailed description of the EFIC System is discussed in Section 7.2.4.

Safety-grade power supplies and manual override of the EFIC system have been provided for valves that must operate to supply water to the steam generators (EFV-11, EFV-14, EFV-32, and EFV-33). The pumps tie into a common discharge header and the emergency feedwater enters the steam generators through emergency feedwater nozzles in the superheater region of the once through steam generators. The pump discharge pressure from either pump deliver the minimum emergency feedwater flow against a steam generator pressure existing during main steam safety valve operation. Thus, the EFW pumps can deliver the required flow under all reactor and steam generator transient conditions. Emergency feedwater flow is controlled by steam generator level and total flow via the EFIC System. Separate EFW flow transmitters (EF-23-FT through EF-26-FT) to each OTSG are provided. Redundant measurement is provided by OTSG level instrumentation transmitters. Each pump is protected by flow recirculation lines back to the Dedicated Emergency Feedwater (EFT-2) Tank. The motor-driven emergency feedwater pump (EFP-1) receives power from the 4160 volt Engineered Safeguards (ES) Bus 3A. The motor is tripped on an AC power failure at the bus. In the event of this condition, the motor-driven emergency feedwater pump will automatically start five seconds after ES bus is loaded onto the "A" train diesel generator. The turbine-driven emergency feedwater pump is independent of AC power and starts by opening the 125V DC motor operated steam admission valves (ASV-5 and ASV-204) when activated by the EFIC System.

Delete

CR3 005939

10.5.2.1 Motor-Driven Emergency Feedwater Pump

The Motor-Driven Emergency Feedwater Pump (EFP-1) is powered from emergency diesel generator 3A when normal AC power is not available. It has a rated capacity of 750 gpm at 1300 psig with a design recirculation flowrate of 200 gpm with the pump discharge closed. Motor, gear cooling, and lube oil cooling are supplied from the Nuclear Services Closed Cycle Cooling System (SW). Manual controls (both locally and on the main control board) provide the operator with the capability of shutting down the pump if the extra feedwater capacity is not required.

10.5.2.2 Turbine-Driven Emergency Feedwater Pump

The Turbine-Driven Emergency Feedwater Pump (EFP-2) provides redundant 100% capacity to the EF System and automatically starts on the same initiating signals as the motor-driven emergency feedwater pump (EFP-1). It has a rated capacity of 750 gpm at 1300 psig with a design recirculation flowrate of 200 gpm with the pump discharge closed. Manual controls (both locally and on the main control board) provide the operator with the capability of shutting down the pump is the extra feedwater capacity is not required. Lube oil cooling is supplied by the pump discharge water.

10.5.2.3 Emergency Feedwater Pump Turbine Drive

The Turbine Drive (EFTB-1) provides rapid start-up for EFP-2. The steam supply for EFTB-1 is obtained from both OTSGs through six-inch lines containing normally-open DC motor operated stop-check valves (MSV-55 and MSV-56) and check valves (MSV-186 and MSV-187) with one exception. The check valves and motor operated valves provide redundant isolation capability to preclude blowing down the operable steam generator in the event of main steam line or main feedwater line break. The exception is the piping between MSV-55/MSV-56 and the EF pump turbine. A rupture in this piping would result in simultaneous blowdown of both steam generators and would cause the turbine driven EF pump to be inoperable. B&W Analysis 86-1176873-00 (FPC Calc M-89-0056) shows this event to be within the plant design basis. Downstream of check valves (MSV-186 and MSV-187) the lines join to form a common supply to the pump turbine. Upstream of the turbine are redundant, normally closed DC motor operated valves (ASV-5 and ASV-204), ~~which are~~ SV-5 opens upon actuation from the EFIC System. A description of the controls for these valves is contained in Section 7.2.4.

Turbine exhaust is vented to the atmosphere.

Two relief valves (EFV-99 and EFV-100) are provided to protect the tank from overpressurization. The relief valve discharge is piped to the tank enclosure sump. Two vacuum breakers (EFV-97 and EFV-98) are provided to protect the tank from negative pressure due to EFW drawdown operations.

EFT-2 can also be aligned as a backup water source to the auxiliary feedwater pump (FWP-7), as discussed in Section 10.6.

10.5.3 Safety Evaluation

For information on the following items, refer to the indicated section:

1. Protection from the effects of pipe rupture - Section 5.4.4.

2. Seismic analysis - Section 5.1.2.

3. Flood design - Section 2.4.2.

4. Tornado design - Section 5.2.1.2.6.

The EF System is also single failure proof to isolate an affected OTSG for those loss of main feedwater events where such isolation is required (e.g., steamline break inside Primary Containment). Use of turbine-driven and motor-driven pumps results in diverse sources of power to the system.

The preferred source and various secondary (or alternate) sources of emergency feedwater as listed in Table 10-2 provide a redundant quality source of water to back up the dedicated emergency feedwater tank. The minimum quantity of usable condensate maintained in the Seismic Category I Dedicated Emergency Feedwater Tank is 150,000 gallons. This amount is sufficient to remove nuclear decay heat for a period of 18 hours at HOT STANDBY conditions, during which time the RCS metal and contained water would not be cooled except by ambient losses to the containment. If plant cooldown to 280°F is required, additional usable condensate (some of which is not seismically protected) is available at CR-3 and on the Crystal River site from fossil-fueled power generating stations as listed in Table 10-2 under "Other EFW Sources."

The EFW turbine-driven pump and turbine are a self-contained unit operating independently of secondary support systems. The bearings on the turbine and pump are lubricated by slinging oil from reservoirs near the bearings. Lube oil cooling is accomplished by heat transfer to the pumped fluid.

To ensure EFW flow in the event of a loss of all AC power, the turbine-driven pump train derives its motive power from the main steam lines through valves MSV-55 and MSV-56. Valves EFV-2 (locked closed), EFV-11, EFV-32, EFV-57, and EFV-58 and ASV-204 receive power from Battery 'A'. Valves EFV-1 (locked closed), EFV-14, EFV-33, EFV-55, EFV-56, MSV-55, MSV-56, and ASV-5 receive power from Battery 'B'.

Delete

A. GENERAL INSTALLATION INSTRUCTIONS

1. Accounting Information

   Charge all time and materials to NU9604120100.

2. Material

   None.

3. Interfaces

   None.

4. Precautions/Cautions

   Perform all work in accordance with FPC Accident Prevention Manual and applicable procedures.

B. SPECIFIC INSTALLATION INSTRUCTIONS

1. Open DC power source at DPDP-8A, Fuse #15, and Fuse #17, to deenergize control circuit and motor. (ASV-204)

2. In Relay Rack 3A, as shown on Interim drawing EC-210-624, disconnect and remove or spare in place internal wiring as shown in the following table:(ASV-204)

From	To
TB15-17	AR-5
TB15-18	AR-6
TB15-19	AR-3
TB15-20	AR-4

3. In Relay Rack 3A, as shown on Interim drawing EC-210-624, disconnect and spare in place conductors 1, 2, 3, and 4 of field cable ASE37 from TB15-17, 20, 18, and 19 respectively. (ASV-204)

4. In Terminal Box AS-9, as shown on Interim drawing EC-209-008, Sheet AS-06, disconnect and spare in place conductors 1, 2, 3, and 4 of field cable ASE-37 on TBA 12, 4, 6, and 3 respectively. (ASV-204)

5. Field to install in AS-9, as shown on Interim drawing EC-209-008, Sheet AS-06, #14AWG EQ safety related SIS Switchboard wire, FMIS #01270197, between TBA terminals 3 and 4. (ASV-204)

6. Reestablish power at DPDP-8A, Fuse #15 and Fuse #17.

CR3 005942

6.1 Field to wire mark conductor 4 of circuit ASE36 wire mark 11.

7. Examination/Testing/ISI Instructions

A. Examination Requirements

None.

B. Testing Requirements

Verify proper operation of ASV-204 from Main Control Board, PSA Section, and Local Control Station AS-9.

C. ISI Instructions

Operations to time the stroke of ASV-204, including remote position indication verification per SP-349B.

Florida Power

DESIGN DATA SHEET

Crystal River Unit 3

MAR NUMBER: 96-04-12-01

	YES	NO	APPLICABLE DESIGN INPUT REQUIREMENTS
1.	[X]	[ ]	Performance requirements such as capacity, rating, system output.
2.	[ ]	[X]	Codes, standards, and regulatory requirements including the applicable issue and/or addenda.
3.	[ ]	[X]	Design conditions such as pressure, temperature, fluid chemistry, and voltage.
4.	[ ]	[X]	Loads such as seismic, wind, thermal, and dynamic.
5.	[X]	[ ]	The modification will add, delete or modify essential equipment.
6.	[ ]	[X]	Environmental conditions anticipated during storage, construction, and operation such as pressure, temperature, humidity, corrosiveness, site elevation, wind direction, nuclear radiation, electromagnetic radiation and duration of exposure including 10CFR50.49 applicability.
7.	[ ]	[X]	Interface requirements including definition of the functional and physical interfaces involving structures, systems, and components.
8.	[ ]	[X]	Material requirements including such items as compatibility, electrical insulation properties, protective coating, corrosion resistance (including erosion-corrosion susceptibility), and the presence of asbestos or cobalt.
9.	[ ]	[X]	Mechanical requirements such as vibration, stress, shock, and reaction forces.
10.	[ ]	[X]	Structural requirements covering such items as equipment foundations and pipe supports.
11.	[ ]	[X]	Hydraulic requirements such as pump net positive suction heads (NPSH), allowable pressure drops, and allowable fluid velocities.
12.	[ ]	[X]	Chemistry requirements such as provisions for sampling and limitations on water chemistry.
13.	[X]	[ ]	Electrical requirements such as source of power, voltage, raceway requirements, electrical insulation, and motor requirements.
14.	[ ]	[X]	Layout and arrangement requirements, to include potential adverse affects of non-seismically qualified masonry walls.
15.	[X]	[ ]	Operational requirements under various conditions such as plant startup, normal plant operation, plant shutdown, plant emergency operation, special or infrequent operation, and system abnormal or emergency operation.
16.	[ ]	[X]	Instrumentation and control requirements including indicating instruments, controls, and alarms required for operation, testing, and maintenance. Other requirements such as the type of instrument, installed spares, range of measurement, and location of indication should also be included.

CR3 005944

Florida Power

DESIGN DATA SHEET

Crystal River Unit 3

MAR NUMBER

96-04-12-01

	YES	NO	APPLICABLE DESIGN INPUT REQUIREMENTS
7.	☐	☒	Access and administrative control requirements for plant security.
8.	☒	☐	Redundancy, diversity, and separation requirements of structures, systems, and components.
9.	☐	☒	Failure effects requirements of structures, systems, and components, including a definition of those events and accidents (e.g., flooding) which they must be designed to withstand.
0.	☐	☒	Test requirements including in-plant tests and the conditions under which they will be performed.
1.	☐	☒	Accessibility, maintenance, repair, and inservice inspection requirements for the plant including the conditions under which these will be performed.
2.	☒	☐	The modification will be within the following ALARA ZONE(S): Check as applicable: ☒ Zone III - White ☐ Zone II - Green ☐ Zone I - Yellow
3.	☐	☒	Transportability requirements such as size and shipping weight, limitations, I.C.C. regulations.
.	☐	☒	Fire protection or resistance requirements: (Check applicable letter(s)) ☐ a. Changes or additions/deletion of fire detection/suppression systems or equipment. ☐ b. Changes or additions to the plant configuration that change the effectiveness of existing fire detection/suppression systems.
.	☐	☒	Handling, storage, and shipping requirements. Include equipment protection requirements for in-place storage or system lay-up as applicable.
.	☐	☒	Other requirements to prevent undue risk to the health and safety of the public.
.	☐	☒	Materials, processes, parts, and equipment suitable for application.
.	☐	☒	Safety requirements for preventing personnel injury including such items as control of dangerous and hazardous materials, such as asbestos, escape provisions from enclosures, grounding of electrical systems, and the identification of appropriate caution statements for installation, removal, or use of hazardous and/or dangerous material. In addition, consideration must be included for the removal and disposition of radioactive waste.
.	☐	☒	Addition or relocation of safe shutdown equipment, systems, components, or circuits that require compliance with the separation criteria stated in 10CFR50, Appendix R.

CHECKLIST FOR EQ DOCUMENT REVIEW

Crystal River Unit 3

DOCUMENT NUMBER	DOCUMENT TITLE
MAR96-04-12-01	ASV-204, EFIC Automatic Opening Removal CBD 4/2/96

NOTE TO REVIEWER: Review documents for additions, deletions or changes to any original qualification (NEP-222) and changes to equipment or environmental parameters (normal or post accident).

1. Does the equipment being installed or modified require environmental qualification per 10CFR50.49?
   ☒ Yes ☐ No

2. If within the scope of 10CFR50.49, is the equipment: (Check NA if the answer to question 1 is "No.")
   a. a safety related electric equipment 1E that is relied upon to remain functional during and following design basis events?
      ☒ Yes ☐ No ☐ NA

   b. a nonsafety-related electric equipment whose failure under postulated environmental conditions could prevent satisfactory performance of equipment within the scope of 10CFR50.49? ☐ Yes ☒ No ☐ NA

   c. a post accident monitoring equipment (Reg. Guide 1.97, Cat I or II)? ☐ Yes ☐ No ☒ NA

3. If the document addressed one of the following, will it have an impact on equipment in the scope of 10CFR50.49:
   a. temperature, pressure, radiation or humidity at ANY time (normal, abnormal, DBE) during plant operation?
      ☐ Yes ☒ No ☐ NA
   b. chemical or building spray? ☐ Yes ☒ No ☐ NA
   c. equipment location? (moving to a different location, nearer or farther from pipeline, nearer or farther from wall)
      ☐ Yes ☒ No ☐ NA
   d. fire barrier, piping insulation shielding? ☐ Yes ☒ No ☐ NA
   e. DBE operating time change for equipment that is in or will result in the need to be added to the scope of 10CFR50.49? ☐ Yes ☒ No ☐ NA
   f. loop accuracy change? ☐ Yes ☒ No ☐ NA

4. Will any of the topics described in item 2 above force the requirements of 10CFR50.49 on electric equipment? (Identify which topics(s)) ☐ Yes ☒ No

5. Is equipment being removed from the scope of 10CFR50.49 with justification provided? ☐ Yes ☒ No

CR3 005946

DOCUMENT NUMBER
MAR-56-04-12-01

Is appropriate EQ documentation on file to support MAR/FCN installation for 10CFR50.49 requirements?

☑ Yes ☐ No ☐ NA

If no, identify the document which is required to support the installation change.

For all questions checked "yes," does the MAR/FCN adequately address 10CFR50.49 requirements?

☑ Yes ☐ No ☐ NA

DOCUMENT NUMBER

NCM/EQ to walk down AS9 term box to ensure EQ acceptability prior to MAR closure.

Florida Power CORPORATION

INTEROFFICE CORRESPONDENCE

Nuclear Engineering Design — NA1E — 240-3383
OFFICE — MAC — TELEPHONE

SUBJECT: Crystal River Unit 3
FSAR Change for MAR 96-04-12-01
File: DXREF MAR 96-04-12-01

TO: B. Gutherman

DATE: April 26, 1996
NED96-0272

MAR 96-04-12-01 removes the automatic opening capability on an EFIC actuation from ASV-204. This capability is being removed due to concerns raised from Operations about the ability of EFP-2 to run without any flow controls or indication, and the fact that the original capability was installed before the Diesel Generators were upgraded, so that EGDG-1A is now capable of handling a fully loaded EFP-1 during a LOCA, LOOP and a "B" Battery failure. Attached are 4 pages o..t of the FSAR with reference to the automatic opening that need to be change to delete the reference. Also attached is a copy of the 10CFR50.59 Evaluation from the MAR and the Regulatory/Environmental Review Form.

C. B. Doyel
Sr. Nuclear I&C Engineer

D. A. Shook
Nuclear Engineering Supervisor

cc: S. F. Ulm
Records Management

MAR 96-04-12-01

Attachment 10 Page 1 of 5

Florida Power CORPORATION

REGULATORY/ENVIRONMENTAL REVIEW

Crystal River Unit 3

DOCUMENT NUMBER
MAR96-04-12-01

REF NUCLEAR OPERATIONS DEPARTMENT PROCEDURE: NOD-11)

1. 10 CFR 50.54 Reviews

Does this modification or document revision reduce the commitment to, scope of, or effectiveness of what is described in any of the following plans/programs which are required by 10CFR50.54 or CR3 License conditions? If unable to determine, contact responsible person designated below.

Quality Program Description (FSAR Section 1.7) Contact: Director, Quality Programs	☐ Yes	☑ No
Licensed Operator Requalification Program (FSAR Section 12.2.3.4 and Training Department Procedure 203) Contact: Manager, Nuclear Operation Training	☐ Yes	☑ No
Physical Security Plan Contact: Nuclear Security Superintendent	☐ Yes	☑ No
Safeguards Contingency Plan Contact: Nuclear Security Superintendent	☐ Yes	☑ No
Radiological Emergency Response Plan Contact: Manager, Site Nuclear Services	☐ Yes	☑ No
Fire Protection Plan Contact: Manager, Site Nuclear Services	☐ Yes	☑ No

If any answers are "Yes"

a. Contact appropriate responsible person identified above to perform on the evaluation and attach.	☐ Evaluation Complete	☑ NA
b. Contact the Manager, Nuclear Licensing for NRC submittal.	☐ Submitted	☑ NA
c. NRC approval received prior to change and attach.	☐ Complete	☑ NA

2. Tech Spec Program Reviews

Does this modification or document revision change what is described in any of the following programs/plans which are required by Tech Spec 5.6.2?

Offsite Dose Calculation Manual Contact: Manager, Nuclear Chemistry	☐ Yes	☑ No
Primary Coolant Sources Outside Containment Program Contact: Manager, Nuclear Plant Technical Support	☐ Yes	☑ No
Component Cyclic or Transient Limit Program Contact: Supervisor, Operations and Engineering Support	☐ Yes	☑ No
Post Accident Sampling Program Contact: Manager, Nuclear Chemistry	☐ Yes	☑ No
Containment Tendon Surveillance Program Contact: Manager, Nuclear Plant Technical Support	☐ Yes	☑ No

CR3 005949

Inservice Inspection Program
Contact: Manager, Nuclear Plant Technical Support — ☐ Yes ☑ No

Inservice Testing Program
Contact: Manager, Nuclear Plant Technical Support — ☐ Yes ☑ No

Secondary Water Chemistry Program
Contact: Manager, Nuclear Chemistry — ☐ Yes ☑ No

Ventilation Filter Testing Program
Contact: Manager, Nuclear Plant Technical Support — ☐ Yes ☑ No

Explosive Gas & Storage Tank Radioactivity Monitoring Program
Contact: Manager, Nuclear Chemistry — ☐ Yes ☑ No

Diesel Fuel Oil Testing Program
Contact: Manager, Nuclear Plant Technical Support — ☐ Yes ☑ No

Core Operating Limits Report
Contact: Manager, Safety Analysis & Fuel Management — ☐ Yes ☑ No

Reactor Coolant System/Pressure Temperature Limits Report
Contact: Manager, Safety Analysis & Fuel Management — ☐ Yes ☑ No

If any are "yes:"

a. Contact appropriate responsible person identified above to perform the evaluation and attach. — ☐ Evaluation Complete ☑ N/A

3. Environmental Protection Plan Review

a. Could this change affect the environment in a non-radiological way? That is, could this change affect:

(1) The environment in a non-radiological way? — ☐ Yes ☑ No
(2) An increase in licensed power level? — ☐ Yes ☑ No
(3) An increase in non-radiological effluents (Volume or chemicals - liquid or gas)? — ☐ Yes ☑ No
(4) A change in Circulating or Raw Water flows? — ☐ Yes ☑ No
(5) Dredging, filling or new construction on previously undisturbed areas? — ☐ Yes ☑ No

If "yes," contact the Manager, Nuclear Licensing to perform evaluation and attach. — ☐ Complete ☑ N/A

b. Does the modification affect the National Pollutant Discharge Elimination System permit or associated state permit? — ☐ Yes ☑ No

If "yes," contact the Supervisor, Water Programs Section to evaluate reporting requirements. Notification of such action must also be made to the Manager, Nuclear Licensing. — ☐ Submitted ☑ N/A

EPA and/or FDER approval received, if needed, and attached. — ☐ Yes ☐ No ☑ N/A

If any answer to 3a or 3b is "yes," a copy of the design document must be sent to the NGRC.

Is NGRC copy required? — ☐ Yes ☑ No

CR3 005950

Florida Power CORPORATION

REGULATORY/ENVIRONMENTAL REVIEW

Crystal River Unit 3

DOCUMENT NUMBER
MAR96-04-12-01

REF NUCLEAR OPERATIONS DEPARTMENT PROCEDURE: NOD-11)

4. Review for change to Radioactive Waste System (10 CFR 50.34a and Appendix I)

Is this a change to a radioactive waste system (liquid, gaseous or solid) that could result in an increase of radioactive material released to the environment?

☐ Yes ☑ No ☐ N/A

If "yes," submit change to Manager, Nuclear Licensing to evaluate the reporting requirements. This is for reporting evaluation only; the response is not part of the modification or document approval process for the change.

☐ Submitted ☑ N/A

5. Review for Implementation of Regulatory Guide 1.33 Activity

Does this document implement any activity in Appendix A of Regulatory Guide 1.33 (NOD-12) or any other procedures required by Technical Specification 5.6.1.1?

☐ Yes ☑ No ☐ N/A

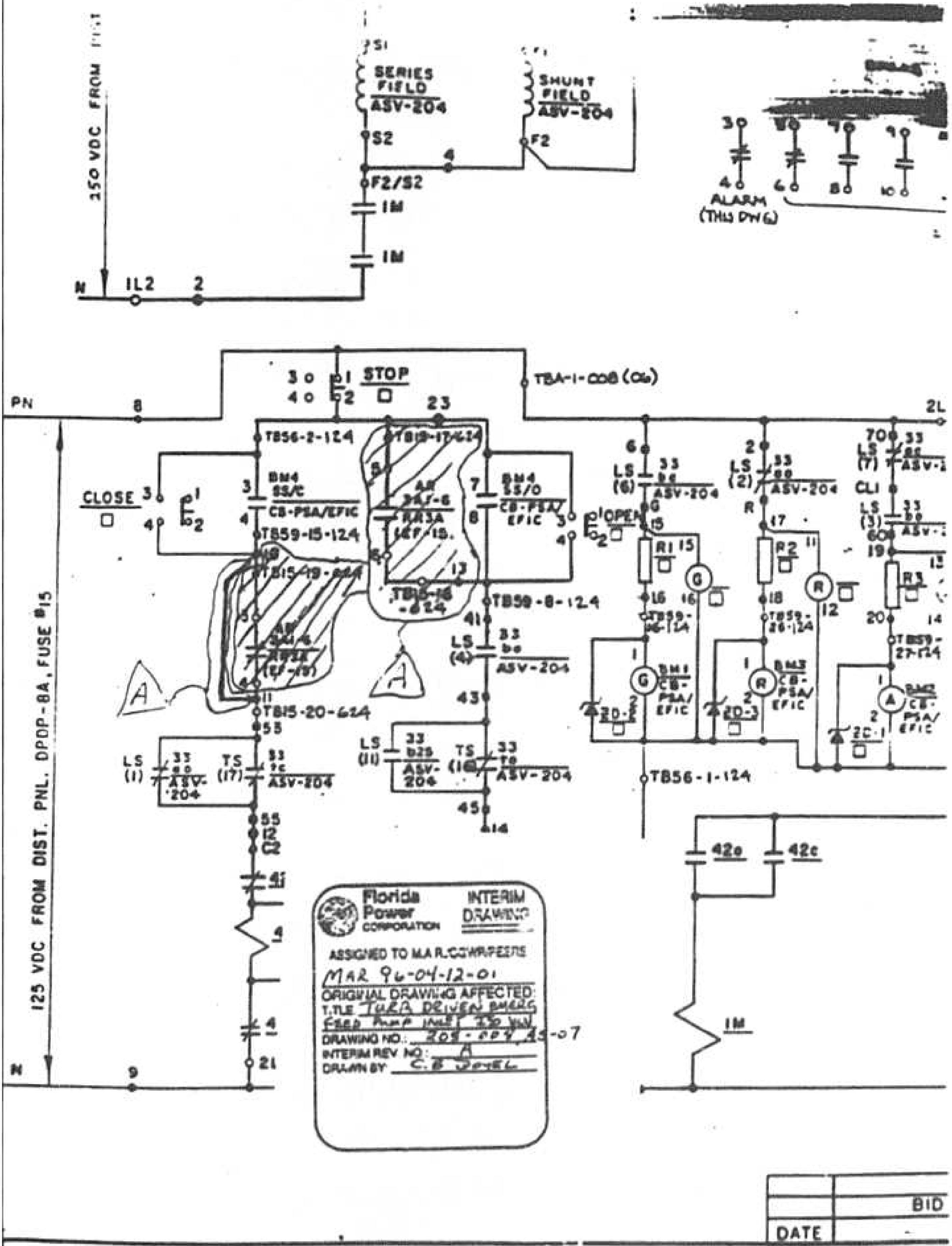
If "Yes," ensure review and approval cycle includes requirements of the Quality Programs Plan

☐ Included ☑ N/A

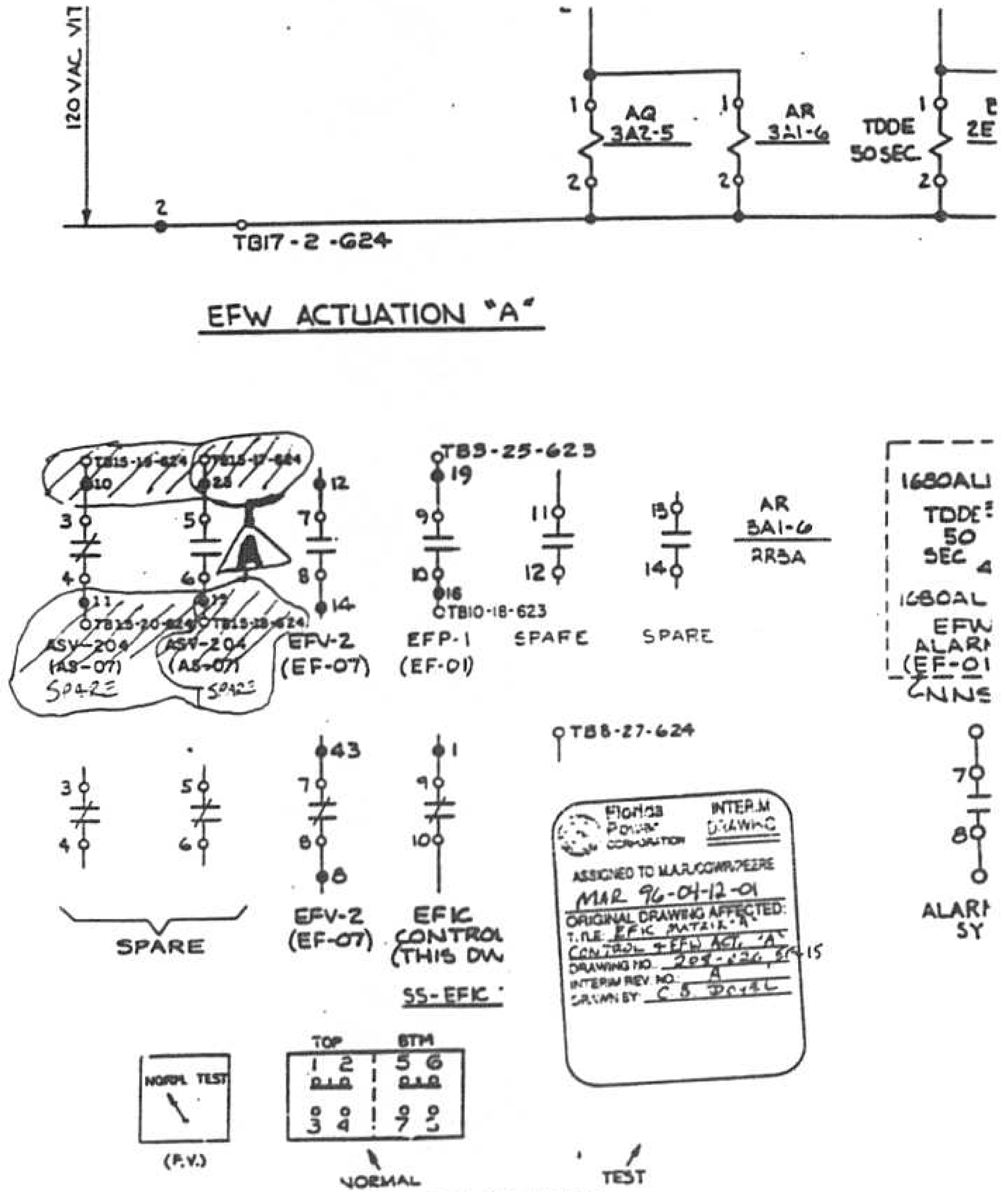
IF ALL OF ABOVE ARE CHECKED NO, NO FURTHER REVIEWS ARE REQUIRED.

PREPARED BY C. B. Doyel DATE 4/26/96	APPROVED BY D A Shook DATE 4/26/96
Printed Name of Preparer C. B DOYEL	Printed Name of Approver D.A. SHOOK

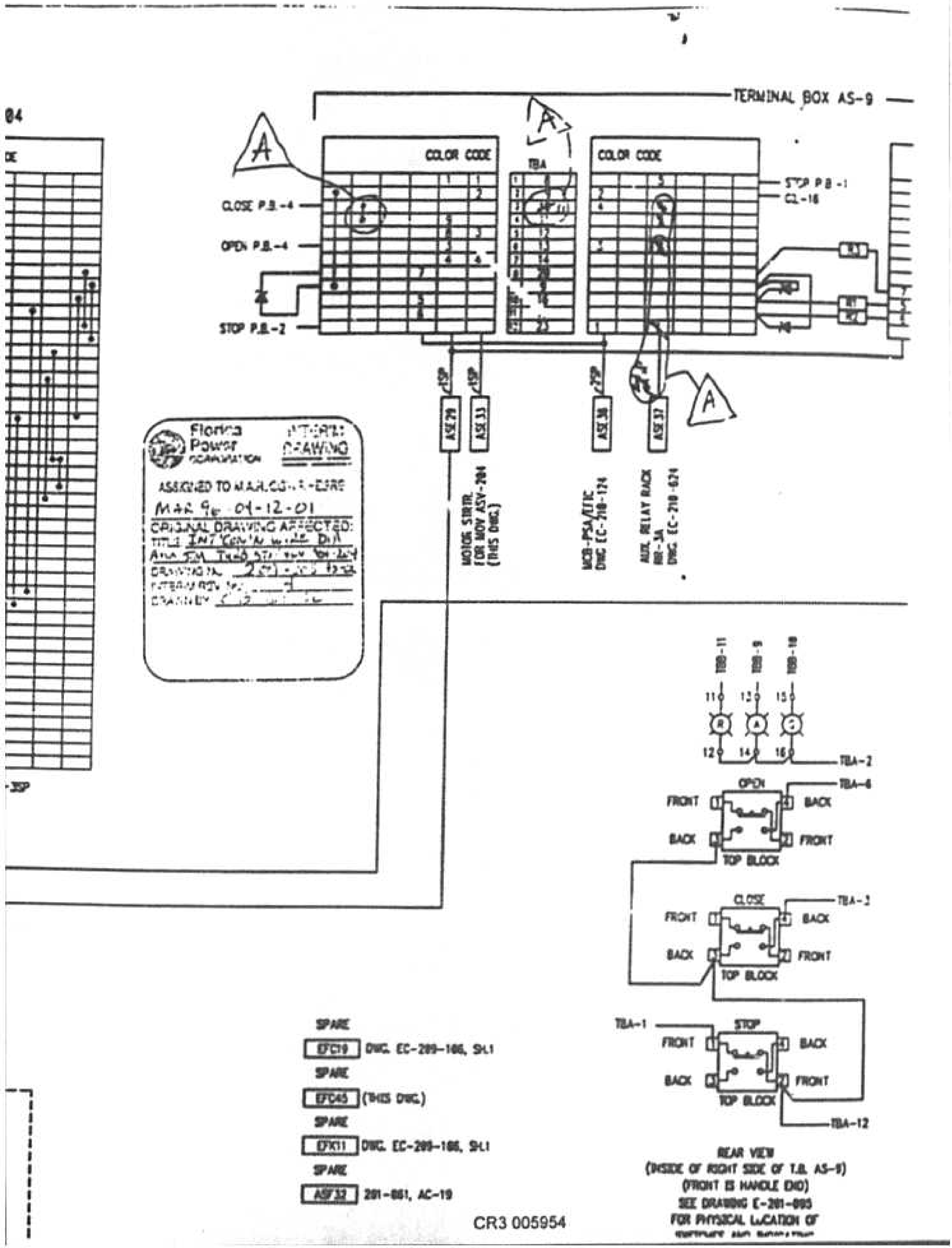
CR3 005951

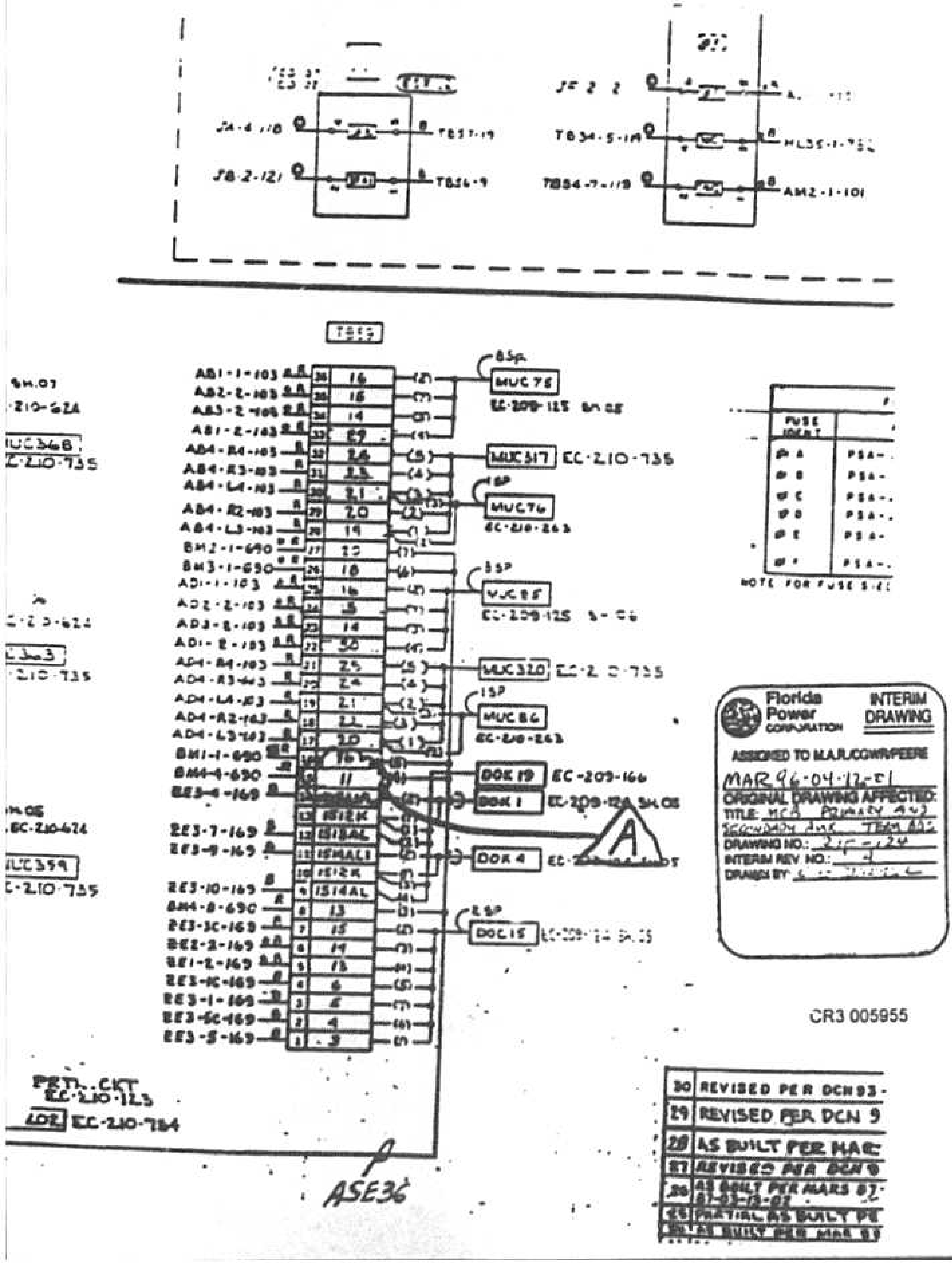


250 VDC FROM
SERIES FIELD ASV-204
SHUNT FIELD ASV-204
S1
S2
F1
F2
F2/S2
1M
N
1L2
ALARM (THIS DWG)
PN
STOP
TBA-1-008 (06)
2L
CLOSE
OPEN
TB56-2-124
TB59-15-124
TB15-19-624
TB15-20-624
TB15-17-624
TB15-18-624
TB59-8-124
TB56-1-124
BM4 SS/C CB-PSA/EFIC
BM4 SS/O CB-PSA/EFIC
LS (6) 33 bc ASV-204
LS (2) 33 aa ASV-204
LS (3) 33 bb ASV-204
LS (4) 33 bc ASV-204
LS (11) 33 b25 ASV-204
LS (1) 33 aa ASV-204
TS (17) 33 tc ASV-204
TS (16) 33 tb ASV-204
R1
R2
R3
BM1 CB-PSA/EFIC
BM3 CB-PSA/EFIC
BM2 CB-PSA/EFIC
ZD-2
ZD-3
ZD-1
42a
42c
125 VDC FROM DIST. PNL. DPDP-8A, FUSE #15
A
Florida Power Corporation
INTERIM DRAWING
ASSIGNED TO M.A.R./CONVR/EERS
MAR 96-04-12-01
ORIGINAL DRAWING AFFECTED:
TITLE TURB DRIVEN EMERG FEED PUMP INLET ISO VALV
DRAWING NO. 208-004 AS-07
INTERIM REV NO. A
DRAWN BY C.B. JOYEL
BID
DATE

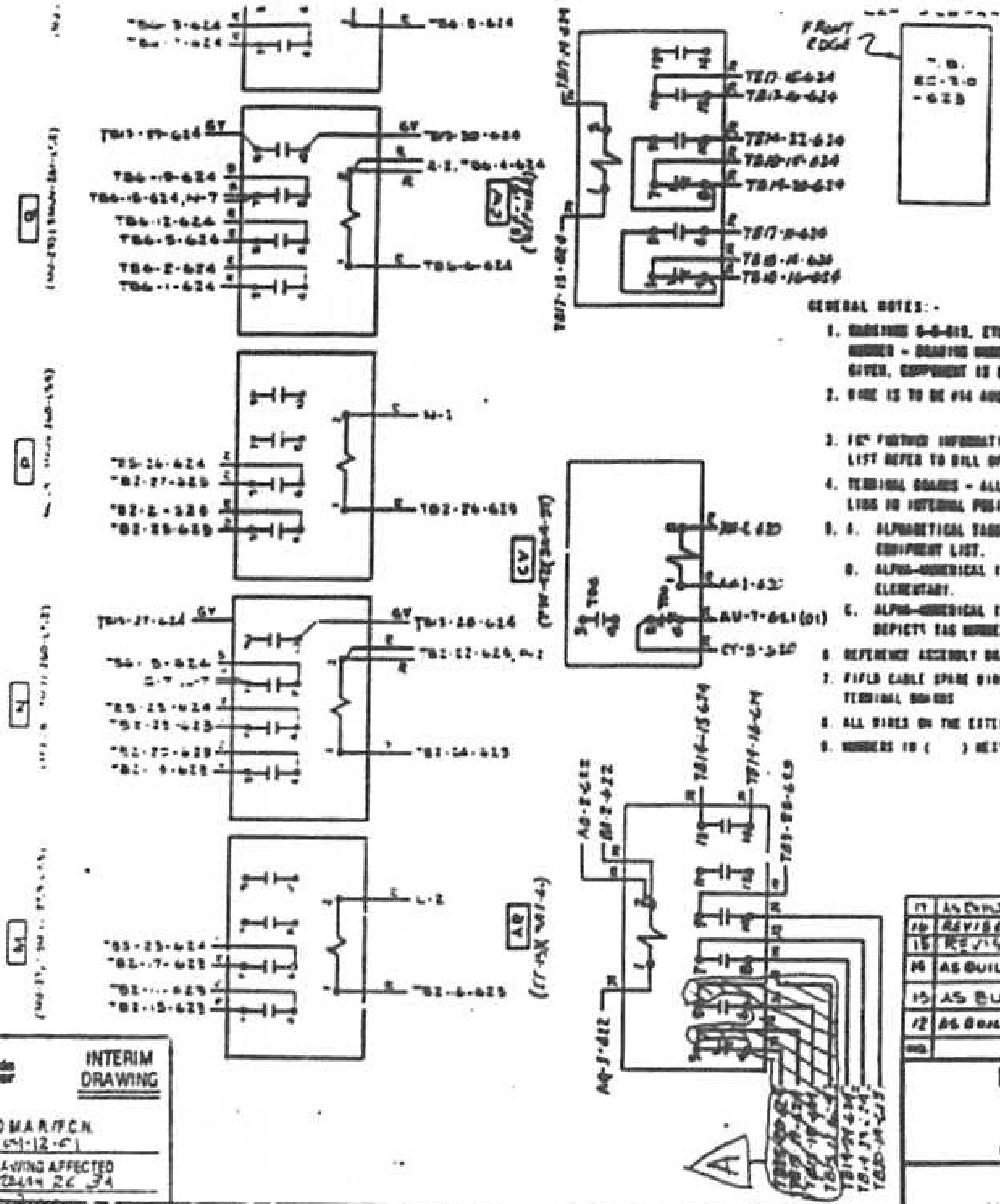


G.E. TYPE CR2940 CYLINDER LOCK SELECTOR
SWITCH MODEL NO. CR2940UN20CDW/CR2940U208 DOUBLE
BLOCK MAINTAINED CONTACTS. KEY REMOVEABLE IN "NORMAL" POSITION ONLY


TERMINAL BOX AS-9
COLOR CODE
TBA
CLOSE P.B.-4
OPEN P.B.-4
STOP P.B.-2
STOP P.B.-1
CL-16
ASE29
ASE33
ASE36
ASE37
MOTOR STRTR. FOR MOV ASV-204 (THIS DWG.)
MCB-PSA/ETC. DWG EC-210-124
AUX. RELAY RACK RR-3A DWG EC-210-624
Florida Power CORPORATION
INTERIM DRAWING
ASSIGNED TO MAR/CONR-EDRS
MAR 96-04-12-01
ORIGINAL DRAWING AFFECTED:
DRAWING NO.
DRAWN BY
TBA-2
TBA-4
TBA-3
TBA-1
TBA-12
OPEN
CLOSE
STOP
FRONT
BACK
TOP BLOCK
SPARE
EFC19 DWG. EC-209-166, SH.1
EFC45 (THIS DWG.)
EFX11 DWG. EC-209-166, SH.1
ASF32 201-061, AC-19
REAR VIEW
(INSIDE OF RIGHT SIDE OF T.B. AS-9)
(FRONT IS HANDLE END)
SEE DRAWING E-201-095
FOR PHYSICAL LOCATION OF


TB59
MUC 75
EC-209-125 SH 05
MUC 317
EC-210-735
MUC 76
EC-210-263
MUC 320
EC-210-735
MUC 86
EC-210-263
DOK 19
EC-209-166
DOK 1
DOK 4
DOC 15
MUC 368
EC-210-735
MUC 359
EC-210-735
PETL. CKT
EC-210-123
402 EC-210-784
A
ASE36
NOTE FOR FUSE
Florida Power CORPORATION
INTERIM DRAWING
ASSIGNED TO M.A.R./CONV/PEERE
MAR 96-04-12-01
ORIGINAL DRAWING AFFECTED:
TITLE:
DRAWING NO.:
INTERIM REV. NO.: 4
DRAWN BY:
30 REVISED PER DCN 93-
29 REVISED PER DCN 9
28 AS BUILT PER MAR
27 REVISED PER DCN
26 AS BUILT PER MARS 87-
25 PARTIAL AS BUILT PE
24 AS BUILT PER MAR 8

CR3 005955



GENERAL NOTES:-

1. MARKINGS ... ETC. NUMBER - DRAWING NUMBER GIVEN, COMPONENT IS LO
2. WIRE IS TO BE #14 AWG.
3. FOR FURTHER INFORMATIC LIST REFER TO BILL OF
4. TERMINAL BOARDS - ALL LUGS IN INTERNAL POSIT
5. A. ALPHABETICAL TAGS EQUIPMENT LIST.
   B. ALPHA-NUMERICAL IS ELEMENTARY.
   C. ALPHA-NUMERICAL IS DEPICTS TAG NUMBER
6. REFERENCE ASSEMBLY DRA
7. FIELD CABLE SPARE WIRE TERMINAL BOARDS
8. ALL WIRES ON THE EXTER
9. NUMBERS IN ( ) NEXT

Florida Power

INTERIM DRAWING

ASSIGNED TO M.A.R./F.C.N. MAR 96-04-12-01

ORIGINAL DRAWING AFFECTED

TITLE ...

DRAWING NO 210-619

INTERIM REV. NO A

DRAWN BY C.S. Deitel

~~CHECKED BY~~

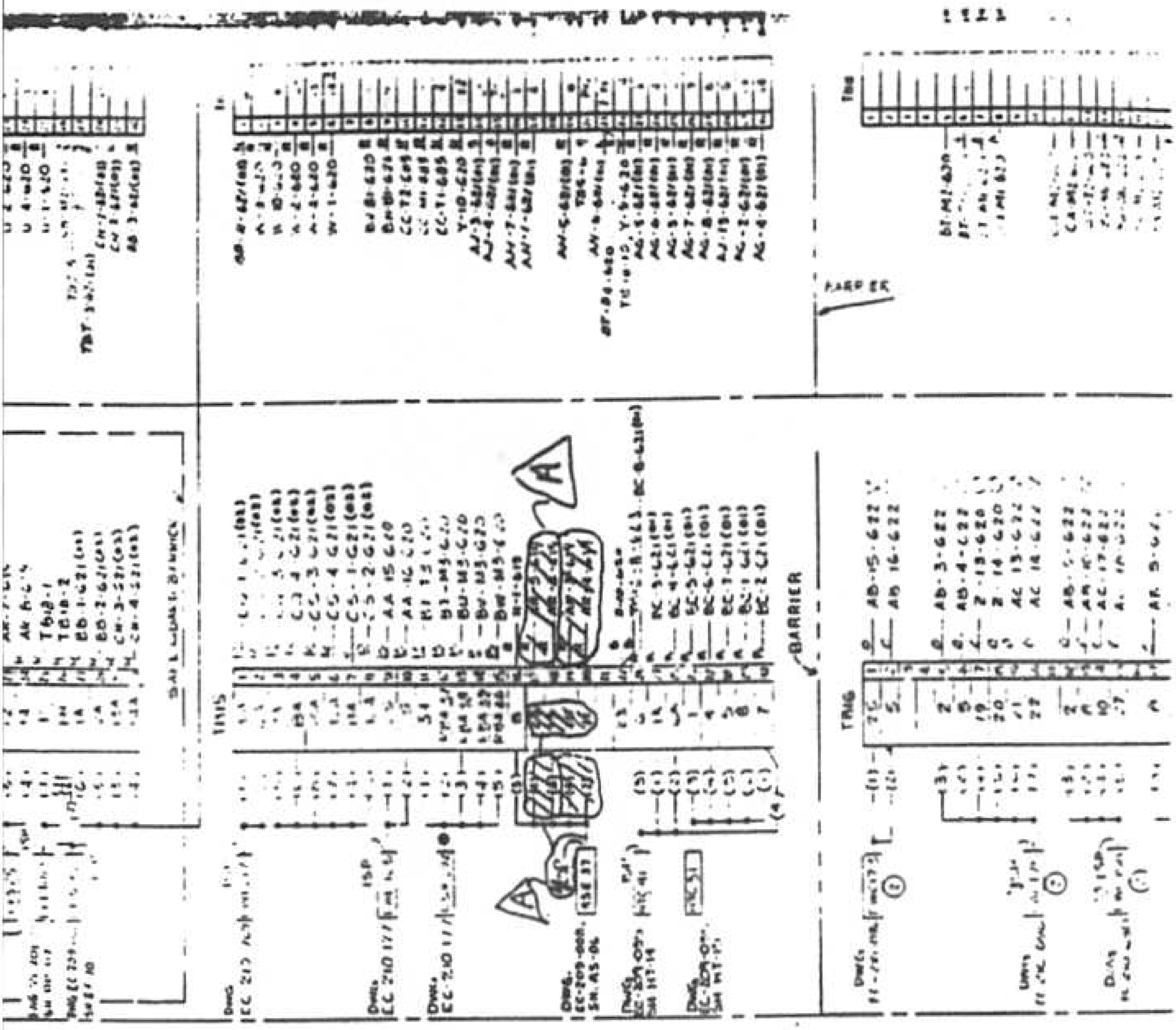
~~APPROVED BY~~

ISSUE DATE

(SEE GENERAL NOTE #2)

17	AS BUILT
16	REVISE
15	REVIS
14	AS BUIL
13	AS BU
12	AS BUIL

DATE	RELEASED FOR	ENGR
	CONSTRUCTION	
	LIMITED CONSTRUCTION: AS NOTED	
	PRELIMINARY NOT FOR CONSTRUCTION	
	BIDDING PURPOSES ONLY	



RIGHT SIDE OF PANEL

Florida Power Corporation — INTERIM DRAWING

ASSIGNED TO M.A.R. COORD/PERFE

MAR 96-04-12-01

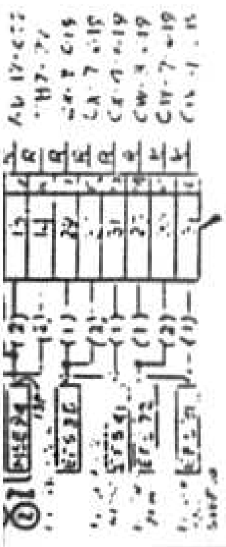
ORIGINAL DRAWING AFFECTED:

TITLE: Aux Relay Rack 3A - Terminal Boards

DRAWING NO.: 210-620

INTERIM REV. NO.: A

DRAWN BY: C. B. Doyel



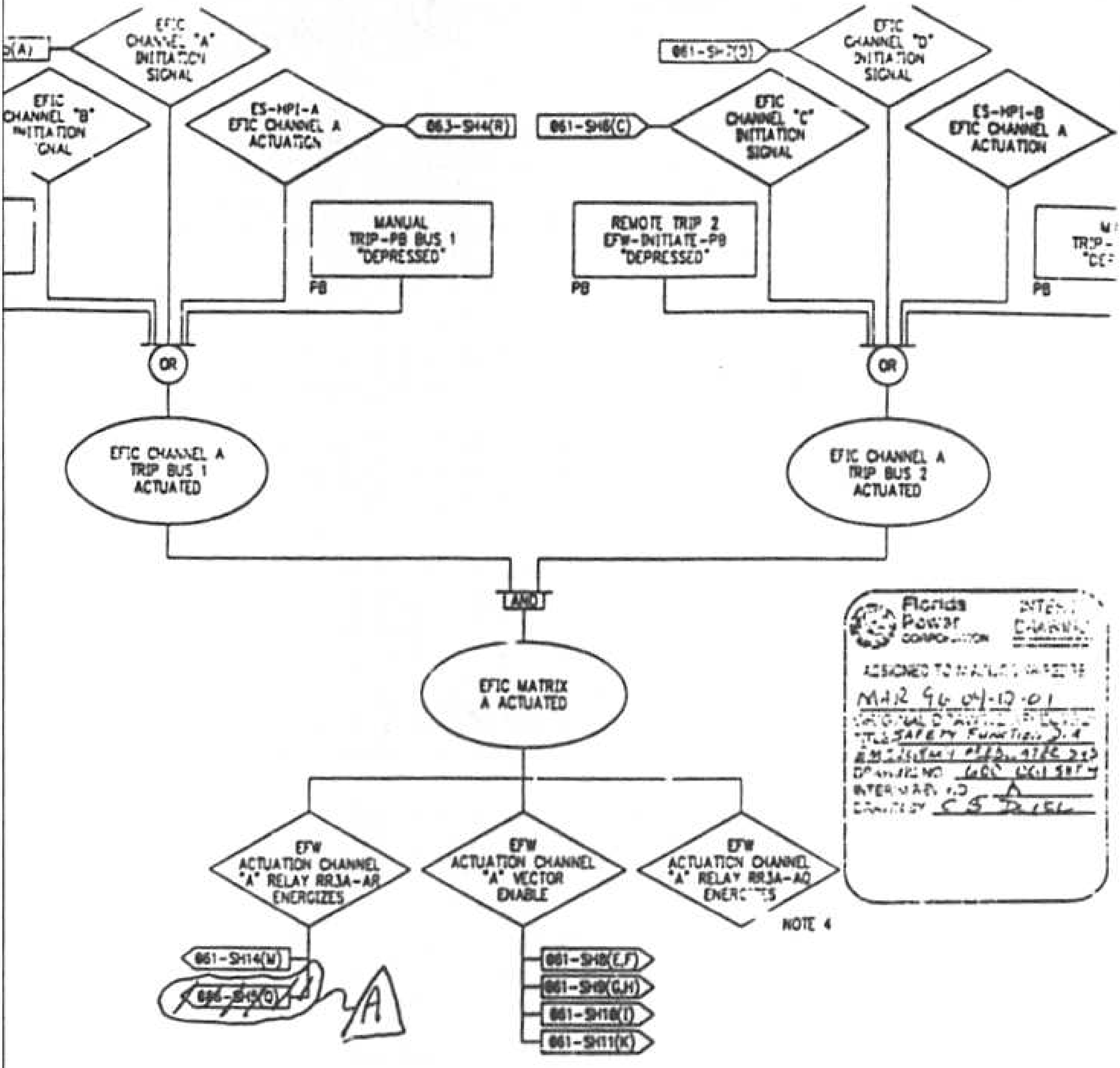
TERM. BLOCKS TB17 & TB10 ARE PHYSICALLY LOCATED BETWEEN ITEMS [W] & [AA] ON DWG. EC-210-620

CR3 005957

	CONSTRUCTION
	LIMITED CONSTRUCTION: AS NOTED
	PRELIMINARY NOT FOR CONSTRUCTION
	BIDDING PURPOSES ONLY

44	REVISED
40	REVISED

42



Florida Power CORPORATION

ASSIGNED TO MANUAL: MAR 96-04-10-01

ORIGINAL DRAWING ... SAFETY FUNCTION ... EMERGENCY FEEDWATER SYS

DRAWING NO ... INTERIM REV. NO A

DRAWN BY C S ...

KEY	PRESSURE TRANSMITTER	BISTABLE	KEY	LEVEL TRANSMITTER	BISTABLE	KEY	NUCLEAR INSTRUMENT	BISTABLE
A	MS-106-PT	MS-106-PS1	C	SP-025-LT	SP-025- 3	E	NI-5-A13	NI-5-A19
B	MS-110-PT	MS-110-PS1	D	SP-029-LT	SP-029-LS			

4 5 6

3. VALVES THAT ARE NORMALLY IN THEIR SAFETY POSITION BUT WHICH RECEIVE A CONFIRMATORY ACTUATION SIGNAL ARE SHOWN AS ACTIVE, BUT ARE ASSUMED TO BE IN THEIR NORMAL POSITION FOR MODELING.

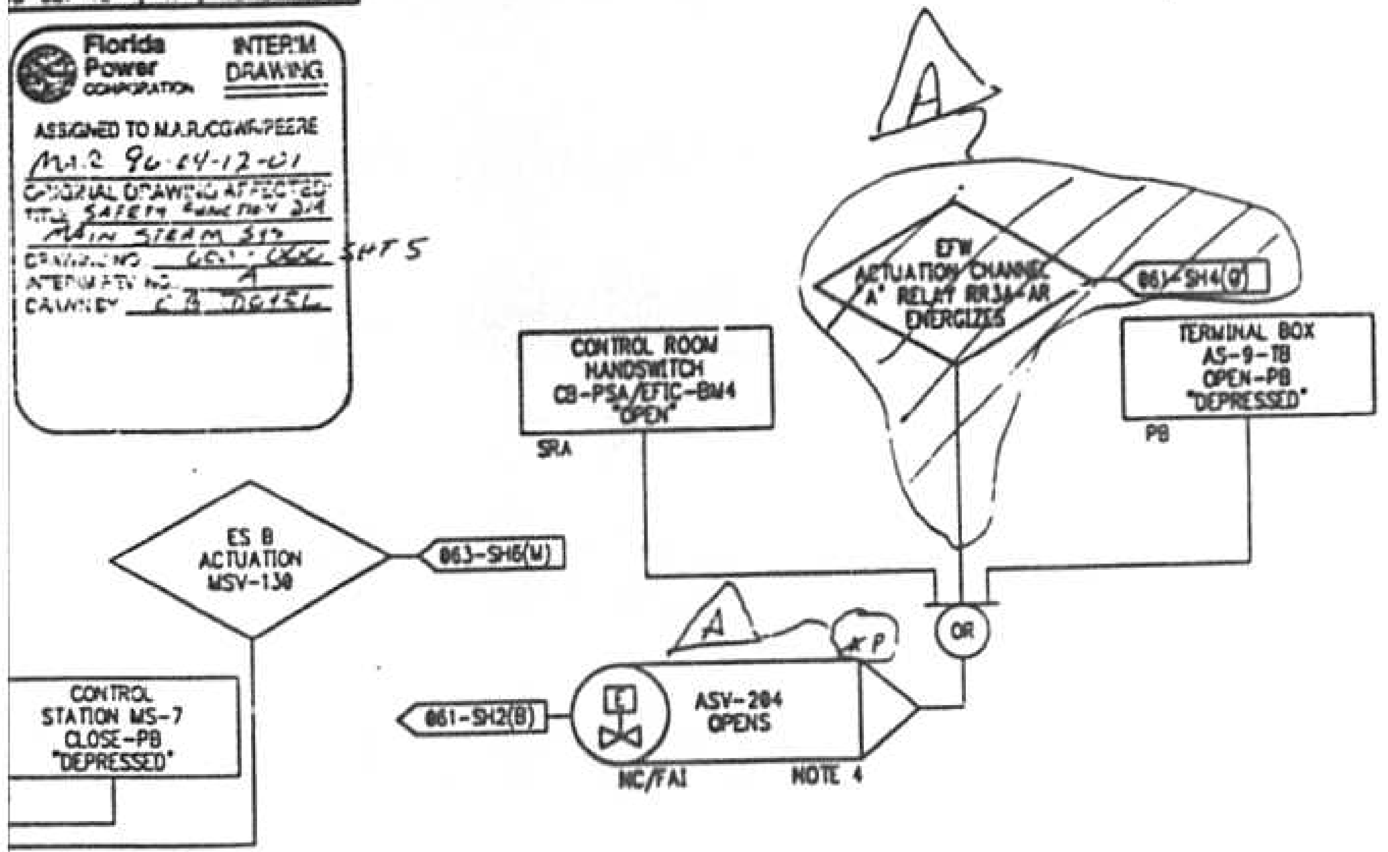
4. THESE VALVES ARE DEPICTED ON THE COMPOSITE FOR THE EMERGENCY FEEDWATER (EF) SYSTEM SFD AS THEY SUPPORT THE OPERATION OF THE EF PUMP TURBINE. HOWEVER, THEIR DETAILS ARE INCLUDED IN THIS STEAM SYSTEM DRAWING SERIES.

IS DETAIL IS TYPICAL OF THE VALVES IN TABLE 3.

MPERATURE ELEMENT	KEY	TEMPERATURE ELEMENT
B-327-TE1	E	TB-326-TE1
B-327-TE1	F	TB-326-TE1
B-327-TE1	G	TB-326-TE1
B-327-TE1	H	TB-326-TE1

Florida Power CORPORATION — INTERIM DRAWING

ASSIGNED TO M.A.R./CGWR/PEERE
MAR 96-04-12-01
ORIGINAL DRAWING AFFECTED:
TITLE SAFETY FUNCTION DIA MAIN STEAM SYS
DRAWING NO. ... SHT 5
INTERIM REV NO. A
DRAWN BY C B DOISL



ALVE
SV
ED

PICAL OF THE VALVES IN TABLE 6.

NO.	REVISIONS	BY	CHECKED	APPROVED	DA
2.	AS BUILT PER MAR 97-10-09-01A	KLK			
1.	ISSUED PER DCN-90-337	KRL	KRC	DAS	10/

FLORIDA POWER CORPORATION
ST. PETERSBURG, FLORIDA
CRYSTAL RIVER PLANT
UNIT NO.3 — 855,000 KW

SAFETY FUNCTION DIAGRAM
MAIN STEAM SYSTEM
DETAILS

DRAWN BY: — DATE:
CHECKED BY: — DATE:
APPROVED BY: — DATE:

ABB

B-KS2
B-KS2

CR3 005959

The trip module located in the "A" cabinet starts the "A" train motor-driven EFW pump and the "B" train turbine-driven EFW pump. The trip module located in the "B" cabinet starts only the "B" train turbine-driven EFW pump. The starting of both EFW pumps on "A" train EFIC actuation is necessary to assure that the turbine driven pump will be operable in the event of a failure of the ES "B" 250/125V DC system coincident with loss of off-site power and a ES actuation. Under this scenario EFP-2 will be relied upon to share the emergency feedwater load with the motor driven emergency feedwater pump in order to decrease the electrical load on diesel generator EDG-3A. In order to cause a train of EFW to initiate, both Trip Bus 1 and Trip Bus 2 relays must be energized. For both trip bus relays to be energized, at least 2 channels of EFIC must issue initiate commands. Any 2 channels of EFIC issuing an "initiate" command will result in the actuation of at least one train of EFW. Since all four EFIC channels monitor the same parameters, initiate commands occur at the same time causing both EFW trains to start.

Delete

After EFW has been actuated, flow will be controlled by modulating the EFW control valves (EFV-55, EFV-56, EFV-57, and EFV-58) to maintain OTSG level at the appropriate setpoint. If the control valves hand/auto (H/A) stations are not in the Auto position, the EFW actuation causes them to be placed in Auto. The setpoint to be maintained is automatically determined by the control modules located in the "A" and "B" EFIC cabinets. The modules in the "A" cabinet controls flow from the "A" EFW train to the A and B OTSG and the modules in the "B" cabinet controls flow from the "B" EFW train to the A and B OTSG.

An occurrence of an EFW actuation with at least one RC pump running and one OTSG with an inventory level below 30 inches causes the EFW system to be on EFW flow control. The EFW flow control is to limit flow above 600 gpm by biasing close the control valve(s) 50% and then applying an additional signal to proportionally close the valves to control flow until reaching the OTSG level control setpoint. At this control point, the OTSG level control positions the valve(s) and the flow control is reduced to zero when flow conditions are below 600 gpm. This same flow control scheme limits the amount of EFW flow to either one or both depressurized OTSG's resulting from a main steam line break or main feedwater line break. In both events, the control valve(s) will be partially closed and proportionally controlled up to 80% valve closure travel whenever the OTSG pressure is below 600 psig.

If EFW is actuated and at least 1 RC pump is running, the system will control OTSG level at approximately 30 inches. If no RC pumps are running, the control module will select the NATURAL CIRCULATION setpoint on the EFIC "High Range" | instrument. The control modules also have a ramp rate function that controls the rate of OTSG level increase between approximately 2 and 8 inches per minute. The parameter that determines the actual ramp rate is OTSG outlet pressure. When pressure is at the low end of the control band (about 800 psig), the lower ramp rate is selected.

Florida Power Corporation

INTEROFFICE CORRESPONDENCE

Nuclear Engineering Design — NA1E — 240-3383
OFFICE — MAC — TELEPHONE

SUBJECT: Crystal River Unit 3
FSAR Change for MAR 96-04-12-01
File: DXREF MAR 96-04-12-01

TO: B. Gutherman

DATE: April 26, 1996
NED96-0272

MAR 96-04-12-01 removes the automatic opening capability on an EFIC actuation from ASV-204. This capability is being removed due to concerns raised from Operations about the ability of EFP-2 to run without any flow controls or indication, and the fact that the original capability was installed before the Diesel Generators were upgraded, so that EGDG-1A is now capable of handling a fully loaded EFP-1 during a LOCA, LOOP and a "B" Battery failure. Attached are 4 pages out of the FSAR with reference to the automatic opening that need to be change to delete the reference. Also attached is a copy of the 10CFR50.59 Evaluation from the MAR and the Regulatory/Environmental Review Form.

C. B. Doyel
Sr. Nuclear I&C Engineer

D. A. Shook
Nuclear Engineering Supervisor

cc: S. F. Ulm
Records Management

CR3 005961

The trip module located in the "A" cabinet starts the "A" train motor-driven EFW pump and the "B" train turbine-driven EFW pump. The trip module located in the "B" cabinet starts only the "B" train turbine-driven EFW pump. The starting of both EFW pumps on "A" train EFIC actuation is necessary to assure that the turbine driven pump will be operable in the event of a failure of the ES "B" 250/125V DC system coincident with loss of off-site power and a ES actuation. Under this scenario EFP-2 will be relied upon to share the emergency feedwater load with the motor driven emergency feedwater pump in order to decrease the electrical load on diesel generator EDG-3A. In order to cause a train of EFW to initiate, both Trip Bus 1 and Trip Bus 2 relays must be energized. For both trip bus relays to be energized, at least 2 channels of EFIC must issue initiate commands. Any 2 channels of EFIC issuing an "initiate" command will result in the actuation of at least one train of EFW. Since all four EFIC channels monitor the same parameters, initiate commands occur at the same time causing both EFW trains to start.

Delete

After EFW has been actuated, flow will be controlled by modulating the EFW control valves (EFV-55, EFV-56, EFV-57, and EFV-58) to maintain OTSG level at the appropriate setpoint. If the control valves hand/auto (H/A) stations are not in the Auto position, the EFW actuation causes them to be placed in Auto. The setpoint to be maintained is automatically determined by the control modules located in the "A" and "B" EFIC cabinets. The modules in the "A" cabinet controls flow from the "A" EFW train to the A and B OTSG and the modules in the "B" cabinet controls flow from the "B" EFW train to the A and B OTSG.

An occurrence of an EFW actuation with at least one RC pump running and one OTSG with an inventory level below 30 inches causes the EFW system to be on EFW flow control. The EFW flow control is to limit flow above 500 gpm by biasing close the control valve(s) 50% and then applying an additional signal to proportionally close the valves to control flow until reaching the OTSG level control setpoint. At this control point, the OTSG level control positions the valve(s) and the flow control is reduced to zero when flow conditions are below 500 gpm. This same flow control schema limits the amount of EFW flow to either one or both depressurized OTSG's resulting from a main steam line break or main feedwater line break. In both events, the control valve(s) will be partially closed and proportionally controlled up to 80% valve closure travel whenever the OTSG pressure is below 600 psig.

If EFW is actuated and at least 1 RC pump is running, the system will control OTSG level at approximately 30 inches. If no RC pumps are running, the control module will select the NATURAL CIRCULATION setpoint on the EFIC "High Range" instrument. The control modules also have a ramp rate function that controls the rate of OTSG level increase between approximately 2 and 8 inches per minute. The parameter that determines the actual ramp rate is OTSG outlet pressure. When pressure is at the low end of the control band (about 800 psig), the lower ramp rate is selected.

10.5.2 System Description

The Emergency Feedwater System is required to assure an adequate emergency supply to the OTSGs to remove reactor decay heat until suitable conditions are attained to start the Decay Heat Removal System. The EF System consists primarily of redundant feedwater pumps powered from diverse power sources and the Emergency Feedwater Initiation and Control System (EFIC) which are required to function during periods when main feedwater supply and/or the electrical supply to the vital main feedwater auxiliaries has been lost. The EFIC System, a four channel safety grade control system, activates EF System components upon (1) both main feedwater pumps are tripped and reactor power is greater than 20%, (2) low level in either steam generator, (3) low pressure in either steam generator, (4) all four reactor coolant pumps are tripped, (5) or HPI actuation on both A and B ESAS channels. The EFIC System also controls the atmospheric steam dump valves and isolates the affected OTSG in event of a high energy line break. Related EF flow and tank level instrument signals are processed through EFW Auxiliary Cabinets for operator information in the main control room and remote shutdown panel. The EF System pumps (EFP-1 and EFP-2) are supplied from either the Dedicated Emergency Feedwater Tank (preferred), Condensate Storage Tank or the condenser hotwell (both secondary sources), with the water boiled off by the steam generators during cooldown being vented to the atmosphere. To provide indication of proper system operation, alarms are provided indicating pump auto start, pump failure to auto start, and motor-driven pump discharge pressure low when its control switch is in the normal after start position. A detailed description of the EFIC System is discussed in Section 7.2.4.

Safety-grade power supplies and manual override of the EFIC system have been provided for valves that must operate to supply water to the steam generators (EFV-11, EFV-14, EFV-32, and EFV-33). The pumps tie into a common discharge header and the emergency feedwater enters the steam generators through emergency feedwater nozzles in the superheater region of the once through steam generators. The pump discharge pressure from either pump deliver the minimum emergency feedwater flow against a steam generator pressure existing during main steam safety valve operation. Thus, the EFW pumps can deliver the required flow under all reactor and steam generator transient conditions. Emergency feedwater flow is controlled by steam generator level and total flow via the EFIC System. Separate EFW flow transmitters (EF-23-FT through EF-26-FT) to each OTSG are provided. Redundant measurement is provided by OTSG level instrumentation transmitters. Each pump is protected by flow recirculation lines back to the Dedicated Emergency Feedwater (EFT-2) Tank. The motor-driven emergency feedwater pump (EFP-1) receives power from the 4160 volt Engineered Safeguards (ES) Bus 3A. The motor is tripped on an AC power failure at the bus. In the event of this condition, the motor-driven emergency feedwater pump will automatically start five seconds after ES bus is loaded onto the "A" train diesel generator. The turbine-driven emergency feedwater pump is independent of AC power and starts by opening the 125V DC motor operated steam admission valves (ASV-5 and ASV-204) when activated by the EFIC System.

Delete

10.5.2.1 Motor-Driven Emergency Feedwater Pump

The Motor-Driven Emergency Feedwater Pump (EFP-1) is powered from emergency diesel generator 3A when normal AC power is not available. It has a rated capacity of 750 gpm at 1300 psig with a design recirculation flowrate of 200 gpm with the pump discharge closed. Motor, gear cooling, and lube oil cooling are supplied from the Nuclear Services Closed Cycle Cooling System (SW). Manual controls (both locally and on the main control board) provide the operator with the capability of shutting down the pump if the extra feedwater capacity is not required.

10.5.2.2 Turbine-Driven Emergency Feedwater Pump

The Turbine-Driven Emergency Feedwater Pump (EFP-2) provides redundant 100% capacity to the EF System and automatically starts on the same initiating signals as the motor-driven emergency feedwater pump (EFP-1). It has a rated capacity of 750 gpm at 1300 psig with a design recirculation flowrate of 200 gpm with the pump discharge closed. Manual controls (both locally and on the main control board) provide the operator with the capability of shutting down the pump is the extra feedwater capacity is not required. Lube oil cooling is supplied by the pump discharge water.

10.5.2.3 Emergency Feedwater Pump Turbine Drive

The Turbine Drive (EFTB-1) provides rapid start-up for EFP-2. The steam supply for EFTB-1 is obtained from both OTSGs through six-inch lines containing normally-open DC motor operated stop-check valves (MSV-55 and MSV-56) and check valves (MSV-186 and MSV-187) with one exception. The check valves and motor operated valves provide redundant isolation capability to preclude blowing down the operable steam generator in the event of main steam line or main feedwater line break. The exception is the piping between MSV-55/MSV-56 and the EF pump turbine. A rupture in this piping would result in simultaneous blowdown of both steam generators and would cause the turbine driven EF pump to be inoperable. B&W Analysis 86-1176873-00 (FPC Calc M-89-0056) shows this event to be within the plant design basis. Downstream of check valves (MSV-186 and MSV-187) the lines join to form a common supply to the pump turbine. Upstream of the turbine are redundant, normally closed DC motor operated valves (ASV-5 and ASV-204), ~~which are~~ v-5 opens upon actuation from the EFIC System. A description of the controls for these valves is contained in Section 7.2.4.

Turbine exhaust is vented to the atmosphere.

Two relief valves (EFV-99 and EFV-100) are provided to protect the tank from overpressurization. The relief valve discharge is piped to the tank enclosure sump. Two vacuum breakers (EFV-97 and EFV-98) are provided to protect the tank from negative pressure due to EFW drawdown operations.

EFT-2 can also be aligned as a backup water source to the auxiliary feedwater pump (FWP-7), as discussed in Section 10.6.

10.5.3 Safety Evaluation

For information on the following items, refer to the indicated section:

1. Protection from the effects of pipe rupture - Section 5.4.4.

2. Seismic analysis - Section 5.1.2.

3. Flood design - Section 2.4.2.

4. Tornado design - Section 5.2.1.2.6.

The EF System is also single failure proof to isolate an affected OTSG for those loss of main feedwater events where such isolation is required (e.g., steamline break inside Primary Containment). Use of turbine-driven and motor-driven pumps results in diverse sources of power to the system.

The preferred source and various secondary (or alternate) sources of emergency feedwater as listed in Table 10-2 provide a redundant quality source of water to back up the dedicated emergency feedwater tank. The minimum quantity of usable condensate maintained in the Seismic Category I Dedicated Emergency Feedwater Tank is 150,000 gallons. This amount is sufficient to remove nuclear decay heat for a period of 18 hours at HOT STANDBY conditions, during which time the RCS metal and contained water would not be cooled except by ambient losses to the containment. If plant cooldown to 280°F is required, additional usable condensate (some of which is not seismically protected) is available at CR-3 and on the Crystal River site from fossil-fueled power generating stations as listed in Table 10-2 under "Other EFW Sources."

The EFW turbine-driven pump and turbine are a self-contained unit operating independently of secondary support systems. The bearings on the turbine and pump are lubricated by slinging oil from reservoirs near the bearings. Lube oil cooling is accomplished by heat transfer to the pumped fluid.

To ensure EFW flow in the event of a loss of all AC power, the turbine-driven pump train derives its motive power from the main steam lines through valves MSV-55 and MSV-56. Valves EFV-2 (locked closed), EFV-11, EFV-32, EFV-57, and EFV-58 and ASV-204 receive power from Battery 'A'. Valves EFV-1 (locked closed), EFV-14, EFV-33, EFV-55, EFV-56, MSV-55, MSV-56, and ASV-5 receive power from Battery 'B'.

Delete

Florida Power

10CFR50.59 SAFETY EVALUATION

Document No. MAR 96-04-12-01

PART II - 10CFR50.59 Evaluation

Review NEP 210, Exhibit 4, instructions before completing Part II. All "Yes" and "No" answers must be explained.

1. Is the probability of occurrence of an accident previously evaluated in the FSAR increased?

☐ Yes ☒ No

Eliminating the automatic opening upon an EFIC actuation of ASV-204 will not impact the probability of occurrence of any of the accidents analyzed in Chapter 14 of the FSAR. Those accidents involve the failure of large plant components and none of them are started by failure of the control circuitry for ASV-204. FSAR Chapter 14 reviewed.

2. Are the consequences of an accident previously evaluated in the FSAR increased?

☐ Yes ☒ No

The accidents that the opening of ASV-204 is involved in are any that involve an EFIC initiation. Since they all take a concurrent LOOP, this involves every one of them. The only single failure that may require ASV-204 to automatically open is a failure of ASV-5 to automatically open. Any single failure that causes ASV-5 to not open precludes a single failure that would effect the "A" Emergency Feedwater Train. Since the "A" EFW train would be available to mitigate the consequences of any postulated accident, the consequences of that accident would not be increased. FSAR Chapter 14 reviewed.

3. Is the possibility of an accident of a different type than any previously evaluated in the FSAR created?

☐ Yes ☒ No

Since this MAR takes out the capability of ASV-204 to open automatically, it actually decreases the possibility of an accident of a different type than any previously evaluated in the FSAR, as ASV-204 cannot spuriously open as a result of a failure in that circuit. FSAR Section 10.5.2 reviewed.

4. Is the probability of occurrence or malfunction of equipment previously evaluated in the FSAR increased?

☐ Yes ☒ No

ASV-204 becomes a redundant manually operated steam admission valve to EFP-2. This does not impact the redundancy of the Emergency Feedwater system. FSAR Section 10.5.2 reviewed.

Florida Power Corporation

10CFR50.59 SAFETY EVALUATION

Document No. MAR 96-04-12-01

5. Are the consequences of malfunction of equipment previously evaluated in the FSAR increased?

☐ Yes ☒ No

ASV-204 becomes a redundant manually operated steam admission valve to EFP-2. The Emergency Feedwater system remains fully redundant.

FSAR Section 10.5.2 reviewed.

6. Is the possibility for malfunction of equipment of a different type than any previously evaluated in the FSAR created?

☐ Yes ☒ No

Removal of the automatic opening on EFIC actuation on ASV-204 actually decreases the possibilities of malfunctions of this equipment of a different type than previously evaluated in the FSAR since it removes two circuits that could fail from the plant. FSAR Section 10.5.3 reviewed.

7. Is the margin of safety, as defined in the basis for any Technical Specification, reduced?

☐ Yes ☒ No

The margin of safety, as defined in the basis for Technical Specification 3.7.5, Emergency Feedwater System, is satisfied by one of the two trains remaining operational in the event of the defined accidents even with a single failure. Removal of the automatic opening of ASV-204 does not change the capability of the EFW System to meet that requirement. Technical Specification 3.7.5 and its Bases reviewed.

If the answer is "Yes" to any of the questions in Part II an unreviewed safety question (USQ) is involved. Contact the Manager, Nuclear Licensing per NEP 210, Exhibit 4.

PREPARED BY	DATE	APPROVED BY	DATE
C. B. Doyel	4/26/96	D. A. Shook	4/26/96
Printed Name of Preparer: C. B. DOYEL		Printed Name of Approver: D. A. SHOOK	

Florida Power

NUCLEAR ENGINEERING SAFETY IMPACT ASSESSMENT

DOCUMENT NO. AND/OR REVISION
MAR 96-04-12-01

PART I - Screening for Impact on Plant Operations.

A. Could this change potentially reduce the level of safety of Crystal River Unit 3?

☐ Yes ☒ No

Justification: (Do not simply restate the question.)

This MAR improves the level of safety at CR3. Per the 10CFR50.59 Evaluation, it creates no increased probability or possibility of accidents, or no increased consequences of those accidents. It improves the Emergency Feedwater System response to a LOCA, LOOP and "B" Battery failure, by eliminating reliance on EFP-2 when no flow controls or indication is present.

B. Could this change possibly lead to an event that impacts the safe operation of Crystal River Unit 3.?

☐ Yes ☒ No

Justification: (Do not simply restate the question.)

Examination of the events that the EFW system is required to mitigate, and the single failure analysis of the system with the new configuration, shows that this MAR will not impact the availability of the EFW system. Eliminating the capability of ASV-204 to automatically open on EFIC actuation also eliminates possible spurious failures.

If either answer is "Yes," PRC review is required.

PREPARED BY C. B Doyel DATE 4/26/96	APPROVED BY D.A. Shook 4/26/96 DATE
Printed Name of Preparer C. B DOYEL	Printed Name of Approver D.A. SHOOK

Rev. 3/96

REF: Life of Plant RESP: Nuclear Engineering

CR3 005968

MAR FUNCTIONAL TEST PROCEDURE FOR

ASV-204 EFIC AUTO REMOVAL

MAR: 96-04-12-01

TP# 1, REV.0

	YES	NO
Safety Related	[X]	[ ]
EQ Review Required	[X]	[ ]

Prepared By J. Morgan

Date 4-26-94

ORIGINAL

Verified By Roger W. Hall

Date 4/26/96

Supervisor, Functional Test KF Lamont

Date 4/27/96

NPTS Review Copy to NPTS for Review

Date 4/27/96

EQ Review Mark Skimel

Date 4/27/96

CR3 005969

1.0 PURPOSE

1.1 TEST INTENT

1.1.1 MAR deletes EFIC-A Actuation contacts from open & close legs of ASV-204 control circuit.

1.1.2 The test will simulate an output for EFIC Actuation A, to verify that
- ASV-204 does not receive an input, and
- all other functions associated with the actuation signal continue to operate satisfactorily.

1.2 TEST SCOPE

1.2.1 This test will effect;
- relay 3A1-6 (item AR) and relay 3A2-5 (item AQ) in RR3A
- control circuits for ASV-204, EFV-2, and EFP-1

1.3 TEST DESCRIPTION

1.3.1 ASV-204, EFV-2, and EFP-1 will be disabled to prevent their operation during the test.

1.3.2 Control circuits for ASV-204, EFV-2, and EFP-1 will be configured to permit the desired testing to proceed (interlocks, standing trips, etc will be disabled as necessary).

1.3.3 A jumper will be installed in RR3A to simulate an EFIC-A actuation.

2.0 ACCEPTANCE CRITERIA

2.1 Simulation of an EFIC-A actuation signal will energize relay 3A1-6 (item AR) and relay 3A2-5 (item AQ) in RR3A and;

2.1.1 Will NOT energize ASV-204 OPEN contactor.

2.1.2 Will NOT prevent energizing ASV-204 CLOSE contactor.

2.1.3 Will ENERGIZE EFV-2 OPEN contactor.

2.1.4 Close breaker for EFP-1.

CR3 005970

3.0 REFERENCES

3.1 IMPLEMENTING REFERENCES

3.1.1 None

3.2 DEVELOPMENTAL REFERENCES

3.2.1 CP-134, Preparation, Approval and Performance of MAR Functional Test Procedures.

3.2.3 MAR

4.0 SPECIAL TEST EQUIPMENT

4.1 None

5.0 LIMITS AND PRECAUTIONS

5.1 Close coordination with Operations Department will be required while performing this test.

5.2 If any leads are lifted, terminal block links opened or jumpers installed, then equipment alteration and restoration must be documented on Equipment Alteration Log (Attachment III) unless otherwise recorded in Section 7.0 of this procedure.

5.3 If a step or subsection fails to meet the established criteria, then testing may continue to the next step or subsection if the Test Engineer determines that subsequent testing will not be affected by the failed step or subsection.

5.4 In the event of any unanticipated or unusual occurrence, stop testing and restore inoperative systems if possible, and notify the NSSOD/ANSSOD.

5.5 The scope of this procedure includes EQ components, but the test does not impact their EQ characteristics

CR3 005971

6.0 PREREQUISITE CONDITIONS

6.1 Mar installation is complete.

____/____
Initl. Date

6.2 Section 5.0 (Limits and Precautions) have been read and understood.

____/____
Initl. Date

6.3 Obtain a Tagging Order, or if the NSSOD will not permit the use of tags, ensure that the following equipment is aligned as required.

[ ] Blue tag DPDP-8A sw#15 is ON - Control power for ASV-204
[ ] Red tag DPDP-8A sw#17 is OFF - Motor power for ASV-204
[ ] Blue tag DPDP-8A sw#2 is ON - Control power for EFV-2
[ ] Red tag DPDP-8A sw#1 is OFF - Motor power for EFV-2
[ ] Blue tag 4KV ES Bus unit 3A3 racked to TEST - EFP-1 motor
[ ] Blue tag EFV-2 CB-PSA control switch
[ ] Blue tag EFV-2 Local contrc' switches
[ ] Blue tag ASV-204 CB-PSA control switch
[ ] Blue tag ASV-204 Local control switches
[ ] Blue tag EFIC Test-A select switch in RR3A

____/____
Initl. Date

6.4 Ensure that the 4KV A-ES Bus is energized via Bkr 3205, 3207, or 3211.

____/____
Initl. Date

6.5 A pretest briefing has been conducted with all test personnel.

____/____
Initial Date

6.6 The NSSOD/ANSS has given his app' ,val to begin the test.

____/____
Initial Date

6.7 Obtain a key for the EFIC Test-A selector switch (in RR3A).

____/____
Initl. Date

CR3 005972

7.0 TEST INSTRUCTIONS

7.1 Configure EFP-1 control circuit for testing.

7.1.1 Ensure 4KV A-ES bus unit 3A3 is in the TEST position.

____/____
Initl. Date

7.1.2 If both EFV-2 and EFV-3 are closed, perform the next two steps.
If either EFV-2 or EFV-3 are open, skip the next two steps; mark them NA.

7.1.2.1 Eliminate the "EFV-2 and 3 closed" trip by opening link AJ11 in 4KV ES unit 3A3.

____/____
Initl. Date

7.1.2.2 At 4KV A-ES unit 3A3, satisfy the "EFV-2 or 3 open" permissive by installing a jumper from AJ1 to AJ5.

____/____
Initl. Date

7.1.3 Direct the control room operator to close EFP-1 breaker by momentarily placing its control switch in START.

Verify that EFP-1 breaker is closed.

____/____
Initl. Date

7.1.3.1 Direct the control room operator to open EFP-1 breaker by momentarily placing its control switch in STOP.

Verify that EFP-1 breaker is open.

____/____
Initl. Date

7.2 Configure ASV-204 and EFV-2 control circuits for testing.

7.2.1 At DPDP-8A;

[ ] Ensure that switch 1 is OPEN,

[ ] Ensure that switch 17 is OPEN.

____/____
Initl. Date

7.2.2 In RR2, disable the vacuum breaker closed interlock with EFV-2, by opening the link at TB13-30.

____/____
Initl. Date

7.2.3 In terminal box EF-2; bypass EFV-2 open and close limits as follows;

[ ] install a jumper from TBA-6 to TBB-6

[ ] install a jumper from TBA-6 to TBC-1

_____/_____
Initl. Date

7.2.3.1 In the motor starter panel for EFV-2, verify that

[ ] Close contactor is de-energized.

[ ] Open contactor is de-energized.

_____/_____
Initl. Date

7.2.3.2 At control station EF-2; momentarily depress the close push button.

_____/_____
Initl. Date

7.2.3.3 In the motor starter panel for EFV-2, verify that the

[ ] Close contactor is ENERGIZED.

[ ] Open contactor is de-energized

_____/_____
Initl. Date

7.2.3.4 At control station EF-2; momentarily depress the stop pushbutton.

_____/_____
Initl. Date

7.2.3.5 In the motor starter panel for EFV-2, verify that the

[ ] Close contactor is de-energized

[ ] Open contactor is de-energized.

_____/_____
Initl. Date

7.2.3.6 At control station EF-2; momentarily depress the open push button.

_____/_____
Initl. Date

CR3 005974

7.2.3.7 In the motor starter panel for EFV-2, verify that the

[ ] Close contactor is de-energized.

[ ] Open contactor is ENERGIZED.

____/____
Initl. Date

7.2.3.8 At control station EF-2; momentarily depress the stop pushbutton.

____/____
Initl. Date

7.2.3.9 In the motor starter panel for EFV-2, verify that the

[ ] Close contactor is de-energized

[ ] Open contactor is de-energized.

____/____
Initl. Date

7.2.4 In terminal box AS-9; bypass ASV-204 open and close limits as follows;

[ ] install a jumper from TBA-4 to TBA-5

[ ] inatall a jumper from TBA-6 to TBA-7

____/____
Initl. Date

7.2.4.1 In the motor starter panel for ASV-204, verify that

[ ] Close contactor is de-energized.

[ ] Open contactor is de-energized.

____/____
Initl. Date

7.2.4.2 At terminal box AS-9 (on the right side panel); depress and HOLD the close push button.

____/____
Initl. Date

7.2.4.3 In the motor starter panel for ASV-204, verify that the

[ ] Close contactor is ENERGIZED.

[ ] Open contactor is de-energized

____/____
Initl. Date

7.2.4.4 At terminal box AS-9; release the close pushbutton.

____/____
Initl. Date

7.2.4.5 In the motor starter panel for ASV-204, verify that the

[ ] Close contactor is de-energized

[ ] Open contactor is de-energized.

____/____
Initl. Date

7.2.4.6 At terminal box AS-9; depress and HOLD the open push button.

____/____
Initl. Date

7.2.4.7 In the motor starter panel for ASV-204, verify that the

[ ] Close contactor is de-energized.

[ ] Open contactor is ENERGIZED.

____/____
Initl. Date

7.2.4.8 At terminal box AS-9; release the open pushbutton.

____/____
Initl. Date

7.2.4.9 In the motor starter panel for ASV-204, verify that the

[ ] Close contactor is de-energized

[ ] Open contactor is de-energized.

____/____
Initl. Date

7.3 In RR3A, place EFIC Test-A selector switch (item G) in the TEST position.

____/____
Initl. Date

7.3.1 In RR3A, install a jumper from Selector switch (item G) terminal 1 (wire on terminal 1 is labeled CU-6) to TB17-1 (210-624).

____/____
Initl. Date

CR3 005976

7.3.1.1 Inform the control room operator that performance of this test will initiate EFP Start Failure alarms.

____/____
Initl. Date

• 7.3.2 In RR3A, place EFIC Test-A selector switch (item G) in NORMAL position. Verify that -
[ ] relay 3A1-6 (item AR) is ENERGIZED,
[ ] relay 3A2-5 (item AQ) is ENERGIZED.

____/____
Initl. Date

• 7.3.3 Verify that the breaker for EFP-1 has CLOSED.

____/____
Initl. Date

• 7.3.4 In the motor starter panel for EFV-2, verify that the

[ ] Close contactor is de-energized.

[ ] Open contactor is ENERGIZED.

____/____
Initl. Date

• 7.3.5 In the motor starter panel for ASV-204, verify that

[ ] Close contactor is DE-ENERGIZED.

[ ] Open contactor is DE-ENERGIZED.

____/____
Initl. Date

7.3.6 At terminal box AS-9; depress and HOLD the close push button.

____/____
Initl. Date

• 7.3.6.1 In the motor starter panel for ASV-204, verify that the

[ ] Close contactor is ENERGIZED.

[ ] Open contactor is de-energized

____/____
Initl. Date

7.3.6.2 At terminal box AS-9; release the close pushbutton.

____/____
Initl. Date

7.3.6.3 In the motor starter panel for ASV-204, verify that the

[ ] Close contactor is de-energized

[ ] Open contactor is de-energized. ____/____ Initl. Date

7.3.6.4 At terminal box AS-9; depress and HOLD the open push button. ____/____ Initl. Date

* 7.3.6.5 In the motor starter panel for ASV-204, verify that the

[ ] Close contactor is de-energized.

[ ] Open contactor is ENERGIZED. ____/____ Initl. Date

7.3.6.6 At terminal box AS-9; release the open pushbutton. ____/____ Initl. Date

7.3.6.7 In the motor starter panel for ASV-204, verify that the

[ ] Close contactor is de-energized

[ ] Open contactor is de-energized. ____/____ Initl. Date

7.4 In terminal box EF-2;

[ ] Remove jumper from TBA-6 to TBB-6 ____/____ Initl. Date

Independent Verification ____/____ Initl. Date

[ ] Remove jumper from TBA-6 to TBC-1 ____/____ Initl. Date

Independent Verification ____/____ Initl. Date

7.5 In terminal box AS-9;

[ ] Remove jumper from TBA-4 to TBA-5 ____/____ Initl. Date

Independent Verification ____/____ Initl. Date

[ ] Remove jumper from TBA-6 to TBA-7 ____/____ Initl. Date

Independent Verification ____/____ Initl. Date

7.6 At 4KV A-ES Bus, trip EFP-1 breaker. ____/____ Initl. Date

7.6.1 In 4KV ES unit 3A3, close link AJ11.
If it was not opened NA this step. ____/____ Initl. Date

Independent Verification ____/____ Initl. Date

7.6.2 In 4KV ES unit 3A3, remove jumper from AJ1 to AJ5.
If it was not installed, NA this step. ____/____ Initl. Date

Independent Verification ____/____ Initl. Date

7.7 In RR2, close link at TB13-30. ____/____ Initl. Date

Independent Verification ____/____ Initl. Date

7.8 In RR3A, place EFIC Test-A selector switch (item G) in the TEST position. ____/____ Initl. Date

7.8.1 In RR3A, remove jumper from Selector switch (item G) terminal 1 to TB17-1. ____/____ Initl. Date

Independent Verification ____/____ Initl. Date

7.8.2 In RR3A, place EFIC Test-A selector switch (item G) in NORMAL position.
Verify that -
[ ] relay 3A1-6 (item AR) is de-energized,
[ ] relay 3A2-5 (item AQ) is de-energized. ____/____ Initl. Date

CR3 005979

8.0 RESTORATION

8.1 Ensure that all panels and terminal boxes are closed;
[ ] Term Box AS-9
[ ] Term Box EF-2
[ ] ASV-204 motor starter panel
[ ] EFV-2 motor starter panel

_____/_____
Initial Date

8.2 Notify NSSOD/ANSS that test is complete.

_____/_____
Initial Date

8.3 Release tags acquired for this test.

_____/_____
Initl. Date

9.0 ATTACHMENTS

9.1 Attachment I - Test Log

9.2 Attachment II - Test Results Review And Approval Form

9.3 Attachment III - Equipment Alteration Log

9.4 Attachment IV - Test Exception Report

9.5 Attachment V - Test Exception Log

9.6 Attachment VI - Infrequently Performed Test Or Evolution Checklist

Attachment I

TEST LOG

MAR No. 96-04-12-01 TP No. 1

DATE/TIME	TEST LOG ENTRY

Attachment II

TEST RESULTS REVIEW AND APPROVAL FORM

MAR No. 96-04-12-01 TP No. 1

1. Is this a partial or full review of MAR Test Procedure results?

PARTIAL ___ FULL ___ If a partial review, describe the portion of the test completed and the remaining test requirements.

2. Has all established acceptance criteria been satisfied?

YES ___ NO ___ If no, explain corrective actions taken to resolve the failure to meet acceptance criteria.

3. Have all test exceptions noted on the TEST EXCEPTION LOG been resolved and tested or otherwise satisfied? If no test exceptions were noted check the N/A block.

YES ___ NO ___ N/A ___ If the answer is NO, then explain what corrective actions were taken to resolve open items.

Prepared by: ______________________ Date: ____________
Test Director

Reviewed by: ______________________ Date: ____________
NSSOD/ANSSOD

Approved by: ______________________ Date: ____________
SNES Supv., Funct. Test

EQUIPMENT ALTERATION LOG

WORK REQUEST NUMBER		PROCEDURE NUMBER	TP-1	MAR NUMBER	96-04-12-01

DESCRIPTION				ALTERATION		RESTORATION	
ITEM	STATE OF ALTERATION		EFFECTS OF ALTERATION UPON SYSTEM / EQUIPMENT	PERFORMED BY INIT/DATE	INDEP. VER. BY INIT/DATE	PERFORMED BY INIT/DATE	INDEP. VER. BY INIT/DATE
	FROM	TO					

NOTE: IF an equipment alteration must remain in effect beyond the shift during which it was implemented and active maintenance activities will not continue in the next shift, THEN the Work Supervisor and the NSSOD must be notified that the equipment alteration will remain in place beyond the present shift.

NOTE: Prior to any equipment alteration being installed, the first three spaces under DESCRIPTION (ITEM, STATE OF ALTERATION and EFFECTS OF ALTERATION) on the Equipment Alteration Log must be completed.

COMPLETED BY ______________ DATE ________ SUPERVISOR CONCURRENCE _____________ DATE _______

Attachment IV

TEST EXCEPTION REPORT

MAR 96-04-12-01 TP # 1 TER # ________

Acceptance Criteria affected: YES or NO Procedure Step No.: ________

DESCRIPTION OF EXCEPTION: ________

Test Engineer ________ Date ________

IMMEDIATE ACTION TAKEN: ________

Test Engineer ________ Date ________

PLANNED RESOLUTION: ________

Supervisor, Functional Test ________ Date ________

REVIEW AND CLOSURE ACTION: ________

Supervisor, Functional Test ________ Date ________

Attachment V

TEST EXCEPTION LOG

MAR # 96-04-12-01 TP # 1

TER No.	DESCRIPTION OF TEST EXCEPTION	DATE INITIATED	DATE CLOSED

SNES SUPERVISOR, FUNCTIONAL TEST ______________________ DATE __________

CR3 005985

MAR 96-04-12-01, TP-1, Rev.1

INFREQUENTLY PERFORMED TEST OR EVOLUTION CHECKLIST

Answer the following questions to determine if this procedure describes an infrequently performed test or evolution.

1. Does this procedure create a situation that can affect the core reactivity control or the reactor protection systems?

   YES__________ NO____X_____

Why? This procedure will test relays associated with EFP-1, with all end devices disabled. It affects neither core reactivity nor reactor protection or control systems.

If the answer is no, do not complete any more of this checklist

2. Does this procedure create an evolution not covered by an existing normal or abnormal operating procedure?

   YES__________ NO__________

3. Does this procedure create an evolution that will seldom be performed, even though it is covered by an existing normal or abnormal operating procedure?

   YES__________ NO__________

4. Does this procedure create an infrequently performed surveillance test that involves complicated sequencing, or placing the plant in an unusual configuration?

   YES__________ NO__________

5. Does this procedure require the use of special test procedure in conjunction with existing operating or testing procedures?

   YES__________ NO__________

IF the answer to questions 1 AND at least one other is "YES", THEN this procedure is an infrequently performed test or evolution, and requires a briefing in accordance with AI-500 prior to being performed. The procedure shall have a sign off step, either a prerequisite or its first step, that documents this briefing.

PROCEDURE DESCRIPTION, REASONS, AND REFERENCES
Procedure: MAR 96-04-12-01, TP-1, Rev # 0 PRR # Title: MAR Functional Test Procedure for ASV-204 EFIC AUTO REMOVAL
DESCRIPTION OF NEW PROCEDURE OR CHANGE (Detailed) New procedure which will simulate an EFIC-A actuation for relay 3A2-5 and 3A1-6, by energizing them with a jumper placed in RR3A. The relays will effect the control circuits for EFV-2 and EFP-1; however, neither component will operate because the motor power for EFV-2 will be tagged OFF, and EFP-1 breaker will be in the TEST position.
REASON AND REFERENCES FOR PROCEDURE CHANGE To ensure correct operation of the EFIC Actuation A relay 3A1-6, following deletion of its associated contacts from ASV-204 control circuit. The test will prove that ASV-204 is no longer effected by EFIC actuation, and that EFP-1 and EFV-2 have retained their EFIC Auto actuation functions.

Check As Appropriate:	Reference#
[X] Modification [ ] FCN [ ] Other Design Related Item	
[ ] Problem Report [ ] Precursor Card [ ] NRC Violation	
[ ] ITS Amendment [ ] ITS Bases Change [ ] FSAR Change	

Screening for Applicability of 10 CFR 50.59		
Procedure: MAR 96041201, TP-1, Rev.0 PRR# Title: MAR Functional Test Procedure for ASV-204 EFIC AUTO REMOVAL		
	YES (✓)	NO (✓)
1. Is this a change to the facility as described in the FSAR? (Use NOD-11, Preparation of Safety, Regulatory, and Environmental Compliance Reviews-Attachment A, as guidance to adequately answer this question). **Because:** This procedure will verify that control circuits affected by the associated MAR, operate per design.		X
2. Does this affect or change the Improved Technical Specifications or their bases? (If yes, contact the Manager Nuclear Licensing before proceeding) **Because:** This procedure does not change the configuration of any equipment that is being relied upon to perform a safety function.		X
3. Is this a new procedure or the ODCM?	N/A	
4. A yes answer to any question above requires completion of Enclosure 4, 10 CFR 50.59 Evaluation and PRC/DNPO reviews prior to implementation. Are these required?		X
5. Can this potentially reduce the level of safety of the plant? (If yes, provide contingency actions <u>AND</u> PRC/DNPO reviews are required prior to implementation) **Because:** This procedure will be performed immediately following completion of the modification activities for ASV-204 control circuit, while it is out of service and not performing any safety function.		X
6. Review Enclosure 5, Examples of Events Which Impact Plant Operations. Can this possibly lead to an event that would impact plant operation? (If yes, provide contingency actions <u>AND</u> PRC/DNPO reviews are required prior to implementation) **Because:** The end devices associated with the control circuits being tested will be disabled (EFV-2 motor power will be OFF and EFP-1 breaker will be in the test position), to ensure that the testing can have no effect on status of plant equipment.		X

ORIGINATOR'S CHECKLIST

Procedure#: 96-04-12-01, TP-1 Current Revision#: 1 PRR#:

Procedure Title: ASV-204 EFIC AUTO REMOVAL

		YES (✓)	N/A (✓)
1	Obtain the current revision of the procedure from Document Control or electronically. For temporary procedure changes, the working copy may be used.		X
2	Obtain PRR#, new procedure#, or temporary procedure change (TPC) # from Document Control or electronically. For work instructions, use a work request number or use another identifying number.	X	
3	Write a new procedure, work instruction, or mark the pages applicable to any changes by using red ink or electronic red line/strikeout methods. For procedures, use the appropriate procedure format in AI-402B, Procedure Writing (Except for EOP/AP/VP).	X	
4*	Attach changed pages to the cover page of the procedure. Other pages may be attached so the continuity of the changes will be understood by persons performing further reviews. For new procedures or work instructions, all pages are included.	* X	
5	Complete Enclosure 2, Procedure Description, Reasons, and References.	X	
6*	Obtain the Procedure Review Package (PRP) from Nuclear Licensing or electronically. For work instructions or new procedures, contact Nuclear Licensing for any related commitments associated with the work instructions or new procedures.	* NA	
	a. Complete the Procedure Cross Reference sheet.	*	* X
	b. Ensure the NOCS commitments remain satisfied.	*	* X
	1. Change the implementing references of the commitments as necessary.	*	* X
7	Complete Enclosure 3, Screening for Applicability of 10 CFR 50.59.	X	

* These steps should not be performed and the YES or N/A blanks should not be checked (✓) for a temporary procedure change. These steps are required to be performed and checked (✓) YES or N/A at a later time if the temporary change is to become permanent.

ORIGINATOR'S CHECKLIST

Procedure# 96-04-12-01TP-1, Rev. 1 PRR# _____		YES (✓)	N/A (✓)
8•	IF a 10 CFR 50.59 Evaluation is required, as determined by the completion of Enclosure 3, Screening for Applicability of 10 CFR 50.59, THEN complete Enclosure 4, 10 CFR 50.59 Evaluation.	•	• X
9•	Complete Enclosure 6, 10 CFR 50.54 Evaluation.	• X	
10•	IF this can affect plant safety, the core, reactivity control, the Reactor Protection System, the engineered safeguards systems, or the plant design basis, THEN complete and comply with Enclosure 7, Infrequently Performed Test or Evolution Checklist.	•	• X
11•	IF any plant equipment tag numbers are being added to or deleted from this procedure, THEN complete and mail Enclosure 8, CMIS Database Update Request Form, to the Manager Nuclear Configuration Management.	•	• X
12•	Complete Enclosure 9, Notification of New/Revised Procedure (NORP) by determining which Nuclear Operations departments need to be notified of the new procedure/permanent procedure change.	• NA	
13•	IF this is a SP, new procedure, or an infrequently performed test or evolution, THEN complete and attach Enclosure 23, Procedure Review Checklist of AI-402B, Procedure Writing (Except for EOP/AP/VP).	•	• X
14	Ensure that correction techniques such as "whiteout," correction tape, paste overs, or similar techniques have not been used.	X	
15•	Internal references to data sheets, enclosures, other sections of the procedure, etc., have been verified to be correct.	• X	
16•	Ensure that Non-Quality documents are identified as "Optional Records Non-Quality" within the procedure as per AI-1100, Retention of Plant Operating Records.	•	• X

• These steps should not be performed and the YES or N/A blanks should not be checked (✓) for a temporary procedure change. These steps are required to be performed and checked (✓) YES or N/A at a later time if the temporary change is to become permanent.

ORIGINATOR'S CHECKLIST

Procedure# 96-04-12-01JP-1, Rev.1 PRR# _____		YES (✓)	N/A (✓)
17*	Forward the original or copies of this document and supporting enclosures to Qualified Reviewers, as determined from below, using Enclosure 10, Qualified Review/Technical Review.	• X	
	a. A Qualified Review, independent of the originator, is required by the department that the procedure's Interpretation Contact is a member of.	• NA	
	b. IF interfacing departments' actions or procedures will be impacted, THEN the interfacing departments must perform a Qualified Review.	•	• X
	c. IF any item on Enclosure 11, Guidelines for Identification of Design Changes, is applicable, THEN Nuclear Engineering Design must perform a Qualified Review.	•	• X
	d. Nuclear Quality Control must be a Qualified Reviewer for FPs, MPs, PMs, PTs, SPs, WP-101, and WP-102 that contain changes to or additions of inspection points to ensure inspection points are appropriate.	•	• X
	e. IF this: 1. Involves a change to an established scheduled performance of a procedure as listed in SP-443, Master Surveillance Plan, OR 2. Is related to a change to the Improved Technical Specification (ITS) or the ITS bases, THEN the Senior Nuclear Scheduling Coordinator responsible for surveillance scheduling must be a Technical Reviewer.	•	• X
	NOTE: EQ equipment is designated in CMIS as QQ. f. IF this involves repair, replacement, or alteration of EQ equipment, THEN a Qualified Review is required by Nuclear Configuration Management, EQ Group.	•	•
	1. IF this procedure involves repair, replacement, or alteration of EQ equipment, THEN the cover page of the procedure must be designated "Environmentally Qualified Related."	•	• X
18*	Ensure Qualified Reviews are completed and any comments have been addressed.	• X	
19	IF this is a temporary procedure change, THEN complete Enclosure 12, Authorization of a Temporary Procedure Change.		X
20*	Forward this enclosure and the preceding package of documentation, as applicable, to the Interpretation Contact for approval of Enclosure 13, Interpretation Contact's Checklist.	• X	

I have completed this checklist, as applicable, and all appropriate YES or NA blanks are completed.

J. Morgar 4/26/96

ORIGINATOR DATE

* These steps should not be performed and the YES or N/A blanks should not be checked (✓) for a temporary procedure change. These steps are required to be performed and checked (✓) YES or N/A at a later time if the temporary change is to become permanent.

INTERPRETATION CONTACT'S CHECKLIST

Procedure#: 96-04-12-01, TP-1, Rev.0 PRR#:		YES (✓)	N/A (✓)
1	Originator's Checklist, Enclosure 1 is adequately completed.	✓	
2	IF this is a new SP, an SP that has received a major technical rewrite, or a procedure identified as an infrequent test or evolution, THEN ensure a simulation, walk through (no manipulation of plant equipment), table talk, or comparison validation has been performed before the procedure is approved by the Interpretation Contact.		✓
3	IF this is a procedure identified as an infrequent test or evolution, THEN ensure the procedure will be walk through validated with an end-user during the first performance after the procedure is issued.		✓
4	IF this is a new SP, or is a SP that has received a major technical rewrite, THEN notify the Senior Nuclear Scheduling Coordinator responsible for surveillance scheduling that the procedure must be walk through validated with an end user on the next scheduled performance after the procedure is issued.		✓
5	IF this is an OP that involves valve changes, THEN ensure adequate administrative controls are in place to control valve positions.		✓
6	IF the Procedure Cross Reference sheet had any questions answered yes, THEN forward the Procedure Cross Reference sheet to Records Management.		✓
7	IF any commitments of the NOCS have any changes to the implementation section, THEN forward the applicable commitments to Nuclear Licensing.		✓
8	IF inspection planning is required per CP-113C, Inspection Planning, THEN ensure inspection planning methods are incorporated within this procedure or work instruction.		✓

INTERPRETATION CONTACT'S CHECKLIST

Procedure #96-04-12-01, TP-1, Rev.0 PRR #

****

NOTE: The Interpretation Contacts can send any item to the PRC to review prior to implementation, especially if the Interpretation Contact has a plant operation or safety concern.

****

1. Check ONE of the following:

[X] All questions of Enclosure 3, Screening for Applicability of 10 CFR 50.59 are answered "No". PRC or DNPO review is not required, procedure can be issued. I am forwarding the applicable completed enclosures to Document Control.
[ ] Any question of Enclosure 3, Screening for Applicability of 10 CFR 50.59, was answered "Yes" or the Interpretation Contact desires a PRC and DNPO review. PRC and DNPO review is required prior to implementation. I am forwarding the applicable completed enclosures to the PRC.

2. Complete ALL of the following ONLY for Temporary Procedure Changes to become permanent:

[ ] Obtain PRR# from Document Control or electronically. PRR#
[ ] The activities as designated on Enclosure 1, Originator's Checklist, including those on Enclosure 1 with an asterisk, have been completed and are attached.
[ ] I am forwarding the applicable completed enclosures to Document Control for issuance of this temporary procedure change as a permanent procedure revision.

This and other applicable enclosures have been properly completed and are being forwarded as stated above.

K.F. [signature] / 4/27/96

Interpretation Contact / Date

PRC and DNPO Review*

PRC Chairman Date	PRC Mtg No.	DNPO Date

* If required

CR3 005993

EXHIBITS TO THE TESTIMONY OF
PAUL F. MCKEE

EXHIBIT NO. ___ (PFM-7)

TECHNICAL SPECIFICATIONS LIMITING CONDITIONS

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1 LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2.

LCO 3.0.2 Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and 3.0.6.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.

LCO 3.0.3 When an LCO is not met, except as provided in the associated ACTIONS, and an associated ACTION is not met or provided, the unit shall be placed in a MODE or other specified condition in which the Specification is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:

a. MODE 3 within 7 hours;

b. MODE 4 within 13 hours; and

c. MODE 5 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

LCO 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued

(continued)

3.0 LCO APPLICABILITY

LCO 3.0.4 (continued)	operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. Exceptions to this Specification are stated in the individual Specifications. These exceptions allow entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered allow unit operation in the MODE or other specified condition in the Applicability only for a limited period of time.
LCO 3.0.5	Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY, the OPERABILITY of other equipment, or variables to be within limits. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the required testing.
LCO 3.0.6	When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system Specification ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, additional evaluations and limitations may be required in accordance with Specification 5.6.2.16, "Safety Function Determination Program." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the Specification in which the loss of safety function exists are required to be entered.

(continued)

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS—Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more trains inoperable. AND At least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available.	A.1	Restore train(s) to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 4.	12 hours

3.7 PLANT SYSTEMS

3.7.5 Emergency Feedwater (EFW) System

LCO 3.7.5 Two EFW trains shall be OPERABLE.

-----------------------------NOTE-----------------------------
Only one EFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 3 with steam generator pressure < 200 psig.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One steam supply to the turbine driven EFW pump inoperable.	A.1	Restore steam supply to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO
B. One EFW train inoperable for reasons other than Condition A.	B.1	Restore EFW train to OPERABLE status.	72 hours AND 10 days from discovery of failure to meet the LCO

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met.	C.1	Be in MODE 3.	6 hours
	AND		
	C.2	Be in MODE 4.	12 hours
D. Two EFW trains inoperable.	D.1	Initiate action to restore one EFW train to OPERABLE status.	Immediately

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources—Operating

LCO 3.8.1 The following AC electrical power sources shall be OPERABLE:

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System; and

b. Two emergency diesel generators (EDGs) each capable of supplying one train of the onsite Class 1E AC Electrical Power Distribution System.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required offsite circuit inoperable.	A.1	Perform SR 3.8.1.1 for OPERABLE required offsite circuit.	1 hour __AND__ Once per 8 hours thereafter
	__AND__		
	A.2	Declare required feature(s), with no offsite power available, inoperable when its redundant required feature(s) are inoperable.	24 hours from discovery of no offsite power to one train concurrent with inoperability of redundant required feature(s)
	__AND__		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.3	Restore required offsite circuit to OPERABLE status.	72 hours AND 6 days from discovery of failure to meet LCO
B. One EDG inoperable.	B.1	Perform SR 3.8.1.1 for OPERABLE offsite circuit(s).	1 hour AND Once per 8 hours thereafter
	AND		
	B.2	Declare required feature(s), supported by the inoperable EDG, inoperable when its redundant required feature(s) are inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	AND		
	B.3.1	Determine OPERABLE EDG is not inoperable due to common cause failure.	24 hours
	OR		
	B.3.2	Perform SR 3.8.1.2 for OPERABLE EDG.	24 hours
	AND		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.4	Restore EDG to OPERABLE status.	72 hours AND 6 days from discovery of failure to meet LCO
C. Two required offsite circuits inoperable.	C.1	Declare required feature(s) inoperable when its redundant required feature(s) are inoperable.	12 hours from discovery of Condition C concurrent with inoperability of redundant required feature(s)
	AND		
	C.2	Restore one required offsite circuit to OPERABLE status.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One required offsite circuit inoperable. AND One EDG inoperable.	------------NOTE------------ Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems—Operating," when Condition D is entered with no AC power source to one train. ----------------------------	
	D.1 Restore required offsite circuit to OPERABLE status.	12 hours
	OR	
	D.2 Restore EDG to OPERABLE status.	12 hours
E. Two EDGs inoperable.	E.1 Restore one EDG to OPERABLE status.	2 hours
F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met.	F.1 Be in MODE 3.	12 hours
	AND	
	F.2 Be in MODE 5.	36 hours
G. Three or more required AC sources inoperable.	G.1 Enter LCO 3.0.3.	Immediately