UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30323

Report No.: 50-302/89-27

Licensee: Florida Power Corporation
3201 34th Street, South
St. Petersburg, FL 33733

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

Facility Name: Crystal River 3

Inspection Conducted: November 13-17, 1989

Inspector: C. Smith 12-01-89
Date Signed

Team Members: K. Poertner
R. Wright

Approved by: F. Jape, Section Chief 12/1/89
Date Signed
Quality Performance Section
Operations Branch
Division of Reactor Safety

SUMMARY

Scope:

This routine, unannounced inspection was in the areas of design, design changes, and modifications.

Results:

In the areas inspected, violations or deviations were not identified. Independent design reviews of modification approval records (MARs) packages revealed that plant modifications are made in a technically adequate manner. Safety evaluations performed in connection with plant modifications demonstrated indepth review and analysis of design changes. Necessary engineering evaluations and calculations required to support the designs activities were performed. Inspection of installed hardware revealed no deficiencies.

REPORT DETAILS

1. Persons Contacted

Licensee Employees

D. Beach, Electrical Design Supervisor
*J. Colby, Nuclear Principal Mechanical Engineer
*C. Doyel, Manager, Nuclear Mechanical Structural Engineer
D. Green, Licensing Engineer
K. Hudak, MAR Manager
*R. Jones, Nuclear Projects Specialist
*J. Kraiker, Management Support Superintendent
*P. McKee, Director Nuclear Plant Operations
R. McLaughlin, MAR Manager
T. Montgomery, MAR Manager
*A. Petrowsky, Supervisor, Site Nuclear Enginering Services
*W. Rossfeld, Manager, Nuclear Compliance
*P. Rubio, Nuclear Instrumentation and Control Engineering Superisor
*F. Sullivan, Manager NPSE
K. Vogel, Nuclear Operations Engineer
*M. Williams, Nuclear Regulatory Specialist
*R. Widell, Director, Nuclear Operations Site Support

Other licensee employees contacted during this inspection included engineers, and administrative personnel.

NRC Resident Inspectors

P. Holmes-Ray, Senior Resident Inspector
*W. Bradford, Resident Inspector

*Attended exit interview

2. The inspectors performed independent design reviews of the following MARs to determine the technical adequacy of the design changes; to verify that safety evaluations were performed and met 10 CFR 50.59 requirements; to verify that the MARs were reviewed and approved in accordance with TS and administrative controls; to ensure that the subject modifications were installed (for those physically inspectable) in accordance with the MAR packages; to verify that applicable plant operating documents (drawings, plant procedures, FSAR, TS, etc.) were revised to reflect the subject modifications; and that post modification test requirements were specified and adequate testing was performed as necessary.

a. MAR 87-11-08-01, Replacement of Valve SWV-632

The purpose of the MAR was to replace a CRDM drain valve SWV-632, Velan brand with a similar valve of Handcock manufacture. The existing valve was damaged and parts were no longer available due to Velan discontinuing the existing model.

The nuclear service cooling water system design (discussed in the FSAR Section 9.5) is not altered in any way by the MAR and will continue to provide the required CRDM cooling. Performance characteristics, capacity, rating and system output of the SW system are unaffected by this modification. The materials of the replacement valve are identical to those of the existing valve (SSA182-F316) and the replacement valve is the same class (B31.1ES) and seismic catagory (1) as the original. The difference in weight between the new valve and existing was two pounds and no seismic re-analysis was required. Existing pipe supports were unaffected by the replacement modification. No licensee revision of the FSAR, TS, or notification to the NRC was required for this modification. The welds for this installation were visually examined in accordance with ANSI B31.1.0, 1967 edition of the Power Piping Code. [illegible] was performed on the new valve installed at full operating [illegible] per paragraph 137.1.2 of the subject code to verify [illegible] acceptability. WR NOS. 94946, 96925, and 96926 cover [illegible] procurement/issue, replacement and testing of the [illegible] modification. A sampling of As-Built drawings [illegible] revised due to this modification were examine[illegible] acceptable. The inspectors review of this modi[illegible] verified procedual compliance with the design engi[illegible] and the TS requirements.

b. MAR T86-10-19-01, Temporary Repair and Eventual Replacement of Decay Heat Spool NO. RW-57

This temporary modification was written to install a new pressure boundary enclosure around an existing leaking spool piece assembly (RW-57) which had developed a 2-inch hole due to corrosion. The enclosure material used was identical to the specified original material (carbon steel) for the system. The void between the enclosure and the process pipe was filled with Master Flow 713 Grout. This enclosure was capable of performing under all postulated design and seismic loading conditions that the original spool was designed to withstand. Surface examinations were performed on the enclosure welds to provide confidence in the structure and its pressure retaining ability. This modification was written on an emergency basis and its Engineering Instructions specified that at the first available opportunity (Mode 5) this temporary fabrication should be replaced with one of original design. The temporary MAR was initiated by WR No. 82188 on August 14, 1986, and carried an expiration date of October 31, 1987. WR NOS. 82189, 89317 and 89318 completed around October 26, 1987, removed the temporary MAR, installed a replacement spool per original vendor specifications, and hydro pressure tested the assembly in accordance with Maintenance Procedure MP-137. No FSAR or TS changes were required by this MAR since the temporary enclosure would continue to perform as originally evaluated and defined in the TS and the subject temporary MAR was subsequently replaced by a spool piece assembly of original design.

c. MAR SP82-105, Intake Canal Survey and Dredging

The intake canal serves as the ultimate heat sink for CR3. The intake canal performs two principal functions. These functions are the dissipation of residual heat after a reactor shutdown and the dissipation of residual heat after an accident. Maintenance of the intake canal performance characteristics ensures that the two safety functions can be performed. The subject MAR provided for a canal survey and dredging efforts to restore the canal to its original design condition and to effectively meet the more conservative acceptance criteria of TS change request No. 155. Rather than be concerned with maintaining rigid cross-sectional shapes for various sections of the canal from the gulf to the bend at Unit 3's intake structure as specified in the FSAR, the G/C Inc. study (Report No. 2669) proposes applying a cross-section index concept that assures the maximum flow rate required (34,900 gpm) to the safety related sea water pumps for the nuclear services and decay heat systems. From the study the minimum allowable value for this cross-section index, computed using a flow rate of 34,900 gpm, is 600,000. The study also requires a smooth bottom elevation less than or equal to 67.2 feet (CRPD) at the intake bend sections. The study concludes that if the intake canal surface is kept free of floating transportable materials at the intake structure, and the above canal characteristics are maintained, dissipation of residual heat will be satisfactory. The inspectors examined the post-maintenance dredging survey report which was completed around October 1987. This report determined that the intake canal conditions were acceptable. Ultimate heat sink surveillance requirement for operatibly, T. S. 4.7.5-1a, requires the inlet water temperature and water level to be verified to be within their limits at least once per 24 hours. The inspectors verified that the surveillance was performed on November 15, 1989, in accordance with surveillance procedure SP-300 and the results were acceptable.

d. MAR T87 10-19-01, EDG-3A Cumulative Timers Installation

This temporary modification provided for the removal of the existing EDG-3A timer and alarm scheme and replacing it with a new cumulative timer and alarm scheme and an elapse time indicator located on the main control board for operator convenience. The timers, alarm module and associated reset push buttons are mounted in a Hoffman enclosure and mounted on a blank panel on the rear of the main control board. This MAR was implemented to provide a more accurate means of monitoring the time EDG-3A operates above 3000KW. An alarm is provided each time the EDG goes above 3000KW for more than ten seconds. Additionally, alarms are provided after five minutes, twenty-four minutes, and twenty-nine minutes of EDG operation above 3000KW. The alarm module gets its imput from the watts transducer, located in relay rack 3A, which is monitoring the output of the EDG. Examination of the licensee's Safety Classification, 10 CFR 50.49 and 10 CFR 50.59 reviews for the timers and their actual physical installation in the control room identified no problems. This temporary MARs' expiration date has been extended from December 31, 1989, to around March 1990 at which time major diesel modifications are planned during refueling outage VII.

MAR 87-10-19-01A was issued to make certain changes to the above temporary MAR installation and make this final installation a permanent MAR for EDG-3A.

Inspection of modifications 2a. through 2d. to the above listed criteria resulted in no violations or derviations being identified.

e. Mar 87-11-27-01, Replace Valve MSV-303

This modification replaced the original Velan W5-274B 600#, one inch gate valve with a Hancock 950W, 600#, one inch gate valve. The modification was performed because the valve required replacement and an exact replacement was not available. The replacement valve was B31.1, sesmic category I, and the same material as the original valve. The inspector reviewed the completed modification package and the post modification test performed. No deficiencies were identified.

f. MAR 86-12-10-02T, BSP Flow Alarm Setpoint Change.

This temporary modification changed the flow control setpoint of the ractor building spray system from 1550 gpm to 1970 gpm. This modification was performed to ensure that the required reactor building spray flow rates were obtained after an actuation signal when the ES actuation signal was bypassed. This modification required the flow instrument loops to be respanned to 2500gpm; adjustment of the flow control setpoint; changeout of the remote indicator spans for the flow meters located in the control room; and adjustment of the High Flow alarm setpoint. This temporary modification was removed when MAR 86-12-10-03 was implemented. MAR 86-12-10-03 provided a seal in circuit for the ES actuation signal to the reactor buildings spray actuation logic. The inspectors reviewed the completed modification package and verified that the remote flow indicators setpoint and span had been returned to the original values prior to the temporary modification. No deficiencies were identifed.

g. MAR 85-09-04-01, Replace SFV 18 and 19.

This modification replaced spent fuel cooling valves 18 and 19 with new valves. SFV-18 was replaced with an Anchor Darling 10 inch, 150#, stainless steel globe valve and SFV-19 was replaced with an Anchor Darling, 10 inch, 150#, stainless steel flexible wedge gate valve. The original valves were Crane Co. solid wedge, 10 inch, 150#, stainless steel gate valves. The valves are manual containment isolation valves and were replaced because they had a history of failing local leak rate tests. The inspector reviewed the completed modification package, the post modification test performed and verified that SFV-19 was an Anchor Darling valve. NO deficiencies were identifed.

h. MAR 87-11-22-01, ISI Code Class Change to CHV-19.

This modification package changed the ISI code class boundary from chilled water valve CHV-18 to CHV-19. This modification was a documentation only change to provide a more practical boundary for ISI surveillance requirements. CHV-18 is a check value and CHV-19 is a manual isolation valve, immediately upstream of CHV-18. The inspectors reviewed the completed modification package, verified that the system P&ID had been updated to reflect the new ISI code boundary and verified that CHV-19 had been included in the plant surveillance procedures. No deficiencies were identified.

i. MAR 86-11-15-01, Repari MUT-1 Inlet Nozzle.

This modification repaired a small leak at the inlet pipe connection to the makeup tank. A pair of equalizing holes were drilled in the top of the tank then a pipe cap was welded onto the inlet nozzle and the tank wall forming the new pressure boundary of the nozzle connection rather than the original tank to nozzle weld. The inspectors reviewed the completed modification package and the post modification test performed. No deficiencies were identified.

j. MAR 86-06-25-01, Relocate RCV-142

This modification relocated valve RCV-142, a manually operated high point vent, to a more accessible location and added a pipe support to the piping arrangement. The inspectors reviewed the completed modification package and the post modification test performed. No deficiencies were identified.

k. MAR 85-06-14-01, SWHE Relief Valves.

This modificatin replaced all the relief valves on the Nuclear Services Heat Exchangers and the relief valves on the tube side of the Decay Heat closed cycle Heat Exchangers. The valves were replaced because the existing valves had deteriorated to a point where it was no longer practical to maintain them. The original valves were manufactured by Dresser and the replacement valves were manufactured by Crosby. The inspectors reviewed the completed modification package and verified that the installed relief valves were Crosby valves. During review of the package the inspectors determined that valves RWV 61 and 62 were replaced with relief valves with a relief capacity of 45gpm @ 10 percent overpressure and set pressure of 75 psig. The original valves had a relief capacity of 111 gpm. RWV61 and 62 are thermal relief valves and are utilized to protect the heat exchangers from overpressurization when the heat exchangers are isolated. The modification package did not continain sufficient justification for use of relief valves with less than 50 percent of the relieving capacity of the original valves installed. However, review of the system P&ID determined that the heat exchanger is continuously vented through a one inch line even when the heat exchanger is isolated for maintenance. Based on the fact that the heat exchanger is constantly vented the inspectors considered that the installed relief valves were adequate.

l. MAR 85-12-01-01T, CAV 434 Position Lights.

THis modification modified the wiring to the indication circuit for valve CAV 434 located inside the reactor building so that the correct indication logic for valve position could be monitored in the control room and at the PASS mimic panel in the count room. Subsequent to implementation of this temporary modification it was determined that wire number eight and nine of circuit CAE-56 were swapped inside the containment penetration and a MAR was initiated to make this temporary modification permanent. The inspectors reviewed the modification package and verified that valve CAV 434 did indicate properly in the control room. No deficienies were noted.

m. MAR 87-03-113-02, MOV Space Heater and Torque Bypass Switch Modification.

This design change modified the wiring of the limit switches on safety related MOVs. The design basis for the plant modification required that the valve opening torque switch bypass contact be set to open at 25 percent of valve full open position instead of the previous setting of 10 percent. This setting assured that the opening torgue switch would remain closed for sufficient valve travel in order to preclude the torque switch from de-energising the valve operator when the valve is opening from the full closed position under high differential pressure conditions. The scope of the plant modification included removal of other logic functions from the limit switch rotor containing the opening torgue switch bypass contact and disconnection of the limit switch compartment space heaters. MOVs within the following process systems were affected by this design change; AN, AS, BS, CA, CF, DH, DW, EF, FW, MS, MT, MU, and WD.

The inspectors reviewed the design output drawings and verified for select MOVs that the hardware changes included:

- °Rewiring the limit switch functions to spare contacts on other rotors to isolate the opening torque switch bypass contact on its own dedicated rotor.

- °Removal of the MOV position indicator lights from the MCC 120V control power supply and replacing these indicator lights with 24V lamps wired in paralled with the MCC 24V control power supply.

- °Installing auxilliary relays to multiply the limit switch contacts as required to obtain the proper limit switch configuration.

Additional reviews were performed of the Safety Evaluations, Design Verification Report, and Purchase Requisition prepared for procurement of auxilliary relays and 24V full voltage inicator lights. Post-modification tests and test acceptance criteria were verified as having been specified and the tests were performed prior to declaring the MOVs operable. Additionally, engineering evaluations completed in support of the design change to address seismic requirements and electrical load on the MCC control power supply were reviewed.

Within this area no violations or deviations were identified.

n. MAR 80-09-13-03, Diesel Generator Differential Protection Relaying Equipment.

This plant modification replaced existing G. E. Model 12 CFD relays used in the EDG differential protection relaying scheme with G. E. Model 1JD52A relays. The design change was made in response to IEN 85-83 which identified possible misoperation of the 12CFD relays when they are subjected to an impact in a de-energized state. The inspectors reviewed the completed MAR package and verified that the engineering instructions contained within the package accurately reflected information shown on referenced vendor drawings; that procurement requisition #2524 specified applicable technical standard for the replacement relays; that applicable surveillance and test procedures were revised; and that post installation calibration and functional tests were completed.

Within this area no violations or deviations were identified.

3. Exit Interview

The inspection scope and results were summarized on November 17, 1989, with those persons indicated in paragraph 1. Proprietary information is not contained in this report. Dissenting comments were not received from the licensee.

4. Acronyms and Initialisms

AN	Air Handling Vent and Cooling
ANSI	American National Standards Institute
AS	Auxiliary Steam
BS	Reactor Building Spray
CA	Chemical Additive and Sampling
CF	Core Flood
CR3	Crystal River Unit 3
CRDM	Control Rod Drive Mechanism
CRPD	Crystal River Plant Datum
DH	Decay Heat
DW	Demineralized Water
EDG	Emergency Diesel Generator
EF	Emergency Feedwater
FSAR	Final Safety Analysis Report
FW	Feedwater
G/C Inc.	Gilbert/Commonwealth Inc.
ISLT	Initial Service Leak Test
MAR	Modification Approval Record
MOV	Motor Operated Valve
MS	Main Steam
MT	Auxiliary Electrical Power
MU	Make Up

P&ID	Piping and Instrumentation Diagram
PASS	Post Accident Sampling System
SWV	Service Water Valve
TS	Technical Specification
WD	Waste Disposal
WR	Work Request