UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
61 FORSYTH STREET, SW, SUITE 23T85
ATLANTA, GEORGIA 30303-8931

October 31, 2007

Mr. Dale E. Young, Vice President
Crystal River Nuclear Plant (NA1B)
ATTN: Supervisor, Licensing & Regulatory Programs
15760 West Power Line Street
Crystal River, FL 34428-6708

SUBJECT: CRYSTAL RIVER NUCLEAR PLANT- NRC COMPONENT DESIGN BASIS INSPECTION REPORT 05000302/2007006

Dear Mr. Young:

On October 5, 2007, the U. S. Nuclear Regulatory Commission (NRC) completed an inspection at your Crystal River Nuclear Plant. The enclosed inspection report documents the inspection findings which were discussed with you on October 5, 2007, with Mr. M. Annacone and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, the inspectors identified two findings of very low safety significance (Green). These two findings were determined to involve violations of NRC requirements. However, because of their very low safety significance and because they are entered into your corrective action program, the NRC is treating these findings as Non-Cited Violations (NCVs) consistent with Section VI.A.1 of the NRC's Enforcement Policy. If you contest any of these NCVs you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the United States Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, U. S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at Crystal River.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Binoy B. Desai, Chief
Engineering Branch 1
Division of Reactor Safety

Docket Nos.: 50-302
License Nos.: DPR-72

Enclosure: NRC Inspection Report 05000302/2007006
w/Attachment: Supplemental Information

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ADAMS: ☐ Yes ACCESSION NUMBER:___

OFFICE	RII:DRS	RII:DRS	RII:DRS	RII:DRP	contractor	contractor	
SIGNATURE	RA	RA	RA	RA	RA	RA	
NAME	RBerryman	RAiello	WFowler	SVias	MShlyamberg	JLeivo	
DATE	10/25/2007	10/22/2007	10/18/2007	10/22/2007	10/24/2007	10/19/2007	10/ /2007
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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No.:	50-302
License No.:	DPR-72
Report No.:	05000302/2007006
Licensee:	Florida Power Corporation
Facility:	Crystal River Nuclear Plant
Location:	Crystal River, FL 34428
Dates:	September 10 - October 5, 2007
Inspectors:	R. Berryman, P.E., Senior Reactor Inspector (Lead) R. Aiello, Senior Operations Engineer W. Fowler, Reactor Inspector M. Shlyamberg, Contractor J. Leivo, Contractor
Accompanying personnel:	J. Helm, Reactor Inspector (trainee)
Approved by:	Binoy B. Desai, Chief Engineering Branch 1 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000302/2007006; 9/10/2007 - 9/14/2007, 9/24/2007 - 9/28/2007, 10/1/2007 - 10/5/2007; Crystal River Nuclear Plant; Component Design Bases Inspection.

This inspection was conducted by a team of three NRC inspectors and two NRC contractors. Two Green findings, all of which were Non-Cited Violations (NCVs) were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very low safety significance involving a violation of Technical Specifications (TS) 5.6.1 for failure to implement an adequate procedure for manual starting of the Control Complex Chilled Water Chiller Units (CHHE-1A/1B) following a Large Break Loss of Coolant Accident (LBLOCA). The chiller units are required to be restarted prior to 127 minutes after the accident to ensure adequate cooling to components within the control complex.

  This finding is more than minor because it affects the Procedure Quality attribute of the Mitigating Systems Cornerstone. It impacts the cornerstone objective of ensuring the availability, reliability, and operability of CHHE-1A/1B to perform the intended safety function during a design basis event. The vendor for CHHE-1A/1B provided a maximum temperature for restarting the chiller units of 104 degrees Fahrenheit (°F). The basis for this limitation is to prevent an inadvertent chiller unit trip due to high chiller freon condenser pressure. The inspectors assessed the finding using the SDP and determined that the finding was of very low safety significance (Green) because the inspectors found that Nuclear Services Closed Cycle Cooling (SW) temperature falls below 104 °F no later than 84 minutes after a LBLOCA. This affords operators at least 40 minutes to successfully restart the chiller units. This issue is documented in the corrective action program as nuclear condition report (NCR) 247908. This finding was reviewed for cross-cutting aspects and none were identified.(Section 1R21.2.3)

- Green. The inspectors identified a finding of very low safety significance involving a violation of 10 CFR 50, Appendix B, Criterion XI, Test Control, for failure to implement a test program which accounted for the effects of instrument uncertainty on surveillance testing of Emergency Feedwater Pump (EFP)-2 in accordance with the approved In-service Testing (IST) program.

  This finding is more than minor because it affects the Procedure Quality attribute of the Mitigating Systems Cornerstone. It impacts the cornerstone objective of ensuring the availability, reliability, and operability of EFP-2 to perform the intended safety function during a design basis event. The inspectors assessed the finding using the SDP and determined that the finding was of very low safety significance (Green) because the inspectors found no documented history of in-service failures of EFP-2 rendering safety-related equipment inoperative. This issue is documented in the corrective actions program as NCR 248036. This finding was reviewed for cross-cutting aspects and none were identified. (Section 1R21.2.7)

B. Licensee-Identified Violations

None

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Mitigating Systems and Barrier Integrity

1R21 Component Design Bases Inspection (71111.21)

.1 Inspection Sample Selection Process

The inspectors selected risk significant components and operator actions for review using information contained in the licensee's Probabilistic Risk Assessment (PRA). In general, this included components and operator actions that had a risk achievement worth factor greater than two or Birnbaum value greater than 1 X10E-6. The components selected were located within the emergency feedwater (EFW) system, SW system, emergency diesel generators (EGDG), 4160 VAC electrical system, 125 VDC electrical system, and the offsite power system. The sample selection included 19 components, five operator actions, and five operating experience items. Additionally, the inspectors reviewed one modification by performing activities identified in IP 71111.17, "Permanent Plant Modifications," Section 02.02.a. and IP 71111.02, "Evaluations of Changes, Tests, or Experiments."

The inspectors performed a margin assessment and detailed review of the selected risk-significant components to verify that the design bases have been correctly implemented and maintained. This design margin assessment considered original design issues, margin reductions due to modification, or margin reductions identified as a result of material condition issues. Equipment reliability issues were also considered in the selection of components for detailed review. These included items such as failed performance test results, significant corrective action, repeated maintenance, maintenance rule (a)1 status, Regulatory Issue Summary 05-020 (formerly Generic Letter 91-18) conditions, NRC resident inspector input of problem equipment, system health reports, industry operating experience and licensee problem equipment lists. Consideration was also given to the uniqueness and complexity of the design, operating experience, and the available defense in depth margins. An overall summary of the reviews performed and the specific inspection findings identified are included in the following sections of the report.

.2 Results of Detailed Reviews

.2.1 Shutdown Reactor Coolant System (RCS) Level Instrumentation

a. Inspection Scope

The team reviewed Design Basis Documents (DBDs), Generic Letter (GL) responses, Operating Procedures (OPs), calculations, plant layout drawings, calibration requirements, and plant operating curves to verify the shutdown RCS level instrumentation is capable of performing the design basis function of providing both local and remote level indications during shutdown conditions. In addition, alarm setpoints associated with the remote instrumentation and Decay Heat Pump (DHP) amperage fluctuations were reviewed to verify the ability to provide prompt indication of vortexing or inadequate Net Positive Suction Head (NPSH) when in reduced RCS inventory conditions. Calculations used to develop plant operating curves for the minimum allowable RCS level to protect against vortexing or exceeding DHP NPSH limits during midloop conditions were reviewed to verify instrument inaccuracies were taken into account. Instrument uncertainties for the remote level indications were reviewed to verify that instrument calibrations were maintaining the validity of the curves and that component aging is not affecting the effectiveness of the calibration frequency. Procedural guidance associated with a Loss Of Offsite Power (LOOP) event while in a midloop or reduced inventory conditions were reviewed to verify the ability to maintain indications of RCS level if power to remote level indication was lost.

b. Findings

No findings of significance were identified.

.2.2 Reactor Vessel Internals

a. Inspection Scope

The team reviewed DBDs, system descriptions, vendor manuals, Annunciator Response Procedures (ARP), OPs, and corrective action documentation to verify that the loose parts monitoring system could provide indication of loose parts in the RCS and that structural integrity of reactor vessel internals required during accident conditions was being maintained. Equipment calibration and environmental qualifications were reviewed to verify that equipment deficiencies for loose part monitoring equipment were being tracked and that equipment was qualified for the postulated radiation fields. System layout drawings were reviewed to verify that accelerometer locations were not obstructed by installed equipment. Condition investigation documents and contingency plans associated with baffle-to-baffle, baffle-to-former plate, and former plate to core barrel bolts were reviewed to verify that corrective actions were adequate and that RCS monitoring equipment was utilized to identify any current fuel cycle issues such as neutron noise measurements. Structural analyses on baffle, former plate, and core barrel stresses experienced during faulted loading conditions were reviewed to verify their applicability and that the results would bound the design requirements identified in the safety analysis including maintaining the core within a coolable geometry

b. Findings

No findings of significance were identified.

.2.3 Chilled Water Pump - CHP-1A

a. Inspection Scope

The team reviewed DBDs, Emergency Operating Procedures (EOPs), calculations, and corrective action documents to verify that the chilled water pump and its associated chiller package could provide adequate flow of chilled water to various air handling units during emergency conditions. Calculations for the control complex and SW system transient temperature responses after a LBLOCA were reviewed to verify that Environmental Qualification (EQ) limits within the control complex would not be exceeded prior to restoration of the control complex chiller units. OPs were reviewed to verify that they contained guidance regarding the maximum allowable time to restore chilled water flow and the time delay interlocks associated with the chiller trips that were expected immediately following certain LBLOCA scenarios. Additionally, the OPs were reviewed to verify that the maximum SW cooling system temperature for starting a chiller was stated. Walkdowns were conducted of control building chiller packages to verify that the chilled water pumps were not being impacted by corrosion and that controls were in place and being maintained to address any condensation on piping or components.

b. Findings

Introduction. The inspectors identified a finding of very low safety significance involving a violation of TS 5.6.1 for failure to implement an adequate procedure for manual starting of CHHE-1A/1B following a LBLOCA. The chiller units are required to be restarted prior to 127 minutes after the accident to ensure adequate cooling to components within the control complex.

Description. CHHE-1A/1B receives cooling water flow from the SW system. The SW system also provides cooling water flow to the Reactor Building Cooling Units (RBCUs). Following a postulated LBLOCA, the heat load from the RBCUs will cause an increase in SW temperature to a peak of near 108 °F. This increase in SW temperature is sufficient to cause CHHE-1A/1B to trip due to high condenser pressure. Heat loads within the Control Complex require CHHE-1A/1B to be restarted by 127 minutes following a LBLOCA to ensure adequate cooling to components. The vendor for CHHE-1A/1B provided a maximum temperature for restarting the chiller units of 104 °F. The basis for this limitation is to prevent an inadvertent chiller unit trip due to high chiller freon condenser pressure which would result in the chiller being unavailable for 30 minutes due to the anti-recycle protective feature on the chiller units. Following a postulated LBLOCA, SW temperature will exceed this value for up to 84 minutes.

EOP-14, Emergency Operating Procedure Enclosures, Enclosure 17, directs operators to manually start the chiller units following a LBLOCA. However, this procedure does not provide any limitations to prevent attempting to start the chiller units until SW temperature is below 104 °F. Additionally, this procedure did not address the potential for an inadvertent chiller unit trip and subsequent 30 minutes of unavailability due to the anti-recycle timer.

Analysis. Progress Energy's failure to develop an adequate procedure for restart of CHHE-1A/1B following a LBLOCA was a performance deficiency. This finding is more than minor because it affects the Procedure Quality attribute of the Mitigating Systems Cornerstone. The EOP was validated to take 57 minutes for operators to reach the step to restart the chiller units. This would afford the operators up to a 70 minute window to perform this action. An inadvertent chiller unit trip would result in a subsequent 30 minutes of chiller unit unavailability during a design basis event during which they are credited to ensure adequate cooling of Mitigating Systems equipment. This impacts the cornerstone objective of ensuring the availability, reliability, and operability of CHHE-1A/1B to perform the intended safety function during a design basis event by effectively reducing the time window for operators to be able to perform the action to restart the chiller units from as much as 70 to as low as 40 minutes. This represents up to a 42% reduction in the time window for operators to restart the chiller units. Additionally, this condition resulted in a potential inadvertent shutdown and subsequent restart of the chiller units which would provide additional opportunity for component failures. The inspectors assessed the finding using the SDP and determined that the finding was of very low safety significance (Green) because the inspectors found that SW temperature decays below 104 F no later than 84 minutes after a LBLOCA. Operator training and a selected sample of on the spot interviews with operators confirmed solid understanding of the chiller unit indications and the ability to immediately recognize a 30 minute anti-recycle lockout of a chiller unit. Additionally, the only duties assigned to the operator assigned to perform EOP-14, Enclosure 17, are to restore control complex ventilation and cooling following a postulated LBLOCA. An operator attempting to start the chiller at 57 minutes after a LBLOCA was assumed to cause a 30 minute anti-recycle lockout of the chiller units. This would make the chillers unavailable until 87 minutes after a LBLOCA. This affords operators at least 40 minutes to successfully restart the chiller units. This finding was reviewed for cross-cutting aspects and none were identified.

Enforcement. TS 5.6.1 “Procedures”, requires in part that written procedures be established, implemented and maintained per Regulatory Guide (RG) 1.33, Rev. 2, “Quality Assurance Program Requirements”. Appendix A of RG 1.33 states, in part, that procedures for combating emergencies and other significant events, such as loss of cooling to individual components, shall be covered by written procedures. Contrary to the above, Progress Energy did not provide sufficient guidance in EOP-14, Enclosure 17 to restart the chiller units following a LBLOCA. This condition has existed since plant initial operation. Because this finding is of very low safety significance and was entered into Progress Energy's corrective action program as NCR 247908, this finding is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000302/2007006-01, Violation of Technical Specification 5.6.1 for Failure to Implement an Adequate Procedure for Manual Starting of the Control Complex Chilled Water Chiller Units (CHHE-1A/1B) Following a LBLOCA)

.2.4 Auxiliary Building Flood Isolation Valves

a. Inspection Scope

The team reviewed plant layout drawings, abnormal operating procedures, and pipe rupture analyses for line breaks outside containment to verify that critical actions can be implemented to combat the effects of a moderate energy line break in the auxiliary building. OPs for isolating line breaks were reviewed and compared to moderate energy line rupture analyses to verify that the inclusion of isolating critical lines was identified. Availability of process valves utilized by procedures for isolating ruptured lines were reviewed to verify they would be available during flooding scenarios. Auxiliary building walkdowns were performed to verify various sump level instruments and associated sump pumps were being properly maintained.

b. Findings

No findings of significance were identified.

.2.5 Emergency Diesel Generators - EGDG-1A & EGDG-1B (Mechanical)

a. Inspection Scope

The team reviewed DBDs, drawings, calculations and engineering change packages to verify the capability of the EGDGs to supply onsite AC electrical power during design basis events. Engineering change packages and procurement requirements associated with the conversion to ultra low sulfur diesel fuel oil were also reviewed to verify that EGDG operation was not adversely impacted. EGDG building heat load and ventilation calculations were reviewed to verify that the ventilation system was adequately sized. Diesel engine radiator and turbo charger intercooler heat removal calculations and chemistry requirements were reviewed to verify that adequate cooling was available to meet EGDG loading requirements with consideration given to EGDG room ventilation recirculation air flow. Corrective actions associated with fuel header depletion during standby conditions were reviewed to verify that modifications were adequate and that changes to the plant are being performed under the appropriate procedures. Walkdowns and interviews with plant personnel were performed to verify that the capability of the EGDGs to perform their design basis function was being maintained.

b. Findings

No findings of significance were identified.

.2.6 Nuclear Services Closed Cycle Cooling Pump (SWP-1A) and Nuclear Service Closed Cycle Cooling Heat Exchanger (SWHE-1B)

a. Inspection Scope

The team reviewed DBDs, calculations, and drawings to verify that design requirements were satisfied related to flow, developed head, available NPSH, vortex formation, minimum flow requirements, and runout protection for all SW system operating conditions. The SW flow assumptions in the Updated Final Safety Analysis Report (UFSAR) accident analysis were also reviewed to verify that they were appropriate. Design calculations were reviewed to verify that the SW system design performance requirements were satisfied. Calculations were also reviewed to verify that adequate NPSH was available and that appropriate measures were taken to prevent suction vortexing. The team reviewed completed pump surveillance tests to assess the adequacy of American Society of Mechanical Engineers (ASME) Section XI testing for SWP-1A to ensure compliance with the approved IST Program. Maintenance, corrective actions, and design change history were reviewed to assess the system for potential component degradation and subsequent impacts on design margins or performance. The team also reviewed the heat transfer calculations, past testing results of SWHE-1B, periodic inspections, tube plugging limits, and current tube plugging to verify that design basis heat removal requirements, capability, and flow rates were satisfied.

b. Findings

No findings of significance were identified.

.2.7 Turbine Driven Emergency Feedwater Pump - EFP-2

a. Inspection Scope

The team DBDs, calculations, and drawings to verify that design requirements related to flow, developed head, available NPSH, vortex formation, minimum flow and runout protection for all Emergency Feedwater (EF) system operating conditions were appropriate. The EF flow assumptions in the UFSAR accident analysis were also reviewed to verify that the pump was capable of performing the intended safety functions. Calculations were also reviewed to verify that available NPSH and measures taken to prevent suction vortexing were adequate. The team reviewed completed pump surveillance tests to assess the adequacy of ASME Section XI IST for EFP-2 and to verify compliance with approved IST program. Maintenance, corrective actions, and design change history were reviewed to assess the system for potential component degradation and subsequent impacts on design margins or performance.

b. Findings

Introduction. The inspectors identified a finding of very low safety significance involving a violation of 10 CFR 50, Appendix B, Criterion XI, Test Control, for failure to implement a test program which accounted for the effects of instrument uncertainty on surveillance testing of EFP-2 in accordance with the approved IST program.

Description. The review of design calculations M-96-0063, EF System Hydraulic Analysis, Rev. 6 and M-02-0002, Minimum Allowable Differential Pressure (dP) for EF Pump Performance, Rev. 5; and surveillance procedure SP-640B, EFP-2 Full-Flow Test, Rev. 16 identified that calculation M-02-0002 incorrectly translated the required EF surveillance requirements into SP-640B as 580 gpm at 1236 psid without accounting for any instrument uncertainty. Calculation M-96-0063 established the minimum required condition for the EFP-2 surveillance based on the maximum EF flow rate used in the accident analysis without accounting for any instrument uncertainty. The NRC issued a Safety Evaluation on January 22, 1999 which provided relief from the requirement to perform flow measurements during quarterly IST testing. The premise of this relief was the rigorous full flow testing that was accomplished by procedure SP-640B.

The licensee initiated NCR 248036 and performed an operability evaluation ED 0000068121R0 to address the team's concerns. The ED revised the instrument uncertainty value previously calculated in calculation M-96-0063 and corrected the acceptance criterion for test SP-640B.

Analysis. Progress Energy's failure to take instrument uncertainty into account during testing of EFP-2 in accordance with the approved IST program was a performance deficiency. This finding is more than minor because it affects the Procedure Quality attribute of the Mitigating Systems Cornerstone. It impacts the cornerstone objective of ensuring the availability, reliability, and operability of EFP-2 to perform the intended safety function during a design basis event. Reasonable doubt existed regarding the operability of EFP-2 when appropriate instrument uncertainty was applied to the results of test SP-640B pending the results of the operability evaluation. The conclusion of the operability evaluation was that EFP-2 was capable of performing the intended safety function. However, only 2.5 psid of margin existed when expressed in terms of pump developed head. This represented less than 0.2% when compared to the latest results of test SP-640B. The inspectors assessed the finding using the SDP and determined that the finding was of very low safety significance (Green) because the inspectors found no documented history of in-service failures of EFP-2 rendering safety-related equipment inoperative. This finding was reviewed for cross-cutting aspects and none were identified.

Enforcement. 10 CFR 50, Appendix B, Criterion XI, Test Control stated, in part, that test programs shall be established to assure that all testing required to demonstrate that structures, systems and components will perform satisfactorily in service and that results shall be documented and evaluated to assure that test requirements have been satisfied. Contrary to the above, Progress Energy did not evaluate the effects of instrument uncertainty on the results of surveillance testing of EFP-2 in accordance with procedure SP-640B. This condition has existed since original plant licensing. Because this finding is of very low safety significance and was entered into Progress Energy's corrective action program as NCR 248036, this violation is being treated as a non-cited violation (NCV), consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000302/2007006/2007006-02, Violation of 10 CFR 50, Appendix B, Criterion XI for Failure to Account for Instrument Uncertainty During EFP-2 Testing)

.2.8 Service Water Surge Tank and Level Instrumentation (SWT-1)

a. Inspection Scope

The team reviewed DBDs to identify the SWT-1 design requirements related to level, pressure, temperature, support of an adequate NPSH for SW pumps, and vortex prevention. The SW flow assumptions in the UFSAR accident analysis were also reviewed to verify that they were appropriate. Process and instrument error calculations were reviewed to verify that level, pressure, and temperature would assure adequate NPSH and vortex prevention. Maintenance, corrective actions, and design change histories were reviewed to assess the system for potential component degradation and subsequent impacts on design margins or performance.

b. Findings

No findings of significance were identified.

.2.9 Vacuum Breaker and Relief Valve for Service Water Surge Tank (SWV-199 & 278)

a. Inspection Scope

The team reviewed the DBDs to identify design requirements related to the function of valves SWV-199 and SWV-278. The team verified that the relief valve setpoint was appropriate. The team reviewed calculations to verify that the relief valve setpoint would not be exceeded during postulated accident conditions. Maintenance, corrective actions, and design change history were reviewed to assess the valves for potential component degradation and subsequent impacts on design margins or performance.

b. Findings

No findings of significance were identified.

.2.10 Service Water Surge Tank Isolation Valve (SWV-1)

a. Inspection Scope

The team reviewed DBDs to identify design requirements related to the function of SWV-1. The team reviewed the design to verify that SWV-1 would allow adequate flow to assure adequate NPSH and vortex prevention to the rest of the SW system. Maintenance, corrective actions, and design change history were reviewed to assess SWV-1 for potential component degradation and subsequent impacts on design margins or performance.

b. Findings

No findings of significance were identified.

.2.11 SWP-1A Discharge Butterfly Valve (SWV-5)

a. Inspection Scope

The team reviewed DBDs to identify design requirements related to the function of SWV-5. The team reviewed the design to verify that SWV-5 would allow adequate flow to assure adequate NPSH and vortex prevention to the rest of the SW system. Maintenance, corrective actions, and design change history were reviewed to assess SWV-5 for potential component degradation and subsequent impacts on design margins or performance.

b. Findings

No findings of significance were identified.

.2.12 Backup Engineered Safeguards Transformer - MTTR-6

a. Inspection Scope

The team reviewed calculations to verify that loading and protective settings for the transformer were appropriate considering the equipment ratings and specifications. The review was also conducted to verify that loading and electrical protection margins were consistent with the design basis conditions that could be postulated during accidents and that the transformer would not be overloaded or prematurely isolated during design basis events. The team also reviewed the electrical protection schemes to verify that the circuits satisfied the design basis for independence of offsite sources. In addition, the team reviewed and discussed with the system engineer the preventive maintenance procedures, most recent Doble electrical testing results, system health reports for the past two years, selected corrective actions reports covering the last three years, and dissolved gas analysis results for the transformer to assess transformer performance. The team also visually inspected external portions of the transformer to observe the visible material condition.

b. Findings

No findings of significance were identified.

.2.13 Startup Transformer - MTTR-2

a. Inspection Scope

The team reviewed the calculations that determined the loading and protective settings for MTTR-2 to verify that they were appropriate considering the ratings and specifications. This review was also conducted to verify that loading margins and electrical protection margins were such that MTTR-6 would not be prematurely isolated as a result of design basis loading and protection conditions for MTTR-2. In addition, the team reviewed and discussed with the system engineer the preventive maintenance procedures, most recent Doble electrical testing, dissolved gas analysis results for the transformer, system health reports for the past two years, and selected corrective action reports covering the past three years to assess transformer performance. The team also visually inspected external portions of the transformer to observe the visible material condition.

b. Findings

No findings of significance were identified.

.2.14 4160 VAC Circuit Breakers 3207, 3210, 3211, 3220, 3221

a. Inspection Scope

The team reviewed the calculations that determined the loading and protective settings for the breakers to verify that they were appropriate considering the equipment ratings and specifications and to verify that loading and electrical protection margins were consistent with the conditions that could be imposed for design basis events and that the breakers would not trip prematurely for those events. In addition, the team visually inspected external portions of the 4160 VAC switchgear to observe visible material condition and potential vulnerabilities to hazards or interactions. The team also reviewed the health reports for the past two years, corrective action history covering the past three years, and reviewed the results of the most recent preventive maintenance activities for the breakers and their electrical protection, including inspection, testing, calibration, and refurbishment to assess breaker performance.

b. Findings

No findings of significance were identified.

.2.15 DC Distribution Panel 1B

a. Inspection Scope

The team reviewed DC loading and short circuit calculations to verify that they were appropriate when considering the panel ratings as well as selected fuse ratings and characteristics. In addition, the team visually inspected external portions of the panel to observe material condition and potential vulnerabilities to hazards or interactions. The team also reviewed and discussed with the system engineer the health reports for the past two years, corrective action history covering the past three years for the panel, and reviewed the results of the most recent preventive maintenance activities for the panel to assess component performance.

b. Findings

No findings of significance were identified.

.2.16 Reactor Coolant System Power Operated Relief Valve (PORV) Control Switch and Circuitry - PS-8

a. Inspection Scope

The team reviewed schematic diagrams and discussed with the system engineer the design, testing procedures, and most recent test results for the PORV control circuits with respect to failure vulnerabilities including the potential for failures undetectable by test. In addition, the team reviewed the calculations that established the basis for the PORV setpoints and instrument uncertainties to verify that they were appropriate. The team also reviewed the basis for selected circuit device ratings, health reports for the past two years, and corrective action history covering the past three years for the circuits and devices to assess component performance.

b. Findings

No findings of significance were identified.

.2.17 ES Actuation Switches and Circuitry

a. Inspection Scope

The team reviewed schematic diagrams for the ES actuation and reset control circuits to verify that previously unevaluated single failure vulnerabilities or the potential for failures undetectable by testing did not exist. The review also included a review and discussion with the system engineer of the most recent test results for the circuitry. The team focused on verifying that the potential for a single failure resulting in the inability to reset an invalid ES signal did not exist. The team reviewed the health reports and the past three years of corrective action history for the circuits and devices to assess component performance. The team also observed the operation of these circuits for a simulated invalid ES actuation event as performed on the training simulator to verify that their representation on the simulator matched those installed in the actual control room.

b. Findings

No findings of significance were identified.

.2.18 Auxiliary Building Flooding Alarms and Indications

a. Inspection Scope

The team reviewed schematic diagrams, installation details, and testing of decay heat pit sumps ‘A’ and ‘B’ level alarms (WD-133-LS, WD-134-LS); auxiliary building sump level alarm (WD-132-LS); nuclear service cooler area sump level alarm (SD-5-LS); and tendon access gallery sump level alarm (SD-6-LS). This review was done to verify the existing level setpoints were appropriate considering the flooding design basis described in the UFSAR. The team also reviewed calibration procedures and results, health reports, and the past three years of corrective action history for the circuits and devices to assess component performance. In addition, the team performed non-intrusive visual inspections of portions of the level instruments and configurations to observe the material conditions.

b. Findings

No findings of significance were identified.

.2.19 Emergency Diesel Generators - EGDG-1A & EGDG-1B (Electrical)

a. Inspection Scope

The team reviewed the EGDG-1B loading analysis for a design basis loss of coolant accident concurrent with a LOOP. This review included the steady-state and transient electrical loading calculations to verify that the design capability of the EGDG to accelerate and support the required loads within specified times was consistent with the design basis and operating procedures. This review included consideration of the available margins with respect to the ratings for continuous operation as well as cumulative 30 minute, 200 hour, and 2000 hour ratings as well as for the starting of motors during block loading. The team reviewed the design input calculations and loading scenarios for the SW pumps, chilled water pumps, raw water pumps, and makeup pumps to verify that design inputs had been correctly translated into the electrical loading calculations. The team reviewed engineering evaluations that accounted for tolerances in EGDG voltage regulator and governor (frequency) setpoints as well as uncertainties in kilowatt testing and measurements used in supporting the analyses. The team reviewed the results of the last three fast-start tests and the last two maximum load tests required by the TS for both EGDGs to verify that the acceptance criteria were consistent with the analytical basis and that the test results were properly evaluated. The team reviewed the health reports for the past two years, corrective action history covering the past three years for both EGDGs, and performed non-intrusive visual inspections of the EDGs and electrical auxiliaries to observe the material condition. In addition, the team reviewed calculations that established the bases for protective trips that would be active during postulated accident conditions and reviewed the most recent tests and calibrations of these protective circuits to verify that they were appropriate.

b. Findings

No findings of significance were identified.

.3 Review of Low Margin Operator Actions

a. Inspection Scope

The inspectors performed a margin assessment and detailed review of five risk significant and time critical operator actions. Where possible, margins were determined by the review of the assumed design basis and UFSAR response times and performance times documented by Job Performance Measures (JPMs). For the selected components and operator actions, the inspectors performed an assessment of the Emergency Operating Procedures (EOPs), Abnormal Operating Procedures (APs), Alarm Response Instructions (ARs), and other operations procedures to determine the adequacy of the procedures and availability of equipment required to complete the actions. Operator actions were observed on the plant simulator and during plant walkdowns.

The following operator actions were observed on the licensee's operator training simulator:

- Recovery of offsite power with and without station batteries
- Loss of Decay Heat Removal (DHR) while shutdown
- Termination of Safety Injection (SI) following spurious actuation prior to plant going solid

Additionally, the inspectors walked down, "table-topped" and reviewed the following operational scenarios:

- Alignment of dedicated chilled water system to Emergency Feedwater Initiation Control (EFIC)
- Isolation of auxiliary building during flooding

b. Findings

No findings of significance were identified.

.4 Review of Industry Operating Experience

a. Inspection Scope

The inspectors reviewed selected operating experience issues that had occurred at domestic and foreign nuclear facilities for applicability at the Crystal River Nuclear Plant. The inspectors performed an independent applicability review, issues that were identified as applicable to the Crystal River Nuclear Power Plant were selected for a detailed review. The issues that received a detailed review by the inspectors included:

- NRC Information Notice (IN), IN 86-14, Overspeed Trips of HPCI, RCIC and AFW Turbines
- IN 2005-025, Inadvertent Reactor Trip and Partial Safety Injection Actuation Due to Tin Whisker

- IN 90-25, Loss of Vital AC Power with Subsequent Reactor Coolant System Heat-up

- NRC Bulletin 96-01 and IN 96-12, Control Rod Insertion Problems

- NRC Bulletin 88-04, Potential Safety Related Pump Loss

b. Findings

No findings of significance were identified.

.5 Review of Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed one modification related to the selected risk significant components in detail to verify that the design bases, licensing bases, and performance capability of the components have not been degraded through modifications. The adequacy of design and post modification testing of these modifications was reviewed by performing activities identified in IP 71111.17, Permanent Plant Modifications," Section 02.02.a. Additionally, the inspectors reviewed the modifications in accordance IP 71111.02, "Evaluations of Changes, Tests, or Experiments," to verify the licensee had appropriately evaluated them for 10 CFR 50.59 applicability. The following modification was reviewed:

EC 61028/61030, EGDG-1A/1B Fuel Header Modification to Prevent Fuel Header Depletion

b. Findings

No findings of significance were identified.

4. **OTHER ACTIVITIES**

4OA6 Meetings, Including Exit

Exit Meeting Summary

On October 5, 2007, the inspectors presented the inspection results to Mr. M. Annacone, Plant General Manager, and other members of the licensee staff. The inspectors returned all proprietary information examined to the licensee. No proprietary information is documented in the report.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

M. Annacone, Plant General Manager
S. Cahill, Engineering Manager
E. Cochran, Senior Engineer, Electrical/I&C Systems Engineering
M. Culver, Lead Engineer, Reactor Systems Engineering
B. Doran, Engineer II, Electrical/I&C Systems Engineering
J. Endsley, Lead Engineer, Electrical/I&C Design Engineering
A. Hartwig, Supervisor, Electrical/I&C Systems Engineering
D. Herrin, Supervisor, Licensing & Regulatory Programs (Acting)
T. Howard, Supervisor, Equipment Performance Engineering
C. Miller, Senior Engineer, Mechanical/Civil Design Engineering
B. Morrow, Engineer II, Electrical/I&C Systems Engineering
E. Ortolan, Supervisor, Mechanical/Civil Design Engineering
M. Rahman, Senior Engineer, Electrical/I&C Design Engineering
T. Stewart, Senior Engineering Technical Support Specialist, Electrical/I&C Systems Engineering
R. Tyrie, Superintendent, Shift Operations
D. Wise, Lead Engineer, Mechanical/Civil Design Engineering

NRC

T. Morrissey, Senior Resident Inspector, Crystal River Nuclear Plant
B. Desai, RII, Engineering Branch 1, Chief

ITEMS OPENED, CLOSED, AND DISCUSSED

Open/Closed

05000302/2007006-01	NCV	Violation of Technical Specification 5.6.1 for Failure to Implement an Adequate Procedure for Manual Starting of the Control Complex Chilled Water Chiller Units (CHHE-1A/1B) Following a LBLOCA. (Section 1R21.2.3)
05000302/2007006-02	NCV	Violation of 10 CFR 50, Appendix B, Criterion XI for Failure to Account for Instrument Uncertainty During EFP-2 Testing. (Section 1R21.2.7)

DOCUMENTS REVIEWED

Calculations

E-00-0003, Emergency Diesel Load: Total Pump Motor kW Load Uncertainty Analysis, Rev. 8
E-03-0004, 4.16 kV ES Buses Protective Relay Settings Calculation [for breakers 3207, 3211], Rev. 0
E-90-0070, Protective Relays Setting Calculations for Startup and Unit Auxiliary Transformers, Plant Line 4, and Main Generator 3, Rev. 1
E-90-0071, Protective Relays Setting Calculations for EDG-1A and EDG-1B, Rev. 2
E-90-0077, Safety Related Busses Scenario Based Steady State Voltage Drop and Load Flow Analysis from Offsite Power Sources, Rev. 3
E-90-0078, 480 Volt Protection Device Coordination Study for ES Bus 3A (MTSW-3F), ES Bus 3B (MTSW-3G) and Plant Auxiliary Bus 3 (MTSW-3J) [for breakers 3220, 3221], Rev. 4
E-90-0101, Class 1E DC System Short Circuit Calculation, Rev. 2
E-90-0102, Electrical DC System Safety Related Coordination Calculation
E-90-0105, DC System Revalidation Master Database, Rev. 3
E91-0018, ES Block Loading Transient Motor Starting Analysis under Degraded Voltage Conditions, Rev. 2
E-91-0027, EGDG-1B Scenario Based Loading, Voltage Drop, Frequency Dip, and Transient Motor Starting Analysis, Rev. 4
E-92-0206, CR3 Startup Transformer Differential Relaying, Rev. 0
E-92-0207, Backup Engineered Safeguards Transformer Differential Relaying CTs, Rev. 1
EC-66279, EDG Loading Scenarios Evaluation to Account for Frequency, Voltage Variation, and KW Testing Uncertainties (AR 213617), Rev. 1
F03-0005, Decay Heat Study
F-97-0014, CR-3 Spent Fuel Pool Temperature Rise From Fuel in the Pool Calculation
F-98-0019, Loss of Feedwater Analysis for CR-3, Rev. 3
F-99-0004, Time to Boil, Time to 200°F, Time to Saturate, and Time to Uncovery
H-93-0001, EDG Minimum Ventilation Design Basis, Rev. 2
H-97-0001, Control Complex Transient Temperature Response, Rev. 7
I-85-0004, EFW Tank Level Accuracy, Rev. 7
I-85-0005, EFW Tank Volume, Rev. 5
I-90-1017, Reactor Vessel Shutdown Level Instrument LOOP Tolerances, Scaling, and Setpoints, Rev. 3
I-92-0020, Instrumentation Error Calculation For Condenser 3A and 3B, Rev. 0
I-97-0015, RC Low Range Pressure Loop Accuracy - RC-147-PT, RC-148-PT, RC-132-PT, Rev. 2
M-92-0065, SW System Hydraulic Analysis, Rev. 4
M-96-0063, EF System Hydraulic Analysis, Rev. 6
M-96-0069, ES Pump Maximum Flow for EDG Loading [review limited to translation of BHP design inputs for selected loads into electrical loading calculations], Rev. 2
M-97-0079, DH Pump Vortex and NPSH limits with RCS open to Atmosphere, Rev. 1
M-97-0081, DH Pump Vortex limits with RCS Open to atmosphere, Rev. 1
M-97-0133, SW Heat Loads During LBLOCA and SW Temperature Decay Times, Rev. 11
M-97-0155, EDG Radiator/Fan Analysis – 15 F to 105 F Radiator Room Intake Air Temperature, Rev. 1

M-99-0027, CR-3 Emergency Feedwater System Hydraulic Design Verification Analysis, Rev. 7
M-02-0002, Minimum Allowable dP for ES Pump Performance, Rev. 5
M-02-0005, UHS Temperature and RWP Pressure Limit, Rev. 0
M-06-0002, Control Complex Chiller Load vs. Service Water Temperature, Rev. 0

Operating Procedures

AI-504, Guidelines for Cold Shutdown and Refueling, Rev 21
AP-330, Loss of Nuclear Service Water, Rev 19
AP-404, Loss of Decay Heat Removal, Rev. 12
AP-520, Loss of RCS Coolant or Pressure, Rev. 11
AP-770, Emergency DG Actuation, Rev 36, Section 4.0, Enclosure 1, Recovery of Faulted ES Bus
AP-880, Fire Protection, Rev 25
AP-990, Shutdown form Outside the Control Room, Rev 0, Rev 0
AP-990, Shutdown from Outside the Control Room, Rev 25, Section 4.0, Enclosure 4
AP-1040, Aux Building Flooding, Rev 0
AR-501, Loose Parts Monitoring Trouble, Rev. 33
EOP-12. Station Blackout, Rev 9
EOP-14, Emergency Operating Procedure Enclosure, Rev. 21
OP-103B, Plant Operating Curves, Rev 39
OP-103H, Reactor Coolant System and Spent Fuel Pool Decay Heat Tables and Figures, Rev 7
OP-301, Operation of the Reactor Coolant System, Rev 92
OP-301A, Refueling Outage RCS Drain and Fill Operations, Rev. 8
OP-404, Decay Heat Removal System, Rev 144
OP-506, Loose Parts Monitoring System, Rev. 11
OP-880A, Appendix R Chiller Start, Rev 11
SAF-ESGX-00002,Energy Supply Fall Protection Procedure, Rev 1
SP-149, Power-Operated Relief Valve Channel Calibration / Test, Rev. 4
SP0354A, Monthly Functional Test of the Emergency Diesel Generator EGDG-1A, Sections 4.2 through 4.6, Rev. 83
SP0354B, Monthly Functional Test of the Emergency Diesel Generator EGDG-1B, Sections 4.2 through 4.6, Rev. 76
REG-NGGC-0010, 10 CFR 50.59 and Selected Regulatory Reviews, Rev. 10

Operations Training Related Documents

In-Plant JPM 395, Appendix R Chiller Lineup
Admin JPM 369 and 376, Perform Time to Boil / Core Uncovery Calculations
Simulator JPM 404, Respond to an Inadvertent ES Actuation
Training Lesson Plan OPS-5-78, Rev 4, Operation of the Reactor Coolant System (OP-301)
Training Lesson Plan OPS-5-125, Rev 1, AP-340 Invalid Engineered safeguards Actuation
LOR Scenario Exercise Guide LOR-1-21, Water box Tube failure
LOCT Weekly Wrap Up, Crew A1, B1,B2, C2, D1, D2, E1, and E2,Cycle 05-06-09 (09/26 - 09/29/2006)
Biennial LOCT Exam Results 2005-2006 Cycle

Test Procedures

SP-344A, SWP-1A and Valve Surveillance, Rev. 49
SP-349B, EFP-2 and Valve Surveillance, Rev. 53
SP-640B, EFP-2 Full-Flow Test, Rev. 16
MAR 96-10-02-01, MAR Functional Procedure EF Cavitating Venturis, TP # 1, Rev. 1
MAR 96-10-02-01, MAR Functional Procedure EF Cavitating Venturis, TP # 3, Rev. 0
MNT-TRMX-00022, Use of Doble Power Factor Test Set, Rev. 0
MP-423, Maintenance of Switchboard Meters and Transducers for Emergency Diesel Generators, Rev. 4
MP-461A, ABB 5 HK Medium Voltage Breaker Refurbishment, Rev. 0
MP-461B, ABB 7.5 HK Medium Voltage Breaker Refurbishment, Rev. 0
PM-101, 4.16 kV and 6.9 kV Switchgear Breakers, Rev. 38.
PM-102, Calibration of Protective Electrical Relays, Rev. 24
PM-119, Maintenance of Electrical Panels and Cabinets, Rev. 22

Completed Test Procedures

MP-423, Maintenance of Switchboard Meters and Transducers for Emergency Diesel Generators, performed for breaker 3210 10/18/03.
PM-101, 4.16 kV and 6.9 kV Switchgear Breakers, performed for breaker 3207 5/28/02 (WO 216548); performed for breaker 3210 6/30/04 (WO 221724); performed for breaker 3220 10/19/03 (WO 241233); performed for breaker 3221 11/25/05 (WO 534470).
PM-102, Calibration of Protective Electrical Relays, performed for MTTR-2 protective relays 10/07/01 (WR 366879, WR 366880, WR 366881); performed for MTTR-2 / MTTR-6 protective relays 10/7/01 (WR 366937); performed for MTTR-2 protective relays 11/3/05 (WR 520297), 11/8/05 (WR 520299, WR 520300, WR 520301); performed 10/7/01 for MTTR-6 protective relays (WR 367360); performed 11/8/05 for MTTR-6 (WR 520296).
PM-102, Calibration of Protective Electrical Relays, performed for breaker 3207 11/13/05 (WO 520222); performed for breaker 3210 10/20/03 (WO 221970); performed for breaker 3211 11/12/05 (WO 522721); performed for breaker 3220 10/20/03 (WO 240407); performed for breaker 3221 11/12/05 (WO 522722).
PM-119, Maintenance of Electrical Panels and Cabinets, performed on DC panel DPDP-1B 10/19/03 (WR 241963).
SP-149, Power-Operated Relief Valve Channel Calibration / Test, performed 11/29/05 (WO 550629).
SP0354A, Monthly Functional Test of the Emergency Diesel Generator EGDG-1A, Sections 4.2 through 4.6, performed 7/11/07, 8/8/07, 9/5/07 (monthly); performed 10/15/03, 11/16/05 (24 month maximum load test).
SP0354B, Monthly Functional Test of the Emergency Diesel Generator EGDG-1B, Sections 4.2 through 4.6, performed 6/27/07, 7/25/07, 8/20/07 (monthly); performed 10/24/03, 11/9/05 (24 month maximum load test).

Design Changes/Modifications

EC 61028, EGDG-1B Fuel Header Modification, Rev. 2
EC 61030, EGDG-1A Fuel Header Modification, Rev. 4
EC 65825, ULSD Evaluation and Diesel Fuel Oil Specification Update, Rev. 1

Design Basis Documents

DBD-61, Enhanced Design Basis Document For The Reactor Coolant System, Rev. 18
DBD-815, Chilled Water System, Rev. 7
EDBD SW 6/11, Enhanced Design Basis for the Nuclear Services Water System, Rev. 17
EDBD RW 6/12, Enhanced Design Basis for the Nuclear Services and Decay Sea Water System, Rev. 12
EDBD EF-EFIC 6/13, Enhanced Design Basis for the Emergency Feedwater System, Rev. 16

Nuclear Condition Reports (NCRs)

135746, EGDG-1B Relay Contact Resistance
135773, EGDG-1B Generator Field Did not Flash During SP-354B
149507-01, Significant Adverse Condition Investigation Report regarding vulnerability of loss of offsite power transformer and backup ES transformer to a single failure in metering and protective relaying circuits
154522, EGDG-1A Start Time was Excessive per SP-354A
175996, Significant Adverse Condition Investigation Report regarding invalid ES actuation.
178139, ES system seal-in anomaly during SP-130
198574, Adverse Condition Investigation regarding inadvertent actuation of sudden pressure relay on Backup ES transformer
210288-20, Adverse Condition Investigation regarding spurious ES channel trip.
227952, Adverse Condition Correct and Trend Form regarding unusual noise from startup transformer
2005100563, 1ABF Material condition is degrading
2005101944, Control Room received 1ND3A 125VDC Ground Annunciator
2005102348, Discharge Test of battery 1Nd1B
2005102356, Failure of the K1 relay to reset, maintains diesel generator field shorted
2006110873, RAT 1B (1NXRB) Gassing trend
2006111881, Battery Charger 1AD1CA ac input breaker tripped
2006112084, Battery Charger 1AD1CA ac input breaker tripped

Work Orders

WO 221729, Breaker 3211 Replacement, performed 8/10/04.
WO 409398, Inspection and PM for ‘A’ Decay Heat Pit Sump Level I (WD-133-LS), performed 5/23/05.
WO 414928, Inspection and PM for ‘B’ Decay Heat Pit Sump Level I (WD-134-LS), performed 1/14/05.
WO 461068, Inspection and PM for Sump Pit Level Switch Check / Calibration (SD-5-LS and SD-6-LS), performed 1/26/06.

WO 478216-01, Remove/Install RB Equip. Hatch, 10/24/05
WR 697507, Infrared inspection of DC Panel DPDP-1B, performed 3/16/06.
WO 756957, Inspection and PM for WD-29-LT/FR/LI, WD-132-LS, WD-30-PT/PI, WD-31-PT / PI, performed 3/22/07.
WR 829101, Infrared inspection of DC Panel DPDP-1B, performed 4/9/07.

Drawings

066-45947E900, Diagram Wiring Electronic, Turbopak Liquid Chilling System, Rev. 14
206-011, Electrical One Line Diagram Composite, Rev. 64
206-013, Electrical One Line Diagram, Generation & Relaying, 6900 V Bus, Rev. 15
206-014, Electrical One Line Diagram, Generation & Relaying, 4160 V Bus, Rev. 11
206-015, Sheet 1, Electrical One Line Diagram, Generation & Relaying, 4160 V Engineered Safeguard Bus, Rev. 17
206-015, Sheet 2, Electrical One Line Diagram, Generation & Relaying, 230 - 4.16 kV Backup Engineered Safeguards Transformer, Rev. 4
206-051, Electrical One Line Diagram, 250 / 125 V DC System, Rev. 16
208-028, ES-A20, Elementary Diagram, Engineered Safeguard, Rev. 19
208-040, MT-14, Elementary Diagram, Breaker 3210, Rev. 24
208-040, MT-14A, Elementary Diagram, Breaker 3210, Rev. 3
208-040, MT-86, Elementary Diagram, Startup Transformer #3 Differential Relay, Rev. 12
208-040, MT-137, Elementary Diagram, Backup ES Transformer Neutral Ground and Ground Differential Relays, Rev. 2
208-040, MT-138, Elementary Diagram, Backup ES Transformer Sudden Pressure Relay, Rev. 2
208-040, MT-139, Elementary Diagram, Backup ES Transformer Differential Relay, Rev. 2
208-040, MT-141, Elementary Diagram, Backup ES Transformer Master Trip Relay ‘A’, Rev. 1
208-047, RC-25, Elementary Diagram, Pressurizer RCT-1 Power Actuated Relief Valve to Reactor Coolant Drain Tank (RCV-10), Rev. 22
308-926, Emergency Diesel Generator Rooms Ventilation System AH-XL, Rev. 3
D8034033, Reactor Coolant Control Loop RC-3, RC3A Schematic (Sheet 4 of 8), Rev. 17
FD-301-601, Nuclear Services Closed Cycle Cooling (SW)
FD-302-001, Symbols, Rev. 43
FD-302-011, Main and Reheat Steam, Rev. 63
FD-302-051, Auxiliary Steam, Rev. 72
FD-302-081, Feedwater, F.W. System Flow Diagram, Sheet 1, Rev. 74
FD-302-081, Feedwater, F.W. System Flow Diagram, Sheet 2, Rev. 63
FD-302-081, Feedwater, F.W. System Flow Diagram, Sheet 3, Rev. 64
FD-302-081, Feedwater, F.W. System Flow Diagram, Sheet 4, Rev. 13
FD-302-082, Emergency Feedwater, E.F. System Flow Diagram, Sheet 1, Rev. 62
FD-302-082, Emergency Feedwater, E.F. System Flow Diagram, Sheet 2, Rev. 18
FD-302-082, Emergency Feedwater, E.F. System Flow Diagram, Sheet 3, Rev. 30
FD-302-281, Emergency Diesel Generator Fuel-Oil Transfer, Rev. 40
FD-302-601, Nuclear Services Closed Cooling, S.W. System Flow Diagram, Sheet 1, Rev. 79
FD-302-601, Nuclear Services Closed Cooling, S.W. System Flow Diagram, Sheet 2, Rev. 62
FD-302-601, Nuclear Services Closed Cooling, S.W. System Flow Diagram, Sheet 3, Rev. 88
FD-302-601, Nuclear Services Closed Cooling, S.W. System Flow Diagram, Sheet 4, Rev. 61

FD-302-601, Nuclear Services Closed Cooling, S.W. System Flow Diagram, Sheet 5, Rev. 102
FD-302-611, Nuclear Services and Decay Sea Water, R.W. System Flow Diagram, Sheet 1, Rev. 97
FD-302-611, Nuclear Services and Decay Sea Water, R.W. System Flow Diagram, Sheet 2, Rev. 15
FD-302-611, Nuclear Services and Decay Sea Water, R.W. System Flow Diagram, Sheet 3, Rev. 20
FD-302-611, Nuclear Services and Decay Sea Water, R.W. System Flow Diagram, Sheet 4, Rev. 10
FD-302-631, Decay Heat Closed Cycle Cooling Train “A” (DC)
FD-302-631, Decay Heat Closed Cycle Cooling Train “B” (DC)
FD-302-641, Decay Heat Removal, (DH)
FD-302-711, Reactor Building Spray (BS)
FD-302-756, Chilled Water, Rev. 56
ROT LP 4-89, AC Electrical Distribution, Figure 1, Rev. 2

Miscellaneous Documents

AR 168361, NRC IN 2005-25 Tin Whisker Problem Resulting in Trip, completed 6/7/07
AR 186375, Unplanned entry into Tech Spec due to CHHE-1A tripping, 12/12/06
AR 240484, CHP-1A/B Min Flows from M-92-0078 and DBD 8/15 do not agree, 9/19/07
BAW-2392P, B&W Design Reactor Vessel Internals Baffle Bolt Failure Structural Analysis and Safety Assessment, August 2001
Coltec Engineering Report VTS985960919, Crystal River Unit 3 Diesel Generator Loading Analysis, 9/19/96
CR3 10CFR50 Appendix R Fire Study, Rev 12
Crystal River Unit 3 Neutron Noise Analysis CY-15 333 EFPD, 3/2007
Doble test report for transformer MTTR-6 dated 10/4/01
Doble test report for transformer MTTR-2 dated 10/3/01
FPC to NRC letter on Emergency Feedwater System Program, dated November 18, 1983
FPC to NRC letter on Emergency Feedwater System Evaluation, dated August 22, 1983
FPC to NRC letter on Emergency Feedwater System Evaluation, dated November 19, 1982
FPC to NRC Letter 3F0187-14, W. Wilgus to Document Control Desk, "Generic Letter 81-21 and NUREG-0737, Item II.E.1.1," January 15, 1987.
NRC to FPC Letter 3N0388-28, Natural Circulation Cooldown Clarification, March 25, 1988.
FPC to NRC letter on Supplemental Response to Bulletin 88-04:
Potential Safety-Related Pump Loss, dated April 17, 1989
Engineering Change (ED) 0000068121R0, Operability Evaluation For AR 248036 (EFP-2 Instr/Gov Error), Rev. 0
Impell Report 03-0920-1186, Pipe Rupture Analysis Criteria Outside the Reactor Building, Crystal River, Unit 3, Rev. 0
Ingersoll-Rand Minimum Flow Evaluation for Emergency Feedwater Pumps Turbine Drive and Motor/Gear Drive, dated April 28, 1989
Neutron Noise Measurements and Report Summary, 8/9/07

NRC Safety Evaluation of Emergency Feedwater System Upgrade (NUREG 0737 Item II.E.1.1), dated May 1, 1984
NRC Safety Evaluation of Third 10-Year Interval Inservice Testing Program for Pumps and Valves, dated January 22, 1999
Nuclear Safety Group Industry Operating Experience Review Form for NRC Information Notice 90-25, Loss of Vital AC Power with Subsequent Reactor Coolant System Heat-up, 5/4/90
Nuclear Safety Group Industry Operating Experience Review Form for NRC Information Notice 90-25, Loss of Vital AC Power with Subsequent Reactor Coolant System Heat-up, Supplement 1, 3/22/91
PC-3-C99-2837, Precursor Card Report evaluating applicability of Foxboro Part 21 report regarding tin whiskers on contact output isolator relays, closed 9/29/99
Rockwell Automation Bulletin 700-P and 700-PK Direct Drive Convertible Contact Cartridge Relays
Summary of dissolved gas analysis results for Transformer MTTR-6, 8/25/94 to 5/9/07
Summary of dissolved gas analysis results for Transformer MTTR-2, 2/24/95 to 11/27/06
Transformer Oil Analysis Trending Basis (licensee's summary of acceptance criteria established by IEEE Std C57.104-1991 and Nuclear Electric Insurance Limited (NEIL) Loss Control Standards Section 8-2C).
Woodward Governor product specification sheet 36700, Rev. G
York Marine Systems Engineering Services Report for Florida Power Corporation, dated May 6, 1998

Corrective Action documents initiated due to CDBI activity:

NCR 246196, Labeling error on EOP/AP equipment
NCR 246442, EFP-3 Duplex Fuel Filter Valve Position Evaluation
NCR 246446, Revise M96-0063 to reflect I85-0005 for 4596 Gallons/FT Value
NCR 246605, EFP-2 and EFP-3 Pump Performance due to variations in RPM
NCR 246632, Tornado protection for EFP-2 exhaust
NCR 246634, Flourescent lights over batteries
NCR 246635, Question on “RC INVENTORY LOW” Setpoint
NCR 246864, Response to RIS 2006-023 was less than adequate
NCR 247402, Labeling for emergency feedwater motor control center 3B was not labeled to current licensee standards with respect to color code
NCR 247409, Evaluate AP-1040 for use of BWST transfer to RB sump
NCR 247418, Evaluate quality record retention for OP-103H sheets
NCR 247663, F99-0004 time to boil calculation error in vent diameter
NCR 247674, Evaluate documentation of system P/T
NCR 247778, Ladder tied off for scheduled work not labeled
NCR 247815, Calculation M97-0133 has unclear design inputs and assumptions
NCR 247839, Technical specifications bases does not match calculation results
NCR 247908, Question on chiller operation during design basis accident
NCR 248036, EFP-2 IST test acceptance criteria non-conservative
NCR 248137, Enhancement opportunity for NAS ISI/IST assessment outline
NCR 248232, Air filter unit installed in EFT-2 room without proper evaluation
NCR 248659, Document acceptability of observed condition of 480V cubicle doors
NCR 248829, EOP-14 Enhancement for starting control complex chiller

NCR 248830, EFP-2 minimum flow reference document enhancement
NCR 248884, FSAR Section 5.4.3.2.2.f needs additional clarification
NCR 248980, AP-1040 implementation training enhancement
NCR 249072, NEIL trigger level exceeded for ethane on startup transformer
NCR 249077, Heating value in DBD 6/15 misleading
NCR 249181, EFP-1 test acceptance criteria non-conservative
NCR 249191, Inadequate calculation support for assumed maximum SW
NCR 249251, Non-conservative assumptions in calculation M97-0133
NCR 249328, EFP-2 minimum flow test (SP349B) and TS SR 3.7.5.2